1.0 Title:

PTP/PSL CORE DAMAGE ASSESSMENT - OFF-SITE EMERGENCY ORGANIZATION

2.0 Approval and List of Effective Pages:

2.1 Approvals

Reviewed by J Maisler Emergency Planning Manager
21 August, 1987

Approved by: [illegible] Chief Engineer

Power Plants 8-21-87, 1987.

2.2 List of Effective Pages

Page	Date
1 through 4 inclusive	August 18, 1987
Appendix I p. 1 through 77	August, 1987
Appendix II p. 1 through 30	December, 1986

Superseded pages On Rev. 6 EPIPs
5/20/95
9506220037
6/16/95

3.0 Scope:

3.1 Purpose:

This procedure identifies the responsibility and methodology to perform core damage assessment for both the Turkey Point and St. Lucie Plants. Methods for estimating core damage assessment are based upon post accident radionuclide concentrations within the reactor coolant system and containment, and auxiliary indicators, including core exit thermocouple, hydrogen, and containment high range radiation monitors.

An estimate of core damage can then be used to assist in evaluating protective action recommendations, severity of plant conditions, and/or recovery plan operations.

.2 Discussion:

The Off-Site Emergency Organization provides an expanded emergency response capability to assist the plant in administration, communications, engineering, technical support, security, and public relations. This organization, which consists of the Emergency Technical Manager and his staff provides engineering and technical support at the request of the Emergency Control Officer (ECO) and/or Recovery Manager (RM). Specifically, this support includes estimating core damage, using the methodology provided in the applicable appendix, to differentiate among four major fuel conditions. These are: No Damage, Fuel Overheating, Cladding Failures, Core Melt

AO45
1/1

The methodology attached is site specific and is based upon Westinghouse and Combustion Engineering Owners Group (generic) core damage assessment guidelines.

3.3 Authority

At the request of the ECO or RM, the Emergency Technical Manager will direct his staff to perform core damage assessment using the applicable guidelines in the attached appendix. The ECO or RM will request that appropriate input parameters be provided by the plant in order for the ETM's staff to perform the assessment.

4.0 Precautions

4.1 The assessment of core damage obtained by using the attached methodology is only an estimate. The techniques employed are only accurate to locate the core condition within one or more of the 10 categories of core damage described in the methodology.

4.2 Core damage assessment using indicators that are readily available (e.g., containment high range radiation monitor) represent only preliminary estimates. Other plant indicators (e.g., radionuclide concentrations) should be obtained to improve upon estimation of core damage.

4.3 Measurements obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage. If deemed necessary, these pertinent indicators should be measured within a minimum time period particularly during rapidly changing conditions.

It is recommended that measurements be made, if possible, when plant conditions stabilize.

5.0 Responsibilities

5.1 The Emergency Control Officer or Recovery Manager will request the Emergency Technical Manager to perform core damage assessment using the methodology attached.

5.2 The Emergency Control Officer or Recovery Manager will request the plant to provide appropriate data in order to perform the assessment.

5.3 The Emergency Technical Manager will direct his staff to perform core damage assessment (when staffed accordingly) using the attached methodology. A QA approved computer program may be used, if available.

5.4 The RM will use this information as deemed appropriate in evaluating severity of plant conditions, protective action recommendations, and/or recovery operation.

6.0 References

6.1 Turkey Point Plant Radiological Emergency Plan.

6.2 St. Lucie Plant Radiological Emergency Plan.

6.3 Procedure 1101, Duties of the Emergency Control Officer, Off-Site Emergency Organization.

6.4 Procedure 1102, Duties of the Recovery Manager, Off-Site Emergency Organization.

6.5 Procedure 1105, Duties of the Emergency Technical Manager, Off-Site Emergency Organization.

6.5 Appendix I, St. Lucie Units 1 and 2 Core Damage Assessment Guidelines, Rev. 2, August, 1987.

6.7 Appendix II Turkey Point Units 3 & 4 Core Damage Assessment Guidelines, Rev. 0, December, 1986.

7.0 Records

All information used to estimate core damage, including appropriate worksheets will be documented by the Emergency Technical manager or his staff designee.

8.0 Instructions

8.1 The ECO or RM can request that an estimate of core damage be performed by the Emergency Technical Manager's staff, when deemed appropriate. The ECO or RM will request pertinent data from the plant to perform the assessment.

8.2 The Emergency Technical Manager will direct his staff designee to perform the estimate using the methodology provided in the applicable appendix.

8.3 The staff designee will perform the estimate using this methodology and the assistance of a QA approved computer program (when available). Available pertinent plant data needed to perform the assessment will be provided to the staff designee through the ECO or RM.

8.4 All pertinent data available should be used in estimating core damage. This includes radionuclide data and auxiliary indicators, including core exit thermocouple, hydrogen, and containment high range radiation monitor.

8.5 Results in terms of fuel condition should be provided to the Emergency Technical Manager and Recovery Manager (and/or Emergency Control Officer) as timely as possible.

8.6 Updated estimates to core damage may be requested periodically by the ECO or RM as plant conditions change and/or stabilize. These updates should be performed using the most recent available data and the methodology. Results should continue to be reported to the ETM and RM (or ECO).

APPENDIX I
TO
GO EPIP 1302
"CORE DAMAGE ASSESSMENT"
FOR
ST. LUCIE UNITS 1 & 2

The core damage assessment methodology is divided into four main sections. Each section contains its own table of contents and list of enclosures. A fifth section, summary of results, has been added as a guideline for a comprehensive evaluation of results.

INDEX

SECTION A

RADIOLOGICAL ANALYSIS OF SAMPLES

TABLE OF CONTENTS

LIST OF ENCLOSURES

TABLE OF CONTENTS (CONTINUED)

1.0 PURPOSE

This section provides a method under post-accident plant conditions to determine the type and degree of reactor core damage which may have occurred by using fission product isotopes measured in samples obtained from the Post Accident Sampling System (PASS). There are three factors considered in this section which are related to the specific activity of the samples. These are the identity of those isotopes which are released from the core, the respective ratios of the specific activity of those isotopes, and the percent of the source inventory at the time of the accident which is observed to be present in the samples. The resulting observation of core damage is described by one or more of the ten categories of core damage in Enclosure A1.

2.0 REFERENCES

2.1 Development of the comprehensive procedure guideline for core damage assessment, CE Owners Group Task 467, July 1983.

2.2 Post Accident Sampling System Operating Procedures. I-C-112, PSL-1 and 2-C-113, PSL-2.

3.0 DEFINITIONS

3.1 Fuel Damage: For the purpose of this methodology, fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the reactor coolant starting with a penetration in the zircaloy cladding. The type of fuel damage as determined by this methodology is reported in terms of four (4) major categories which are: no damage; cladding failure; fuel overheat; and fuel melt. Each of these categories is characterized by the identity of the fission products released, the mechanism by which they are released, and the source inventory within the fuel rod from which they are released. The degree of fuel damage is measured by the percent of the fission product source inventory which has been released into fluid media and therefore available for immediate release to the environment. The degree of fuel damage as determined by this methodology is reported in terms of three levels which are: initial; intermediate; and major. This results in a total of ten possible categories as characterized in Enclosure A1.

3.2 Source Inventory: The source inventory is the total quantity of fission products expressed in curies of each isotope present in either source: the fuel pellets or the fuel rod gas gap.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using the methodology in this section is only an estimate. The techniques employed in this section are only accurate to locate the core condition within one or more of the 10 categories of core damage described in Enclosure A1. The methodology is based on radiological data. Other plant indications may be available which can improve upon estimation of core damage. These include incore temperature indicators, the total quantity of hydrogen released from zirconium degradation and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 The methodology in this section relies upon samples taken from multiple locations inside the containment building to determine the total quantity of fission products available for release to the environment. The amount of fission products present at each sample location may be changing rapidly due to transient plant conditions. Therefore, it is recommended that the samples should be obtained within a minimum time period and if possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

4.3 A number of factors influence the reliability of the chemistry samples upon which this section is based. Reliability is influenced by the ability to obtain representative samples due to incomplete mixing of the fluids and equipment limitations. The accuracy achieved in the radiological analyses are also influenced by a number of factors. The equipment employed in the analysis may be subjected to high levels of radiation exposure over extended periods of time. Chemists are recommended to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources and they may not result from a uniform distribution of the sample fluid. Cooling or reactions may take place in the long sample lines. Therefore, the results obtained may not be representative of plant conditions. To minimize these effects multiple samples should be obtained over an extended time period from each location, conditions permitting.

5.0 INSTRUCTIONS

5.1 Obtain the following plant indications and source of indication. Record on Enclosure A2. Because of transient conditions the values should be recorded as close as possible to the time at which the radiological samples are obtained.

5.1.1 Reactor Coolant System: SOURCE:

Pressure	________ PSIG	________
Temperature (T_{avg})	________ °F	________
Reactor Vessel Level Shows: (Full, Void or Below Recorder)	________	________
Pressurizer Level	________ %	________

5.1.2 Containment Building:

Atmosphere Pressure	________ PSIG	________
Atmosphere Temperature	________ °F	________

5.1.3 Prior 30 days Power History

POWER, PERCENT	DURATION, DAYS
________	________
________	________
________	________

Estimated average power level during last 30 days:____%

Estimated average power level during last 4 days:____%

5.1.4 Time of Reactor Trip

Date: ____________ Time: ____________

5.1.5 Change in Refueling Water Tank (RWT) Volume
____________gal. Time: ____________

5.1.6 Change in Boric Acid Makeup Tank (BAMT) volume
____________gal. Time: ____________

5.1.7 Safety Injection Tanks injected (yes/no): ____

5.2 Select the most appropriate sample locations required for core damage assessment using the guidelines provided in Enclosure A3.

5.3 Obtain and analyze the selected samples for fission product specific activity using the procedures for Post Accident Sample System operation described in Reference 2.2. Record the required sample data.

corrected to Standard Temperature and Pressure (STP), and time of sample collection on Enclosure A2. All of the isotopes listed in the enclosure may not be observed in the sample.

5.4 Correct the sample specific activity at STP for decay back to the time of reactor trip which is recorded in step 5.1.4 using the following equation. Enclosure A4 is provided as a worksheet.

$$A_o = \frac{A}{e^{-\lambda t}}$$

Where: A_o = the specific activity of the sample corrected back to the time of reactor trip, uci/cc.

A = the measured specific activity, uci/cc

λ = the radioactive decay constant, 1/sec.

t = the time period from reactor trip to sample analysis, sec.

5.5 Identification of the Fission Product Release Source.

5.5.1 Calculate the following ratios for each noble gas and iodine isotope using the specific activities obtained in step 5.4. Enclosure A5 is provided as a worksheet.

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Specific Activity}}{\text{Xe 133 Specific Activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Isotope Specific Activity}}{\text{I-131 Specific Activity}}$$

5.5.2 Determine the source of release (gas gap or fuel pellet) by comparing the results obtained in step 5.5.1 to the predicted ratios provided in Enclosure A5. An accurate comparison is not anticipated. Within the accuracy of this methodology it is appropriate to select as the source of release that ratio which is closest to the value obtained in step 5.5.1.

5.6 Quantitative Release Assessment

5.6.1 Calculate the total quantity of fission products available for release to the environment. Enclosure A6 is provided as a worksheet.

5.6.1.1 If the water level in the reactor vessel recorded in step 5.1.1 indicates that the vessel is full, the quantity of fission products found in the reactor coolant is calculated by the following equation:

Total Activity, $A_{T,\ RCS}$ (Ci) =

A_o (uci/cc) X RCS volume X 1.0 (-6)

Where: A_o = the specific activity of the reactor coolant sample corrected to time of Reactor trip obtained in step 5.4, uci/cc.

RCS Volume = in units of cc, the full reactor coolant system water volume corrected to standard temperature and pressure using Enclosure A8.

RCS Volume = Water Volume X density ratio (Enclosure A8).

SL1 Water Volume is 2.945 (8) cc
SL2 Water Volume is 2.883 (8) cc

1.0 (-6) = Ci/uci

5.6.1.2 If the water levels in the reactor vessel and pressurizer recorded in step 5.1.1 indicates that a steam void is present in the reactor vessel, then the quantity of fission products found in the reactor coolant is also calculated by step 5.6.1.1. However, it must be recognized that the value obtained will overestimate the actual quantity released. Therefore, this sample should be repeated at such time when the plant operators have removed the void from the reactor vessel.

5.6.1.3 If the water level in the reactor vessel recorded in step 5.1.1 is below the low end capability of the indicator, it is not possible to determine the quantity of fission products from this sample because the volume of water in the reactor coolant system is

unknown. Under this condition, assessment of core damage is obtained using the containment sump sample.

5.6.2 The quantity of fission products found in the containment building sump is determined as follows:

5.6.2.1 The water volume in the containment building sump is determined from the sum of the following sources as applicable:

________(gal) RCS Volume

+ ________(gal) the injected S.I.T. tanks volume (step 5.1.7)

+ ________(gal) the Delta volume change in B.A.M.T. (step 5.1.6)

+ ________(gal) the Delta volume change of the RWT (step 5.1.5)

Sump Volume = ________(gal) Total gal X 3785 cc/gal = ________cc

Maximum Values in gallons for each unit from applicable FSAR Ch. 6.

	SL1	SL2
RCS Volume (cold)(gal)	53,300	57,400
Safety Injection Tanks (SIT) volume (gal)	34,049	46,564

Caution: Values reported indicate maximum volumes in applicable FSAR. Water volume in containment building sump is only applicable in recirculation mode.

5.6.2.2 The quantity of fission products in the sump is calculated by the following equation:.

Total Activity, $A_{T,\ sump}$ (Ci) = A_o (uci/cc) X Sump Volume (cc) X 1.0 (-6)

Where:. A_o = the specific activity of the containment sump sample corrected to the time of reactor trip obtained in step 5.4, uci/cc.

5.6.3 The quantity of fission products found in the containment building atmosphere is determined as follows:

5.6.3.1 The volume of gas in the containment building, at the time of the accident is corrected to standard temperature and pressure using the following equation:

$$\text{Gas Volume (STP)} = \text{Gas Volume} \times \frac{(P_2 + P_1)}{P_2} \times \frac{(T_2 + 460)}{(T_1 + 460)}$$

Where:

Gas Volume = 7.096×10^{10} cc

T_1, P_1 = Containment Atmosphere temperature and pressure recorded in step 5.1.2

T_2, P_2 = Standard temperture, $32^{o}F$, and Standard pressure 14.7 psia.

5.6.3.2 The quantity of fission products (Ci) in the containment atmosphere is calculated by:

Total Activity $A_{T\ cont}$ (Ci) = Ao (uci/cc) X Gas Volume (STP,cc) X $1.0^{(-6)}$ (Ci/uci)

Where: A_o = The specific activity of the containment atmosphere sample corrected to Standard Temperture and Pressure and for decay to the time of reactor trip. (Enclosure A4)

5.6.4 The total quantity of fission products available for release to the environment is equal to the sum of the values obtained from each sample location (liquid and gas). Enclosure A7 is provided as a worksheet.

5.7 Plant Power Correction

The quantitative release of the fission products is expressed as the percent of the source inventory at the time of the accident. The equilibrium source inventories are to be corrected for plant power history.

5.7.1 To correct the source inventory for the case in which plant power level has remained constant for a period greater than four radioactive half lives the following procedure is employed. --Enclosure A9 is provided as a worksheet.

5.7.1.1 The fission products are divided into two groups based upon the radioactive half lives. Group 1 isotopes are to be employed in the case where core power had not changed greater than $\pm$ 10% within the last 30 days prior to the reactor trip. Group 2 isotopes are to be employed in the case where core power had not changed greater than $\pm$ 10% within the last 4 days prior to the reactor trip

5.7.1.2 The following equation may be applied to the fission product Group which meets the criteria stated in 5.7.1.1.

Group 1 Power Correction Factor =

$$\frac{\text{Average Power Level For Prior 30 Days}}{100}$$

Group 2 Power Correction Factor =

$$\frac{\text{Average Power Level For Prior 4 Days}}{100}$$

5.7.2 To correct the source inventory for the case in which plant power level has not remained constant prior to reactor trip, the following equation is employed. The entire 30 days power history should be employed. Enclosure A10 is provided as a worksheet.

Power Correction Factor =

$$\frac{\sum_j P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^o}}{100}$$

Where: P_j = Steady reactor power in period j

t_j = duration of period j (sec)

t_j^o = time from end of period j to reactor trip (sec)

λ = decay constant (Enclosure A4)

5.8 Comparison of Measured Data with Source Inventory

The total quantity of fission products available for release to the environment obtained in step 5.6.4 (Enclosure A7) is compared to the source inventory corrected for plant power history obtained in step 5.7 (Enclosure A9 or A10). This comparison is made by dividing the total quantity available for release by the power corrected source inventory. Enclosure A11 is provided as a worksheet.

5.9 Core Damage Assessment

The conclusion on core damage is made using the three parameters developed above. These are:

1. Identification of the fission product isotopes which most characterize a given sample, step 5.3 (Enclosure A2).

2. Identification of the source of the release, step 5.5 (Enclosure A5).

3. Quantity of fission product available for release to the environment expressed as a percent of source inventory, step 5.8 (Enclosure A11).

Knowledgeable judgement is used to compare the above three parameters to the definitions of the 10 NRC categories of fuel damage found in Enclosure A1. Core damage is not anticipated to take place uniformly. Therefore when evaluating the three parameters listed above the methodology in this section is anticipated to yield a combination of one or more of the 10 categories defined in Enclosure A1. These categories will exist simultaneously.

The type of core damage is described in terms of the 10 NRC categories defined in Enclosure A1. The degree of core damage is described as the percent of the fission products in the source inventory at the time of the accident which is now in the sampled fluid and therefore available for release to the environment.

