ENCLOSURE 1

EDP-ZZ-00005
March 19, 1984
Revision 0

CALLAWAY PLANT

Engineering Departmental Procedure

EDP-ZZ-00005

ASSESSING CORE DAMAGE

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APPROVED BY W. R. Campbell DATE 3-20-84

DATE ISSUED 4/10/84

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This procedure contains the following:

Pages	1	through	5
Attachments	1	through	23
Appendices		through	
Checklist		through	

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ASSESSING CORE DAMAGE

1.0 PURPOSE AND SCOPE

1.1 This procedure provides a methodology for determining the extent of core damage following an accident using the Post Accident Sampling System (PASS). Preliminary estimates may also be made based on H_2 concentration in the containment, core exit thermocouple readings, reactor vessel water level, and containment radiation readings.

2.0 DEFINITIONS

2.1 Clad damage - Clad damage is characterized by the release of fission products which have accumulated in the gap between the clad and the fuel. The fission products which diffuse to this gap are the volatile ones such as the noble gases, the iodines, and the cesiums.

2.2 Fuel overheating - Fuel overheating is characterized by grain boundary release and diffusion from the UO_2 grains. This is estimated to be 20-40% of the noble gas, iodine and cesium inventories.

2.3 Fuel melt - Fuel melt leads to rapid release of many noble gases, halides and cesiums remaining in the fuel after overheating. Significant release of the strontium and barium - lanthanum groups distinguishes this condition.

3.0 NOTES AND PRECAUTIONS

3.1 This procedure may be copied so that it can be used more than once. Attachment 1 will have to be copied for each isotope to be used in the analysis.

3.2 During accident conditions, it is not known in what order that information will become available. Therefore, this procedure does not have to be completed in the order that it is written.

3.3 If hydrogen recombiners or the hydrogen purge system are operating, core damage estimates based on hydrogen in the containment may be inaccurate.

3.4 Use as many indications as possible to differentiate between the various core damage states. Because of overlapping values of release and potential simultaneous conditions of clad damage, overtemperature, and core melt, considerable judgement needs to be applied.

4.0 PROCEDURE

4.1 Obtain an estimate of core damage using containment hydrogen concentration, core exit thermocouple readings, reactor vessel water level, and the containment radiation monitor.

4.1.1 Hydrogen Concentration

4.1.1.1 Record containment hydrogen concentration.

____

4.1.1.2 From Attachment 9, obtain the % zirconium-water reaction and record here.

____

4.1.2 Core Exit Thermocouple Readings

4.1.2.1 From Attachment 8, estimate the core damage based on core exit thermocouple readings. Core damage: ____________________

4.1.3 Reactor Vessel Water Level

4.1.3.1 Record the duration and extent of core uncovery.

Duration: __________ minutes
Extent: __________ %

4.1.3.2 From Attachment 8, estimate the core damage based on core uncovery.

Core damage: ____________________

4.1.4 Containment Radiation Monitor

4.1.4.1 Record the Containment Radiation Monitor level R = __________R/hr.

4.1.4.2 Record the average power during entire period of operation (from Attachment 2) P=________%

4.1.4.3 Calculate the normalized dose rate

$$\text{Normalized Dose Rate} = 2.34 \times 10^{-4} \times R \times \frac{100\%}{P} =$$
_______R/hr -MWt

4.1.4.4 Record the time since the accident _______ hours.

4.1.4.5 Using Attachment 10, estimate the core damage.

Core damage: ____________________

4.2 Estimation of core damage using PASS sample results.

4.2.1 As sample results become available, complete a copy of Attachment 1 for each isotope. If an estimation of core damage was made in 4.1, then preference should be given to those isotopes which are indicative of that type of core damage. Attachment 3 provides a list for this purpose.

4.2.2 Using the percentage of inventory released and the fission product ratio from Attachment 1, and using Attachment 8 and 11 to 23, estimate the damage and record below.

Sample Time	Fission Product	Percentage Released	Fission Product Ratio	Estimated Core Damage

5.0 REFERENCES

5.1 WOG-84-111, Draft CDA Methodology

5.2 FSAR Table 6.2.2-6

5.3 Table of Isotopes; Lederer, Hollander & Perlman

CALCULATION OF PERCENT OF CORE INVENTORY RELEASED

1.0 Isotope ______

1.1 Decay constant (from Attachment 3) λ = ______

1.2 Half-life (from Attachment 3) $T_1/_2$ = ______

2.0 Time and date of shutdown __________.

3.0 POWER CORRECTION FACTOR

3.1 Determine the power history using Attachment 2.

3.2 For steady-state power (except Cs-134), complete the appropriate section of 3.3. For transient power history (except Cs-134), complete the appropriate section of 3.4. For Cs-134, complete 3.5

3.3 STEADY STATE EXCEPT Cs-134

3.3.1 Half Life <1 day

Power Correction Factor (PCF) =

$$\frac{\text{Steady state power percentage for prior 4 days}}{100} = \_\_\_\_\_\_$$

3.3.2 Half Life >1 day

Power Correction Factor (PCF) =

$$\frac{\text{Steady state power percentage for prior 30 days}}{100} = \_\_\_\_\_\_$$

3.4 TRANSIENT EXCEPT Cs-134

3.4.1 Isotope with $T_1/_2 \geq 1$ year

$$\text{Power Correction Factor (PCF)} = \frac{\text{EFPD}}{\text{Total days of operation}} = \_\_\_\_\_\_$$

3.4.2 Total period of operation > 4 x $T_1/_2$

$$\text{Power Correction Factor (PCF)} = \frac{\Sigma j[Pj\ (1-e^{-\lambda tj})e^{-\lambda t^{o}j}]}{100} = \_\_\_\_\_\_$$

where tj = operating period in days at power Pj where power does not vary more than ±10 percent power from time average value (Pj)
Pj = percent power during operating period tj
$t°j$ = time between end of period j and time of reactor shutdown in days.

