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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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10CFR50.4

April 14, 2003

U S Nuclear Regulatory Commission
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
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PRAIRIE ISLAND EMERGENCY PLAN IMPLEMENTING PROCEDURES

Furnished with this letter are the recent changes to the Prairie Island Nuclear Generating Plant Emergency Plan Implementing Procedures F3. This submittal includes the following documents:

INDEX:
Emergency Plan Implementing Procedures TOC

REVISIONS

F3-17	Core Damage Assessment	Rev. 11
F3-22	Prairie Island Radiation Protection Group Response to a Monticello Emergency	Rev. 17
F3-23.1	Emergency Hotcell Procedure	Rev. 13

DELETIONS:
None

TEMPORARY CHANGE DELETIONS:
2003 0080 F3-23.1 Emergency Hotcell Procedure

INSTRUCTIONS:
Instructions for updating the manual are included.

This letter contains no new commitments and no revisions to existing commitments.

A045

As per 10 CFR 50.4, two copies have also been provided to the NRC Region III Office and one to the NRC Resident Inspector. If you have any questions, please contact Mel Agen at 651-388-1121 Extension 7210.

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Steve Orth, USNRC, Region III (2 copies)
NRC Resident Inspector- Prairie Island Nuclear Generating Plant
(w/o attachment)

Attachment

Mfst Num: 2003 - 0259 Date : 04/02/03
FROM : Bruce Loesch/Mary Gadient Loc : Prairie Island
TO : UNDERWOOD, BETTY J
Copy Num: 515 Holder : US NRC DOC CONTROL DESK
SUBJECT : Revisions to CONTROLLED DOCUMENTS

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Procedure #	Rev	Title
Revisions:		
F3-22	17	PRAIRIE ISLAND RADIATION PROTECTION GROUP TO A MONTICELLO EMERGENCY
F3-23.1	13	EMERGENCY HOTCELL PROCEDURE
F3-17	11	CORE DAMAGE ASSESSMENT

Temporary Change Deletions:

2003 0080 F3-23.1 EMERGENCY HOTCELL PROCEDURE

UPDATING INSTRUCTIONS

Place this material in your Prairie Island Controlled Manual or File. Remove revised or cancelled material and recycle it. Sign and date this letter in the space provided below within ten working days and return to Bruce Loesch or Mary Gadient, Prairie Island Nuclear Plant, 1717 Wakonade Drive E., Welch, MN 55089.
Contact Bruce Loesch (ext 4664) or Mary Gadient (ext 4478) if you have any questions.

Received the material stated above and complied with the updating instructions

_______________________________________ Date ______________

PRAIRIE ISLAND NUCLEAR GENERATING PLANT	Title: Emergency Plan Implementing Procedures TOC
Approved By: Joyce Chitty /MG BPS Supt	Effective Date : 04/02/03 NOTE: This set may contain a partial distribution of this Document Type. Please refer to the CHAMPS Module for specific Copy Holder Contents.

F3	CORE DAMAGE ASSESSMENT	NUMBER: F3-17
		REV: 11

REFERENCE USE
• *Procedure segments may be performed from memory.*
• *Use the procedure to verify segments are complete.*
• *Mark off steps within segment before continuing.*
• *Procedure should be available at the work location.*

O.C. REVIEW DATE:	OWNER:	EFFECTIVE DATE
031803 SC	M. Werner	4-2-03

1.0 PURPOSE

The purpose of this procedure is to provide a means to best estimate the degree of reactor core damage from the measured fission product concentrations in water and gas samples taken for the primary system and containment under accident conditions.

2.0 APPLICABILITY

This procedure **SHALL** apply to the Nuclear Engineering Staff.

3.0 PRECAUTIONS

3.1 The numbers obtained using this procedure are at best, estimates only.

3.2 When making core damage calculations as per this procedure, considerations should be given to other plant indicators, for example:

3.2.1 Incore Thermocouples.

3.2.2 Reactor Coolant Loop Radiation Monitors (R70/71).

3.2.3 Containment Radiation Monitors (R48/49).

3.2.4 Hydrogen Concentration in the Containment Atmosphere

3.3 Spiking may occur after a shutdown or significant power change, usually during the 2 to 6 hour period following the power change. Iodine spiking is a characteristic of the condition where an increase in the normal primary coolant activity is noted, but no damage to the cladding has occurred.

4.0 RESPONSIBILITIES

The Nuclear Engineering Group is responsible to estimate the degree of reactor core damage according to the guidance provided in this procedure.

5.0 DISCUSSION

The approach utilized in this methodology of core damage assessment is measurement of fission product concentrations in the primary coolant system, and containment, when applicable, utilizing the post accident sampling system.

Certain nuclides have been selected to be associated with each particular core damage state, i.e., clad damage, fuel overheat and fuel melt. These nuclides reach equilibrium quickly within the fuel cycle. Once equilibrium condition are reached, a fixed inventory of the nuclides is assumed to exist within the fuel pellet. For these nuclides which reach equilibrium, their relative ratios within the fuel pellet can also be considered to be constant. During operation, certain volatile fission products collect in the gap. The relative ratios in the gap can also be considered to be constant, however, the distribution of the nuclides in the gap is not in the same proportion as the fuel pellet inventory since the migration of each nuclide into the gap is dependent on its particular diffusion rate. The relative ratios of the nuclides analyzed during an accident may be compared to the predicted relative ratios existing in the gap and fuel pellet to determine the source of the fission product release, i.e., gap release or fuel pellet.

Clad damage is characterized by the release of these fission products, i.e., isotopes of the noble gases, iodine, and cesium which have accumulated in the gap and during the operation of the plant. When the cladding ruptures, it is assumed that the fission product gap inventory of the damaged fuel rods is instantaneously released to the primary system. For this methodology it is assumed that the noble gases will escape through the break of the primary system boundary to the containment atmosphere and the iodines will stay in solution and travel with primary system water during the accident.

Fission product release associated with overtemperature fuel conditions arises initially from the portion of the noble gas, cesium and iodine inventories that was previously accumulated in grain boundaries. In addition, small amounts of the more refractory elements, barium-lanthanum, and strontium are also released.

Fuel pellet melting leads to rapid release of many noble gases, halides, and cesiums remaining in the fuel after overheat conditions. Significant release of the strontium, barium-lanthanum chemical groups is perhaps the most distinguishing feature of melt release conditions.

Auxiliary indicators such as core exit thermocouples, reactor vessel water level, reactor coolant loop radiation monitors, containment radiation monitors, and the containment hydrogen concentration are available for estimating core damage. These indications should confirm the core damage estimates which in turn are based on the radionuclide analysis.

6.0 PREREQUISITES

An emergency of an Alert, Site Alert, or General Emergency has been declared.

7.0 PROCEDURE

NOTE:	The program B80DAMASS may be used whenever core damage estimates are desired.

7.1 Request the Radiation Protection Group to obtain the applicable samples to enable an adequate assessment of core damage. See Table 1 for suggested sampling locations.

7.2 Obtain the following plant data at the approximate sample time:

7.2.1 Incore Thermocouple Map

7.2.2 Containment Pressure

7.2.3 Containment Temperature

7.2.4 Containment Hydrogen Concentration

7.2.5 Containment Radiation Level

7.2.6 Containment Sump Level

7.2.7 RVLIS Level

7.3 Perform B80DAMASS according to the instructions in SWI-NE-5 (23) to obtain core damage estimates. Continue with Step 7.15 of this procedure when the B80DAMASS run is complete.

NOTE:	If the computer is not available, perform the following manual calculations to obtain core damage estimates.

7.4 Decay correct the specific activities determined by the sample analysis, back to the time of reactor shutdown, as follows:

NOTE:	The decay correction may have been accomplished by the computer during the spectrum analysis. Therefore, this step may not need to be completed.

$$A_0 = \frac{A}{e^{-\lambda_i t}}$$

Where:

A = measured specific activity, µCi/gm or µCi/cc

λ_i = decay constant of isotope i, sec^{-1}

t = time elapsed from reactor shutdown to time of sampling, sec.

A_0 = decay corrected specific activity µCi/gm or µCi/cc

7.5 If a parent-daughter relationship exists for a specific isotope, the following steps should be followed to calculate the fraction of the measured activity due to the decay of the daughter that was released and then to calculate the activity of the daughter released at shutdown.

7.5.1 Calculate the hypothetical daughter concentration (Q_B) at the time of the sample analysis assuming 100 percent release of the parent and daughter source inventory:

$$Q_B(t) = K_i \frac{\lambda_B}{\lambda_B - \lambda_{Ai}} Q^0_{Ai}\left(e^{-\lambda_{Ai}t} - e^{-\lambda_B t}\right) + Q^0_B\, e^{-\lambda_B t}$$

Where:

Q^0_{Ai} = 100% source inventory (Ci) of parent i, Table 2 or Table 4.

Q^0_B = 100% source inventory (Ci) of daughter, Table 2 or Table 4.

$Q_B(t)$ = hypothetical daughter activity (Ci) at sample time.

K_i = if parent has 2 daughters, K_i is the branching factor, Table 3.

λ_{Ai} = decay constant of parent i, sec^{-1}

λ_B = daughter decay constant, sec^{-1}

t = time period from shutdown to time sample, sec.

7.5.2 Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time:

$$Fr = \frac{Q^0_B \, e^{-\lambda_B t}}{Q_B\,(t)}$$

7.5.3 Calculate the amount of decay corrected sample specific activity associated with just the daughter that was released.

$$M^0_B = Fr \times A_0$$

Where: A_0 = decay corrected specific activity (µCi/gm or µCi/cc) as determined by the analysis.

7.6 Determine the total volume or mass of the medium which was sampled.

7.6.1 Containment Volume:

V = containment free volume (cc's)

= 3.74×10^{10} cc's

7.6.2 Liquid Mass:

A. Liquid temperature < 200°F

$$\text{Mass (gms)} = \text{volume (ft3)} \times \rho_{STP} \times \frac{28.3 \times 10^3 \text{cc}}{\text{ft}^3}$$

Where: ρ_{STP} = water density at STP = 1.0 gm/cc

B. Liquid temperature > 200°F

$$\text{Mass (gms)} = \text{volume (ft}^3\text{)} \times \frac{\rho}{\rho_{STP}} \text{(2)} \times \rho_{STP} \times \frac{28.3 \times 10^3 \text{cc}}{\text{ft}^3}$$

Where: $\frac{\rho}{\rho_{STP}}$ (2) = water density ratio at medium temperature, from Figure 1

ρ_{STP} = water density at STP = 1.0 gm/cc

7.7 Determine the total activity of each isotope in each medium.

7.7.1 Containment Atmosphere:

$$\text{Total containment Activity (curies)} = A_0\ (\mu Ci/cc) \times V\ (cc's) \times \frac{\text{Curie}}{1 \times 10^6 \mu Ci}$$

Where: A_0 = Specific activity of containment atmosphere ($\mu Ci/cc$), decay corrected to time of reactor shutdown and temperature/pressure corrected.

V = containment free volume (cc's)

= 3.74×10^{10} cc's

7.7.2 Liquid Sample:

$$\text{Total Liquid Activity (Curies)} = \text{Liquid MASS (gms)} \times A_0\ (\mu Ci/cc) \times \frac{\text{Curie}}{1 \times 10^6 \mu Ci}$$

Where: A_0 = Specific activity of liquid sample ($\mu Ci/gm$), decay corrected to time of reactor shutdown.

7.8 The approximate total activity of each isotope in the liquid samples can now be calculated.

Total Water Activity = RCS Activity + Sump Activity + Activity Leaked to Secondary System.

7.9 Now the total activity of each isotope released at the time of the accident can be determined:

Total Activity Released = Total Water Activity + Containment Atmosphere Activity

7.10 Utilizing the total activity of each isotope released, calculate the activity ratios of the released fission products.

7.10.1 Noble Gas Ratio $= \frac{\text{Noble Gas Activity}}{\text{Xe}-133 \text{ Activity}}$

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

NOTE: **Steady state power conditions may be assumed where power does not vary by more than ± 10% of rated power level from time averaged value.**

7.11 Determine the power history prior to reactor shutdown.

7.12 Using the power history, determine a power correction factor for each isotope, in accordance with the following guidelines:

NOTE: **Steady state power condition is assumed where the power does not vary by more than ± 10% of rated power level from time averaged value.**

7.12.1 Steady State power prior to shutdown.

A. Half-life of nuclide < 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 4 Days}}{\text{Rated Power Level (Mwt)}}$$

B. Half-life of nuclide > 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 30 Days}}{\text{Rated Power Level (Mwt)}}$$

C. Half-life of nuclide ~ 1 year

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 1 year}}{\text{Rated Power Level (Mwt)}}$$

7.12.2 Transient power history in which the power has not remained constant prior to reactor shutdown.

NOTE:	For the majority of the selected nuclides, the 30-day power history prior to shutdown is sufficient to calculate a power correction factor.

A. Power Correction Factor $= \dfrac{\Sigma_j P_j \left(1 - e^{-\lambda_i t_j}\right) e^{-\lambda_i t_j^0}}{RP}$

P_j = average power level (Mwt) during operating period t_j

RP = rated power level of the core (Mwt)

t_j = operating period in days at power P_j where power does not vary more than ±10 percent power of rated power level from time averaged value (P_j).

λ_i = decay constant of nuclide i in inverse days.

t_j^0 = time between end of period j and time of reactor shutdown in days.

B. For the few nuclides with half-lives around one year or longer, a power correction factor which ratios effective full power days to total calendar days of cycle operation is applied.

Power Correction Factor $= \dfrac{\text{Actual Operating EFPD of equilibrium cycle}}{\text{Total expected EFPD of equilibrium cycle operation}}$

Where: Equilibrium Cycle = three (3) cycles of core operation (approximately 1050 EFPD)

7.12.3 For Cs-134, Figure 2 is used to determine the power correction factor. To use Figure 2, the average power during the entire operating period is required.