ENCLOSURE A1

RADIOLOGICAL CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

NRC CATEGORY OF FUEL DAMAGE	MECHANISM OF RELEASE	SOURCE OF RELEASE	CHARACTERISTIC ISOTOPE	RELEASE OF CHARACTERISTIC ISOTOPE EXPRESSED AS A PERCENT OF SOURCE INVENTORY
1. No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137, Rb 88	Less than 1
2. Initial Cladding Failure		Gas Gap		Less than 10
3. Intermediate Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Xe-131m, Xe-133, I-131, I-133	10 to 50
4. Major Cladding Failure		Gas Gap		Greater than 50
5. Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Cs-134, Rb-88 Te-129, Te-132	Less than 10
6. Intermediate Fuel Pellet Overheating		Fuel Pellet		10 to 50
7. Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet		Greater than 50
8. Fuel Pellet Melt		Fuel Pellet		Less than 10
9. Intermediate Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet	Ba-140, La-140, La-142, Pr-144	10 to 50
10. Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

ENCLOSURE A2 (sheet 1 of 2)

INPUT PARAMETERS (ref. step 5.1)

Unit: ____________

Reactor Coolant System:

		SOURCE:
Pressure	_______ PSIG	_______
Temperature (T_{avg})	_______ ^{o}F	_______
Reactor Vessel Level Shows (Circle One)	Full Void Below Recorder	_______
Pressurizer Level	_______ %	_______

Containment Building:

Atmosphere Pressure	_______ PSIG	_______
Atmosphere Temperature	_______ ^{o}F	_______

Prior 30 Days Power History

Power, Percent	Duration, Days
_______________	_______________
_______________	_______________
_______________	_______________
_______________	_______________
_______________	_______________

Estimated Average Power Level During Last 30 Days ________%
Estimated Average Power Level During Last 4 Days ________%

Time of Reactor Trip: Date: _________ Time: _______

Change in volume of RWT ________gal. Time: __________

Change in volume of BAMT________gal. Time: __________

SIT injected (yes/no): ____

ENCLOSURE A2 (sheet 2 of 2)

INPUT PARAMETERS - RADIONUCLIDE DATA (ref. step 5.3)

UNIT: __________
SAMPLE NUMBER: __________
SAMPLE LOCATION (RCS, SUMP, CONTAINMENT): __________
TIME OF SAMPLE COLLECTION: __________

ISOTOPE	MEASURED SPECIFIC ACTIVITY @ STP A(uCi/cc)
Kr 87	
Xe-131m	
Xe-133	
I-131	
I-132	
I-133	
I-135	
Cs-134	
Rb-88	
Te-129	
Te-132	
Sr-89	
Ba-140	
La-140	
La-142	
Pr-144	

NOTE: NI means not identified

Performed by: __________

Date: __________

ENCLOSURE A3

SAMPLE LOCATIONS RECOMMENDED FOR CORE DAMAGE ASSESSMENT (ref. step 5.2)

ACCIDENT SCENARIO KNOWN	RCS HOT LEG	RCS PRESSURIZER	CONTAINMENT SUMP *	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM	STEAM GENERATOR SECONDARY
Small Break LOCA, Reactor Power >1%	YES	YES	---	YES	YES	---
Small Break LOCA, Reactor Power <1%	YES	YES	---	---	YES	---
Small Steam Line Break	YES	YES	---	---	---	---
Large Break LOCA, Reactor Power >1%	YES	---	YES	YES	YES	---
Large Break LOCA, Reactor Power <1%	---	---	YES	YES	YES	---
Large Steam Line Break	YES	---	---	YES	---	---
Steam Generator Tube Rupture	YES	---	---	YES	---	YES

* available only on recirculation

ENCLOSURE A4 (REF. STEPS 5.3 AND 5.4)

RECORD OF MEASURED SPECIFIC ACTIVITY (DECAY CORRECTED)

UNIT: ______________
TIME OF REACTOR TRIP, ENCLOSURE A2: ______________
SAMPLE NUMBER: ______________
SAMPLE LOCATION (RCS, SUMP, CONTAINMENT): ______________
TIME OF SAMPLE COLLECTION: ______________

ISOTOPE	DECAY CONSTANT, λ(1/sec)	MEASURED SPECIFIC ACTIVITY @ STP A (uci/cc)	DECAY CORRECTED SPECIFIC ACTIVITY, Ao (uci/cc)
Kr 87	1.5 (-4)		
Xe 131m	6.7 (-7)		
Xe 133	1.5 (-6)		
I 131	9.9 (-7)		
I 132	8.4 (-5)		
I 133	9.3 (-6)		
I 135	2.9 (-5)		
Cs 134	1.1 (-8)		
Rb 88	6.5 (-4)		
Te 129	1.7 (-4)		
Te 132	2.5 (-6)		
Sr 89	1.6 (-7)		
Ba 140	6.3 (-7)		
La 140	4.8 (-6)		
La 142	1.2 (-4)		
Pr 144	6.7 (-4)		

$Ao = \frac{A}{e^{-\lambda t}}$, where A and λ are as above, and t = time period in seconds from reactor trip to sample collected.

Performed by: ______________

Date: ______________

NOTE: NI means not identified

ENCLOSURE A5 (ref. step 5.5)

RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

Unit: ____________

Sample Number: ____________

Location: ____________

ISOTOPE	DECAY CORRECTED SPECIFIC ACTIVITY (ENCLOSURE A4), uci/cc	CALCULATED ISOTOPE RATIO*	Activity Ratio in FUEL PELLET INVENTORY**	ACTIVITY RATIO IN GAS GAP INVENTORY **	IDENTIFIED SOURCE (GAS GAP OR FUEL PELLET)
Kr 87			0.2	< 0.001	
Xe 131m			0.003	0.001 - 0.003	
Xe 133		1.0	1.0	1.0	NA
I 131		1.0	1.0	1.0	NA
I 132			1.4	0.01 - 0.05	
I 133			2.0	0.5 - 1.0	
I 135			1.8	0.1 - 0.5	

NA = NOT APPLICABLE

* Noble Gas Ratio = $\frac{\text{Decay Corrected Noble Gas Specific Activity}}{\text{Decay Corrected Xe 133 Specific Activity}}$

Iodine Ratio = $\frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I-131 Specific Activity}}$

** Table 3.3 of Reference 2.1

Performed By: ____________

Date: ____________

ENCLOSURE A6 (Sheet 1 of 2)

QUANTITATIVE RELEASE ASSESSMENT WORKSHEET (ref. step 5.6)

RCS ACTIVITY ($A_{T,\ RCS}$)

RCS T_{avg} __________ ^{o}F (ref. step 5.1.1, Enclosure A2)

Vessel Level Indication (Full, Void, Below Recorder)
__________ (ref. step 5.1.1, Enclosure A2)

IF FULL OR VOID, perform the following calculation for each isotope measured:

$A_{T,\ RCS}$ (C_i) = Ao (uci/cc) X RCS Volume X 1.0 (-6) (Ci/uci)

Where: Ao = decay corrected specific activity of RCS sample (Enclosure A4)

RCS volume = Water Volume X Density Ratio at RCS Tavg (Enclosure A8). PSL1 water volume is 2.945 (8) cc and PSL2 water volume is 2.388 (8) cc.

Enter results in Enclosure A7 (A_T, RCS)

IF BELOW RECORDER, Use Containment Sump Calculation Below.

SUMP ACTIVITY ($A_{T,\ sump}$)

Determine sump water volume by adding the following (ref. step 5.6.2).

				SL1	SL2
RCS Volume	=	__________	gal	58,300	57,400
SIT Injected Volume	= +	__________	gal	34,049	46,564
BAMT Injected Volume	= +	__________	gal	(Enclosure A2)	
RWT Volume Change	= +	__________	gal	(Enclosure A2)	

V_s = Total Sump Volume = __________ gal x 3785 cc/gal= ______ cc

$A_{T,\ sump}$ = A_o uci/cc) X Vs X 1.0 (-6) (Ci/uci)

Where A_o = decay corrected specific activity of SUMP sample (Enclosure A4)

Enter results in Enclosure A7 ($A_{T,\ sump}$)

ENCLOSURE A6 (Sheet 2 of 2)

CONTAINMENT ACTIVITY ($A_{T\ cont}$)

Calculate Containment Volume in cc, including pressure and temperature corrections (Ref. step 5.6.3)

V_c = Containment Volume (cc) =

$$7.096\ (10) \cdot \times \frac{14.7 + P1}{14.7} \times \frac{32 + 460}{T1 + 460}$$

Where: P1 = Containment pressure in psig (ref. step 5.1.2, Enclosure A2)
T1 = Containment temperature in ^{o}F (ref. step 5.1.2, Enclosure A2)

$A_{T,\ cont}$ = Ao (uci/cc) X V_c X 1.0 (-6)

Where: Ao = decay corrected specific activity for containment sample (Enclosure A4)

Enter results in Enclosure A7 ($A_{T,\ cont}$)

Performed by: ______________

Date: ____________________

ENCLOSURE A7

RECORD OF CORE RELEASE INVENTORY (ref. step 5.6.4)

UNIT: __________

ISOTOPE	REACTOR COOLANT SAMPLE NUMBER, $A_{T, RCS}$ (Ci) (ENCLOSURE A5) +	CONTAINMENT SUMP SAMPLE NUMBER, $A_{T, SUMP}$ (Ci) (ENCLOSURE A6) +	CONTAINMENT ATMOSPHERE SAMPLE NUMBER, $A_{T, CONT}$ (Ci) (ENCLOSURE A6) =	TOTAL QUANTITY (Ci)
Kr 87				
Xe 131m				
Xe 133				
I 131				
I 132				
I 133				
I 135				
Cs 134				
Rb 88				
Te 129				
Te 132				
Sr 89				
Ba 140				
La 140				
La 142				
Pr 144				

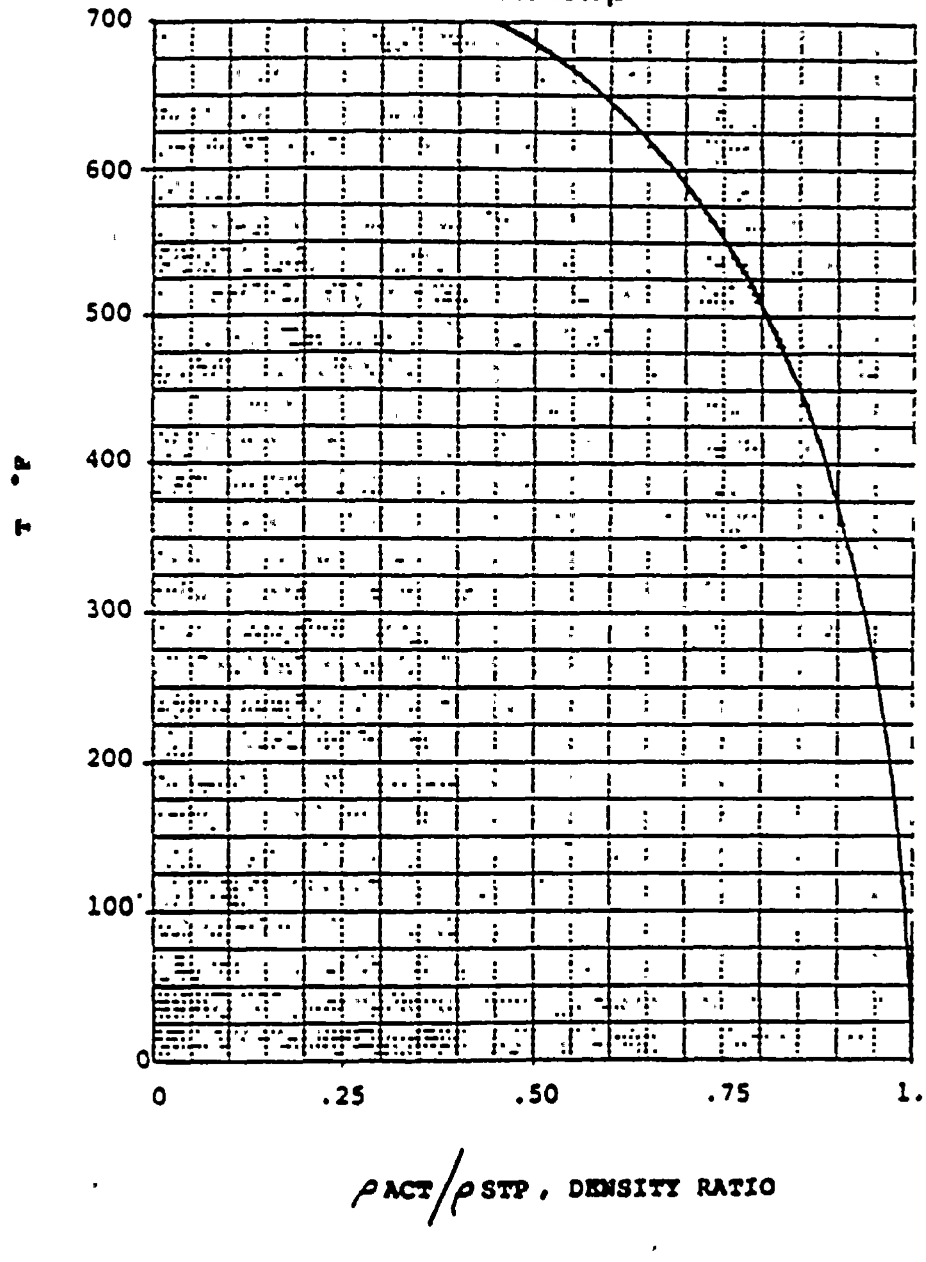
Total Quantity (Ci) = $A_{T, RCS} + A_{T, SUMP} + A_{T, CONT}$

Performed by: __________

Date: __________

ENCLOSURE A8

RATIO OF H_2O DENSITY TO H_2O DENSITY
AT STP vs. TEMPERATURE



ENCLOSURE A9-PSL1 (ref. step 5.7)

RECORD OF STEADY STATE POWER CORRECTION

UNIT: ______________ Performed By: ______________

AVERAGE 30 DAYS POWER LEVEL: ______________ Date: ______________
AVERAGE 4 DAYS POWER LEVEL: ______________

ISOTOPE	FUEL HISTORY GROUPING	POWER CORRECTION FACTOR X	SL1 EQUILIBRIUM SOURCE INVENTORY* =	CORRECTED SOURCE INVENTORY
GAS GAP INVENTORY				
Kr 87	2		6.5 (0)	
Xe 131m	1		4.6 (4)	
Xe 133	1		1.3 (7)	
I 131	1		7.0 (6)	
I 132	2		7.7 (3)	
I 133	2		6.7 (6)	
I 135	2		1.1 (6)	
FUEL PELLET INVENTORY				
Kr 87	2		3.2 (7)	
Xe 131m	1		4.9 (5)	
Xe 133	1		1.5 (8)	
I 131	1		7.6 (7)	
I 132	2		1.1 (8)	
I 133	2		1.5 (8)	
I 135	2		1.4 (8)	
CS 134	1		1.2 (7)	
Rb 88	2		4.8 (7)	
Te 129	2		2.5 (7)	
Te 132	1		1.2 (8)	
Sr 89	1		6.6 (7)	
Ba 140	1		1.4 (8)	
La 140	1		1.4 (8)	
La 142	2		1.5 (8)	
Pr 144	2		9.6 (7)	

Corrected source Inventory = Power Correction Factor X Equilibrium Source Inventory

*Values from Table 3.4 and 3.5 of Reference 2.1

$$\text{Group 1 Power Correction, Factor} = \frac{\text{Average Level for Prior 30 days}}{100}$$

$$\text{Group 2 Power Correction, Factor} = \frac{\text{Average Level for Prior 4 Days}}{100}$$

ENCLOSURE A9-PSL 2 (ref. step 5.7)

RECORD OF STEADY STATE POWER CORRECTION

UNIT: ____________ Performed By: ____________

AVERAGE 30 DAYS POWER LEVEL: ____________ Date: ____________
AVERAGE 4 DAYS POWER LEVEL: ____________

ISOTOPE	FUEL HISTORY GROUPING	POWER CORRECTION FACTOR X	SL 2 EQUILIBRIUM SOURCE INVENTORY* =	CORRECTED SOURCE INVENTORY
GAS GAP INVENTORY				
Kr 87	2		5.5 (0)	
Xe 131m	1		4.6 (4)	
Xe 133	1		1.3 (7)	
I 131	1		7.0 (6)	
I 132	2		7.7 (3)	
I 133	2		6.7 (6)	
I 135	2		1.1 (6)	
FUEL PELLET INVENTORY				
Kr 87	2		3.2 (7)	
Xe 131m	1		4.9 (5)	
Xe 133	1		1.5 (8)	
I 131	1		7.6 (7)	
I 132	2		1.1 (8)	
I 133	2		1.5 (8)	
I 135	2		1.4 (8)	
CS 134	1		1.2 (7)	
Rb 88	2		4.8 (7)	
Te 129	2		2.5 (7)	
Te 132	1		1.2 (8)	
Sr 89	1		6.6 (7)	
Ba 140	1		1.4 (8)	
La 140	1		1.4 (8)	
La 142	2		1.6 (8)	
Pr 144	2		9.6 (7)	

Corrected source Inventory = Power Correction Factor X Equilibrium Source Inventory

*Values from Table 3.4 and 3.5 of Reference 2.1

Group 1 Power Correction, Factor = $\frac{\text{Average Level for Prior 30 days}}{100}$

Group 2 Power Correction, Factor = $\frac{\text{Average Level for Prior 4 Days}}{100}$

ENCLOSURE A10-PSL1 (ref. step 5.7.2)

RECORD OF TRANSIENT POWER CORRECTION

UNIT: ______________________________ Performed By: __________

Date: ____________________

Prior 30 Days Power History:

POWER, %	DURATION, Days (t_j)	TIME TO TRIP, Days (t_j^o)

ISOTOPE	POWER CORRECTION FACTOR	X	SL1 EQUILIBRIUM SOURCE INVENTORY	=	CORRECTED SOURCE INVENTORY
GAS GAP INVENTORY					
Kr 87			6.5 (0)		
Xe 131m			4.6 (4)		
Xe 133			1.3 (7)		
I 131			7.0 (6)		
I 132			7.7 (3)		
I 133			6.7 (5)		
I 135			1.1 (6)		
FUEL PELLET INVENTORY					
Kr 87			3.2 (7)		
Xe 131m			4.9 (5)		
Xe 133			1.5 (8)		
I 131			7.6 (7)		
I 132			1.1 (8)		
I 133			1.5 (8)		
I 135			1.4 (8)		
CS 134			1.2 (7)		
Rb 88			4.8 (7)		
Te 129			2.5 (7)		
Te 132			1.2 (8)		
Sr 89			3.6 (7)		
Ba 140			1.4 (8)		
La 140			1.4 (8)		
La 142			1.6 (8)		
Pr 144			9.6 (7)		

Corrected Source Inventory = Power Correction Factor X Equilibrium Source Inventory

*Values from Table 3.4 and 3.5 of Reference 2.1

ENCLOSURE A10-PSL 2 (ref. step 5.7.2)

RECORD OF TRANSIENT POWER CORRECTION

UNIT: ______________________ Performed By: __________

Date: ______________

Prior 30 Days Power History:

POWER, %	DURATION, Days (t_j)	TIME TO TRIP, Days (t_j^o)

ISOTOPE	POWER CORRECTION FACTOR	X SL 2 EQUILIBRIUM SOURCE INVENTORY	= CORRECTED SOURCE INVENTORY
GAS GAP INVENTORY			
Kr 87		6.5 (0)	
Xe 131m		4.6 (4)	
Xe 133		1.3 (7)	
I 131		7.0 (6)	
I 132		7.7 (3)	
I 133		6.7 (5)	
I 135		1.1 (6)	
FUEL PELLET INVENTORY			
Kr 87		3.2 (7)	
Xe 131m		4.9 (5)	
Xe 133		1.5 (8)	
I 131		7.6 (7)	
I 132		1.1 (8)	
I 133		1.5 (8)	
I 135		1.4 (8)	
CS 134		1.2 (7)	
Rb 88		4.8 (7)	
Te 129		2.5 (7)	
Te 132		1.2 (8)	
Sr 89		6.6 (7)	
Ba 140		1.4 (8)	
La 140		1.4 (9)	
La 142		1.6 (8)	
Pr 144		9.6 (7)	

Corrected Source Inventory = Power Correction Factor X Equilibrium Source Inventory

*Values from Table 3.4 and 3.5 of Reference 2.1

ENCLOSURE A11 (ref. step 5.8)

RECORD OF PERCENT RELEASE

UNIT: ______________ Performed By: __________

Date: __________________

ISOTOPE	TOTAL QUANTITY AVAILABLE FOR RELEASE (ENCLOSURE A7), Ci	POWER CORRECTED SOURCE INVENTORY Ci (ENCLOSURE A9 OR A10)	PERCENT*
GAS GAP INVENTORY			
Kr 87			
Xe 131m			
Xe 133			
I 131			
I 132			
I 133			
I 135			
FUEL PELLET INVENTORY			
Kr 87			
Xe 131m			
Xe 133			
I 131			
I 132			
I 133			
I 135			
CS 134			
Rb 88			
Te 129			
Te 132			
Sr 89			
Ba 140			
La 140			
La 142			
Pr 144			

$$^{*}\text{Percent} = \frac{\text{Total Quantity Available for Release}}{\text{Power Corrected Source Inventory}} \times 100$$

SECTION B

CORE DAMAGE ASSESSMENT USING HYDROGEN

TABLE OF CONTENTS

LIST OF ENCLOSURES

TABLE OF CONTENTS (CONTINUED)

1.0 PURPOSE:

This section provides the methodology for use under post accident plant conditions to determine the extent of fuel clad damage which may have occurred. It utilizes hydrogen measured in samples obtained with the Post Accident Sampling System (PASS) and containment hydrogen analyzers. The measured hydrogen is related to the amount of fuel clad oxidation. Clad oxidation is in turn related to clad damage which is expressed in terms of the percent of fuel rods which are ruptured and the percent which are embrittled. The resulting observation of damage is described by one or more of the seven categories of core damage in Enclosure B1.

2.0 REFERENCES:

2.1 Development of the comprehensive procedure guideline for core damage assessment, CE Owners Group Task 467, July 1983.

2.2 Operation of the CE Post Accident Sampling System (PASS). Chemistry Procedure No. 1-C112 for PSL1 and No. 2-C113 for PSL-2.

2.3 Clarification of TMI action plan requirements. NUREG 0737, Item II.B.3.

2.4 Determination of Hydrogen gas in containment. Chemistry Procedure No. 1-C-80 for PSL-1 and 2-C-80 for PSL-2.

3.0 DEFINITIONS:

3.1 Clad Rupture: The fuel clad ruptures when the internal gas pressure exceeds the external coolant pressure and the clad yield strength is reduced because of elevated temperatures. Clad rupture results in release of gaseous fission products from the gas gap and possibly some fragments of fuel pellets but does not otherwise destroy the structure of the fuel assembly.

3.2 Clad Embrittlement: At temperature above the rupture temperature significant oxidation of the clad occurs. If the oxidation exceeds the embrittlement threshold, fragmentation of embrittled clad may subsequently occur from thermal shock, hydraulic pressure forces or handling such that the structure of the fuel assembly is destroyed and substantial fuel pellet fragments are released to the coolant.

4.0 PRECAUTIONS AND LIMITATIONS:

4.1 The assessment of core damage obtained by using this methodology is only an estimate. The techniques employed in this section are only accurate to locate the core condition within one or more of the 7 categories of core damage in Enclosure B1. The methodology is based on hydrogen data. Other plant indications may be available which can improve upon estimation of core damage. These include radiological sample characteristics, incore temperature indicators, and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 This methodology relies upon hydrogen samples taken from the containment atmosphere and the reactor coolant system hot leg. Those samples may contain a mixture of hydrogen generated within the core by clad oxidation and also hydrogen from radiolytic dissociation of water and oxidation of aluminum and zinc in the containment. The estimate of clad damage is influenced by the amount of hydrogen generated by ex-core sources and by the ability to identify plant conditions conducive to such hydrogen generation. Therefore, a hydrogen measurement is not a unique indicator of the amount of core clad oxidation.

4.3 There are areas of aluminum components in the containment building. This aluminum would oxidize rapidly at temperatures about 200°F and would be consumed within about two hours. The remainder of the aluminum and other oxidizing material react at a rate determined by temperature and over a longer time. The methodology in this section assumes all of the short term transient hydrogen is generated within the first two hours and is added to the slower accumulation as a function of time. Hence, the methodology is valid for hydrogen samples taken after about two hours with temperatures about 200°F, or after the short term oxidation is complete.

4.4 The methodology in this section yields estimates of the percentages of fuel rods with ruptured clad and embrittled clad. Simultaneous with embrittling of the clad, there may be clad melting and pellet overheating occurring. This section provides an estimate of only the percentage of rods which have progressed to at least clad rupture or clad embrittlement, and does not attempt to predict and physical configuration of those rods which have progressed beyond local clad fragmentation.

4.5 Depending on the accident scenario, a given total amount of hydrogen produced by oxidation of fuel clad

can represent varying local amounts and distributions of clad damage.

4.6 The methodology in this section is applicable under conditions for which there are no voids measurable by the Reactor Vessel Level Monitoring System. It is assumed that if such voids had been found, their removal would be accomplished by using the Reactor Vessel Vent System as prescribed elsewhere in the actions to mitigate the consequences of accidents. However, if the hydrogen samples are taken under conditions in which measurable void does exist, a guideline for analysis is provided in the addendum attached to this section to estimate the contribution of that source to be added to the total hydrogen measured.

5.0 INSTRUCTIONS:

5.1 Obtain the Following Plant Indicators.

5.1.1 Core damage can occur following reactor trip only when the coolant level within the reactor vessel drops below the top of the active fuel. Several instrument records are available from which an estimate of the core uncovery and recovery times might be made. The instruments are:

Reactor Vessel Level Monitoring System
Core Exit Thermocouple Temperature
Core Exit Thermocouple Saturation Margin

Obtain data from these instruments according to the instructions on the worksheet of Enclosure B2.

5.1.2 The magnitude of Reactor Coolant System (RCS) pressure during the core uncovery period can influence the number of early clad ruptures. Interpret the data from Step 5.1.1 to determine the best estimate for the time period of core uncovery and determine the range of RCS pressure during this time period. Record on the Enclosure B2 worksheet.

5.1.3 The presence of some subcooled inlet flow while the core is uncovering can slow the uncovery and cause greater local clad oxidation for a given total amount of core oxidation, thereby leading to a greater underestimate of the number of damaged rods predicted by this procedure. Observe available instrument records to determine if there was some reactor vessel inlet flow during the rising temperature

portion of the core uncovery period. Include net flow from charging and letdown systems, HPSI, LPSI, spray, etc. Record the data on the Enclosure B2 worksheet.

5.1.4 Record the conditions in the containment and the reactor coolant system at the time the hydrogen samples are obtained in Step 5.2 following. Enter on the worksheet of Enclosure B3.

5.2 Obtain a liquid sample from the RCS hot leg and a gas sample from the containment atmosphere and analyze them for hydrogen concentration using the procedures for Post Accident Sample System operation described in Reference 2.2. Record the results on the worksheet of Enclosure B3. Follow the instructions on Enclosure B3 to obtain the total amount of hydrogen measured in units of cubic feet of hydrogen at standard temperature and pressure.

5.3 The total measured hydrogen in Step 5.2 includes the hydrogen generated by three processes: 1) core clad oxidation, 2) radiolysis of water and 3) oxidation of containment materials such as aluminum and zinc. The amount of hydrogen generated by the last two processes is calculated and then subtracted from the total measured to yield the amount generated by core clad oxidation.

Enclosure B5 is a worksheet for calculating the amount of hydrogen generated by oxidation of materials within the containment. It utilizes measured data for the containment temperature as a function of time up to the sampling time and a plant specific curve of the rate of production as a function of containment temperature in Enclosure B6. Record the data required on Enclosure B5 and complete the indicated calculations to obtain the cubic feet of hydrogen at STP generated by containment materials oxidation.

5.4 The hydrogen generated by radiolysis is a function of operating power and decay time. Record the data required on the worksheet of Enclosure B7, and utilize the curve of Enclosure B8 to obtain the cubic feet of hydrogen at STP generated by radiolysis. The appropriate power is determined as follows:

5.4.1 For the case in which the operating power is constant or has not changed by more than $\pm$ 10 percent for a period greater than 30 days, that power is used.

5.4.2 For the case in which the power has not remained constant during the 30 days prior to the reactor trip engineering judgement is used to determine the most representative power level. The following guidelines should be considered in the determination.

5.4.2.1 The average power during the 30 day time period is not necessarily the most representative value for determining radiolysis by fission products.

5.4.2.2 The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

5.4.2.3 Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

5.4.3 For the case in which the reactor has produced power for less than 30 days, this methodology may be employed. However, the estimate of hydrogen from radiolysis will be too high and therefore the calculated hydrogen by core oxidation will be too low. Hence an underprediction of core damage may result.

5.5 Enter the amounts of hydrogen from Steps 5.2, 5.3 and 5.4 on the worksheet of Enclosure B9. Subtract the amounts in 5.3 and 5.4 from 5.2 as indicated on the worksheet to yield the cubic feet of hydrogen generated by core clad oxidation. Adjust with the plant specific constant as shown on the worksheet to obtain the estimated percent of the core clad which is oxidized. This value represents the quantity of hydrogen produced per percent of zirconium oxidized.

5.6 Enter the abscissa of the curve on Enclosure B10 with the percent of core clad oxidized from Step 5.5. Use the curve labeled with the pressure closest to but greater than the RCS pressure during the core uncovery period as obtained in Step 5.1.2 and recorded on Enclosure B2, e.g. if pressure during core uncovery is 1300 psia, the curve labeled with temperature 1800°F is used. Read on the ordinate of Enclosure B10, the percent of fuel rods with ruptured clad. Record on the worksheet of Enclosure B9. Note that the sensitivity of measurement of hydrogen is comparable to the range of oxidation on Enclosure B10. Hence, small amounts of clad rupture are not reliably predicted by the methodology in this section.

5.7 Enter the abscissa of the curve on Enclosure B11 with the percent of core clad oxidized from Step 5.5. Read on the ordinate the lower and upper values of the range indicated by the curve for the percent of fuel rods which have embrittled clad. Record on the worksheet of Enclosure B9.

5.8 For a given percent oxidation of the core clad, the lower limit estimated of embrittled clad in Step 5.7 is, for most accident scenarios, the least amount of potential fuel structural failure. Actual values are probably greater. The upper limit of the range in Step 5.5 may be interpreted as follows:

5.8.1 When the pressure during uncovery, from Step 5.1.2 and recorded on Enclosure B2, is less than about 100 psia, a rapid core uncovery by blowdown is concluded. Heatup with minimum clad oxidation occurs. The extent of potential clad structural failure by melting may be greater than the upper limit of embrittlement from Step 5.7 as determined by oxidation. Hence, use the upper limit from Step 5.7.

5.8.2 When there is inlet flow while the core is uncovering, the rate of uncovery is slower than assumed in the derivation of the curves on Enclosures B10 and B11. For a measured total amount of oxidation, the local percentage oxidation is probably greater along a shorter length of the upper portions of the fuel. Hence, favor the upper limit from Step 5.7.

5.9 Core Damage Assessment

The conclusion on core damage is made using the two results from above. These are:

1. Percentage of fuel rods with ruptured clad, Step 5.6.

2. Percentage of fuel rods with embrittled or structurally damaged clad, Step 5.7.

Knowledgeable judgement is used to compare the above two results to the definitions of the 7 NRC categories of fuel damage found in Enclosure B1. Core damage does not take place uniformly. Therefore when evaluating damage using these results, Enclosure B1 may yield a combination of categories of damage which exist simultaneously.

ENCLOSURE B1

CLAD DAMAGE CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1. No Fuel Damage	~750	None	--	--	Less Than 1
2. Initial Cladding Failure		Rupture Due to Gas Gap	Maximum Core Exit	<1550°F*	Less Than 10
3. Intermediate Cladding Failure	1200-1800	Overpressurization	Thermocouple Temperature	<1700°F*	10 to 50
4. Major Cladding Failure				$\tilde{<}$2300°F $\tilde{<}$2% Oxidation	Greater Than 50
5. Initial Fuel Pellet Overheating		Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6. Intermediate Fuel Pellet Overheating	1800-3350			<18%	
7. Major Fuel Pellet Overheating				$\tilde{<}$65%	Greater Than 50

* Depends on Reactor Pressure and Fuel Burnup, Valves Given for Pressure ≤1200 psia and Burnup ≥0.

ENCLOSURE B2

CORE UNCOVERY CONDITIONS

Step 5.1.1 Time period of core uncovery. Complete the following table using recorded instrument data.

Instrument	Estimated Core Uncovery Time	Estimated Core Recovery Time
Reactor Vessel Level Monitoring System	Lower Limit Elevation Uncovers (core uncovery) Time ________	Lower Limit Elevation Recovers. Time ________
Core Exit Thermocouple Temperature	Start of Continuous Rise or Exceed 660°F Time ________ Temperature ________	Rapid Temperature Drop to Saturation Time ________ Temperature ________
Core Exit Thermocouple Saturation Margin	Start of Superheat Time ________	Return to Saturation or Subcooling Time ________

Step 5.1.2 Interpret above data to obtain best estimate for time period of core uncovery and obtain pressurizer pressure range during that period. The superheat derived from the thermocouple temperature and corresponding system pressure is considered as the best indicator for core uncovery during boiloff and should be used, but should be compared with the other indicators to help identify possible anomalies. The pressure during uncovery is used later in Enclosure B10, Step 5.6, to determine the appropriate curve for assessment of the number of clad ruptures.

	Core Uncovery	Core Recovery
Time	________	________
Pressure	________	________

Step 5.1.3 Estimate vessel inlet flow rates during core uncovery heatup period, up to approximately the time of peak core exit thermocouple temperature. Net inlet flow indicates that the methodology may have additional bias which underpredicts clad damage.

Charging Flow Rate ________

Letdown Flow Rate ________

HPSI Flow Rate ________

LPSI Flow Rate ________

Other Inlet Flows ________

Net inlet flow = Charging flow + HPSI and LPSI flow + other inlet flow - letdown flow

Performed By: ________ Date: ________

ENCLOSURE B3

SAMPLING CONDITIONS AND MEASURED HYDROGEN

Step 5.1.4 Obtain the RCS and containment conditions at the time of sampling for hydrogen.

Reactor Coolant System		Containment	
Sampling Time	______	Sampling Time	______
Pressure	______ psig	Atmosphere Pressure	______ psig
Tempterature, T_{avg}	______ °F	Atmosphere Temperature	______ °F
Reactor Vessel Coolant Level	______ %	Has Hydrogen Recombiner Operated	Yes/No
Pressurizer Level	______ %	Does Pressure or Temperature History Indicate a Hydrogen Burn	Yes/No

Step 5.2 Hydrogen Sample Data Reduction.

Cont. Sample (Vol. %/100)	x	Cont. Vol. (Ft^3)	x	(32 + 460)	÷	(Normal Temp. + 460)	=	Ft^3 Hydrogen at STP
______	x	2.5 (6)	x	492	÷	460	=	______ ft^3

Hot Leg Sample (cc/kg @ STP)	x	RCS Vol.* (Ft^3)	x	Density Ratio (Enclosure B4)	÷	1000	=	Ft^3 Hydrogen at STP
______	x	______	x	act/ STP	÷	1000	=	______ ft^3

Total = Cont. Sample (ft^3) + Hot Leg Sample = ______ + ______ = ______ ft^3.

Also record total on Enclosure B9.

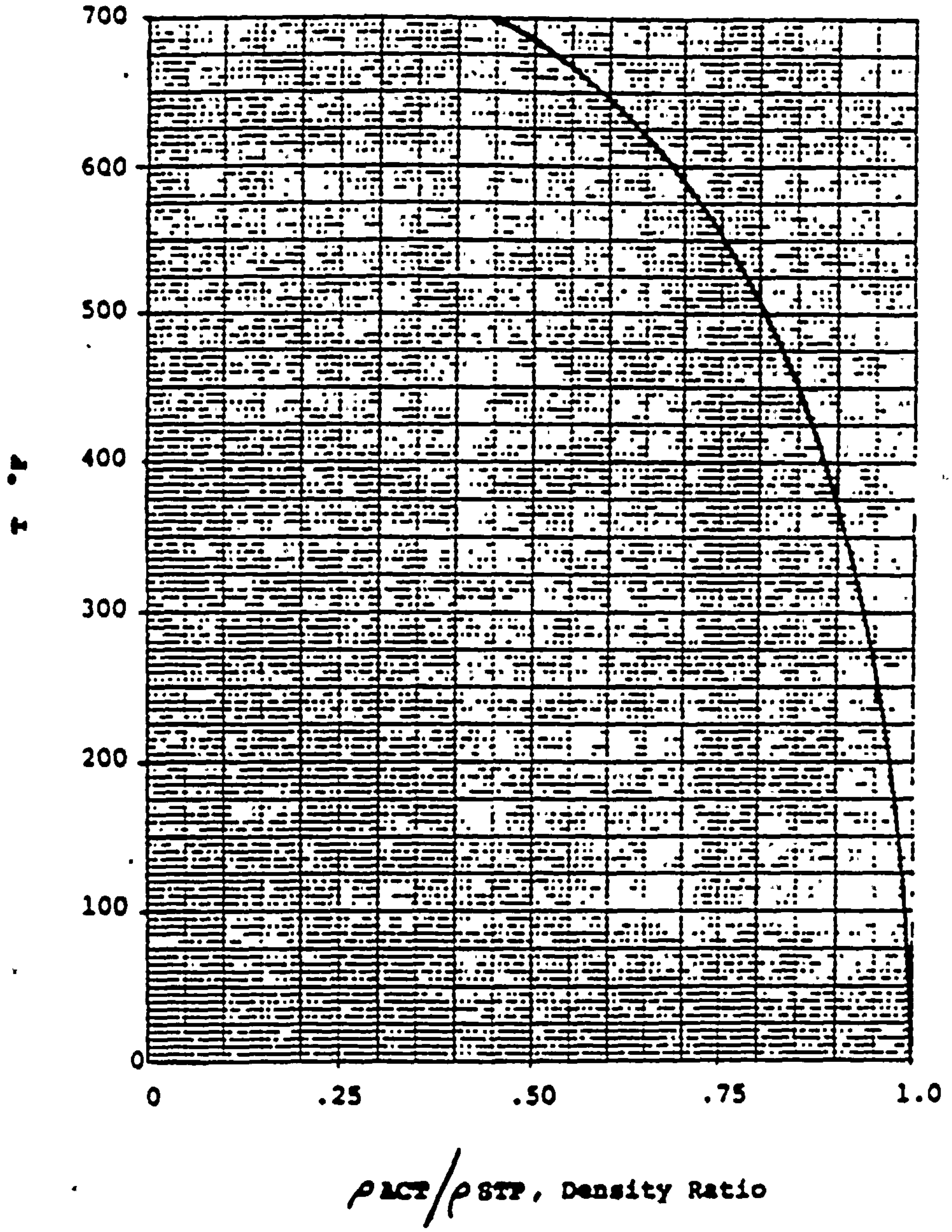
*RCS liquid volume is: SL1 = 10,401 ft^3
SL2 = 10,198 ft^3

Note: If the Reactor Vessel Coolant Level indication shows a measurable void, refer to the addendum to this section. This addendum contains instructions to calculate Hydrogen in the void. The void volume should be subtracted from the RCS volume above. The estimated hydrogen in the void is to be added to the total hydrogen measured above.

Performed By: ______________ Date ____________

ENCLOSURE 64

RATIO OF H_2O DENSITY TO H_2O DENSITY
AT STP vs. TEMPERATURE



ENCLOSURE B5

HYDROGEN GENERATED IN CONTAINMENT

Step 5.3 Record the containment temperature at selected time intervals and calculate the hydrogen generated by oxidation of containment materials utilizing the plant-specific production rates from Enclosure B6.

1 Time at Start of Intervals	2 Containment Temperature (°F)	3 Interval Duration (hr)	4 Avg. Containment Temp. During Interval (°F)	5 H_2 Prod. Rate (ft^3/hr, Enclosure B6)	6 H_2 Produced = (Col 3) X (Col 5)
Accident Starts					
Sample Time					

Long Term Hydrogen Production in Containment, Total (Summation of Column 6) ________ SCF

Short term rapid hydrogen production by containment aluminum, 2,277 ft^3 for SL1 and 5235 ft^3 for SL2 (Reference 2.1 Table 4-3.) + ________ SCF

Total Hydrogen Production in Containment = ________ SCF

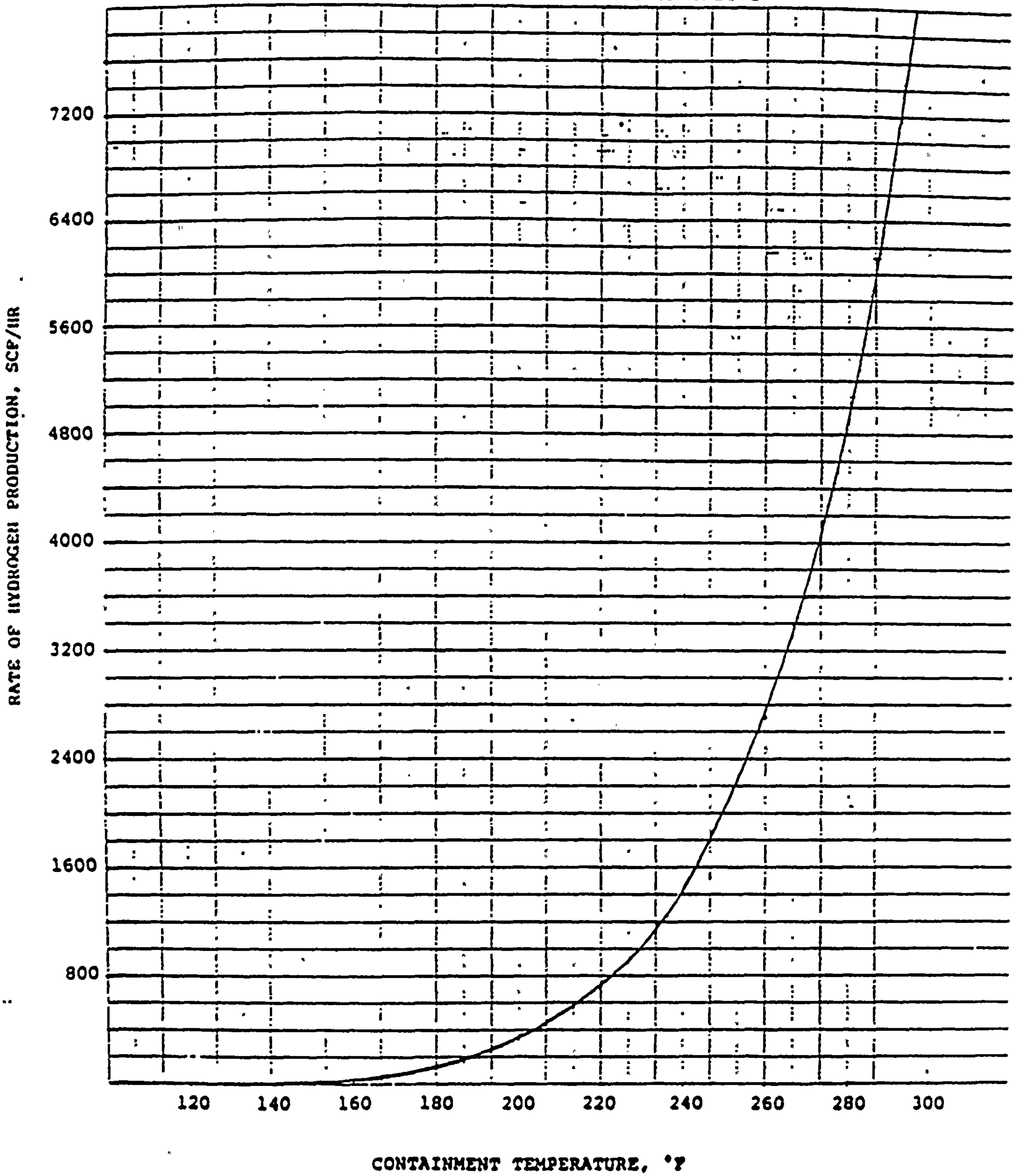
Record total on Enclosure B9 also.