7.13 The total inventory of fission products available for release at reactor shutdown are calculated by applying the power correction factors to the equilibrium, end-of-life core inventories.

$$\text{Corrected Inventory} = \begin{array}{c}\text{Equilibrium Inventory at}\\ \text{end-of-life (Ci)}\\ \text{(Table 2)}\end{array} \times \begin{array}{c}\text{Power}\\ \text{Correction}\\ \text{Factor}\end{array}$$

7.14 Determine the percentage of inventory released, for each isotope.

$$\text{Release Percentage (\%)} = \frac{\text{Total Activity Released (Ci)}}{\text{Corrected Inventory (Ci)}} \times 100$$

7.15 The results of radionuclide analysis may now be used to determine an estimate of the extent of core damage.

7.15.1 From Figure 3 thru 15, estimate the extent of core damage by categorizing the percentage of clad damage, fuel over-temperature, and fuel melt.

7.15.2 Compare the calculated activity ratios with those listed in Table 5. Measured relative ratios greater than the gap activity ratios listed in Table 5 are indicative of more severe failures, e.g., fuel overheat.

7.16 To verify the conclusion of the radionuclide analysis, other indicators should now be used to provide verification of the estimate of core damage.

7.16.1 Containment Hydrogen Concentration:

A. Obtain the containment hydrogen concentration (%).

NOTE:	**Within the accuracy of this methodology, it is assumed that recombiners will have an insignificant effect on the hydrogen concentration when it is indicated that extensive zirconium-steam reaction could have occurred.**

B. From Figure 16, determine the percentage (%) zirconium water reaction.

C. Table 6 can be used to validate the extent of core damage estimate.

7.16.2 Core Exit thermocouple Readings:

A. Obtain as many core exit thermocouple readings as possible for evaluation of core temperature conditions.

NOTE:	If a thermocouple reads greater than 1650°F or is reading considerably different than neighboring thermocouples, thermocouple failure should be considered.

B. Compare the thermocouple readings with those in Table 6 to confirm the core damage estimate.

NOTE:	Radiation Monitors in containment may experience errors during first 4 hours after a DBA LOCA due to thermally induced errors. See Attachment 1 for more information.

7.16.3 Containment Radiation Monitor:

A. Obtain the containment dome monitor readings, R/Hr, from R-48 and/ or R-49.

B. From Figure 17, verify core damage estimate. The exposure rate in Figure 17 is based on the release of only noble gases to the containment. Halogens and other fission products were not considered to be significant contributors to the containment monitor reading.

7.16.4 Reactor Coolant Loop Radiation Monitor:

A. Obtain the reactor coolant loop radiation monitor readings, R/Hr, from R-70 and/or R-71.

B. From Figure 18, determine estimated core damage.

7.17 All indicators should confirm any core damage estimates. If radio-nuclide analysis and auxiliary indicators do not agree on core damage estimates, then recheck of indications may be performed, or certain indicators may be discounted, based on engineering judgment.

Table 1 Suggested Sampling Locations

Scenario	Principal Sampling Locations	Other Sampling Locations
Small Break LOCA		
Reactor Power > 1%*	RCS Hot Leg, Containment Atmosphere	
Reactor Power < 1%*	RCS Hot Leg**	
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,	Containment Atmosphere
Steam Generator Tube Rupture	RCS Hot Leg, Secondary System	
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	
Indication of High Radiation Level in RCS	RCS Hot Leg	

* Assume operating at that level for some appreciable time.

** If a RCS hot leg sample is unavailable and the RHR system is operating, obtain a RHR system sample. However, for a RHR system sample to be a good representation of the RCS, the primary water should be circulating through the system.

Table 2 - Fuel Pellet Inventory

Fuel Pellet Inventory*

Nuclide	Half Life	Inventory Curies**
Kr 85m	4.4	1.0×10^7
Kr 87	.76 m	1.85×10^7
Kr 88	2.8 h	2.69×10^7
Xe 131m	11.8 d	2.94×10^5
Xe 133	5.27 d	9.26×10^7
Xe 133m	2.26 d	1.35×10^7
Xe 135	9.14 h	1.77×10^7
I 131	8.05 d	4.54×10^7
I-132	2.26 h	6.65×10^7
I 133	20.3 h	9.26×10^7
I 135	6.68 h	8.33×10^7
Rb 88	17.8 m	2.69×10^7
Cs 134	2 yr	1.09×10^7
Cs 137	30 yr	4.96×10^6
Te 129	68.7 m	1.51×10^7
Te 132	77.7 h	6.65×10^7
Sr 89	52.7 d	3.70×10^7
Sr 90	28 yr	3.36×10^6
Ba 140	12.8 d	7.91×10^7
La 140	40.22 h	8.33×10^7
La 142	92.5 m	7.07×10^7
Pr 144	17.27 m	5.81×10^7

* Inventory based on ORIGEN run for equilibrium, end-of-life core.
** Westinghouse, 2-Loop, 1650 Mwt Plant

Table 3 Parent-Daughter Relationships

Parent	Parent Half-Life*	Daughter	Daughter Half-Life*	K**
Kr-88	2.8 h	Rb-88	17.8 m	1.00
I-131	8.05 d	Xe-131m	11.8 d	.008
I-133	20.3 h	Xe-133m	2.26 d	.024
I-133	20.3 h	Xe-133	5.27 d	.976
Xe-133m	2.26 d	Xe-133	5.27 d	1.00
I-135	6.68 h	Xe-135	9.14 h	.70
Xe-135m	15.6 m	Xe-135	9.14 h	1.00
I-135	6.68 h	Xe-135m	15.6 m	.30
Te-132	77.7 h	I-132	2.26 h	1.00
Sb-129	4.3 h	Te-129	68.7 m	.827
Te-129m	34.1 d	Te-129	68.7 m	.680
Sb-129	4.3 h	Te-129m	34.1 d	.173
Ba-140	12.8 d	La-140	40.22 h	1.00
Ba-142	11 m	La-142	92.5 m	1.00
Ce-144	284 d	Pr-144	17.27 m	1.00

* Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition

** Branching decay factor

Table 4 Source Inventory of Related Parent Nuclides

Nuclide	Half-Life	Inventory, Curies
Xe-135m	15.6 m	1.97×10^{7}
Sb-129	4.3 h	1.49×10^{7}
Te-129m	34.1 d	3.74×10^{6}
Ba-142	11 m	7.65×10^{7}
Ce-144	284 d	4.83×10^{7}

Table 5 Isotopic Activity Ratios of Fuel Pellet and Gap

Isotopic Activity Ratios of Fuel Pellet and Gap*

Nuclide	Fuel Pellet Activity Ratio	Gap Activity Ratio
Kr-85m	0.11	0.022
Kr-87	0.22	0.022
Kr-88	0.29	0.045
Xe-131m	0.004	0.004
Xe-133	1.0	1.0
Xe-133m	0.14	0.096
Xe-135	0.19	0.051
I-131	1.0	1.0
I-132	1.5	0.17
I-133	2.1	0.73
I-135	1.9	0.39

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Inventory}}{\text{Xe-133 Inventory}}$$

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

* The measured ratios of various nuclides found in reactor coolant during normal operation is a function of the amount of "tramp" uranium on fuel rod cladding, the number and size of "defects" (i.e., "pin holes"), and the location of the fuel rods containing the defects in the core. The ratios derived in this report are based on calculated values of relative concentrations in the fuel or in the gap. The use of these present ratios for post accident damage assessment is restricted to an attempt to differentiate between fuel overtemperature conditions and fuel cladding failure conditions. Thus the ratios derived here are not related to fuel defect levels incurred during normal operation.

Table 6 Characteristics of Categories of Fuel Damage

Core Damage Category	Percent and Type of Fission Products Released	Fission Product Ratio***	Containment Radiogas Monitor R/hr 10 hrs after shutdown**	Core Exit Thermocouples Reading (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)
No clad damage	KR-87 $< 1 \times 10^{-3}$ Xe-133 $< 1 \times 10^{-3}$ I-131 $< 1 \times 10^{-3}$ I-133 $< 1 \times 10^{-3}$	No Applicable	--	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 $10^{-3} - 0.01$ Xe-133 $10^{-3} - 0.1$ I-131 $10^{-3} - 0.3$ I-133 $10^{-3} - 0.1$	Kr-87 = 0.022 I-133 = 0.71	0 – 50	750 – 1300	Core uncovery	0 - 6
50 – 100% clad damage	Kr-87 0.01 – 0.02 Xe-133 0.1 – 0.2 I-131 0.3 – 0.5 I-133 0.1 – 0.2	Kr-87 = 0.022 I-133 = 0.71	50 to 100	1300 – 1650	Core uncovery	6 - 13
0 – 50% fuel pellet overtemperature	Xe-Kr, Cs, I 1 – 20 Sr-Ba 0 – 0.1	Kr-87 = 0.22 I-133 = 2.1	100 to 1.15E4	> 1650	Core uncovery	6 - 13
50-100% fuel pellet overtemperature	Xe-Kr, Cs, I 20 - 40 Sr-Ba 0.1– 0.2	Kr-87 = 0.22 I-133 = 2.1	1.15E4 to 2.3E4	> 1650	Core uncovery	6 - 13
0 – 50% fuel melt	Xe, Kr, Cs, I 40-70 Sr-Ba 0.2 – 0.8 Pr 0.1 – 0.8	Kr-87 = 0.22 I-133 = 2.1	2.3E4 to 2.7E4	> 1650	Core uncovery	6 - 13
50 – 100% fuel melt	Xe, Kr, Cs, I, Te > 70 Sr, Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 2.7E4	> 1650	Core uncovery	6 - 13

Characteristics of Categories of Fuel Damage*

* This table is intended to indicate whether there is fuel damage.

** These values are from Figure 17 and should be revised for times other than 10 hours.

*** $\frac{Kr-87}{Xe-133}, \frac{I-133}{I-131}$

Table 7 Expected Fuel Damage Correlation With Fuel Rod Temperature

For Information Only – See Note Below

Fuel Damage	Temperature °F*
No Damage	< 1300
Clad Damage	1300 - 2000
Ballooning of zircaloy cladding	> 1300
Burst of zircaloy cladding	1300 - 2000
Oxidation of cladding and hydrogen generation	> 1600
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Fuel Melt	> 3450
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
Melting of remaining UO_2	5100

* These temperatures are material property characteristics and are non-specific with respect to locations within the fuel and/or fuel cladding.

Figure 1 Water Density Ratio (Temperature vs. STP)

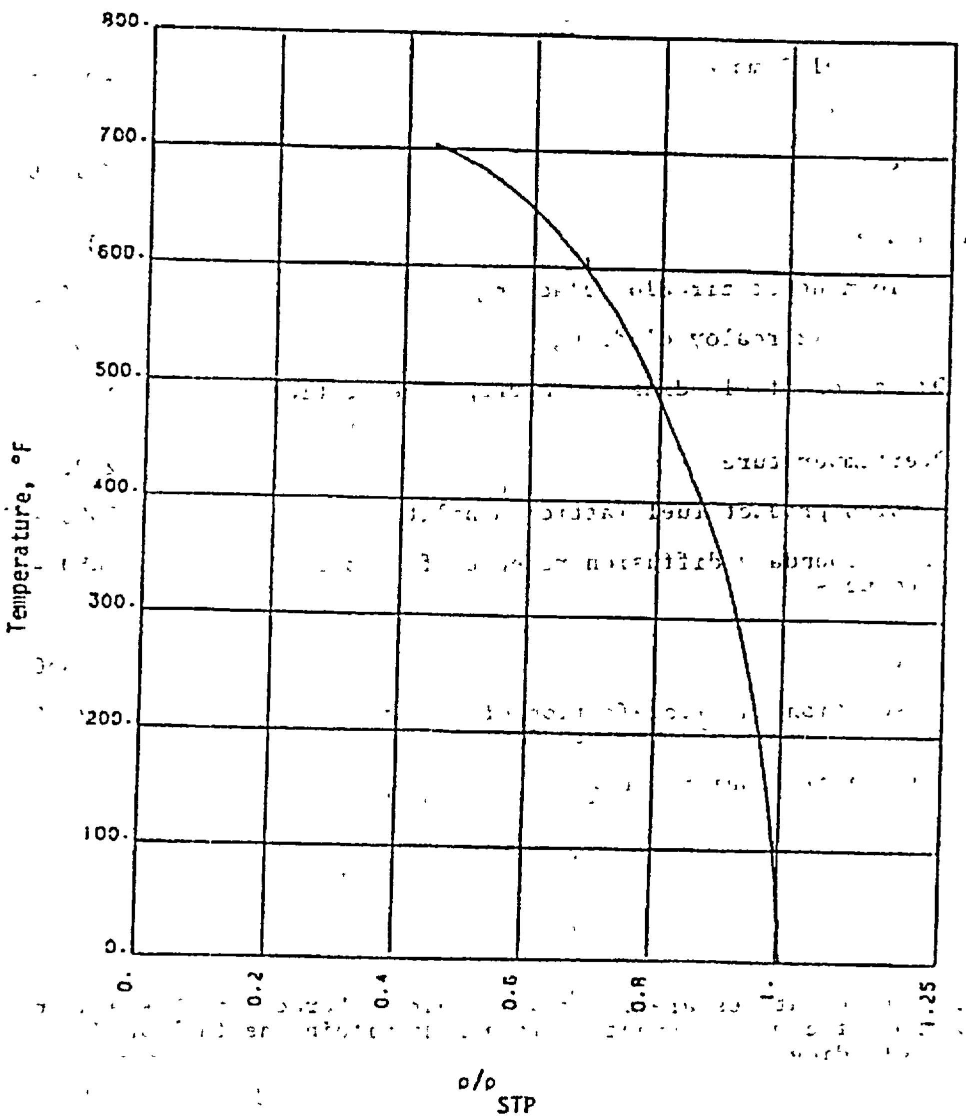


Figure 2 Power Correction Factor For CS-134 Based on Average Power During Operation