1 and 2 Items in Columns 1 and 2 are input plant data
3 Interval Duration is the line difference between consecutive temperture readings.

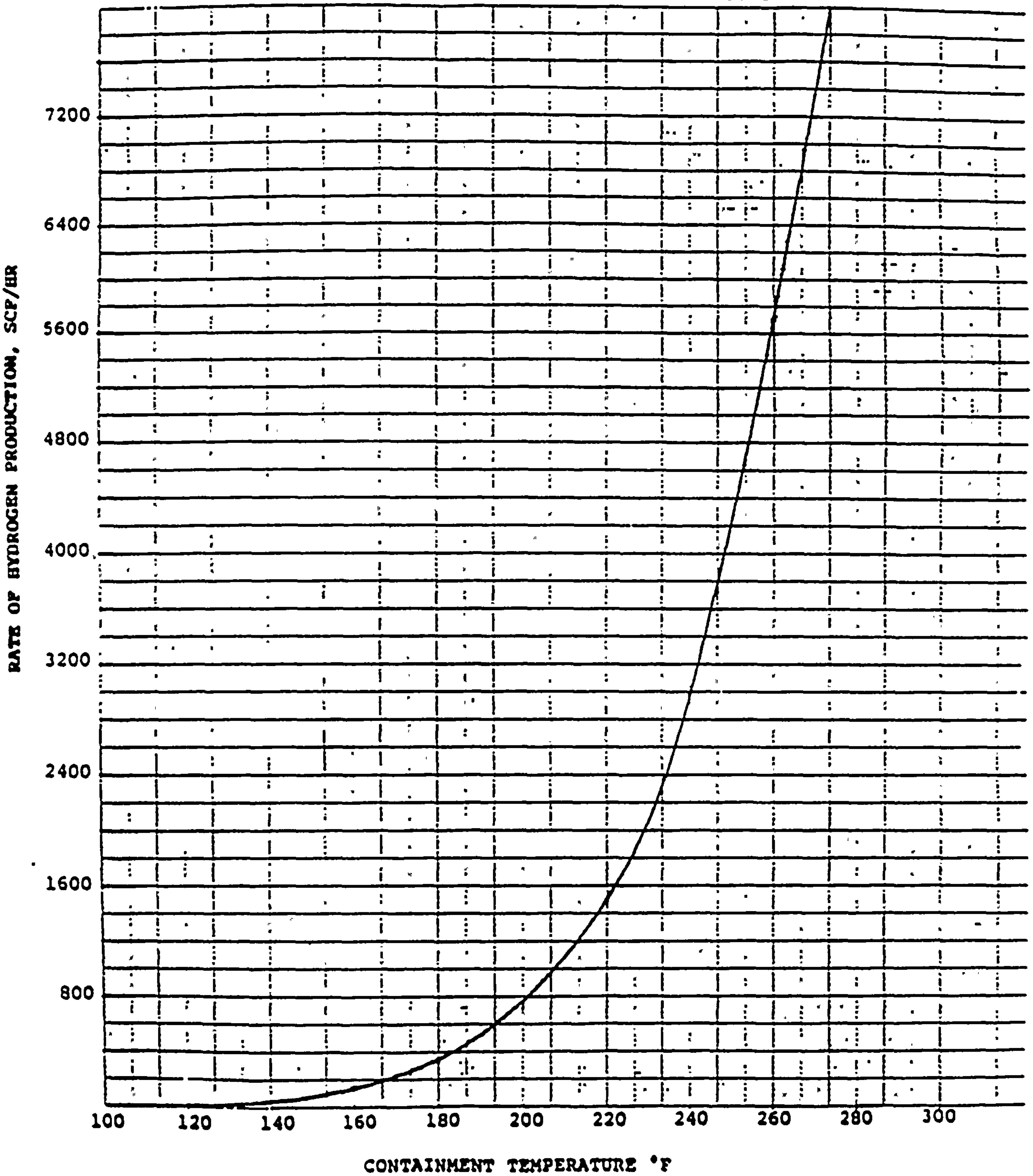
Performed By: ____________________

Date: ____________________

ENCLOSURE B6 - SL1
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND
ZINC VS TEMPERATURE FOR ST. LUCIE UNIT 1



ENCLOSURE B6 - SL2
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC VS TEMPERATURE FOR ST. LUCIE UNIT 2



ENCLOSURE B7

HYDROGEN GENERATED BY RADIOLYSIS

Step 5.4 Record the following data and utilize the curves of Enclosure B8 to determine the hydrogen generated by radiolysis.

Prior 30 days power history	Power, Percent	Duration, Days
Note: No calculation is required to determine power level, guidance on judgement is provided in Step 5.4.	________	________
	________	________
	________	________

Estimated Power Level based on a power history: __________

Operating Power (Mwt):

Power to use in evaluating long term hydrogen production by radiolysis = (Full Power, Mwt) X $\frac{\text{Power Level.}}{100}$

[Full Power:. SL1 = 2700 Mwt; SL2 = 2700 Mwt]

T_o = Time of Reactor Trip Time __________

T_i = Time Sample Taken (see Enclosure B3) __________

Decay Time (Time Interval, $T_i - T_o$) __________ Hours

Enter abscissa on Enclosure B8 with above decay time and read two values of hydrogen produced by radiolysis, one from each curve, in cubic feet of hydrogen at STP per Mwt operating power. Multiply by above power and record as follows:

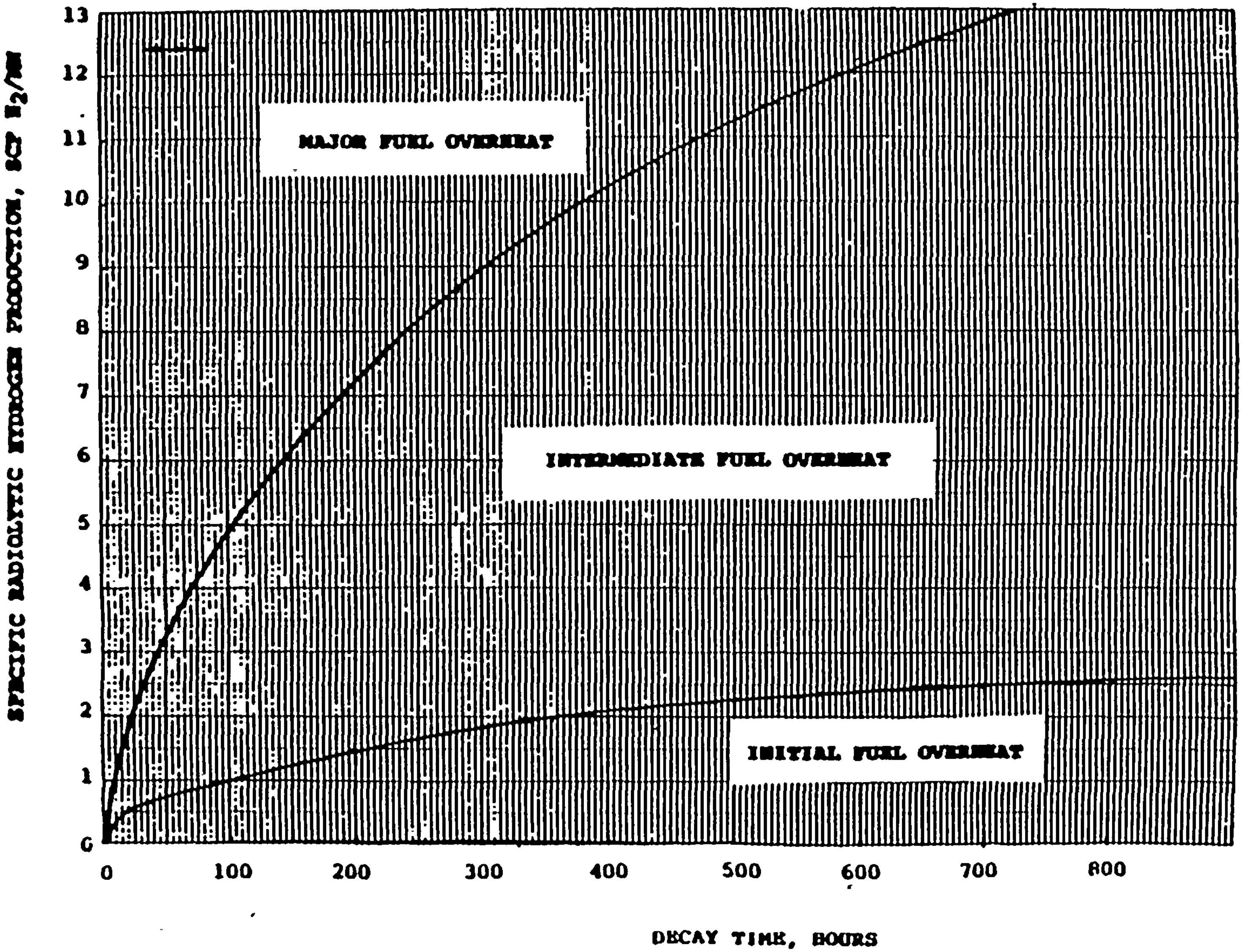
Limit Curve	Hydrogen Produced (SCF/Mwt, Enclosure B3)	x	Operating Power (Mwt)	=	Total Hydrogen Produced (SCF)
Upper	________		________		________
Lower	________		________		________

Results from Radiological Analysis of Samples are used to estimate whether the upper limit for major fuel overheat or lower limit for intermediate fuel overheat is appropriate. Circle corresponding value of hydrogen above and also record on Enclosure B9.

Performed By: __________ Date: ________

ENCLOSURE B8

SPECIFIC RADIOLYTIC HYDROGEN PRODUCTION VS TIME



ENCLOSURE B9

CORE DAMAGE ASSESSMENT FROM
HYDROGEN MEASUREMENT

Step 5.5 Hydrogen Measured, Step 5.2, Enclosure B3(Total) ______________ SCF

Hydrogen Produced in Containment, Step 5.3, Enclosure B5 ______________ SCF

Hydrogen Produced by Radiolysis, Step 5.4, Enclosure B7 ______________ SCF

Subtract Step 5.3 and 5.4 from 5.2 to Get Hydrogen Produced by Core Clad Oxidation ______________ SCF

Divide by [4.21 E3 for PSL1] or [4.64 E3 for PSL2]. These values represent the quantity of hydrogen produced per percent of Zirconium oxidized for St. Lucie Unit 1 and Unit 2, respectively. Reference 2.1, Table 4.2. = ______________ % = % Core Clad Oxidized

Step 5.6 Enter abscissa on Enclosure B10 with "% Core Clad Oxidized" and read ordinate from curve labeled with pressure during core uncovery as given on Enclosure B2, Step 5.1.2. Record here Percent of Fuel Rods with Ruptured Clad ____________ %.

Step 5.7 Enter abscissa on Enclosure B11 with above "% Core Oxidized" and read range of values on ordinate. Record here

Percent of fuel rods embrittled

Range - Upper ______________ %

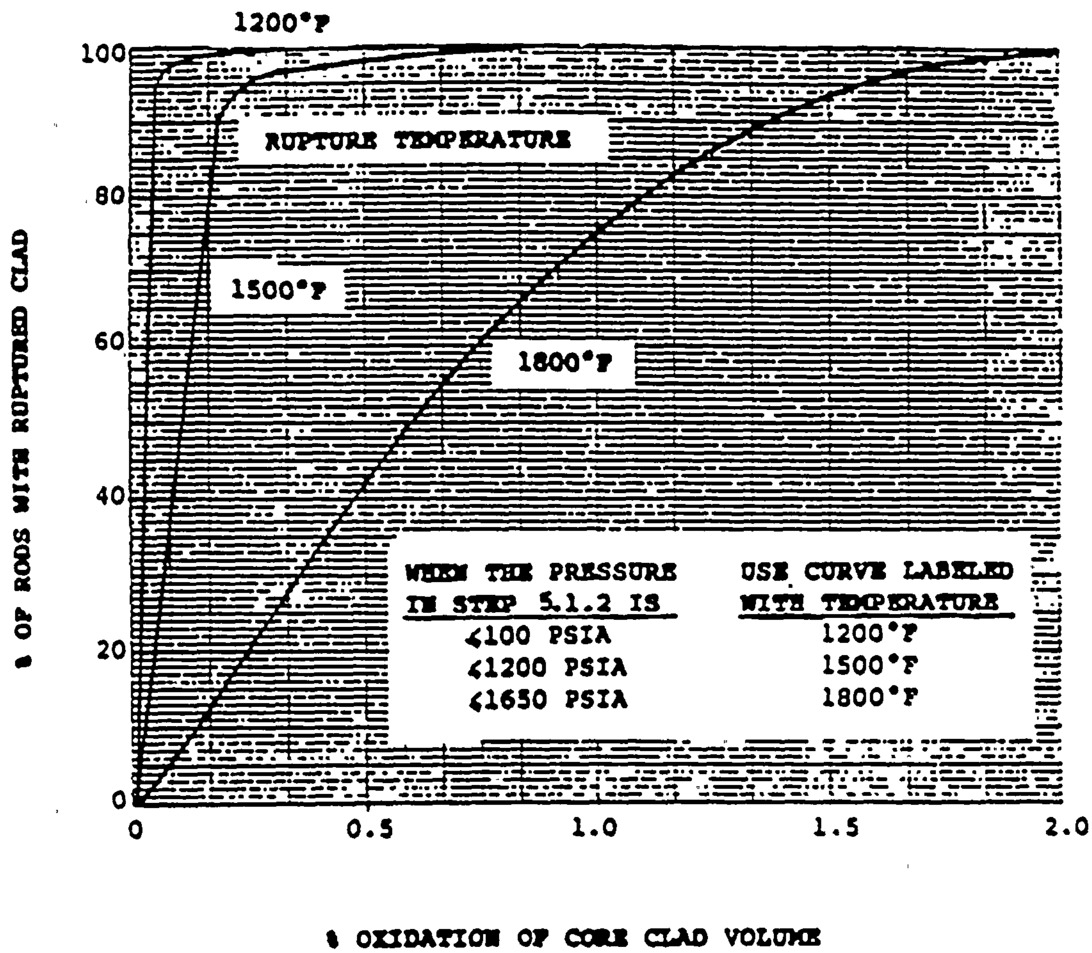
- Lower ______________ %

Step 5.8 Review Step 5.1 of these instructions to determine which of these limits is more likely to be representative of the core damage.

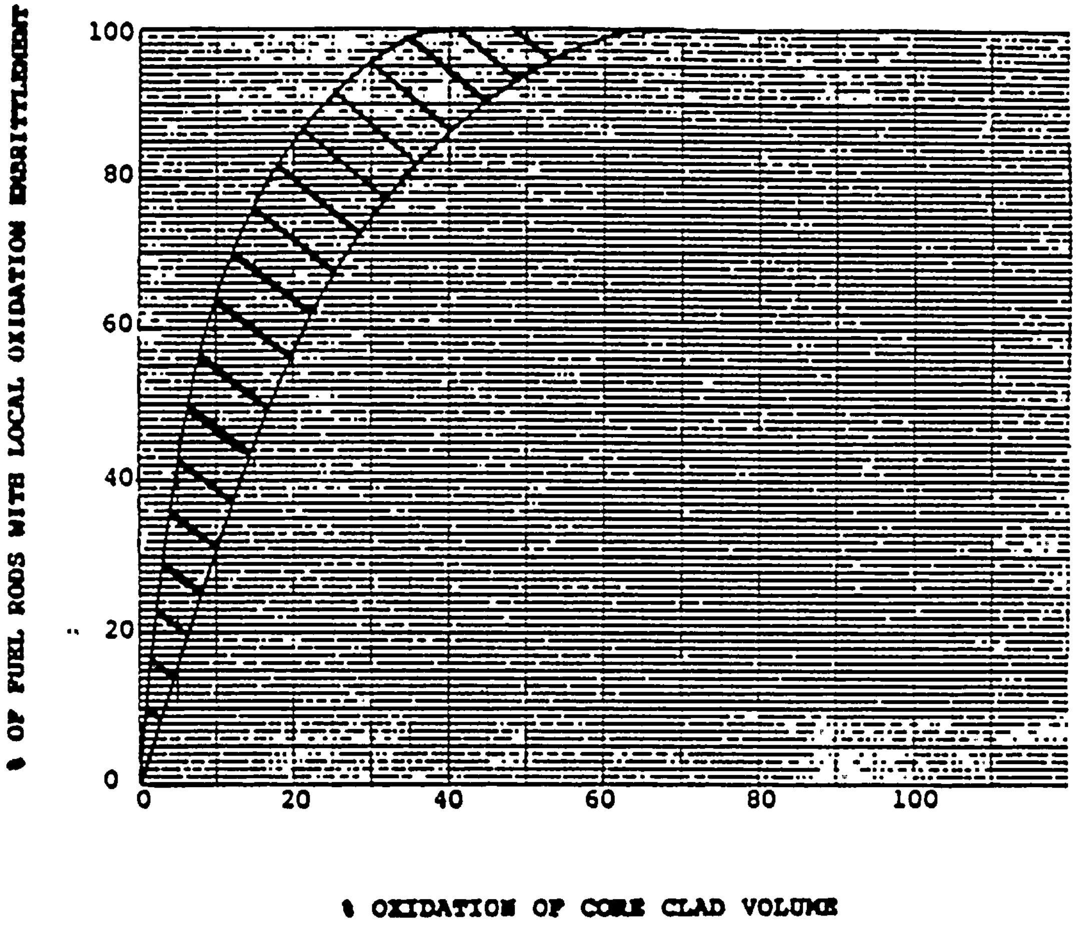
Step 5.9 From Enclosure B1 select the core clad damage categories based on the above percentages of rods ruptured and rods embrittled.

Approved By: __________________ Date: ______________

ENCLOSURE B10
PERCENT OF FUEL RODS WITH RUPTURED
CLAD VS CORE CLAD OXIDATION



ENCLOSURE B11
% OF THE FUEL RODS WITH OXIDATION EMBRITTLEMENT VS
TOTAL CORE OXIDATION
FOR 1% TO 3% DECAY HEAT AND 300 PSIA TO 2500 PSIA
WHEN COOLANT LEVEL DROPS BY BOILOFF WITH
NO INLET FLOW UNTIL CORE IS RAPIDLY QUENCHED



6.0 ADDENDUM TO SECTION B

ESTIMATION OF AMOUNT OF HYDROGEN IN REACTOR VESSEL HEAD VOID

1.0 PURPOSE:

The purpose of this addendum is to provide the methodology to calculate the amount of hydrogen gas contained in a void in the top of the reactor vessel. This hydrogen is added to the measured amount in Step 5.2 of Section B to determine the total hydrogen generated by all sources.

2.0 LIMITATIONS:

2.1 The preferred method of determining the amount of hydrogen in the primary system is to sample liquid from the hot leg when the system is full. However, if the system cannot be filled, a method based on this addendum could be used to estimate the hydrogen which is in the vessel void and which would not be evident from the hot leg liquid sample.

2.2 This method applies when the coolant level is above the hot leg and the remainder of the primary system is filled. Verification that the steam generator tubes are filled can be provided by the existence of natural convection flow in the primary system. If the coolant level is below the hot leg, this method does not apply.

2.3 A reactor vessel level monitoring system is required which can provide the coolant level. The volume of the void is obtained by relating the volume in the vessel above the coolant level to the value of level.

2.4 The methodology in this addendum provides the analytical means for only an estimate of the hydrogen contained in the void. The presence of other gases including helium, nitrogen and fission product gases will add uncertainty to the result.

3.0 INSTRUCTIONS:

3.1 Determine the conditions of the void as follows:

V = Void volume (Ft^3) derived from measurement of coolant level

T_L = Temperature of liquid at coolant surface (^oF) as measured by CET

P_{sat} = Water saturation pressure at temperature T_L (Enclosure BB1)

P_{tot} = Reactor coolant system pressure (psia)

3.2 A first approximation is made assuming the following:

3.2.1 The partial pressure of vapor in the void is assumed equal to saturation pressure at the liquid temperature, T_L. This implies no heating of the void gas by the reactor vessel walls and head. They are normally at reactor outlet temperature and could remain above the temperature of the void causing the vapor to be superheated.

3.2.2 All the non-condensible gas in the void is hydrogen. This implies no helium or fission product gas from ruptured fuel rods and no nitrogen from Safety Injection Tanks. A second approximation which eliminates this assumption is given in 3.4.

3.3 Calculate the amount of hydrogen as follows:.

$$P_{H_2} = P_{tot} - P_{sat};$$

$$Ft^3\ H_2 \ @\ STP = (V)\left(\frac{P_{H_2}}{14.7}\right)\left(\frac{492}{T_L + 460}\right)$$

Add this amount to the total hydrogen in Step 5.2 of Section 3.

3.4 A second approximation can be made in plants with a CE designed PASS (i.e. PSL2) which measures both total gas and hydrogen which are dissolved in the hot leg liquid sample. This approximation includes the following assumptions regarding the ralative solubilities of the non-condensible gases in the liquid.

3.4.1 The gases are assumed to have the same values of Henry's law constant which relates the partial pressure of gas to the amount of gas dissolved in the liquid sample at equilibrium.

3.4.2 When the dissolved gas is not in equilibrium with the gas in the void, the dissolved concentrations are in the same ralative proportion as if equilibrium did exist.

3.5 The partial pressure of hydrogen is calculated from:

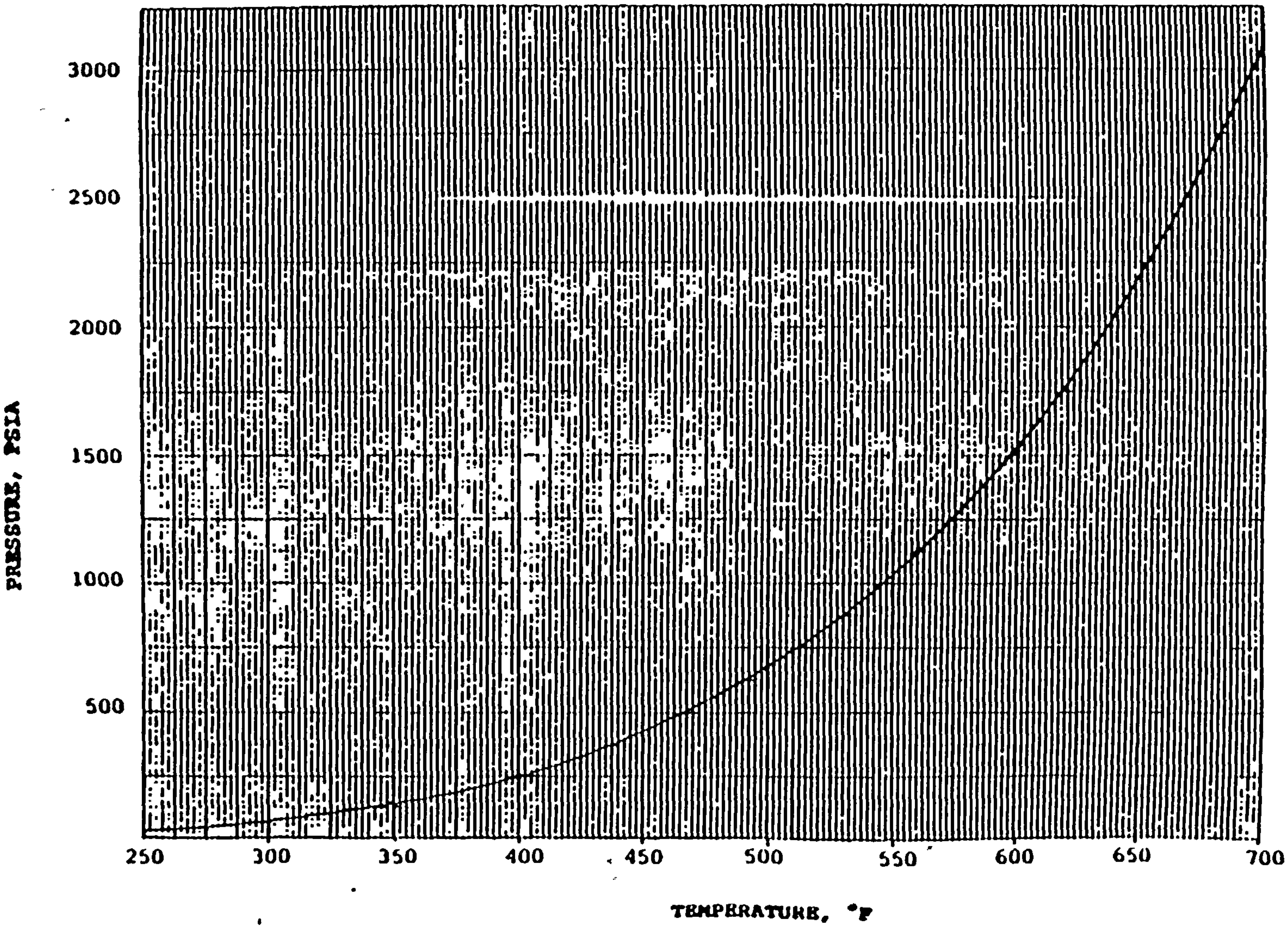
$$P_{H_2} = (P_{tot} - P_{sat})_X \frac{(cc/kg)_{H_2}}{(cc/kg)_{total}}$$

and the amount of hydrogen in the vessel head void is

3.6 This procedure can be extended to include specific values of Henry's law constants but the assumption of equilibrium at the gas liquid interface would still be questionable. Also, to utilize detailed values of the gas constants, the individual gases in the sample would have to be identified and measured. This would require additional measurement capability.

ENCLOSURE BB1

SATURATED WATER PRESSURE AND TEMPERATURE



ENCLOSURE 8B2

RECORD OF HYDROGEN IN VOID

Reactor Vessel Coolant Level Indication (1 to 8):______ = VL

Percent Void Height = $\frac{VL}{8}$ x 100 = _____%

Note: 1% corresponds to approximately 0.135 Ft.

Void height above fuel alignment plate = _________% x $\frac{0.135 \text{ Ft}}{1\%}$ = _________ Ft.

V = Void Volume = 105 $\frac{ft^3}{ft}$ X Void Height (Ft) = ____________

T_L = Temperature of Liquid at coolant surface (°F) as measured by CET. =

P_{sat} = Pressure at T_L from Enclosure 8B1 =

P_{tot} = RCS pressure (psia) =

Amount of Hydrogen Calculation: $P_{H_2} = P_{tot} - P_{sat}$

P_{H_2} = ______________ - ______________

$$Ft^3\ H_2 \text{ @ STP} = V \left(\frac{P_{H_2}}{14.7}\right) \left(\frac{492}{T_L + 460}\right)$$

= Add this amount to the total hydrogen in Step 5.2 of Section 8 (Enclosure B3)

Approved By: ________________ Date: ____________

SECTION C

ASSESSMENT OF CORE DAMAGE USING CORE EXIT THERMOCOUPLES TEMPERATURES

TABLE OF CONTENTS

LIST OF ENCLOSURES

1.0 PURPOSE:

This section provides the methodology for use under post accident plant conditions to determine the number of fuel rods with ruptured clad. It provides an estimate of damage up to about the time when the peak core temperature reaches about 2300°F. At that time most of the rods probably have ruptured clad but little other structural degradation has occurred. Therefore this procedure applies to the relatively less severe accidents although it may be used for other accidents to confirm that damage exceeds this minimum amount. The resulting observation of core damage is described by categories 1 through 4 of the seven NRC categories in Enclosure C1.

2.0 REFERENCES:

2.1 Development of the Comprehensive Procedure Guidelines for Core Damage Assessment, CE Owners Group Task 467, May, 1983.

2.2 Inadequate Core Cooling, St. Lucie Unit 2, Final Safety Analysis Report, Appendix 1.9.B.

2.3 Generic Thermal-Hydraulic Functional Design Objectives for Inadequate Core Cooling Instrumentation, CE-NPSD-199, prepared for the CE Owners Group.

3.0 DEFINITIONS:

3.1 Clad Rupture: Clad rupture is defined as a break in the fuel rod clad at least sufficient to release the internal gas pressure. Rupture may be preceded by ballooning of the clad if the internal gas pressure exceeds the external coolant pressure during an accident, and the temperature is higher than normal.

4.0 PRECAUTIONS AND LIMITATIONS:

4.1 The assessment of core damage obtained by using this method is only an estimate. The techniques employed in this section are only accurate to locate the core condition within the first four of the 7 categories of core damage described in Enclosure C1. The methodology is based on core exit temperature data. Other plant indications may be available which can improve upon estimation of core damage. These include radiological sample characteristics, the total quantity of hydrogen released from zirconium degradation and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 The assessment of damage provided by this procedure extends up to the time of clad rupture on most of the fuel rods. This time occurs early in very severe core uncovery accidents. More severe core damage cannot be quantified by this procedure.

4.3 The relationship between the core exit thermocouple temperature and the clad temperature varies with the core uncovery senario. This procedure applies to slow core uncovery by boiloff of the coolant. For other more rapid uncovery scenarios this procedure could yield a very low estimate of the number of ruptured rods. In general, for core uncovery at pressures below about 1200 psia there is high confidence that at least the predicted estimate of rods are actually ruptured.

5.0 INSTUCTIONS

5.1 Obtain the following from the instrument recordings:

5.1.1 From the recording of maximum core exit thermocouple temperature as a function of time, obtain and record on Enclosure C2 the maximum temperature and the time it occurs. As many thermocouples as possible should be used, in this way equipment malfunction may be detected if a thermocouple reads greater than 1650°F or varies considerably from its neighboring thermocouples.

5.1.2 From the recording of reactor coolant system pressure as a function of time, obtain and record on Enclosure C2 the pressure during the period of maximum thermocouple temperature.

5.2 Select the curve on Enclosure C3 which is labeled with a pressure approximately equal to or greater than the pressure in Step 5.1.2. Enter the abscissa at the maximum temperature from Step 5.1.1 and read on the ordinate the percent of the fuel rods which have ruptured clad. Record on Enclosure C2.

5.3 The result in 5.2 is probably a lower limit estimate of damage. Some judgement on the bias is available as follows.

5.3.1 This procedure applies most directly for relatively slow core uncovery with a maximum temperature below the rapid oxidation temperatures at about 1800°F and above. A smooth core exit thermocouple recording and an uncovery duration of 20 minutes or longer are indicators for a good prediction of clad ruptures.

5.3.2 If the pressure in 5.1.2 drops to less than about 100 psia within less than about two minutes of accident initiation, a large break is indicated. This causes undetected core heatup followed by flashing during refill. Depending on the rate of refill, the thermocouple temperature may rise rapidly then quench when the core is recovered. This procedure could yield a very low estimate for the percent of rods ruptured.

5.3.3 If the pressure in Step 5.1.2 is above about 1650 psia, it could exceed the rod internal gas pressure depending on rod burnup, causing clad collapse onto the fuel pellet instead of outward clad ballooning. The clad rupture criteria are less well defined for such conditions, but at temperatures above 1800°F where the highest pressure curve applies on Enclosure C3, clad failure sufficient to release fission gas is likely and this procedure may be used to obtain estimates of damage.

5.4 Core Damage Assessment

Use the percent of rods ruptured from Step 5.2 and the clad damage characteristics of Enclosure C1 to determine the NRC category of cladding failure. This procedure yields damage estimates in categories 2, 3 or 4 (Enclosure C1).

ENCLOSURE C1

CLAD DAMAGE CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

	NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1.	No Fuel Damage	~750	None	--	--	Less Than 1
2.	Initial Cladding Failure		Rupture Due to Gas Gap	Maximum Core Exit	<1550°F*	Less Than 10
3.	Intermediate Cladding Failure	1200-1800	Overpressurization	Thermocouple Temperature	<1700°F*	10 to 50
4.	Major Cladding Failure				$\lesssim$2300°F $\lesssim$2% Oxidation	Greater Than 50
5.	Initial Fuel Pellet Overheating		Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6.	Intermediate Fuel Pellet Overheating	1800-3350			<10%	
7.	Major Fuel Pellet				$\lesssim$65%	Greater Than 50

* Depends on Reactor Pressure and Fuel Burnup Valves Given for Pressure $\leq$1200 psia and Burnup $\geq$0.

ENCLOSURE C2

RECORD OF TEMPERATURE PRESSURE AND DAMAGE ESTIMATE

Step 5.1 Record the following data

Maximum Core Exit Thermocouple Temperature* ____________ °F
(see Instruction 5.1 in the text for guidelines)

Time of Maximum Temperature ____________

Reactor Coolant System Pressure at Above Time ____________ psia

Step 5.2 From Enclosure C3, at maximum thermocouple temperature and at appropriate pressure

Read percent of ruptured rods ____________ %

Step 5.3 Comment on probable bias of results in 5.2 (see paragraph 5.3 in text).

Step 5.4 NRC category of cladding failure from Enclosure C1 ____________

*As many thermocouple readings as possible should be recorded. In this way equipment malfunction may be detected if a thermocouple reads greater than 1650°F or varies considerably from its neighboring thermocouples.

Maximum Core Exit Thermocouple Temperature

____________ °F

____________ °F

____________ °F

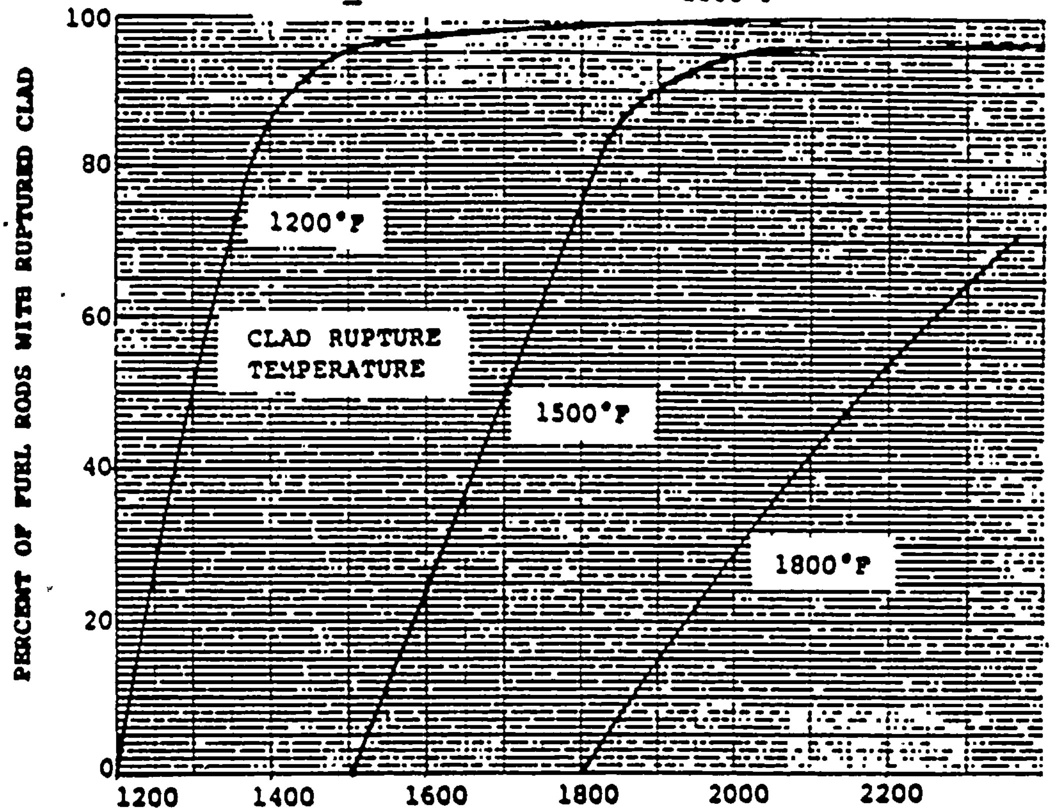
____________ °F

____________ °F

Performed By: ____________ Date: ____________

ENCLOSURE C3
PERCENT OF FUEL RODS WITH RUPTURED CLAD VS
MAXIMUM CORE EXIT THERMOCOUPLE TEMPERATURE

WHEN THE PRESSURE IN STEP 5.1.2 IS	USE CURVE LABELED WITH TEMPERATURE
≤ 100 PSIA	1200°F
≤1200 PSIA	1500°F
≤1650 PSIA	1800°F



MAXIMUM CORE EXIT THERMOCOUPLE TEMPERATURE

SECTION D

ASSESSMENT OF CORE DAMAGE USING RADIATION DOSE RATES

TABLE OF CONTENTS

LIST OF ENCLOSURES

1.0 PURPOSE:

This section provides the methodology for use under post accident plant conditions to determine the type and degree of core damage which may have occurred by using radiation dose rates measured inside the containment building using the containment high range radiation monitor. The radiation dose rate is related to the quantitative release of fission products from the core expressed as the percent of the source inventory at the time of the accident. The resulting observation of core damage is described by one or more of the seven categories of core damage in Enclosure D1.

2.0 REFERENCES:

2.1 Development of the comprehensive procedure guideline for core damage assessment, CE Owners Group Task 467, July 1983.

3.0 DEFINITIONS

3.1 Fuel Damage: For the purpose of this section fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the reactor coolant starting with a penetration in the zircaloy cladding. The type of fuel damage as determined by this methodology is reported in terms of three major categories which are: no damage, cladding failure, and fuel overheat. The categories are characterized by the resulting radiation dose rate inside the containment building. The degree of core damage is determined by making a comparison between dose rates measured following an accident and analytically determined values of the realistic or best estimate dose rates that would correspond to the specific categories of core damage. The degree of core damage as determined by this section is reported in terms of three levels which are: initial; intermediate; and major. This results in a total of seven possible categories as characterized in Enclosure D1.

3.2 Source Inventory: The source inventory is the total quantity of fission products expressed in curies of each isotope present in either source; the fuel pellets or the fuel rod gas gap.

4.0 PRECAUTIONS AND LIMITATIONS:

4.1 The assessment of core damage obtained by using the methodology in this section is only an estimate. The techniques employed in this section are only accurate to locate the core condition within one or more of the 7 categories of core damage described in Enclosure D1. The procedure is based on radiation dose rate. Other plant indications may be available which can improve upon the estimation of core damage. These include sample radiological analysis, incore temperature indicators, and the total quantity of hydrogen released from zirconium degradation. Whenever possible these additional indicators should be factored into the assessment.