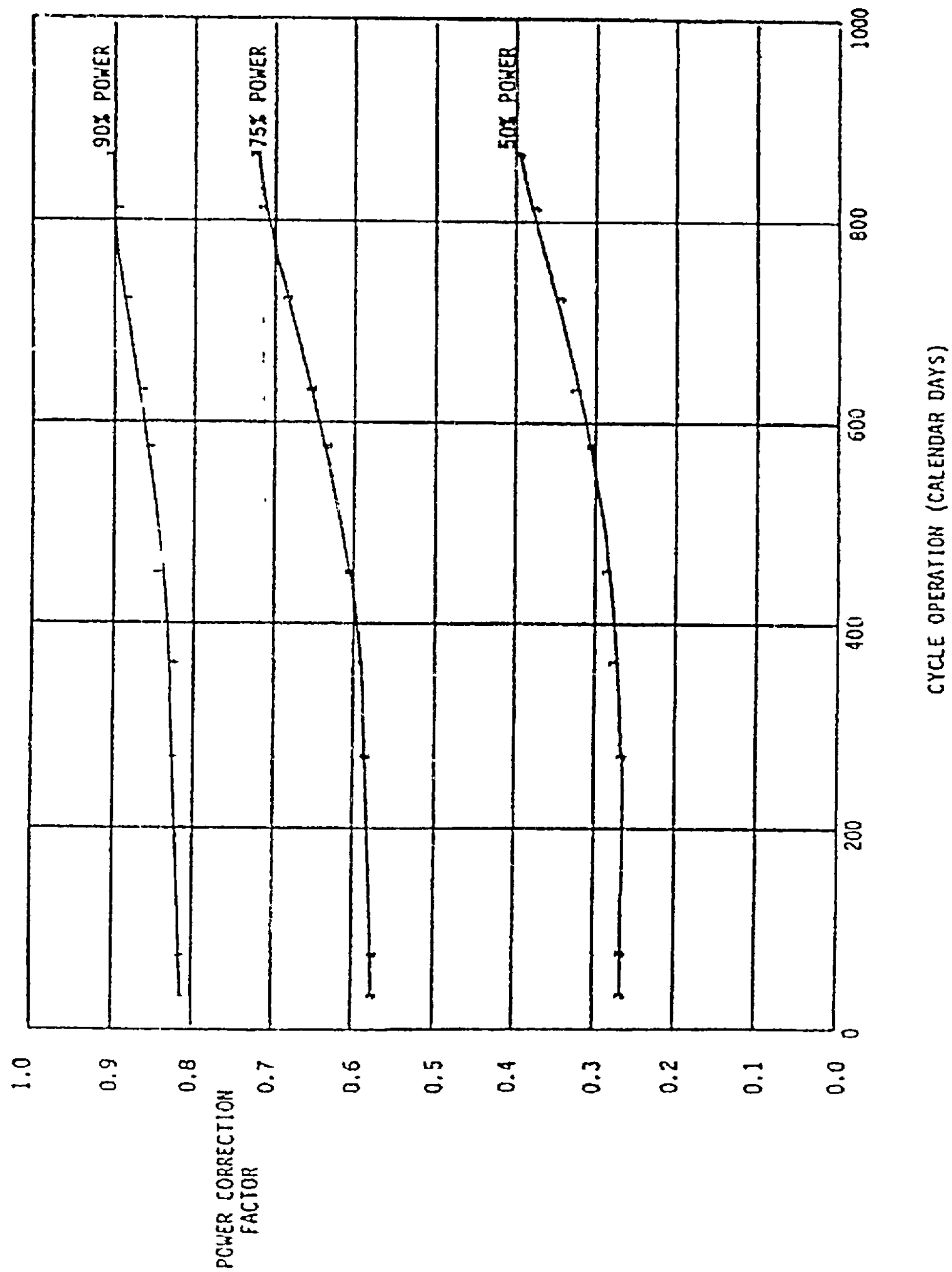


Figure 3 Relationship of % Clad Damage With % Core Inventory Released of XE-133

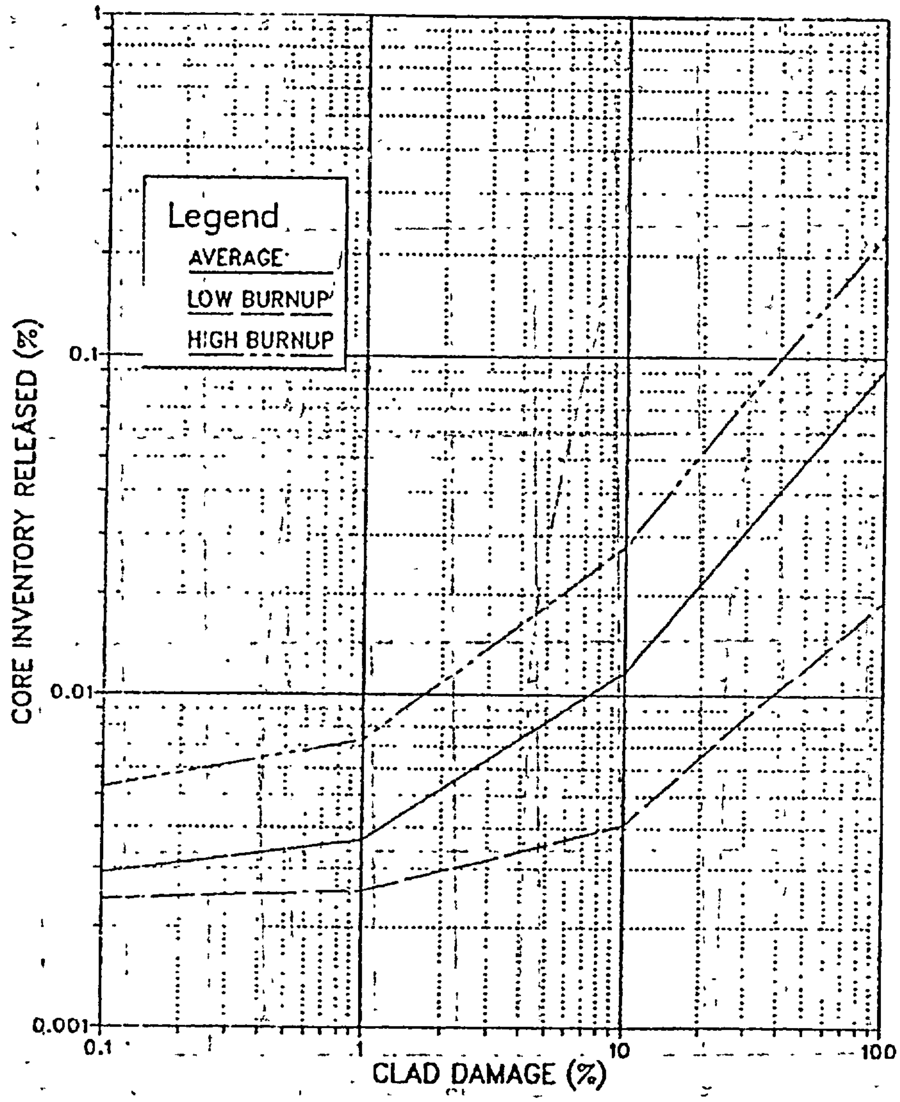


Figure 4 Relationship of % Clad Damage With % Core Inventory Released of I-131

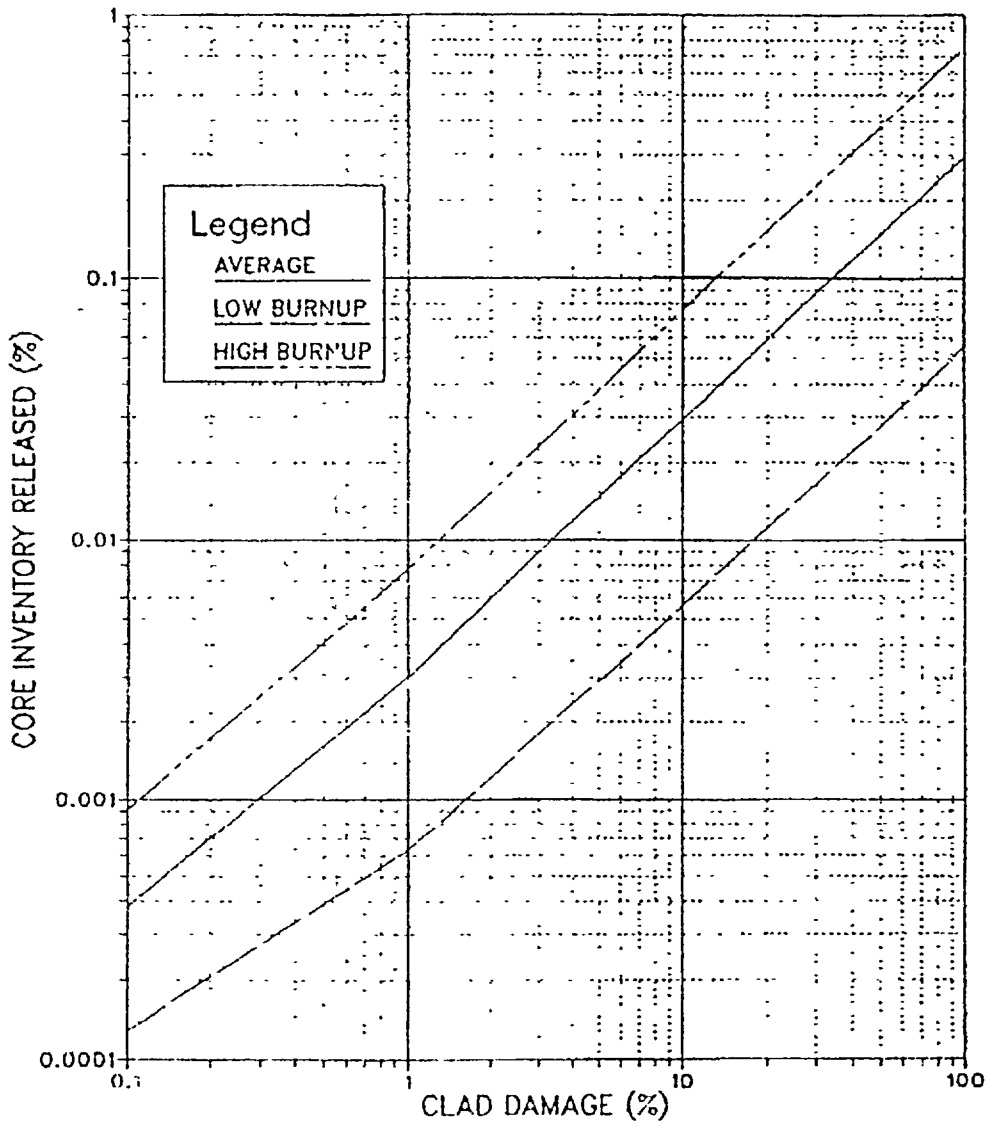


Figure 5 Relationship of % Clad Damage With % Core Inventory Released of I-131 W/Spiking