4.2 This section relies upon radiation dose rate measurements taken from the higher of two high range radiation monitors located inside the containment building to determine the total quantity of fission products released from the core and therefore available for release to the environment. The amount of fission products present at the location of the monitors may be changing rapidly due to transient plant conditions. Therefore, multiple measurements should be obtained within a minimum time period and when possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

4.3 A number of factors influence the reliability of the measured radiation dose rates upon which this procedure is based. Reliability is influenced by the ability to obtain representative measurements due to incomplete mixing of the measured media, and equipment limitations. Additionally the procedure relies upon analytically determined values of the best estimate dose rates that are anticipated to correspond to the specific categories of core damage. These analytical values are based upon assumptions made about the identity and relative proportions of the fission products released from the core and their transport within the containment building. Therefore, the method is only accurate to within the validity of the assumptions.

4.4 The methodology in this section is limited to the upper bound condition of fission product release from the core due to fuel overheat. Simultaneous with fuel overheat, there may be localized fuel pellet melting within the core. The transport of the non-volatile fission products released due to melting is not known. The dose rates measured under conditions of

fuel pellet melting are anticipated to exceed those, shown in Enclosure D3, for major fuel overheat. However, this procedure does not attempt to identify the extent of any potential fuel melting.

4.5 This section is limited to the interpretation of the dose rate measurement resulting from a mix of fission products. The methodology cannot accurately distinguish between the conditions of fuel cladding failure and fuel overheat when the resulting dose rates are the same. The methodology does provide an upper limit estimate of the progressive core damage. Concurrent conditions of cladding failure and overheat should be anticipated due to the radial distribution of heat generation within the core. Distinction between the type of core damage requires the identification of the characteristic fission products. The procedure for core damage assessment using radiological analysis of fluid samples is required to explicitly distinguish between the categories.

4.6 This methodology is limited in applicability to those conditions in which the fission product inventory in the core has had sufficient time to reach equilibrium. Equilibrium fission product inventory is a function of reactor power and burnup. Based upon the fission products of concern equilibrium conditions are achieved after thirty days of operation at constant power. Constant power is considered to include changes of no greater than ± 10 percent. The methodology may be used following non-constant periods of operation by using engineering judgement to select the most representative power level during the period. This method may also be used if the reactor has produced power for less than thirty days, however, the resulting assessment of core damage would be an underprediction of the actual conditions.

5.0 INSTRUCTIONS

5.1 Record the following plant indications.

5.1.1 Containment Building:

Radiation Dose Rate ____________ Rads/hr.

Time of Measurement Date________ Time__________

5.1.2 Prior 30 days power history:

Power, Percent	Duration, Days
______________	______________
______________	______________
______________	______________
______________	______________
______________	______________

5.1.3 Time of reactor trip

Date________ Time_________

Record these values in Enclosure D2.

5.2 Plant Power Correction

The measured radiation dose rate inside the containment building is to be corrected for the plant power history. A correction factor is used to adjust the measured dose rate to the corresponding value had the plant been operating at 100 percent power.

5.2.1 To correct the radiation dose rate for the case in which plant power level has remained constant for a period greater than 30 days a simple ratio of the power may be employed. The reactor power is considered to be constant if it has not changed by ± 10 percent within the last thirty days prior to the reactor trip.

5.2.2 To correct the radiation dose rate for the case in which reactor power level has not remained constant during the 30 days prior to the reactor shutdown engineering, judgement is used to determine the most representative power level. The following guidelines should be considered in the determination.

5.2.2.1 The average power during the 30 day time period is not necessarily the most representative value for correction to equilibrium conditions.

5.2.2.2 The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

5.2.2.3 Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

5.2.3 In the case in which reactor has produced power for less than 30 days the procedure may be employed. However, the estimate of core damage obtained under this condition may be an under prediction of the actual condition.

5.2.4 The following equation is applied to determine the radiation dose rate corresponding to equilibrium full power source inventory conditions.

$$\text{Equilibrium Dose Rate (Rad/Hr)} = \text{Measured Dose Rate (Rad/Hr)} \times \frac{100}{\text{Reactor Power Level (\%)}}$$

The reactor power level and the resulting dose rate correction factor used above will be the same for all subsequent measurement of the dose rate. Record these values to reduce the work required to evaluate the subsequent measurements.

5.3 The decay correction for the radiation dose rate requires the determir tion of the time duration between the reactor trip and the measurement of the dose rate. This is done simply using the time of reactor shutdown recorded in Section 5.1.3.

5.4 The conclusion on the extent of core damage is made using the equilibrium dose rate, the duration of reactor shutdown, and the analytically determined dose rates provided in Enclosure D3. The equilibrium dose rate is plotted on Enclosure D3 as a function of time following reactor shutdown. Engineering judgement is used to determine which category of core damage shown on Enclosure D3 is most representative of the particular value that has been plotted. The following criteria should be considered in the determination.

5.4.1 Dose rate measurements may have been recorded during periods of transient conditions within the plant. Measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.

5.4.2 Dose rates significantly above the lower bound for the category of major fuel overheat may indicate concurrent fuel pellet melting. The methodology in this section may not be emmployed to estimate the degree of fuel pellet melting.

5.4.3 Dose rates within any category of fuel overheating may be anticipated to include concurrent fuel cladding failure. The methodology in this section may not be used to distinguish the relative contributions of the two categories to the total dose rate. The methodology does give the estimate of the highest category of damage.

5.4.4 Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Enclosure D3. The evaluation of other plant parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

Enclosure D1

Radiologic Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release From Core	Source of Release	Percent of Source Inventory Released to Containment	Distribution of Fission Products in Containment
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	Less than 1	Airborne
2.	Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Less than 10	Airborne
3.	Intermediate Cladding Failure		Gas Gap	10 to 50	Airborne
4.	Major Cladding Failure		Gas Gap	Greater than 50	Airborne
5.	Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Less than 10	Airborne: 100% Noble Gas, 25% Halogen; Plated Out: 25% Halogen, 1% Solids
6.	Intermediate Fuel Pellet Overheating		Fuel Pellet	10 to 50	
7.	Major Fuel Pellet Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet	Greater than 50	

ENCLOSURE D2

CONTAINMENT HIGH RANGE RADIATION MONITOR (CORE DAMAGE ASSESSMENT) WORKSHEET

Highest Radiation Dose Rate (CHRRM) ______________ Rads/hr

Time of Measurement Date: ____________ Time: ____________

Prior 30 Days Power History:

Power, Percent	Duration, Days
____________	____________
____________	____________
____________	____________
____________	____________

Time of Reactor Trip Date: ____________ Time: ____________

Equilibrium Dose Rate (Rad/hr) =

$$\text{Measured Dose Rate (Rad/hr)} \times \frac{100}{\text{Reactor Power Level (\%)}}$$

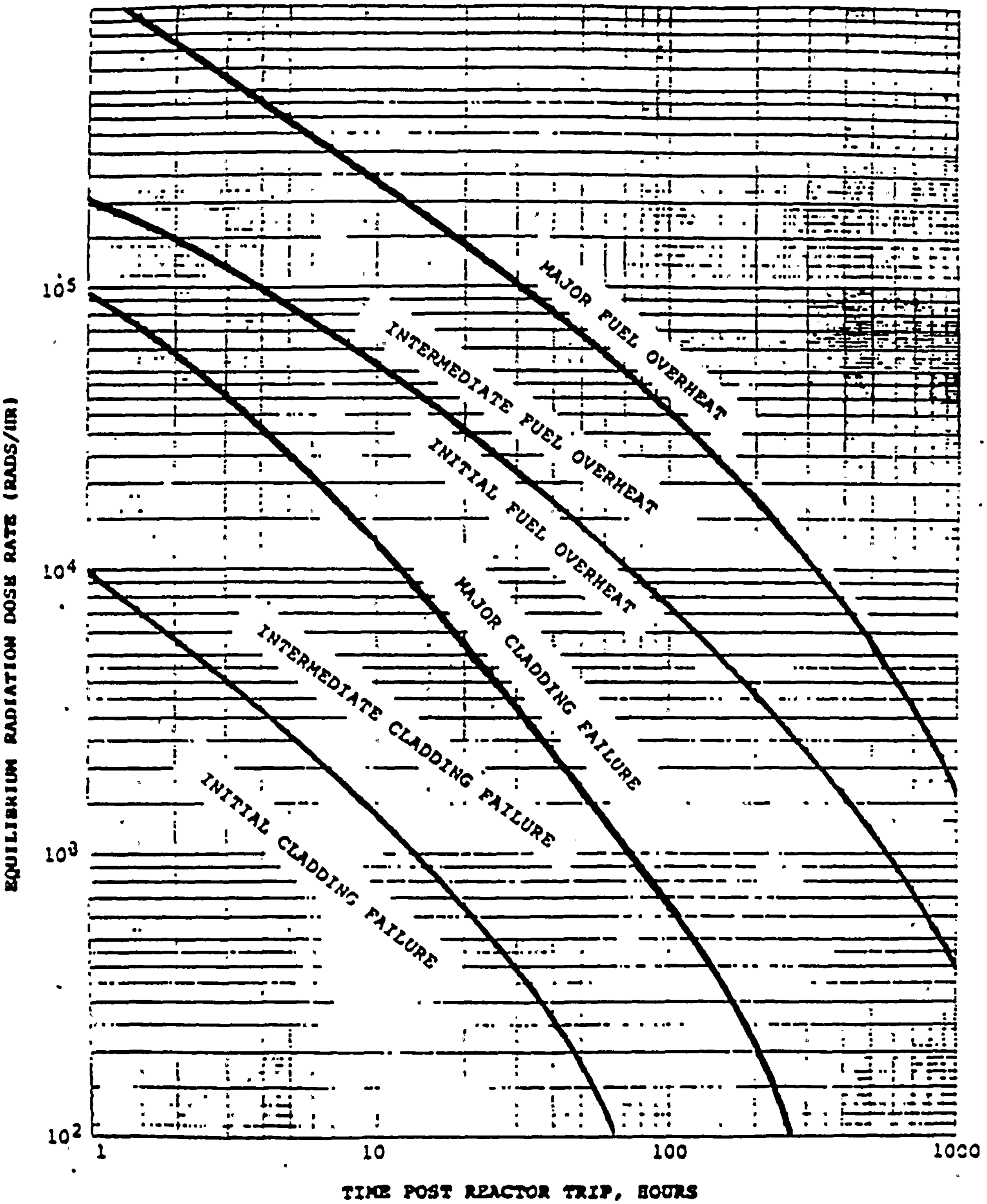
Refer to Enclosure D3 to obtain category of core damage. See step 5.4.

Performed by: ____________

Date: ____________

ENCLOSURE D3

ANALYSIS FOR POST ACCIDENT
DOSE RATE INSIDE CONTAINMENT
(CONTAINMENT HIGH RANGE RADIATION MONITOR)



SECTION E

SUMMARY OF RESULTS

SUMMARY OF RESULTS:

Section A, or radiological analysis of samples, is the most complete and possibly the most accurate of the methods used to assess the degree of core damage. This section of the methodology provides the instructions required to determine core damage up to the major fuel melt category identified in the NRC guidelines for core damage assessment.

Other indicators which are described in Sections B, C and D of these methodology are limited to the fuel overheat category of core damage. Section B, which uses the hydrogen content of both the reactor coolant and containment building atmosphere for an indication of fuel cladding oxidation is most applicable within the fuel overheat category. Section C, which uses the information from reactor coolant core exit temperatures is most applicable within the cladding failure category. While Section D, use information from area dose rates within containment and it is most applicable within both cladding failure and fuel overheat categories of damage.

It is important to note that core damage is not anticipated to take place uniformly. Therefore a combination of one or more of the categories of fuel damage will most likely exist simultaneously. The results obtained from the radionuclide analysis should be compared with the results of the evaluation of other available indicators for a comprehensive assessment. If the results are in agreement, the core damage assessment is complete. If the results are not in agreement, a recheck of both analyses may be performed or certain indications may be discounted or weighted more heavily based on engineering judgement.

Appendix II

To

GO EPIP 1302

"CORE DAMAGE ASSESSMENT"

FOR

TURKEY POINT UNITS 3 & 4

Table of Contents

1.0 DISCUSSION

The purpose of this Appendix is to provide a step-by-step procedure for assessing the extent of core damage at Turkey Point based on post-accident radionuclide concentrations within the RCS and containment. In addition, this procedure provides for the use of auxiliary indicators, such as containment hydrogen, containment high range radiation monitor, and core exit thermocouple data to substantiate the radiological analyses results.

The methodology presented herein conforms to the general guidelines developed by Westinghouse under contract to the Westinghouse Owner's Group, "Westinghouse Owner's Group Post-Accident Core Damage Assessment Methodology" with appropriate plant specific modifications.

The format of the Appendix is "Instruction on Worksheet", and presumes the user is familiar with the basis for the methodology.

The analyses methods and their range of core damage assessment are listed below:

Radiological Analysis:	Percent of clad damage, fuel overheat, or fuel melt
Hydrogen Concentration:	General category of clad damage
Containment High Range: Radiation Monitor	General Category of clad damage, fuel overheat, or fuel melt
Core Exit Thermocouple:	General Category of clad damage
Reactor Vessel Level:	Potential core damage; Yes or No

2.0 SUMMARY OF DATA REQUIRED

In order to complete this procedure, the information listed in the example Plant Data Sheet must be provided.

3.0 RADIOLOGICAL ANALYSIS

3.1 POST ACCIDENTS SAMPLING

Table II - 1 presents a list of sampling locations recommended for different accident scenarios.

Samples of the selected nuclides, which are listed in Table II - 2, should be taken at the indicated locations after the accident is defined.

3.2 DETERMINATION OF SOURCE INVENTORY

Determine the source inventory by performing a power correction of the equilibrium activity. Complete Table II-3.

3.3 DETERMINATION OF RELEASE PERCENTAGES

3.3.1 Calculate decay corrected specific activity for all nuclides by completing Table II - 4. See instructions below:

For those isotopes in Table II - 2 which do not list a parent (i.e., radiosotopes that are not Xenons, Rb-88, La-140, or I-132) Calculate the decay factor as follows:

Decay factor = $e^{\lambda t}$
e = 2.7183

Where: λ = Decay Constant from Table II - 2

t = Time from Reactor shutdown to sample analysis (seconds)

For isotopes in Table II-2 which list a parent, it is necessary to calculate both the decay factor and the Parent-Daughter Factor. This is required because of the "Daughter" isotope measured, some will be from the decay of the daughter concentration at time of shutdown and some will be from the decay of the parent. The Parent-Daughter factor is the theoretical fraction of the total measured daughter isotope which can be attributed to the decay of the Daughter concentration available at time of reactor shutdown.

Calculate the Parent-Daughter Factor (F_r) for the daughter nuclides as follows:

$$Fr = \frac{Q_d^o \, e^{-\lambda dt}}{Q_d}$$

Where:

$$Q_d = K \frac{\lambda_d}{\lambda_d - \lambda_p} Q_p^o \left\{ e^{-\lambda_p t} - e^{-\lambda_d t} \right\} + Q_d^o e^{-\lambda_d t}$$

K = Decay branching factor

λd = Daughter decay constant, See Table II - 2 on Page 10

λp = Parent decay constant, See Table II - 2 on Page 10

Q_p^o = Parent source Inventory, See Table II - 3 on Page 11

Q_d^o = Daughter Source Inventory, See Table II - 3 on page 11

t = Time from post-accident reactor shutdown to sample analysis time. (Seconds)

Q_d - The total theorectical daughter activity at the time of sample.

* Fr is only applicable for daughter nuclides. For parent nuclides this factor is ignored.

Calculate the decay factor for both parent and daughter nuclides.

Decay factor = $e^{\lambda t}$
e = 2.7183

Where: λ = Decay constant from Table II - 2
t = Time from reactor shutdown to sample analysis (seconds)

3.3.2 Containment Activity

Calculate containment activity for samples obtained from the containment atmosphere by completing TABLE II - 5.

3.3.3 Sump Activity

Calculate sump activity for samples obtained from the sump by completing Table II - 6.

3.3.4 RCS Activity

Calculate RCS activity for samples obtained from the reactor coolant system by completing Table II - 7.

3.3.5 Inventory Released

Calculate total activity and % inventory released by completing Table II - 8.

3.4 CORE DAMAGE ASSESSMENT

Determine the level of core damage by completing Table II - 9. Enter results into "Summary Worksheet" on Page 28.

4.0 SUBSTANTIATION OF CORE DAMAGE STATE USING AUXILIARY INDICATORS

Utilize containment hydrogen concentration, Reactor vessel level, core exit thermocouples, and containment radiation readings to substantiate radiological analysis results by completing Tables II - 12, 13, and 14. Enter results into "Summary Worksheet" on page 28.

CORE DAMAGE ASSESSMENT
PLANT DATA SHEET

SAMPLING CONDITIONS

Unit: ______________

Time of Reactor Trip: Date/Time: _______

Operating Point __________

Prior 40 Days Power History	
Power %	Duration, Days
______	____________
______	____________

Reactor Coolant System:

Pressure _______ PSIG

Temperature (T_{avg}) _______ ^{o}F

Reactor Vessel Level Shows _______ ft.
(Lowest Recorded Reading)

Pressurizer Level _______ %

Containment Building:

Atmosphere Pressure _______ PSIG

Atmosphere Temperature _______ ^{o}F

Sump Water Level _______ ft.

Tank Data:

Refueling Water Storage Tank Level _______ ft.

Safety Injection Accumulators Injected (yes/no): _______

Auxiliary Indicators:

Containment Hydrogen Concentration: _______ V/O H_2

Core Exit Thermocouples: _____ ^{o}F _____ ^{o}F _____ ^{o}F
(Three highest reliable readings)

Containment Radiation Monitor RDG: _______ R/hr

Loose Parts Monitoring System Alarm (Yes/No): _______

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RADIONUCLIDE DATA

SAMPLE NUMBER: ________

SAMPLE LOCATION (RCS, SUMP, CONTAINMENT): ________

TIME OF SAMPLE COLLECTION: ________

ISOTOPE	MEASURED SPECIFIC ACTIVITY @ STP A (uCi/cc)
Kr-85m	
Kr-87	
Kr-88	
Xe-131m	
Xe-133	
Xe-133m	
Xe-135	
I-131	
I-132	
I-133	
Cs-134	
Cs-137	
Te-132	
Ba-140	
La-140	
Rb-88	
I-135	

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TABLE II - 1
Suggested Sampling Locations

Scenario	Principal Sampling Locations	Other Sampling Locations
Small Break LOCA		
* Reactor Power > 1%	RCS Hot Leg Containment Atmosphere	RCS Pressurizer
* Reactor Power < 1%	RCS Hot Leg**	RCS Pressurizer
Large Break LOCA		
* Reactor Power > 1%	Containment Sump, Containment Atmosphere, RCS Hot Leg	
* Reactor Power < 1%	Containment Sump, Containment Atmosphere	
Steam Line Break	RCS Hot Leg,	RCS Pressurizer Containment Atmosphere
Steam Generator Tube Rupture	RCS Hot Leg, Secondary System	Containment Atmosphere
Indication of Significant Containment Sump Inventory	Containment Sump, Containment Atmosphere	
Containment Building Radiation Monitor Alarm	Containment Atmosphere, Containment Sump	
Safety Injection Actuated	RCS Hot Leg	RCS Pressurizer
Indication of High Radiation Level in RCS	RCS Hot Leg	RCS Pressurizer

* Assume operating at that level for some appreciable time.
** If a RCS hot leg sample is unavailable and a RCS cold leg sample is available, obtain a RCS cold leg sample. However, for a cold leg sample to be a good representation of the RCS, the primary water should be circulating through the system.