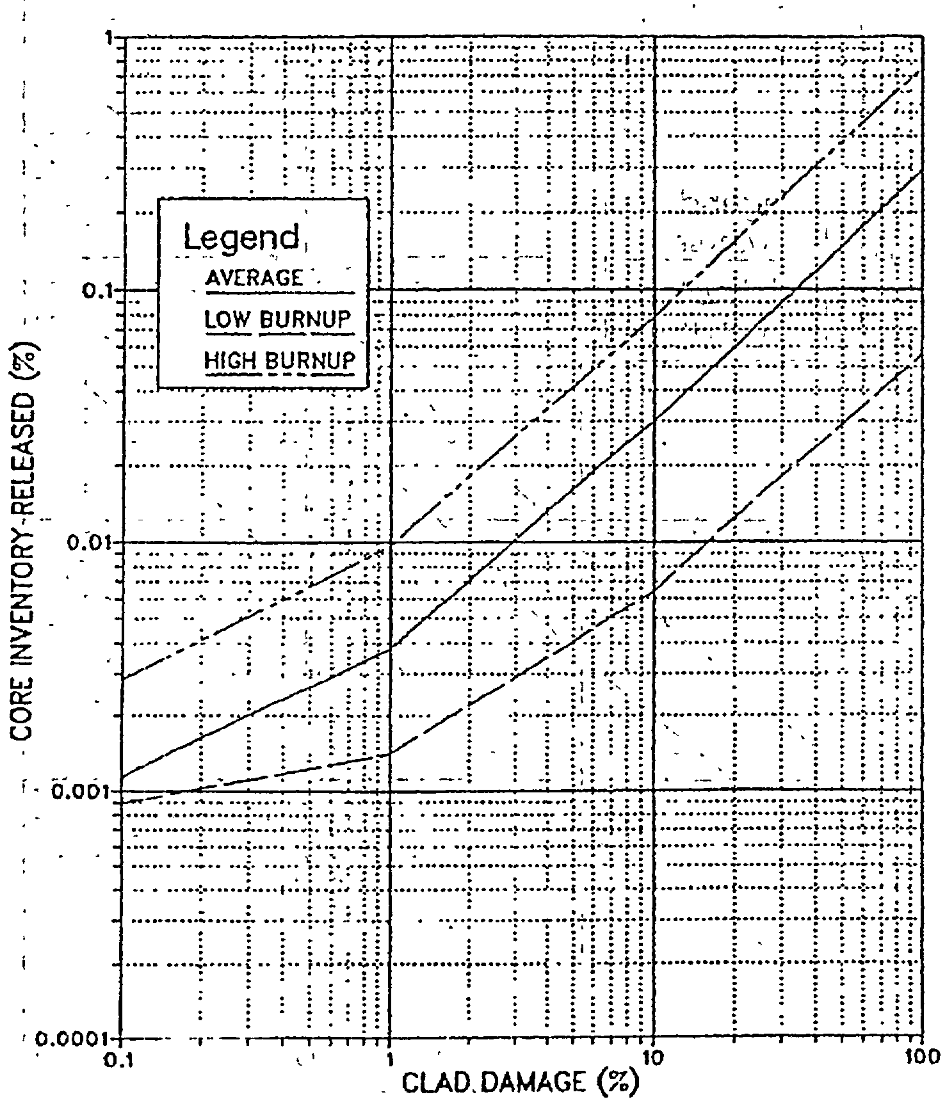


Figure 6 Relationship of % Clad Damage With % Core Inventory Released of KR-87

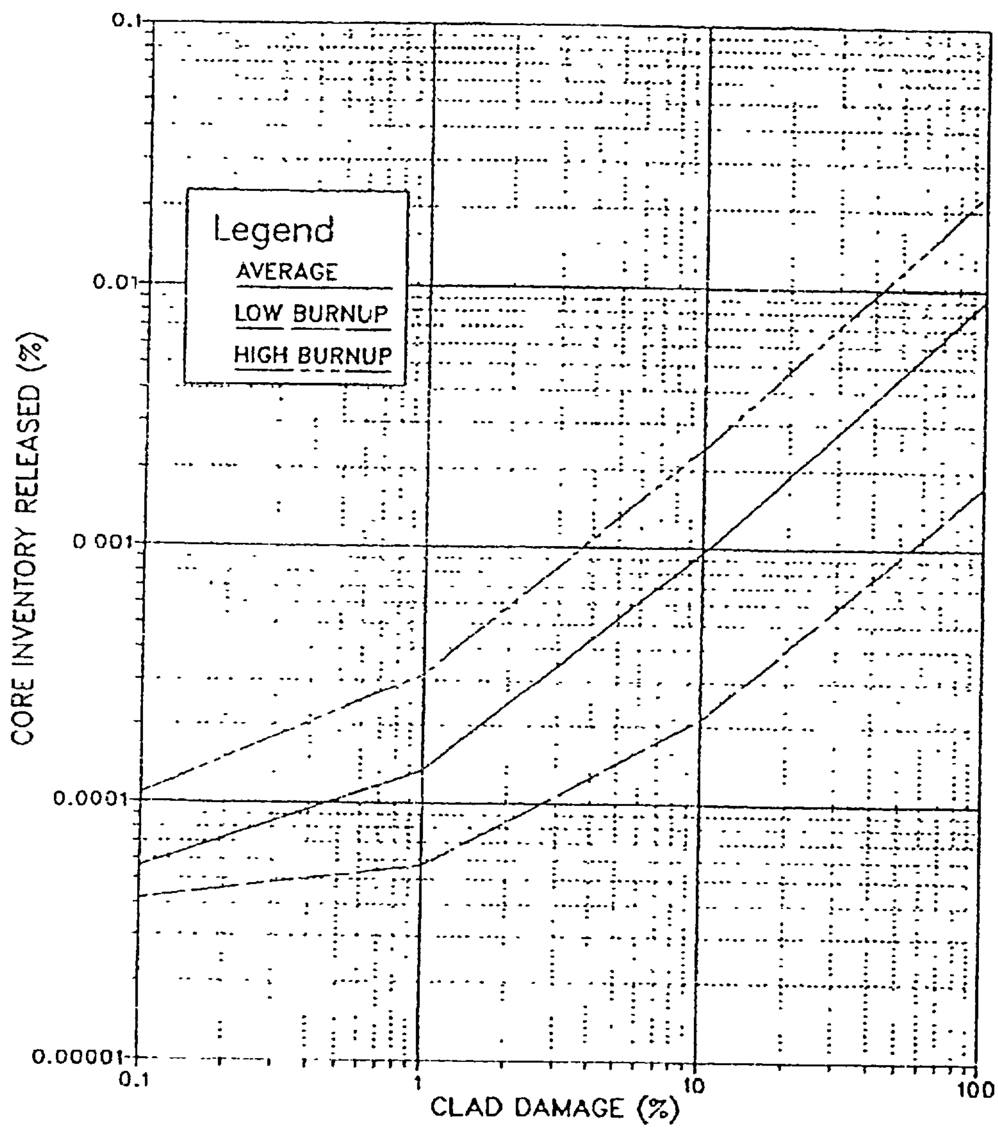


Figure 7 Relationship of % Clad Damage With % Core Inventory Released of XE-131M

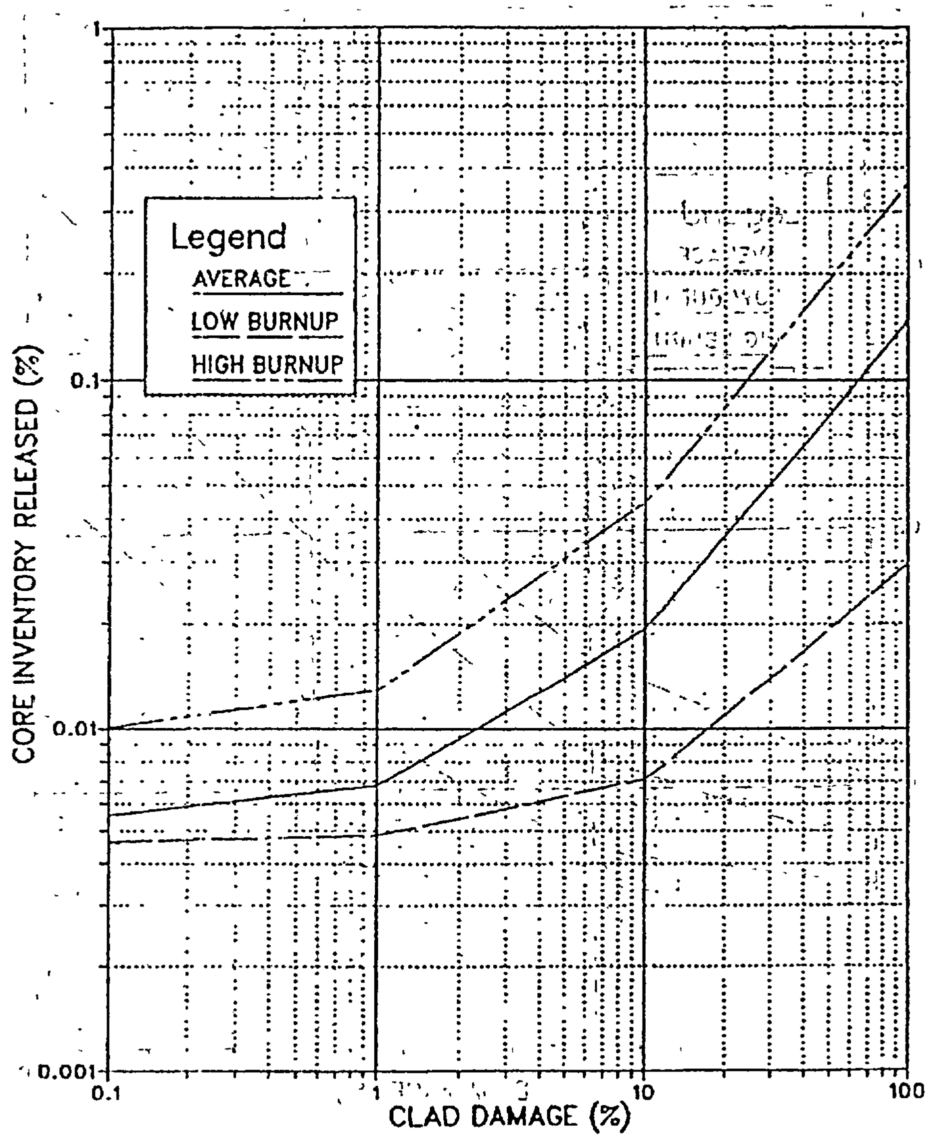


Figure 8 Relationship of % Clad Damage With % Core Inventory Released of I-132

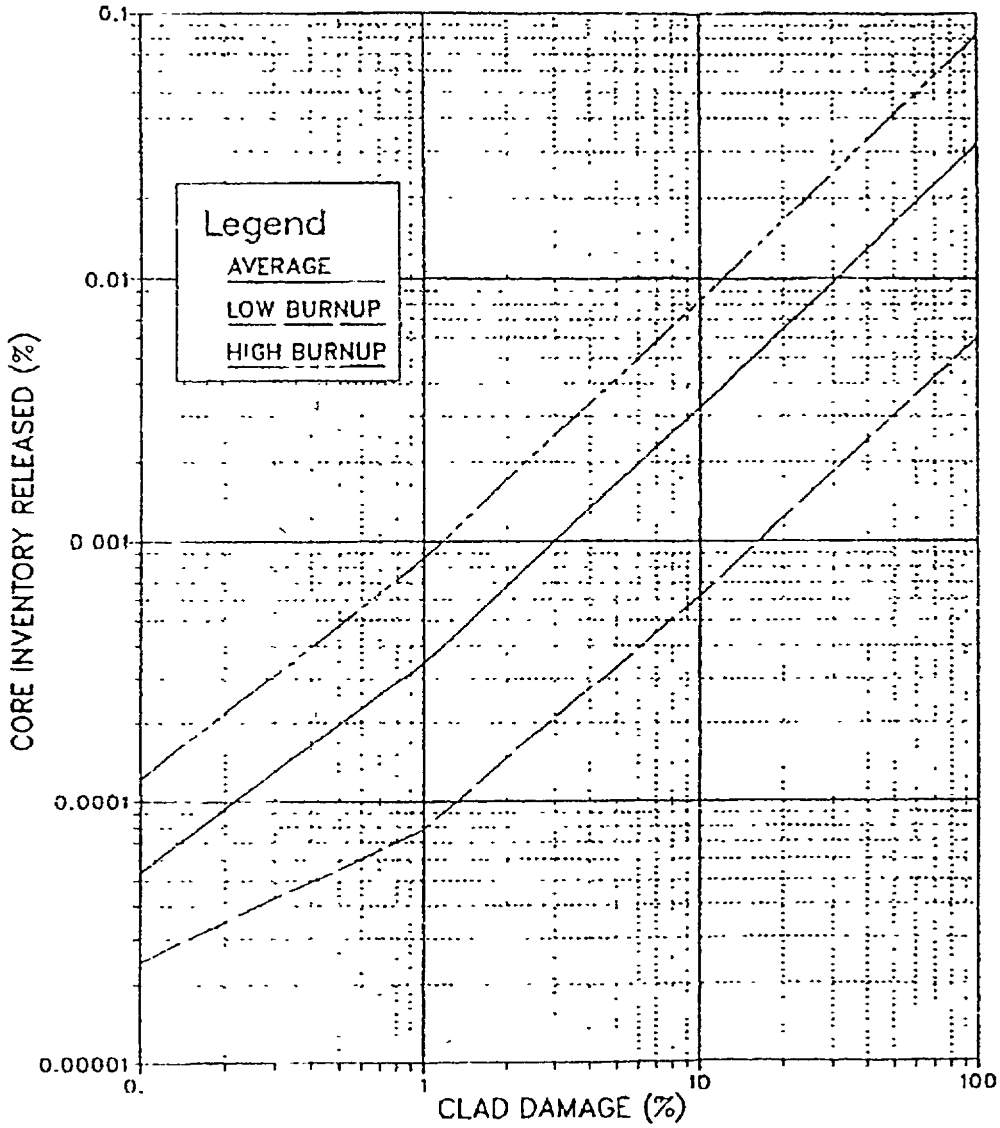


Figure 9 Relationship of % Clad Damage With % Core Inventory Released of I-133

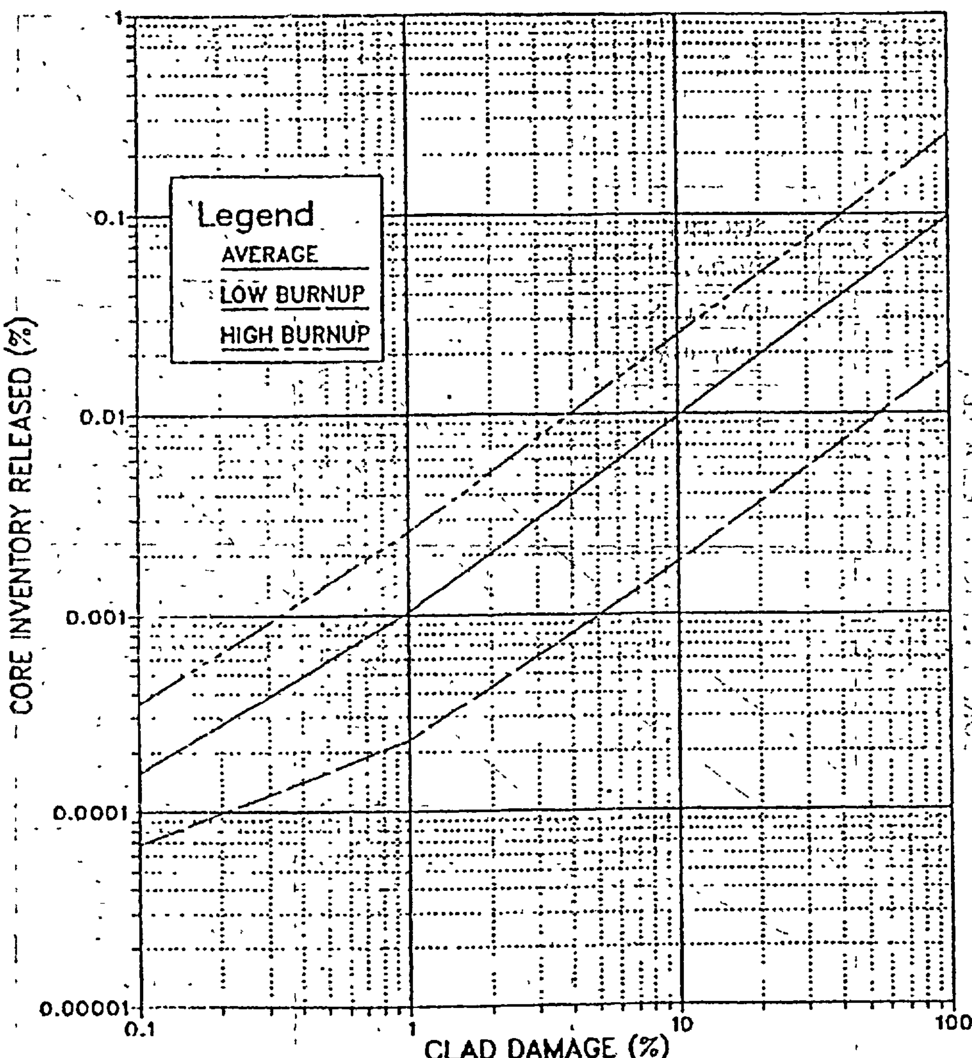


Figure 10 Relationship of % Clad Damage With % Core Inventory Released of I-135

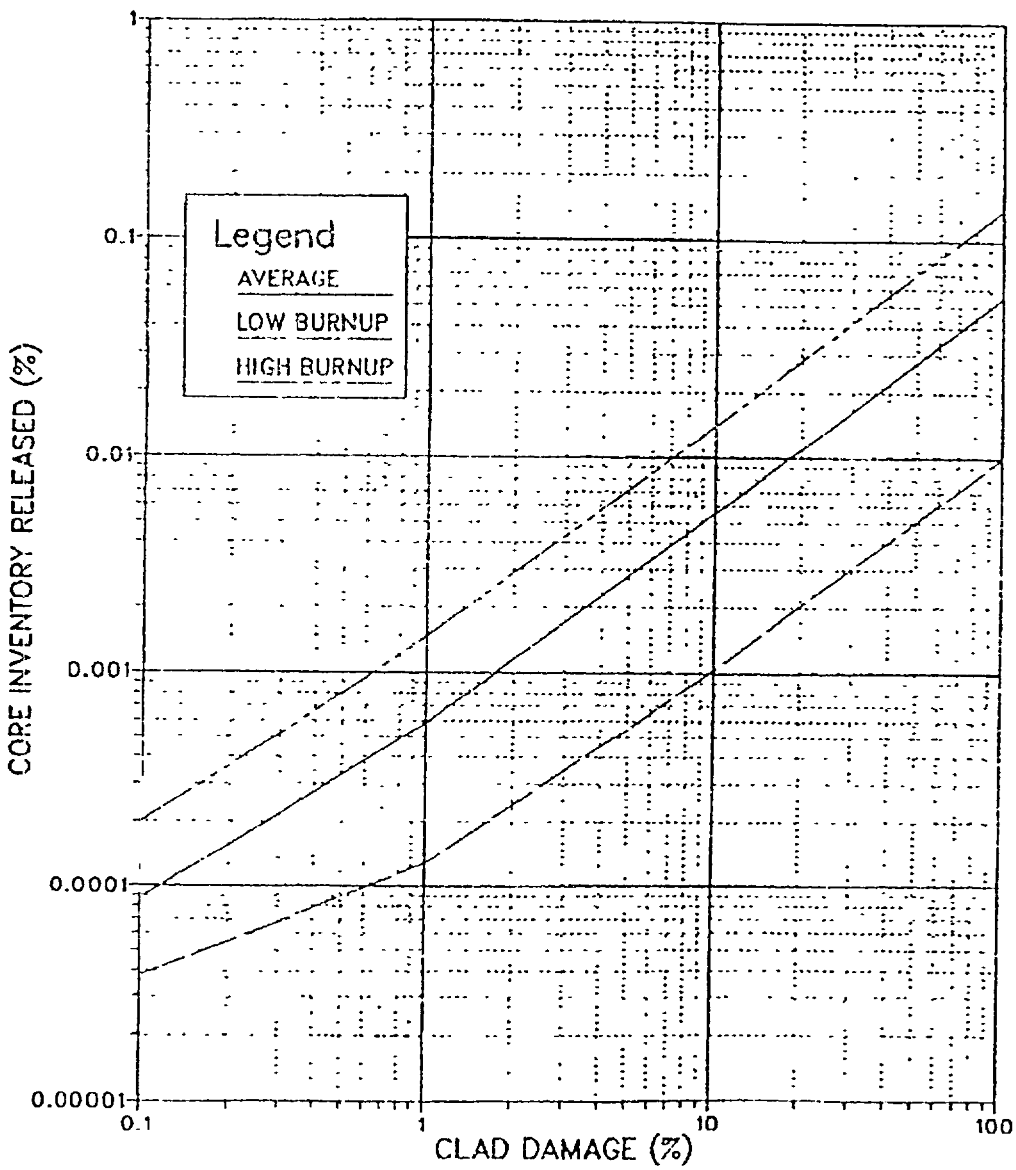


Figure 11 Relationship of % Fuel Over Temperature With % Core Inventory Released of XE, KR, I, or CS

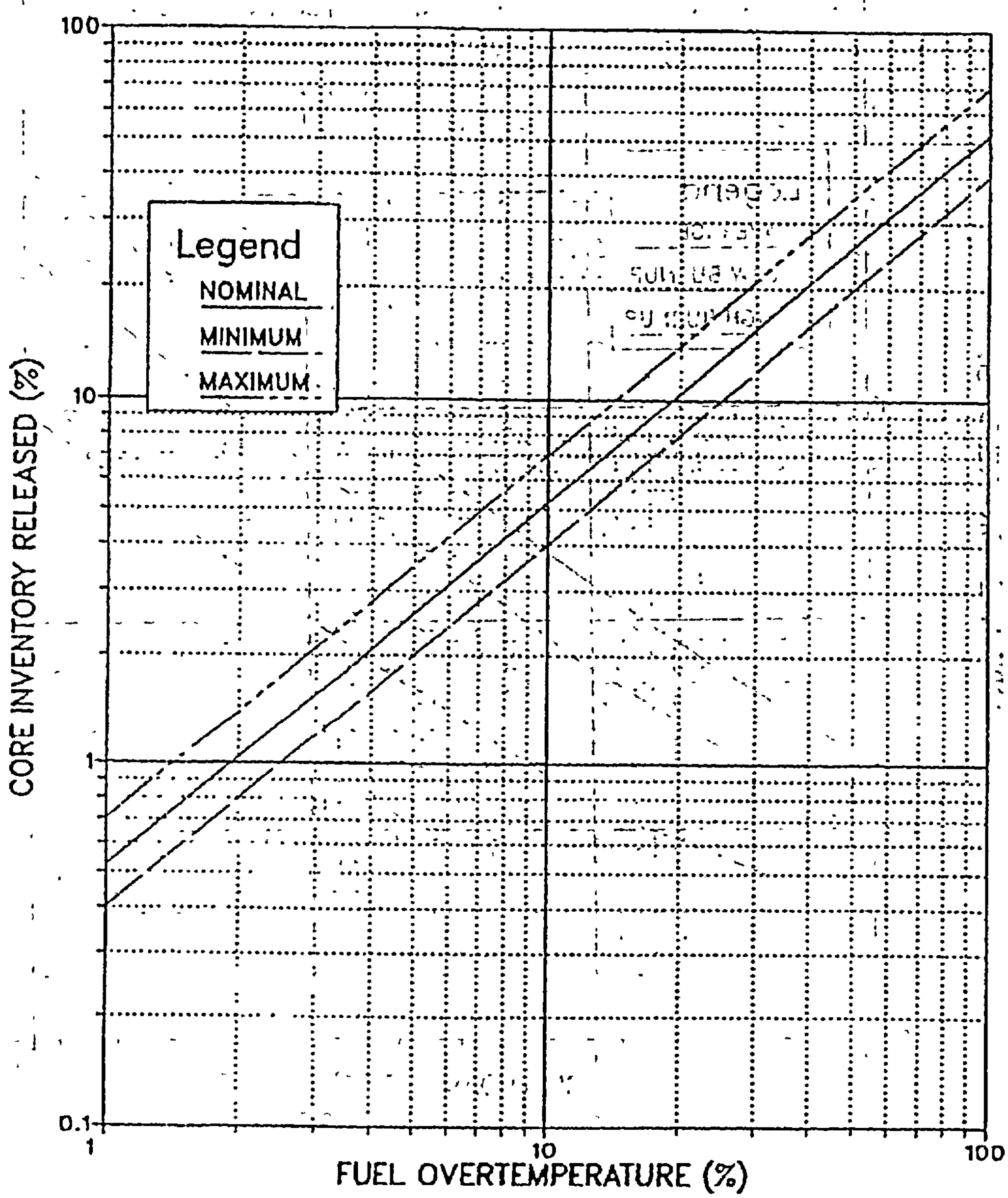


Figure 12 Relationship of % Fuel Over Temperature With % Core Inventory Released of BA or SR

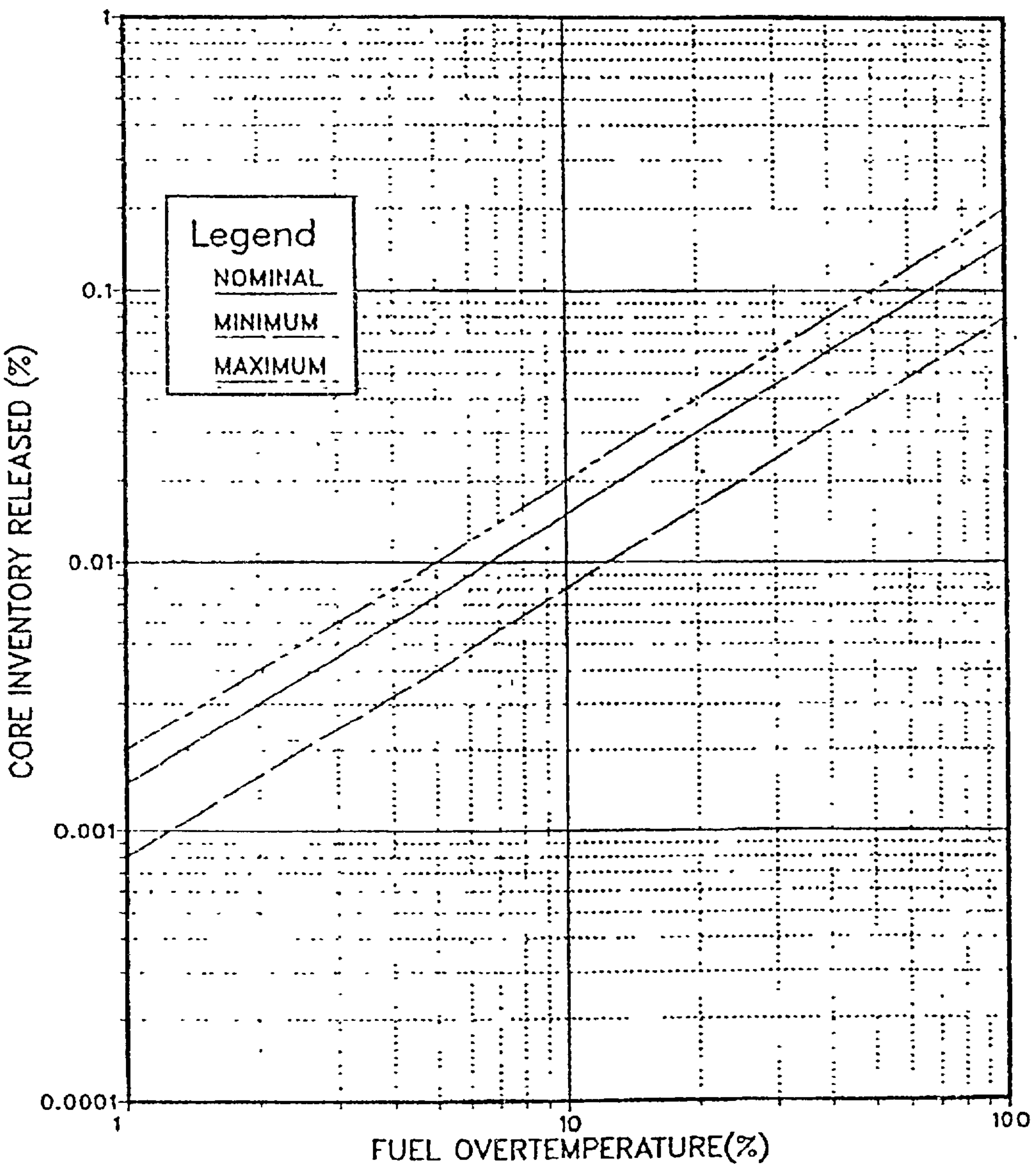


Figure 13 Relationship of % Fuel Melt With % Core Inventory Released of XE, KR, I, CS or TE