TABLE II - 2

CORE DAMAGE CATEGORY VS. RADIOISOTOPIC DATA

ISOTOPE	DECAY CONSTANT, SEC^{-1}	PARENT	DECAY CONSTANT OF PARENT, SEC^{-1}	DECAY BRANCHING FACTOR K
		CLAD FAILURE		
Kr-85m	6.313 E-5			
Kr-87	2.193 E-4			
Kr-88	9.921 E-5			
Xe-131m	9.809 E-7	I-131	1.438 E-6	0.008
Xe-133	2.196 E-6	I-133	1.386 E-5	0.976
Xe-133	2.196 E-6	Xe-133m	5.121 E-6	1.00
Xe-133m	5.121 E-6	I-133	1.386 E-5	0.024
Xe-135	3.039 E-5	I-135	4.158 E-5	0.70
*I-131	1.438 E-6			
I-132	1.229 E-4	TE-132	3.575 E-6	1.00
*I-133	1.386 E-5			
I-135	4.158 E-5			
+Rb-88	9.363 E-4	Kr-88	9.921 E-5	1.00
		FUEL OVERHEAT (ABOVE AND)		
*Cs-134	1.585 E-8			
Cs-137	1.057 E-9			
		FUEL MELT (ABOVE AND)		
*La-140	6.906 E-6	Ba-140	9.042 E-7	1.00
Te-132	3.575 E-6			

* Denotes leading indicator. If a quick assessment of core damage is desired, utilize leading indicator. Otherwise, use as many as are available.

\+ Rb-88 is used only as an indicator of clad failure.

TABLE II - 3

DETERMINATION OF FISSION PRODUCT SOURCE INVENTORY, SI, IN CURIES

NOTE:

This method is based on constant power level for a minimum of 40 days prior to shutdown or trip. Otherwise, see PCF Worksheet Page 29.

ISOTOPE	EQUALIB ACTIVITY, Ci	PCF*	CORRECTED ACTIVITY, Ci
Kr-85m	1.319 (7)		
Kr-87	2.528 (7)		
Kr-88	3.556 (7)		
Rb-88	3.612 (7)		
I-131	5.057 (7)		
Xe 131m	5.628 (5)		
Te 132	7.155 (7)		
I-132	7.271 (7)		
I-133	1.028 (8)		
Xe-133	1.029 (8)		
Xe-133m	2.705 (6)		
#Cs-134			
I-135	9.604 (7)		
Xe-135	2.709 (7)		
#Cs-137			
Ba-140	8.813 (7)		
La-140	9.053 (7)		

For Cs134
Equalib Activity = 2.984 (6) + D x 1.04 (4), Curies

For Cs 137
Equalib Activity = 2.653 (6) + D x 6.91 (3), Curies

Where D = number of EFPD since BOC

* PCF = Power Correction Factor

NOTE: NI means not identified

$$PCF = \frac{\text{LAST 40 DAY RUN MWt}}{\text{PLANT RATED MWt (2200)}}$$

Performed by: ______________
Date: ______________

TABLE II - 4

DECAY CORRECTED SPECIFIC ACTIVITIES OF SAMPLING ANALYSIS

SEE PLANT DATA SHEET			SEE PAGE 5	PAGE 4	CALCULATE
Nuclide	Location	Measured Activity	Parent- x Daughter x Factor (F_r)	Decay Factor =	Decay Corrected Specific Activity (uCi/cc) at time of S/D

N/A = Not Applicable

Performed by: ____________

Date: ____________

Note: See Section 3.3 of procedure for definition of terms.

Calculate Parent-Daughter Factor for Daughter nuclides using the following formula

$$Fr = \frac{Q_d^o \quad e^{-\lambda dt}}{Q_d}$$

Where:

$$Q_d = K \frac{\lambda d}{\lambda d - \lambda p} Q_p^o \left\{ e^{-\lambda_p t} - e^{-\lambda_d t} \right\} + Q_d^o e^{-\lambda_d t}$$

Calculate the Decay Factor for both Parent and Daughter nuclides as follows:

Decay factor = $e^{\lambda t}$

$e = 2.7183$

TABLE II - 5

CONTAINMENT ATMOSPHERE ACTIVITY, CURIES

Performed by: __________
Date: __________

Isotope	Decay Corrected Specific Activity	x	Atm Vol.	=	Activity, Ci

Containment Activity

Step 1: Calculate containment volume in cc, including pressure and temperature corrections.

$$V_c = 4.387 \times 10^{10} \quad \times \quad \frac{P_2}{P_1} \times \frac{(T_1 + 460)}{(T_2 + 460)}$$

$P_2 = 14.7$ psia

$T_2 = 32^oF$

If sample temperature and pressure are different from the temperature and pressure of the medium from which the sample was taken, then

P_1 = sample pressure
T_1 = sample temperature

If they are the same, then

P_1 = containment pressure
T_1 = containment temperature

Step 2: Calculate containment activity for samples obtained from the containment atmosphere

Containment Activity = V_c x (decay corrected specific activity)

TABLE II - 6

CONTAINMENT SUMP ACTIVITY

Isotope	Corrected Specific Activity	x	Sump Volume	=	Activity, Ci

Performed by: __________

Date: __________

Sump Activity

Step 1: Find sump water volume.

$$V_S = \text{Water level Monitor Reading (ft)} \times 11{,}360.7\ ft^2 \times 28.3 \times 10^3 \frac{cc}{ft^3}$$

Step 2: Calculate sump activity for samples obtained from the sump.

Sump activity = V_S x (decay corrected specific activity)

TABLE II - 7

REACTOR COOLANT SYSTEM ACTIVITY, CURIES

Isotope	Decay Corrected Specific	x	Water Density Ratio	=	Density Corrected Specific Activity	x	RCS Volume	=	Activity, Ci

RCS Activity

Performed by: ________

Date: ________

Step 1: Find Refueling Water Storage Tank Volume.

$$V_{RWST} = \text{Water level Monitor Reading (ft)} \times 962.113\ (ft^2) \times 28.3 \times 10^3\ \frac{cc}{ft^3}$$

Step 2: Find RCS water volume.

i) Prior to recirculation,
$V_{RCS} = 2.660 \times 10^8 + (1.211 \times 10^9 - V_{RWST}) + V_{ACCUM} - V_s$

ii) During recirculation
$V_{RCS} = 2.660 \times 10^8 + (1.211 \times 10^9 - V_{RWST}) + V_{ACCUM}$
Where $V_{ACCUM} = 7.56 \times 10^7$ (only if accumulators are injected)

Step 3: Determine the density ratio by entering Figure 1 with RCS temperature.

Step 4: Calculate density corrected specific activity.

Density Corrected Specific Activity = Decay Corrected Specific Activity x Density Ratio

FIGURE 1

WATER DENSITY RATIO

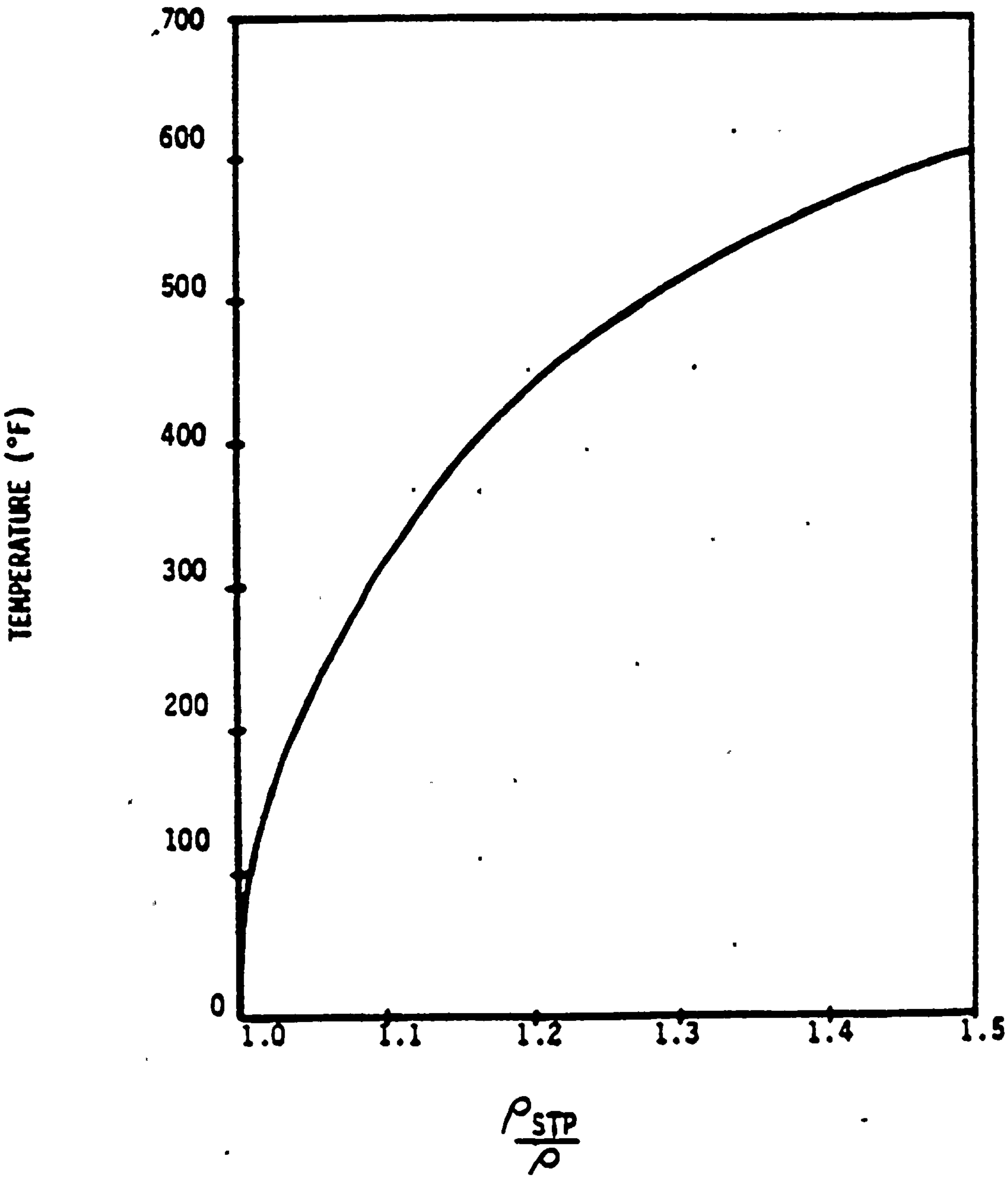


TABLE II - 8

TOTAL ACTIVITY AND
INVENTORY RELEASED

Isotope	See Page 13 Cont. Atm. Activity	+	Page 14 Cont. Sump Activity	+	Page 15 RCS Activity	=	Calculate Total Activity	÷	See Page 11 Source Inventory	x	Calculate 100 = % Inv. Released

Performed by: ____________

Date: ____________

TABLE II - 10

NORMAL OPERATING ACTIVITIES, CURIES*

Nuclide	Activity
Kr-85m	268.84
Kr-87	184.46
Kr-88	662.78
Rb-88	660.18
Xe-131m	325.27
Xe-133	44,185.61
Xe-133m	481.78
Xe-135	1,216.53
I-131	399.27
I-132	147.46
I-133	644.15
I-135	346.03

* These numbers are based upon an RCS volume of 9400ft^3 and specific operating activities in CORE DAMAGE ASSESSMENT PROCEDURE.

TABLE II - 11

DETERMINATION OF FUEL ROD GAS GAP ACTIVITY, GI, IN CURIES

NOTE:

This method is based on constant power level for a minimum of 40 days prior to shutdown or trip. Otherwise, see PCF Worksheet, Page 29.

Isotope	EQUALIB Activity, Ci	x	*PCF	=	Corrected Activity, Ci
Kr-85m	1.673 (3)				
Kr-87	9.734 (3)				
Kr-88	2.029 (4)				
Rb-88	6.732 (3)				
I-131	2.906 (5)				
Xe-131m	3.924 (3)				
I-132	4.493 (4)				
I-133	1.862 (5)				
Xe-133	4.657 (5)				
Xe-133m	9.644 (4)				
I-135	1.003 (5)				
Xe-135	3.107 (4)				

* PCF = Power Correction Factor

$$PCF = \frac{\text{Last 40 day run MW}}{\text{Plant Rated MW} \quad (2200)}$$

Performed by: ________

Date: ________

TABLE II - 9
CORE DAMAGE ASSESSMENT

Isotope	%IR	SI	NOA	GI	%CD	FUEL OVER-HEAT %FO	CORE MELT %CM

Performed by: ______
Date: ______

Clad Damage Assessment

Using the % Inventory Released calculated in TABLE II - 8, calculate % Clad Damage with the following equation where:

$$\% CD = \left[\frac{\left(\frac{\%IR}{100} \times SI\right) - NOA}{GI} \right] \times 100$$

%IR = Percent Inventory Released
%CD = Percent Clad Damage
GI = Gap Inventory (See Page 20)
NOA = Normal Operating Activity within RCS (From TABLE II - 10 on Pg. 19)
SI = Source Inventory (obtained on page 11)

Fuel Overtemperature Assessment

Using the % Inventory Released calculated in TABLE II-8 read % Fuel Overtemperature from Figure 2 or Figure 3. Use Figure 2 for Xe, Kr, I and Cs isotopes and use Figure 3 for Ba and La isotopes. (Use nominal values)

Core Melt Assessment

Using the % Inventory Released calculated in Table II - 8, read % Core Melt from Figure 4 or Figure 5. Use Figure 4 for Xe, Kr, I, Cs, and Te isotopes and use Figure 5 for Ba and La isotopes. (Use nominal values)

FIGURE 2

RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH
% INVENTORY RELEASED OF XE, KR, 'I, CS