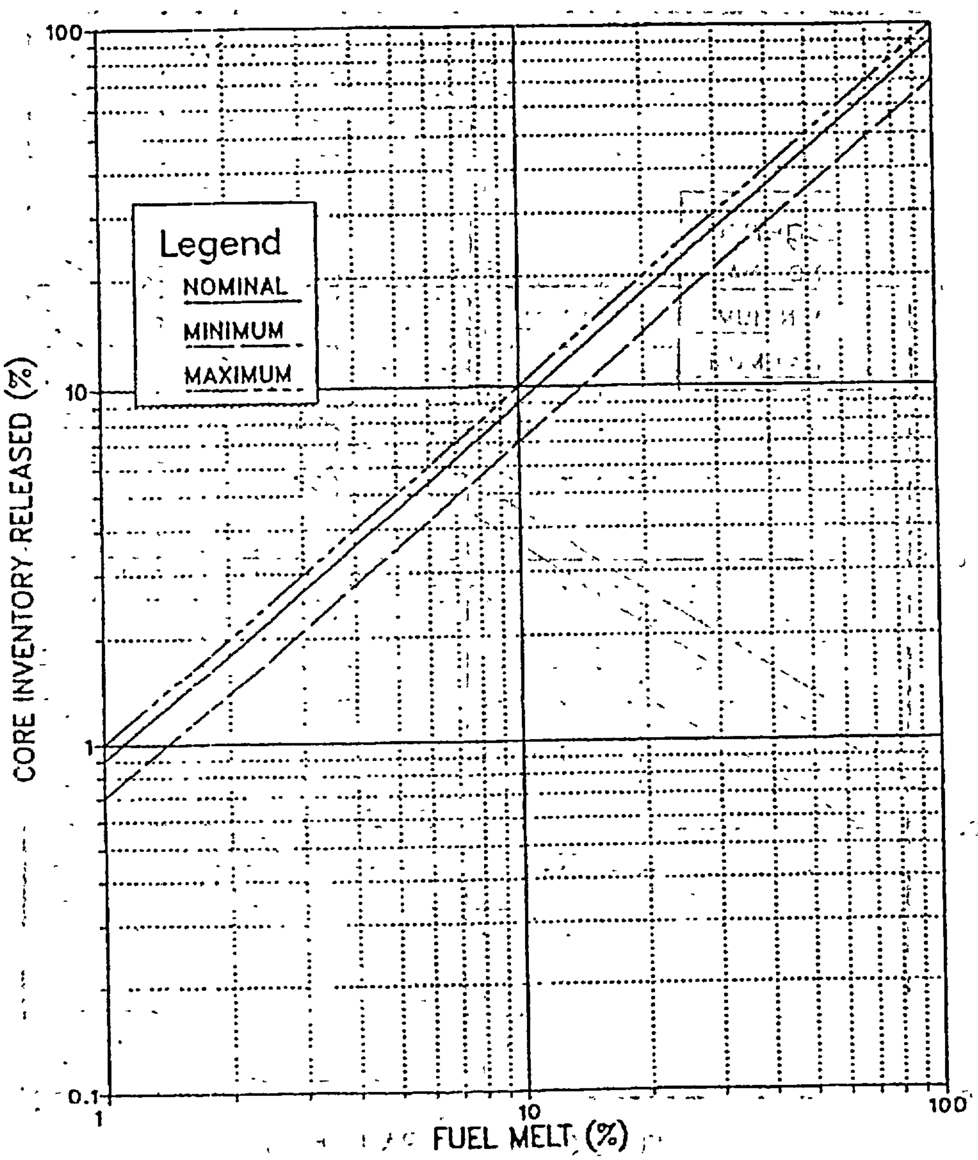


Figure 14 Relationship of % Fuel Melt With % Core Inventory Released of BA or SR

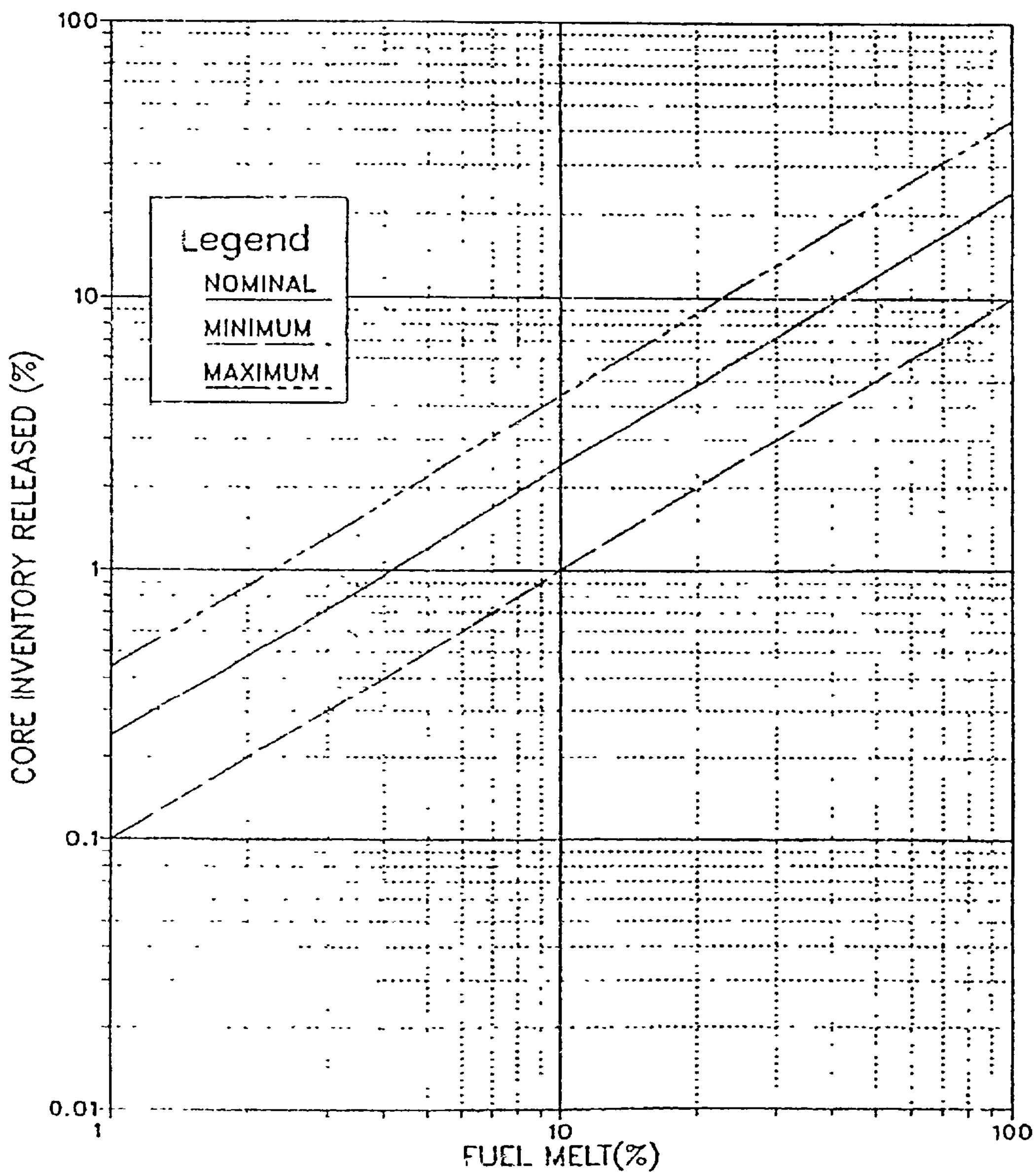


Figure 15 Relationship of % Fuel Melt With % Core Inventory Released of PR

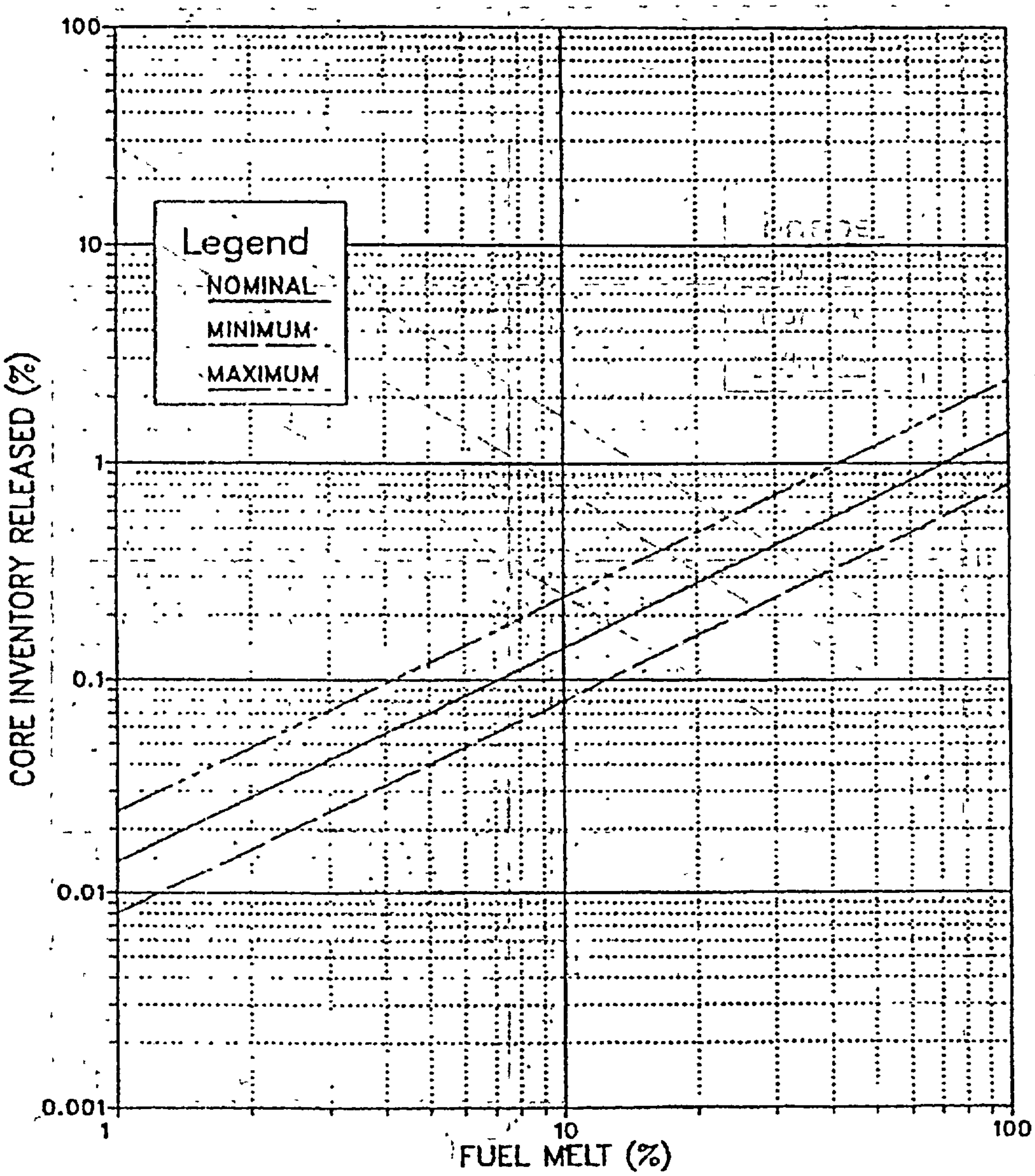


Figure 16 Containment Hydrogen Concentration Based on Zirconium Water Reaction

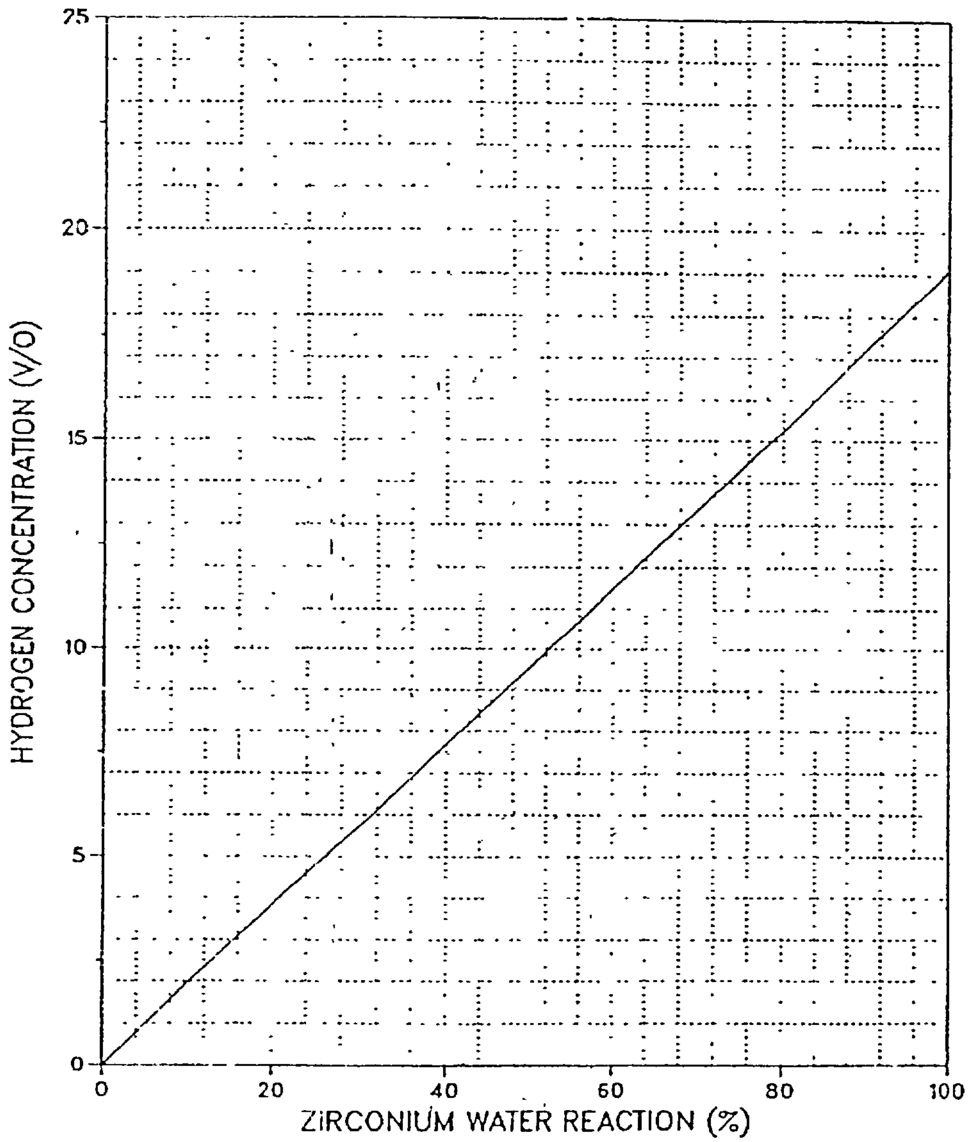


Figure 17 Percent Noble Gases in Containment

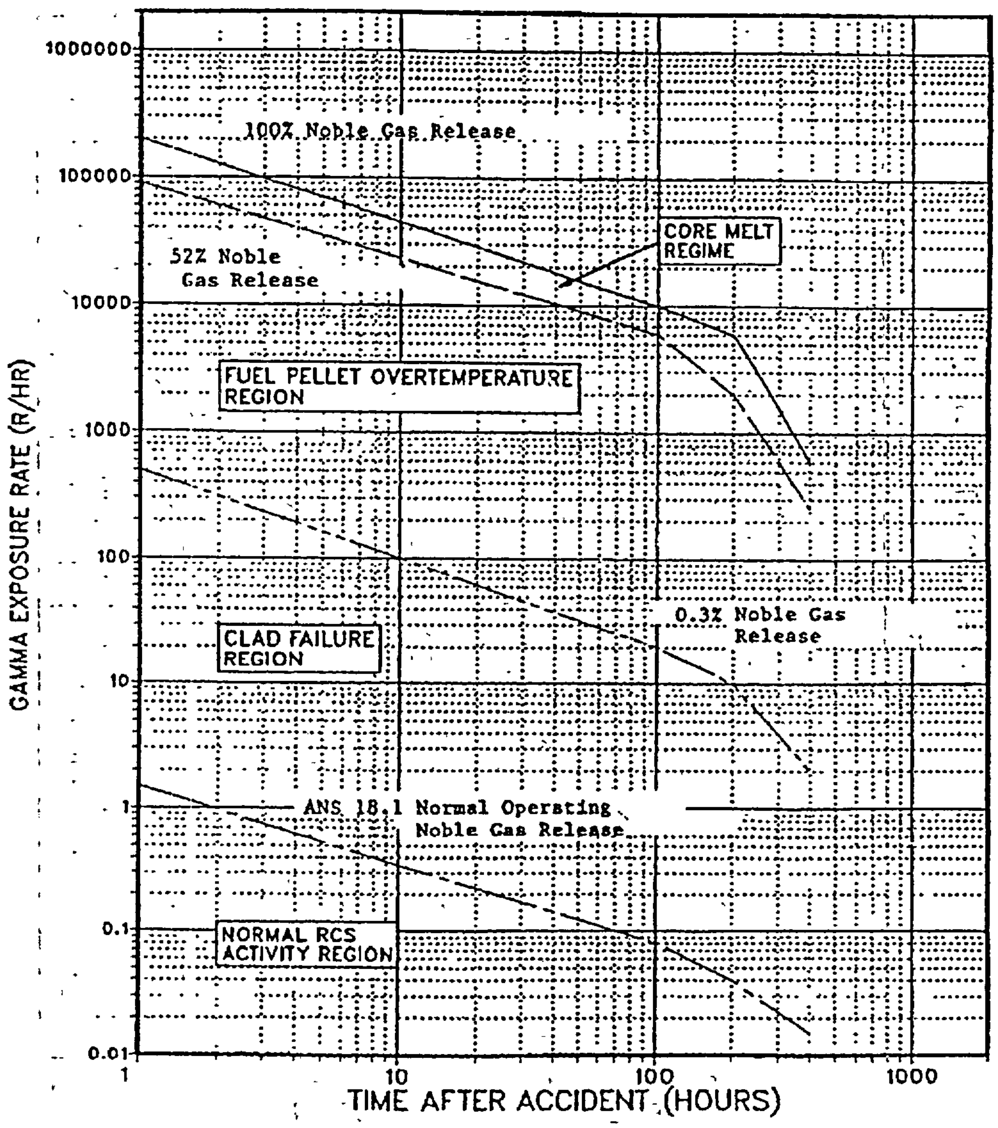
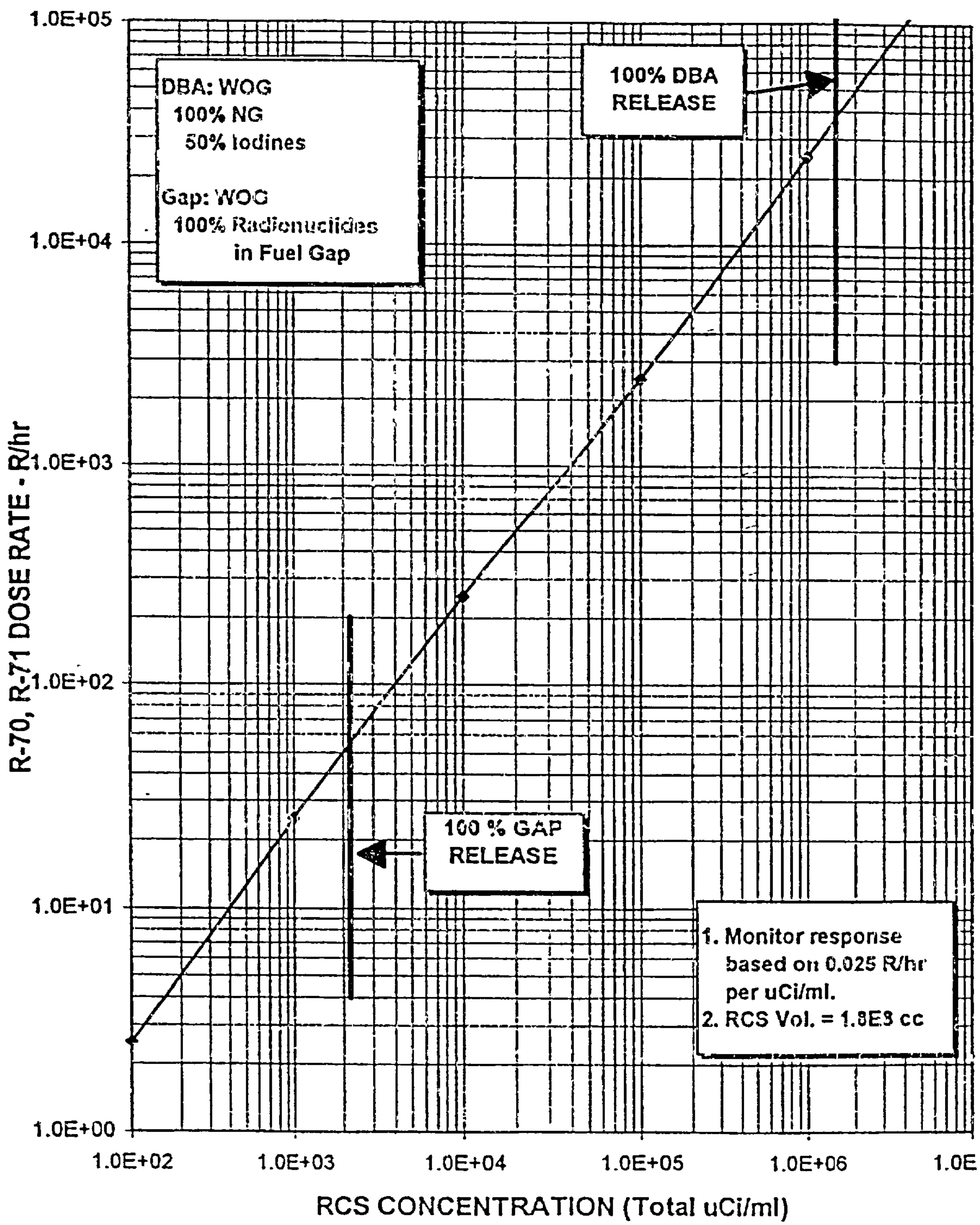


Figure 18 RCS Dose Rate vs. RCS Activity Concentrations 1 Hour After Shutdown



Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

1. R-48/R-49 & R-70/R-71 Thermally Induced Errors

R-48/49 or R-70/71 signals may experience errors during the first 4 hours after a DBA LOCA. Industry testing of high range radiation monitor (HRRM) systems has revealed that signal errors or the loss of signal are the result of thermally induced current (TIC) and/or moisture intrusion into the coaxial connectors. Based on the EPRI Plant Support Engineering study, worst case estimated errors are summarized below:

Time After Postulated DBA	Estimated Errors in Readings
~ 1 minute	> 3000 R/hr
~ 2 minutes	~ 100 R/hr
~ 8 minutes	~ 15 R/hr
~ 2 to 4 hours	~ - 9 R/hr
> 4 hours	No Effect from TIC

More background information concerning thermally induced current in high range radiation monitors is described in Section III.

Please note that errors in the range of ±10 R/hr one hour after a postulated DBA has minimal effect on our assessment of fission product release to containment when we are considering magnitudes of ≥ 100 R/hr reading to be confirmation of fission product released to containment.

2. Background on Thermally Induced Current (TIC) on High Radiation Monitors

<u>Background</u>

Excerpts from: PINGP Response to High Range Radiation Monitor Cable Study: Phase II, Report No. TR-112582 November 2000.

Transient signal errors have been observed in industry testing of the high range radiation monitor (HRRM) system. At PINGP, these are plant radiation monitors RE-48 and RE-49. The investigation into this issue revealed that signal errors or the loss of signal are the result of thermally induced currents (TICs) and/or moisture intrusion into the coaxial connectors. Information Notices, IN 97-45 and IN 97-45 Supplement 1, were issued by the NRC to alert licensees to these potential issues.

Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

EPRI Plant Support Engineering (PSE) was tasked to study the significance of this issue, which resulted in the issuance of TR-112582, "High Range Radiation Monitor Cable Study: Phase II". This study was focused on the thermally induced current phenomena since moisture intrusion issues are well understood within the industry and have more generic applications. Phase I of the EPRI study confirmed that TIC existed and was significant under thermal transients. Phase II of the study identified the sources of the TIC and developed a mathematical model for cable responses to thermal transients.

Study Results and Analysis

Using the developed profiles in the Phase II study, the actual amplitude, duration, and sign of HRRM signal errors to be expected could be determined. From this data, PINGP was able to ascertain the approximate expected signal error for the HRRMs during the postulated DBA. The expected radiation readings due to the TIC phenomena, based upon the worst case cable length, are as follows:

50 seconds	3372 R/hr
100 seconds	38 R/hr
500 seconds	13.2 R/hr
8000-15000 seconds	-8.8 R/hr
>15000 seconds	no effect from TIC

From 8000 seconds until 15000 seconds, the HRRMs could provide a "fail" alarm, based on the required "keep alive" signal current of 1E-11 amps since the current may drop to – 8.8E-11 amps. It should be noted that any significant radiation releases would drive the current back up and the HRRMs would function properly, except for the-8.8 R/hr error that may be present. After 15000 seconds (4.1 hours), there would be no TIC effects on the HRRMs.

The installed HRRM cable at PINGP is the worst case tested cable, Rockbestos RSS-6-104, and is in greater lengths than were tested, 130 feet tested vs. 290 feet installed (worst case). Other variables that could significantly effect the TIC phenomena are, 1) the tested cable was not installed within conduit whereas the PINGP cable runs are installed entirely within conduit, 2) the temperature differential of the test samples, 100 degc, is greater than the temperature differential from the PINGP accident profile, 68 degc, 3) the EPRI mathematical model was developed based on hypothetical LOCA profiles, which are more severe than the PINGP LOCA profile, and 4) consideration regarding whether the test methodology of immersion of the test samples into a ice bath and then to a boiling water plunge is representative of what the cable would experience during an actual transient.

Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

PINGP Response to HRRM Signal Error

During the initial phase of any postulated accident, it would not expected to see indication of actual fuel damage for the first 10-15 minutes. If indeed the alarms would come in for RE-48 and RE 49, Operations would be occupied with accident mitigation and monitoring tasks during this time period and this alarm, even though acknowledged, would be ignored during this period. Other parameters would be available for alarm validation, i.e., core exit temperatures, RVLIS, radiation monitors located in the Auxiliary Building, etc. Due to the nature of the TIC phenomenon, the radiation level readings, even if the alarms have come in, would be decreasing. Again, this is validation of an erroneous signal and not actual core damage.

For emergency plan response and possible SAMG considerations, the TIC phenomenon would no longer be affecting the radiation monitors and/or due to the earlier alarms and decreasing readings that were noted, it would be confirmed that no fuel damage had occurred and these were indeed erroneous readings. A general site emergency alarm would be activated at 1000 R/hr, but as cited previously, this is well after the expected error signal has been significantly reduced. Other variables would be available to verify possible fuel damage and any possible actions required within the emergency plan procedures would not occur until after the TIC phenomena has either passed or has been verified to be erroneous.

F3	PRAIRIE ISLAND RADIATION PROTECTION GROUP RESPONSE TO A MONTICELLO EMERGENCY	NUMBER: F3-22
		REV: 17

REFERENCE USE
• *Procedure segments may be performed from memory.*
• *Use the procedure to verify segments are complete.*
• *Mark off steps within segment before continuing.*
• *Procedure should be available at the work location.*

O.C. REVIEW DATE:	OWNER:	EFFECTIVE DATE
030703 SC	M. Werner	4-2-03

1.0 PURPOSE

When an Alert, Site Area, or General Emergency occurs at Monticello, the Prairie Island Radiation Protection Group **SHALL** be requested to respond with personnel and equipment to support the Monticello Radiation Protection Group. This Prairie Island support allows the Monticello personnel to concentrate their efforts in performing onsite sampling and monitoring and relieves them of the offsite monitoring requirements.

The purpose of this instruction is to describe the personnel, equipment, and procedures required to respond to a Monticello emergency.

2.0 APPLICABILITY

This instruction **SHALL** apply to Shift Supervisors (SS), Shift Emergency Communicators (SEC), Manager of Radiation Protection, and all Radiation Protection Group and Offsite Survey Team members.

3.0 PRECAUTIONS

All meters should be source checked prior to departing the Prairie Island plant site.

4.0 RESPONSIBILITIES

4.1 The Shift Supervisor has the responsibility to ensure the Prairie Island Radiation Protection Group is notified of a Monticello request for assistance per this procedure.

4.2 The Shift Emergency Communicator is responsible to assist the Shift Supervisor as directed by the Shift Supervisor.

4.3 The Manager of Radiation Protection or designee is responsible to ensure Prairie Island provides adequate radiation protection staff in support of a Monticello emergency per this procedure.

4.4 Radiation Protection Group members are responsible to respond to a Monticello emergency per this procedure.

4.5 Offsite Survey Team members are responsible to respond to a Monticello emergency per this procedure.

5.0 DISCUSSION

When an Alert, Site Area, or General Emergency occurs at Monticello, the Prairie Island Radiation Protection Group will be requested to respond with personnel and equipment to the Monticello EOF and if appropriate, the Monticello Area Public Reception Center.

At least two (2) Offsite Survey Team members are required to respond to the Monticello EOF to staff two field survey teams and one (1) Radiation Protection Specialist (RPS) to provide assistance to the Radiation Protection Support Supervisor (RPSS) in the EOF. If available, two (2) additional Offsite Survey Team members should respond to EOF to facilitate staffing two (2) Offsite Survey Team members per field survey team.

In addition, another team, consisting of two (2) RPSs and one REC (if available), should respond to the Monticello Area Public Reception Center if it is activated.

See Appendix A for a listing of required Monticello response equipment.

6.0 PREREQUISITES

The Shift Supervisor has received a request for radiation protection assistance in response to a Monticello emergency.