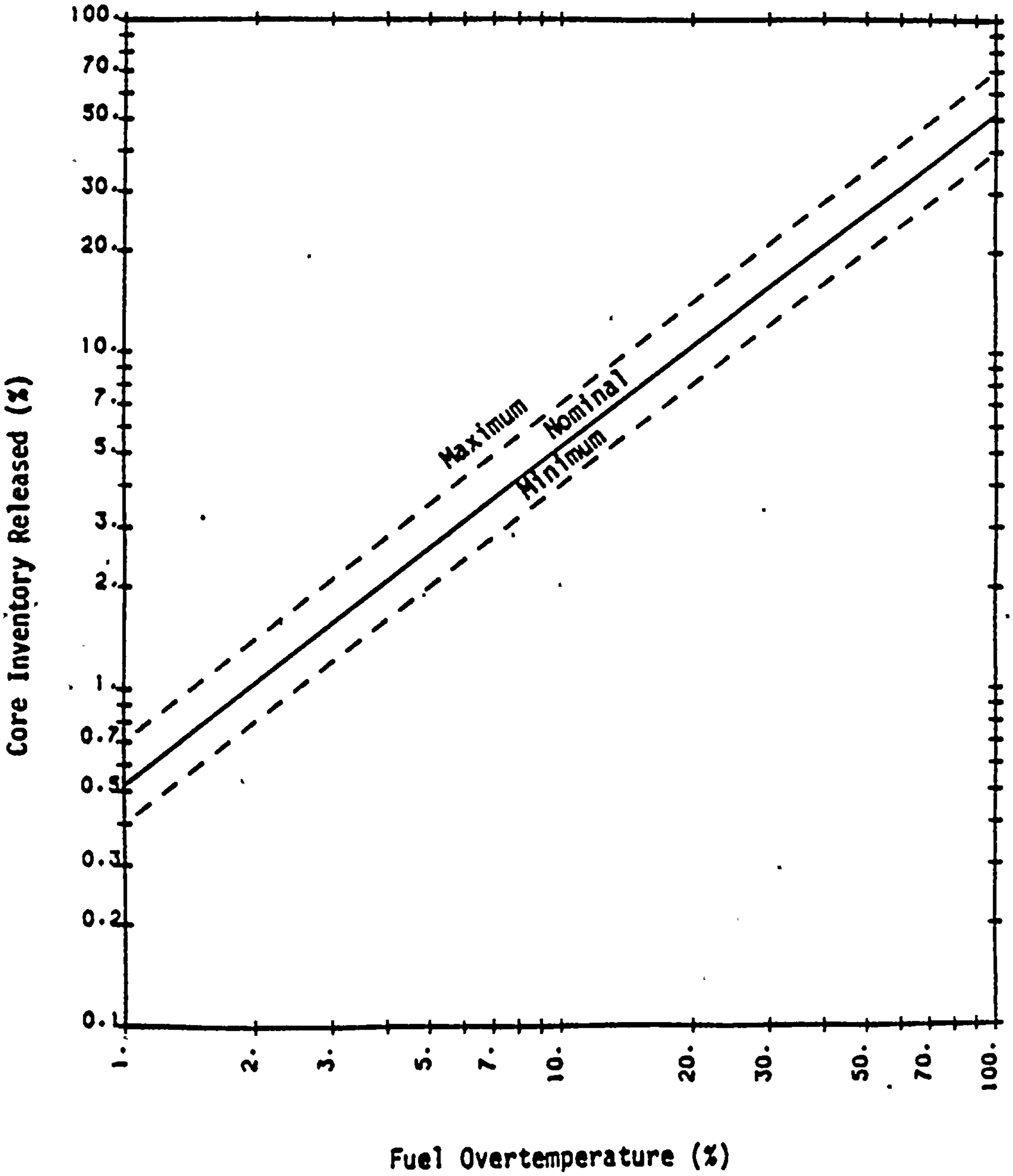


FIGURE 3

RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH

% INVENTORY RELEASED OF BA, La

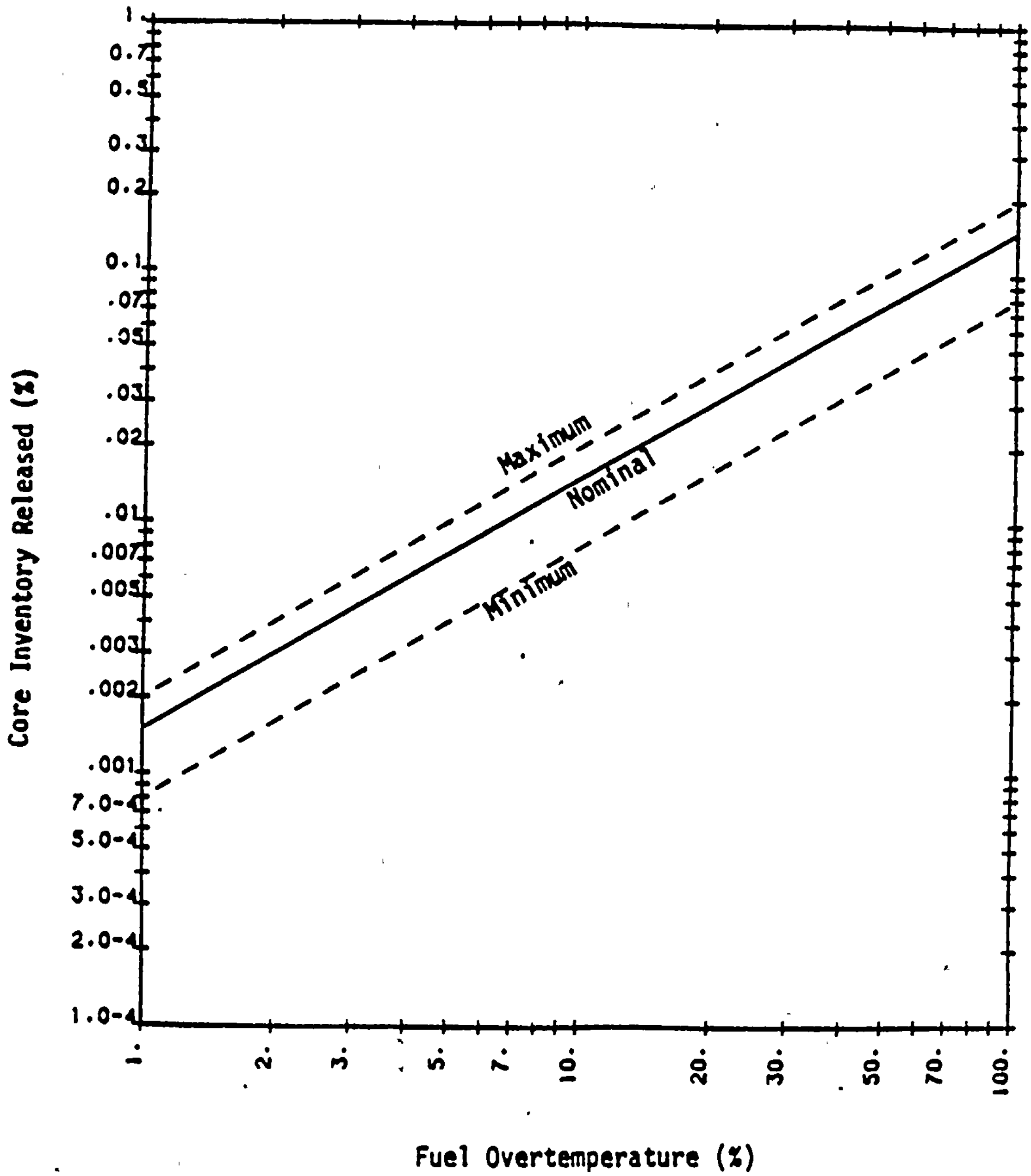


FIGURE 4

RELATIONSHIP OF % FUEL MELT WITH % INVENTORY
RELEASED OF XE, KR, CS, TE, I

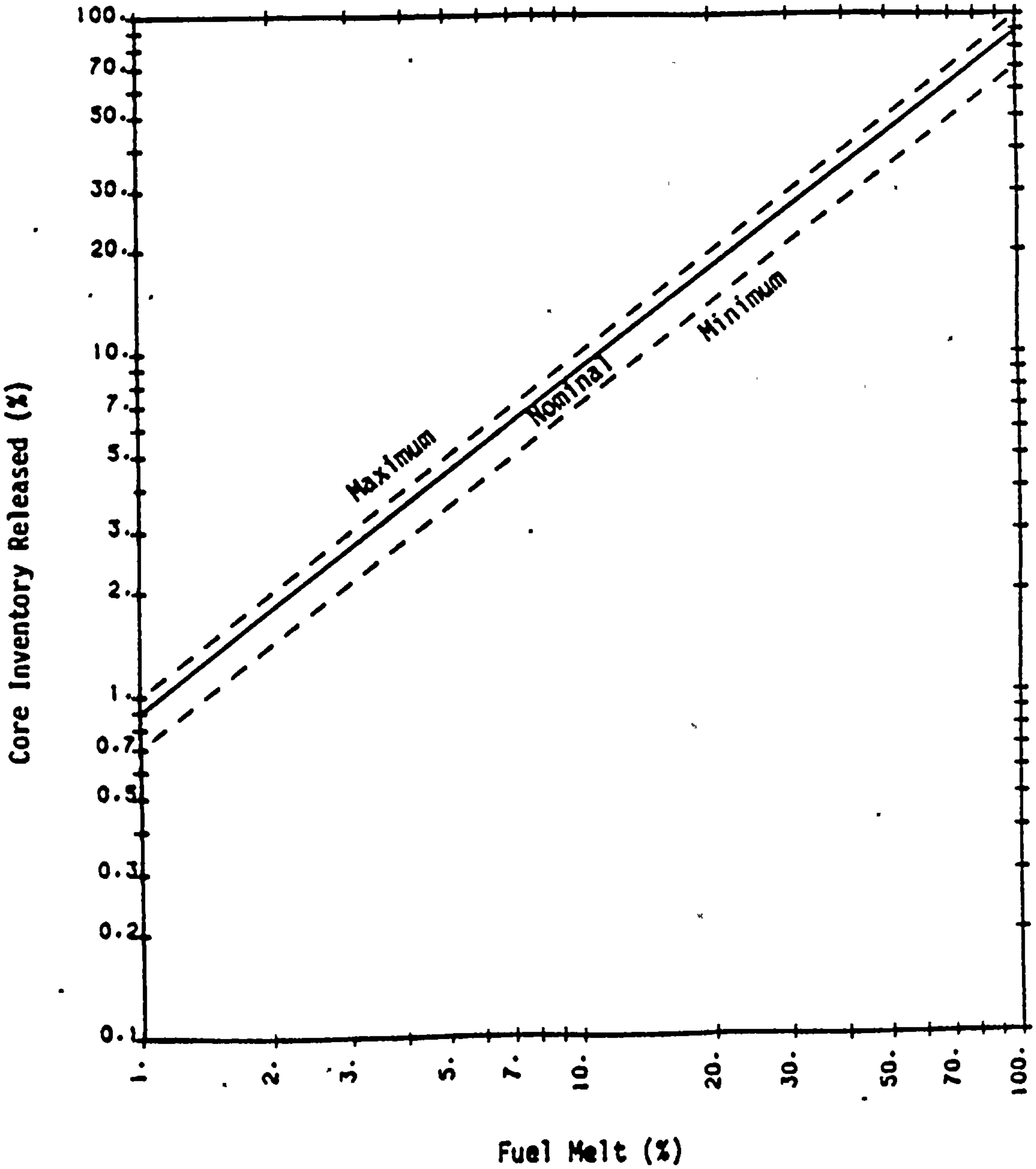


FIGURE 5

RELATIONSHIP OF % FUEL MELT WITH % INVENTORY
RELEASED OF BA, La

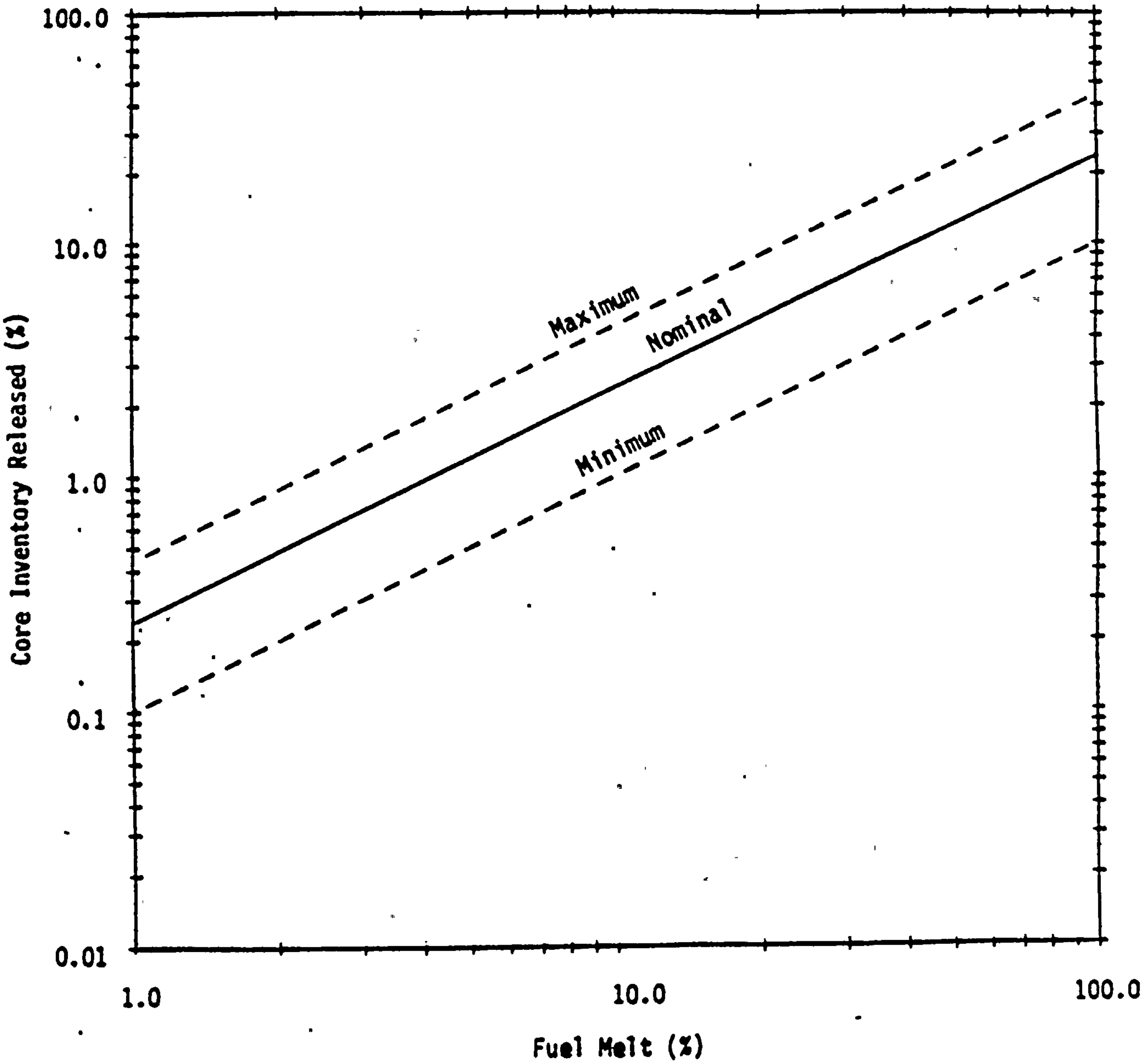


TABLE II - 12

CORE EXIT THERMOCOUPLES

STEP 1: Obtain at least three of the highest reliable CETC readings and average.

-CAUTION-

Reflux cooling from the hot legs may cool the fluid near the vessel walls and the CETC may read depressed temperatures.

-WARNING-

CETC readings >1650°F may indicate equipment malfunction.

STEP 2: Compare the average CETC temperature to the table below

Core Damage Category	Thermocouple Reading (°F)
No Clad Damage	750
0 - 50% Clad Damage	750 - 1300
50 - 100% Clad Damage*	1300 - 1650
Fuel overheat * or fuel melt	1650

* The extent of this damage state must be determined by other parameters.

STEP 3: Record the average temperature and core damage category.

CETC Average Temperature: ________ °F
Core Damage Category: ________

STEP 4: Enter data into "SUMMARY WORKSHEET" on Page 28.

Performed by: ________
Date: ________

TABLE II - 13

CONTAINMENT HIGH RANGE RADIATION MONITOR

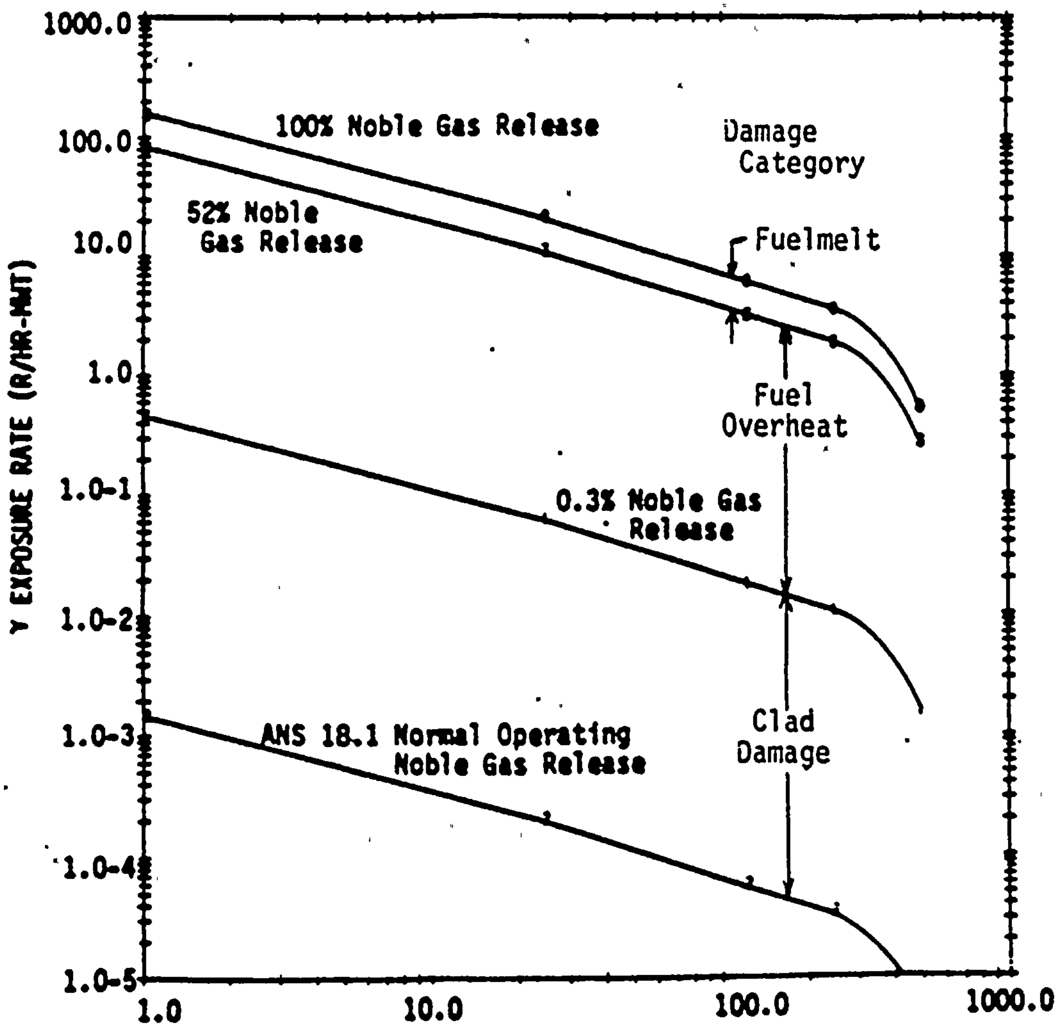
STEP 1: Obtain CHRRM Reading (R/HR) and elasped time from shutdown (or trip) to CHRRM Reading, Hrs.
CHRRM Reading ____________ R/HR Time __________ Hrs.

STEP 2: Calculate Exposure Rate = $\frac{\text{CHRRM R/HR}}{\text{2200 MWt}}$ x 0.775 = ___________

STEP 3: Enter exposure rate and time into graph below and determine

Core Damage Category: ____________________

STEP 4: Enter data into "SUMMARY WORKSHEET" on Page 28



Elapsed time from shutdown (or trip) (HOURS)

Performed by: ________
Date: ________

TABLE II - 14

HYDROGEN IN CONTAINMENT ATMOSPHERE

- NOTE -

The percentage of Zirc - Water reaction does not directly correspond to the percent clad damage. It can be used to verify the range of extent of clad damage estimated from radioisotopic analysis.

STEP 1: Read containment hydrogen monitor to determine the volume percent hydrogen concentration.

V/O H_2 = ___________

STEP 2: Determine the core damage category from the Table below.

Core Damage Category ___________

STEP 3: Enter into "SUMMARY WORKSHEET" on Page 28.

Core Damage Category	Hydrogen Monitor (v/O H_2)
No Clad Damage	negligible
0 - 50% Clad Damage	0 - 7.4
50 - 100% Clad Damage*	7.4 - 14.8

* This Core Damage Category may also include fuel pellet overtemperature or fuel melt. The extent of this damage state must be determined by other parameters.

Performed by: ________

Date: ________

SUMMARY WORKSHEET

Results of Radioisotopic Analysis:

________% Clad Damage ________% Over heat ________% Melt

Results of Auxiliary Indicators

Method			Category
CHRRM	______ $\frac{R}{Hr\text{-}MWt}$	______ ET	__________
H_2 Monitor	______ v/o		__________
CETC Avg.	______ °F		__________

Is RVLS below zero ☐ yes ☐ no

Has RVLS dropped below zero ☐ yes ☐ no

Summary of Results

1. Compare percent clad damage, percent fuel overheat and percent fuel melt results obtained from the radionuclide analysis to those obtained from the auxiliary indicators analyses.

2. If results are in agreement, the core damage assessment is complete. If the results are not in agreement, a recheck of both analyses may be performed or certain indications may be discounted based on engineering judgement.

Performed by ____________ Date ______ Time ______

Checked by ____________ Date ______ Time ______

POWER CORRECTION FACTORS FOR TRANSIENT POWER HISTORIES

A) For those isotopes listed in Table II-2, that have a decay constant, λ, greater than 8.0 E-6 AND the constant power run is greater than four days the PCF is the ratio of the 4 day run power level to the rated power level.

B) For the other isotopes except Cs-134, Cs-137 and power histories, the PCF is defined by the relationship:

$$PCF = \Sigma_j \frac{P_j \left(1 - e^{-\lambda i \, tj}\right) e^{-\lambda i \, t_j^0}}{RP}$$

Where:

P_j is the average MWt for interval t_j

t_j is the duration of P_j

λi is the length of time between the end of t_j and shutdown or trip

RP is the rated MWt

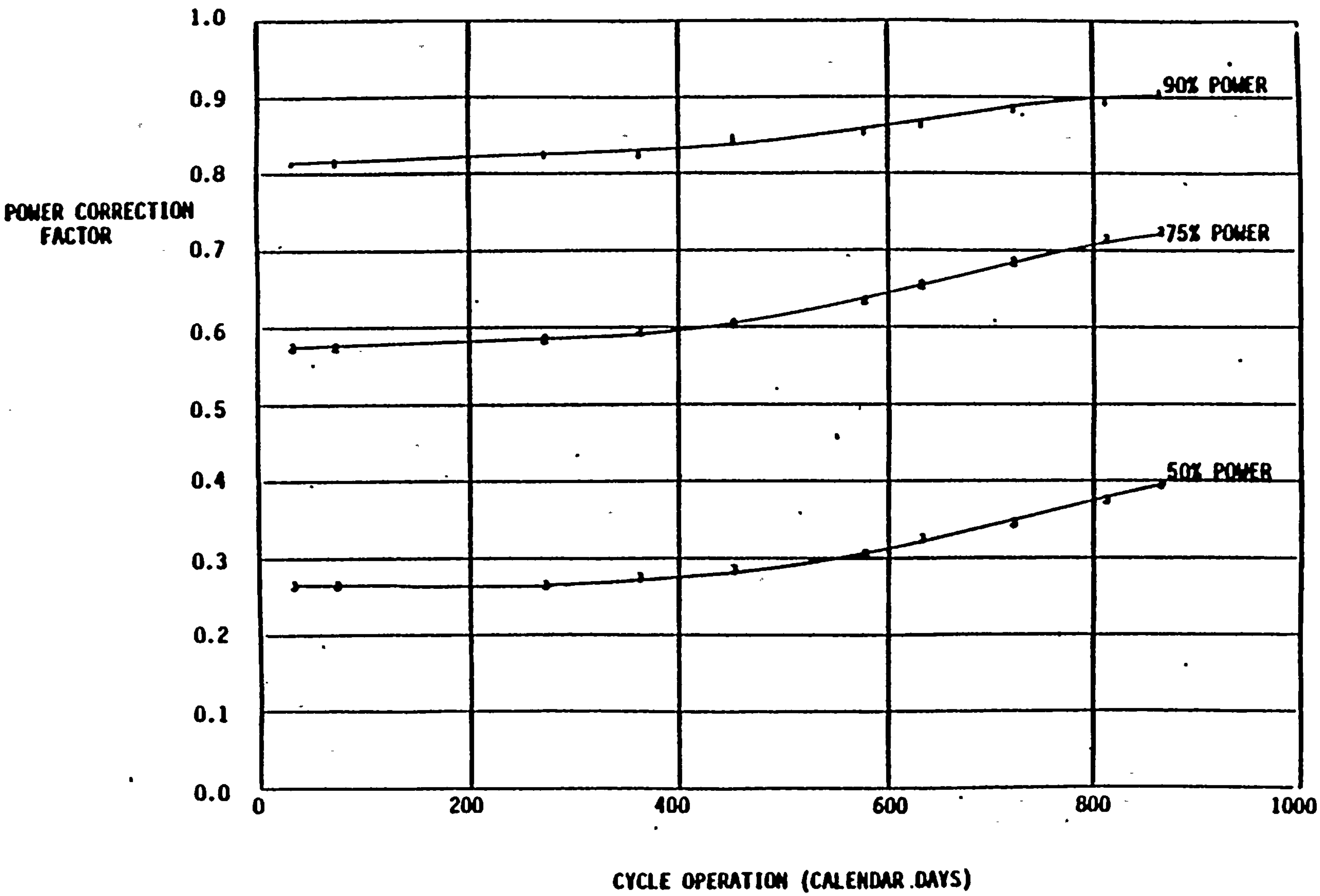
t_j^0 = time from end of period j to reactor trip (sec)

Note: If using percent instead of MWt, the Pj/RP term can be replaced with $\frac{\%\ \text{power}}{100}$

The entire 40 days power history should be employed.

C) For Cs-137, the relationship is the same, 2.653 (6) + D x 6.91 (3), Curies
Where D = number of EFPD since BOC

D) For Cs-134, utilize the figure on the next page.



POWER CORRECTION FACTOR FOR CS-134 BASED ON AVERAGE POWER DURING OPERATION

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