7.0 PROCEDURE

7.1 Initial Notifications Performed by the Shift Manager (SM) or Shift Supervisor (SS)

NOTE:	The SM or SS may request the SEC to assist in notifying the Radiation Protection Group.

7.1.1 Using the RADIATION PROTECTION CALL LIST in the NUCLEAR EMERGENCY PREPAREDNESS TELEPHONE DIRECTORY under PI ERO, **notify** the Manager of Radiation Protection or designee and brief him on what is known of the following:

A. Emergency Classification.

B. Is there a radioactive release.

C. Has there been offsite protective actions issued.

D. **Call** Security Shift Lieutenant and notify him/her of Radiation Protection Personnel emergency call out and need for FFD screening.

7.1.2 WHEN the Offsite Survey Team notifies the Control Room of the expected departure time, THEN the SM or SS should **notify** the Monticello SEC (763) 295-3739 x1216 or x1072 and **inform** them of the PI Radiation Protection Group's departure time.

7.2 Response Team Activation Performed by the Manager of Radiation Protection or Designee

7.2.1 Using RADIATION PROTECTION CALL LIST in the NUCLEAR EMERGENCY PREPAREDNESS TELEPHONE DIRECTORY under PI ERO, or other list of qualified personnel, **mobilize** two (2) response teams to the Monticello EOF made up of at least one (1) Offsite Field Team member per field survey team (two (2) if available) and one (1) additional RPS to assist in the EOF.

7.2.2 During off-normal working hours, **screen** all personnel for Fitness for Duty (FFD).

7.2.3 IF offsite protective action recommendations have been issued OR the classification is a Site Area or General Emergency, THEN **direct** the dispatch of another response team to the Monticello Area Public Reception Center. The team should be made up of two (2) RPSs and one (1) REC (if available). See Appendix B for directions to the Monticello Area Public Reception Center.

NOTE:	**Long term coverage for Offsite Survey Teams should be provided by contract Health Physics personnel. Arrangements for contract services will be handled by the Monticello EOF Organization.**

7.2.4 **Augment** additional personnel as soon as possible to supply short term 24 hours per day coverage.

7.3 Monticello EOF Response Team Activation

7.3.1 Personnel **pick up** vehicles and equipment as listed on Appendix A.

7.3.2 **Perform** meter check on all meters prior to departing the plant.

7.3.3 **Notify** the Shift Supervisor of the expected departure time prior to departing the plant.

7.3.4 **Notify** the SS of any members that may need FFD screening and **have** those personnel report to Security Building (Guardhouse).

NOTE:	The Minnesota road map and Monticello survey map are located in the field team survey kits.

7.3.5 **Review** the Minnesota road map and Monticello survey map for reference to routes to the Monticello EOF.

7.3.6 **Depart** for the Monticello EOF.

7.3.7 WHEN approaching the boundary line of the Monticello 10 mile EPZ, THEN **attempt** to contact the Monticello EOF using the portable radio. **Identify** yourself as the Prairie Island Survey teams.

7.3.8 IF determined from the initial radio contact with the Monticello EOF that the plume may be encountered while enroute, THEN **conduct** a search for the plume, in accordance with F3-15 and **proceed** directly to the EOF.

7.3.9 **Upon** arrival at the Monticello EOF, **contact** the Radiation Protection Support Supervisor or EOF Coordinator at the Nearsite EOF for obtaining field team drivers and receiving further instructions.

7.3.10 **Perform** the requested offsite surveys as required by the Emergency Manager.

NOTE:	All field sample analysis should be performed in the EOF Count Room by Monticello Radiation Protection Specialists.

7.3.11 **Continue** with all assigned duties until relieved by the Monticello Emergency Manager.

Appendix A Monticello Emergency Support

EQUIPMENT LIST

1. Personnel TLD's and self-reading dosimeters (rezero)
2. Two (2) vehicles to be used for offsite monitoring purposes
3. Two (2) emergency survey team kits
4. Two (2) installed radios or portable radios with mag-mount antennas.
5. Two (2) dose rate meters
6. Two (2) count rate meters
7. Two (2) DC Air Samplers

Appendix B Monticello Area Public Reception Center Support

Directions to Osseo Jr. High School

Option 1:

Proceed from plant to northwest side of Minneapolis via I-94 or I-694 to Highway 169 (Exit 31).

Exit onto Highway 169 (North).

Continue North on Highway 169 to 93rd Ave. N. (7th St. or Co. Rd. 30).

Turn left onto 93rd Ave. N. (7th St. or Co. Rd. 30).

Continue West on 93rd Ave. No. (7th St. or Co. Rd. 30) to Osseo Jr. High School (10223 93rd Ave. N.).

Option 2:

Proceed from plant to northwest side of Minneapolis via I-494 to I-94.

Continue West on I-94 to Co. Rd. 109 (Exit 215).

Exit onto Co. Rd. 109 (East).

Continue East on Co. Rd. 109 to Highway 169.

Exit Co. Rd. 109 onto Highway 169 (North).

Continue North on Highway 169 to 93rd Ave. N. (7th St. or Co. Rd. 30).

Turn left onto 93rd Ave. N. (7th St. or Co. Rd. 30).

Continue West on 93rd Ave. No. (7th St. or Co. Rd. 30) to Osseo Jr. High School (10223 93rd Ave. N.).

F3	EMERGENCY HOTCELL PROCEDURE	NUMBER: F3-23.1
		REV: 13

REFERENCE USE

- *Procedure segments may be performed from memory.*
- *Use the procedure to verify segments are complete.*
- *Mark off steps within segment before continuing.*
- *Procedure should be available at the work location.*

O.C. REVIEW DATE:	OWNER:	EFFECTIVE DATE
030703 SC	M. Werner	4-2-03

1.0 PURPOSE

The purpose of this procedure is to provide instructions to the Radiation Protection Group on the use of the Hotcell, to include Hotcell setup, various chemical analysis evolutions and radioactive sample disposal techniques.

2.0 APPLICABILITY

This Instruction is applicable to Chemistry Radiation Protection Specialists.

3.0 PRECAUTIONS

3.1 Monitor the general area of the Hotcell for direct radiation to ensure the habitability of the Hotcell.

3.2 The reactor coolant samples taken in an accident condition have the potential to be highly radioactive. This may give rise to dose rates far in excess of what would normally be encountered. All work involving these samples is to be performed in the Hotcell with the fume hood in operation and with remote handling tools, to minimize radiation exposure, until one of the following is determined:

3.2.1 The sample is determined not to have dose rates in excess of normal values.

3.2.2 The sample has been diluted to the point where the diluted portion does not have dose rates in excess of normal values.

3.3 If a sample is determined to be of normal dose rate values, or is diluted to the point NOT to exceed normal dose rate values, the following should apply:

3.3.1 The instructions specified in this procedure may be completed in an area other than the Hotcell Hood.

3.3.2 Monitor the alternate area for direct radiation to ensure habitability.

3.3.3 Analyze the sample in accordance with the appropriate RPIP, as a normal chemistry sample for the analyte of interest

3.3.4 The instructions for **Post Accident Sample Waste Storage and Disposal** apply.

4.0 RESPONSIBILITIES

The Chemistry Radiation Protection Specialists are responsible to implement this procedure.

5.0 DISCUSSION

The Hot Chem Lab in the Auxiliary Building may not be available due to abnormal radiological conditions. Use of the Hotcell or Alternate Area would be necessary.

6.0 PREREQUISITES

6.1 Hotcell Set-up Procedure or Alternate Area

NOTE:	The following procedure should be completed prior to introducing a hot sample into the Hotcell Area.

6.1.1 **Ensure** that all instrumentation is turned on, warmed up and calibrated.

6.1.2 **Fill** a 1 L volumetric to the mark with demineralized water.

6.1.3 **Fill** a 100 ml volumetric to the mark with demineralized water.

6.1.4 **Remove** 1 ml of demineralized water from each volumetric using a 1 ml pipet.

6.1.5 **Add** a stir bar to each volumetric.

6.1.6 **Turn ON** the two stir plates in the fume hood

NOTE:	<u>IF</u> containment spray has been activated, consider buffering pH meter with 7 and 10 buffer.

6.1.7 **Buffer** the pH electrode.

6.1.8 **Place** a 250 ml beaker of water near the pH probe.

7.0 Procedure

7.1 Sample Preparation

NOTE:	The RPS Sample Team members SHOULD ensure all samples are properly labeled with sample identification, sample size/volume, flowrates, pressures, and sample times, as appropriate to facilitate accurate analysis. As samples are diluted, split, or reduced; the appropriate information needs to be included on new labels attached to the newly created samples. Sample dose rate information should be included on all sample labels, to help ensure personnel awareness of radiological consideration. For ALARA reasons, the sample containers should be prelabeled whenever possible.

7.1.1 **Label** all samples.

7.1.2 **Verify** postings and boundaries for expected radiation and contamination levels.

7.1.3 **Don** a finger ring on each hand.

7.1.4 **Ensure** TLD and dosimeters are worn.

7.1.5 **Place** the 60 ml bottle shielded carrier in the fume hood near the pH probe.

CAUTION:	AVOID PLACING HANDS OVER TOP OF OPEN SHIELDED CARRIER.

7.1.6 IF radiation levels require, THEN **use** the remote handling tool.

7.1.7 **Remove** the lid from the 60 ml bottle shielded carrier.

7.1.8 **Remove** the stopper from the bottle.

7.1.9 **Pipet** 1 ml of coolant from the 60 ml bottle to the 1L volumetric.

7.1.10 **Cap** the volumetric and **agitate** to mix.

7.1.11 **Pipet** 1 ml of coolant from the 60 ml bottle to the 100 ml volumetric.

NOTE: **The 100 ml volumetric is to be saved for the Chloride Analysis, which is to be completed within four days. The undiluted sample must also be saved for 30 days.**

7.1.12 **Cap** the volumetric and **agitate** to mix.

7.1.13 **Label** the volumetric with sample, date, time, and the number of mls of sample in the volumetric.

7.1.14 **Mark** sample "**TO BE SAVED**".

7.1.15 **Store** the 100 ml volumetric in the Hotcell Shielded Area.

7.1.16 IF a pH Analysis is to be determined on the sample, THEN **proceed** to Step 7.2. IF NOT, THEN **replace** the stopper on the 60 ml bottle.

7.1.17 **Replace** the lead cover on the shielded carrier, **place** the shielded carrier in the Hotcell Shielded Area and **proceed** to Step 7.3, Gamma Analysis Preparation.

7.2 pH Analysis - Using the Combination Methods

NOTE: **The pH meter gives a digital readout of sample temperature and will auto-compensate for temperature.**

7.2.1 **Insert** the combination pH probe and temp probe into the 60 ml bottle and **read** pH and temperature of coolant.

7.2.2 **Remove** both probes and **place** in a beaker of demin water.

7.2.3 **Log** sample results on PINGP 655, Post Accident Chemical Analysis Report.

7.2.4 IF radiation levels require, THEN **use** remote handling tools for handling the 60 ml bottle stopper and shielded carrier Lid.

7.2.5 **Replace** the stopper on the 60 ml bottle and the lid on the 60 ml bottle shielded carrier.

7.2.6 **Remove** the shielded carrier and the beaker of rinse water from the fume hood and **store** according to Step 7.6, Post Accident Sample Waste Storage and Disposal.

7.3 Gamma Analysis Preparation

7.3.1 **Pipet** 10 ml of diluted coolant sample from the 1 L volumetric to a 10 ml vial.

NOTE: **Sample should be diluted to give a contact reading of under 1 millirem/hr contact. The diluted sample should NOT exceed 25 millirem/hr contact.**

7.3.2 **Verify** that the indicated dose rate on the 10 ml vial is capable of being counted on extended geometry in EOF Countroom.

7.3.3 **Label** the vial with the sample point, date, time, and dilution factor to the sample prior to sending to EOF Countroom.

7.3.4 **Place** the 10 ml vial in the shielded carrier for transport to the EOF Countroom.

7.3.5 <u>WHEN</u> radioactive gas, charcoal, or particulate samples are received, <u>THEN</u> **ensure** all samples are labeled with date and time of sample, sample point, sample volume and/or correction factor, and flow rate.

7.3.6 **Store** all samples in the Hotcell Shielded Area until transported to the EOF Countroom.

7.4 Boron Analysis

7.4.1 **Using** the 1 L sample prepared in Step 7.1, Sample Preparation, **analyze** in accordance with the appropriate Boron Analysis procedure.

7.4.2 **Log** the results on PINGP 655, Post Accident Chemical Analysis Report.

7.4.3 **Dispose** of all radioactive waste according to Step 7.6, Post Accident Sample Waste Storage and Disposal.

7.5 Chloride Analysis

NOTE:	Chloride analysis SHALL be completed within 4 days of accident.

CAUTION:	THE REACTOR COOLANT SAMPLES TAKEN IN AN ACCIDENT CONDITION HAVE THE POTENTIAL TO BE HIGHLY RADIOACTIVE. THIS MAY GIVE RISE TO DOSE RATES FAR IN EXCESS OF WHAT WOULD NORMALLY BE ENCOUNTERED. THE ION EXCHANGE COLUMNS ON THE ION CHROMATOGRAPH COULD HAVE CONTACT READINGS OF UP TO 10 R/HR.

7.5.1 Using the 100 ml sample prepared in Step 7.1, Sample Preparation analyze for anions IAW the appropriate analysis procedure.

7.5.2 Log the results on PINGP 655, Post Accident Chemical Analysis Report.

7.5.3 Dispose of all radioactive waste according to Step 7.6, Post Accident Sample Waste Storage and Disposal.

7.6 Post Accident Sample Waste Storage and Disposal

NOTE:	Ensure samples are labeled. "TO BE SAVED" or "TO BE DUMPED" before storage in shielded area.

7.6.1 **Place** all capped or covered radioactive sample waste in the Hotcell Shielded Area.

7.6.2 IF additional waste samples are added to the Hotcell Shielded Area, THEN **survey** the Hotcell general area radiation levels. **Add** additional shielding, as necessary.

7.6.3 IF making subsequent entries into Auxiliary Building, THEN **return** the sample waste to the Sample Room for disposal down the affected unit's Sample Hood Drain.