3.4.3 Remaining trasient cases

Power Correction Factor (PCF) =

$$\frac{\Sigma j[Pj(1-e^{-\lambda tj})e^{-\lambda t°j}]}{100(1-e^{-\lambda \Sigma j tj})} = \_\_\_\_\_\_$$

3.4 POWER CORRECTION FACTOR FOR CS-134

Power Correction Factor (from Attachment 6) = ______
(Use average power during entire period of operation from Attachment 2)

4.0 RCS ACTIVITY

4.1 Sample Data

4.1.1 Time and date of RCS sample ______

4.1.2 Time since shutdown
t = ______ (same units as λ)

4.1.3 RCS volume (from Attachment 4) V = ______ ft^3

4.1.4 RCS temperature TI = ______ °F.

4.1.5 RCS water density ratio (from Attachment 7)
$\rho 1/\rho stp$ = ______

4.1.6 Sample result Cm = ______ µCi/cc

4.1.7 Sample temperature T2 = ______ °F

4.1.8 Sample water density ratio (from Attachment 7)
$\rho Z/\rho stp$ = ______

4.2 Decay correction of sample to time of reactor shutdown

4.2.1 $Cc = Cme^{\lambda t}$ = ______ µCi/cc

4.3 Temperature correction of sample

4.3 Temperature correction of sample

4.3.1 $C = Cc \times \frac{\rho_2/\rho stp}{\rho^1/\rho stp} =$ ______ µCi/cc

4.4 RCS Activity A(RC)

4.4.1 $A(RC) = V \times \rho_1/\rho stp \times C \times .02833 =$ ______ Ci

5.0 <u>CONTAINMENT SUMP ACTIVITY</u>

5.1 Sample Data

5.1.1 Time and date of containment sump sample ______

5.1.2 Time since shutdown t= _____ (same units as λ)

5.1.3 Containment sump volume (from Attachment 5)
$V =$ _____ ft^3

5.1.4 Containment sump temperature T1 = ______°F

5.1.5 Containment sump water density ratio (from Attachment 7)

$\rho^1/\rho stp =$ ________

5.1.6 Sample result Cm = _______ µCi/cc

5.1.7 Sample temperature T2 = _______°F

5.1.8 Sample water density ratio (from Attachment 7)
$\rho z/\rho stp =$ _______

5.2 Decay correction of sample to time of reactor shutdown

5.2.1 $Cc = Cm \times e^{\lambda t} =$ ______ µCi/cc

5.3 Temperature correction of sample

5.3.1 $C = Cc \times \frac{\rho_2/\rho stp}{\rho_1/\rho stp} =$ _______ µCi/cc

5.4 Containment Sump Activity A(CS)

5.4.1 $A(CS) = V \times \rho_1/\rho stp \times C \times .02833 =$ ______ Ci

6.0 <u>CONTAINMENT ATMOSPHERE ACTIVITY</u>

6.1 Sample Data

6.1.1 Time and date of containment atmosphere sample ______

6.1.2 Time since shutdown t = ______ (same units as λ)

6.1.3 Containment atmosphere temperature $T1$ = ______ °F

6.1.4 Containment atmosphere pressure $P1$ = ______ psia

6.1.5 Sample result Cm = ______ μCi/cc

6.1.6 Sample temperature $T2$ = ______ °F

6.1.7 Sample pressure $P2$ = ______ psia

6.2 Decay correction of sample to time of reactor shutdown

6.2.1 $Cc = Cme^{\lambda t}$ = ______ μCi/cc

6.3 Temperature and pressure correction of sample

6.3.1 $C = Cc \times \frac{P_1 \times (T_2 + 460)}{P_2 \times (T_1 + 460)}$ = ______ μCi/cc

6.4 Containment Atmosphere Activity A(CA)

6.4.1 $A(CA) = C \times 7.075 \times 10^4$ = ______ Ci

7.0 TOTAL ACTIVITY A

7.1 $A = A(RC) + A(CS) + A(CA)$ = ______ Ci

8.0 INVENTORY AVAILABLE FOR RELEASE

8.1 Uncorrected inventory (from Attachment 3) Iu = ______ Ci

8.2 Power Correction Factor (from section 3) PCF = ______

8.3 Corrected inventory $Ic = PCF \times Iu$ = ______ Ci

9.0 PERCENTAGE OF INVENTORY RELEASED

9.1 Percentage of inventory released = $\frac{A}{I_c} \times 100\%$ = ______ %

10.0 ACTIVITY RATIO

10.1 If the isotope is a noble gas, complete 10.2. If the isotope is an isotope of iodine, complete 10.3. Otherwise don't complete this section.

10.2 Noble gas ratio = A/A(Xe-133) = ______

10.2.1 Iodine ratio = A/A(I-131) = ______

POWER HISTORY

1. 30-day power history

Days Before Shutdown	Average Power
1	______
2	______
3	______
4	______
5	______
6	______
7	______
8	______
9	______
10	______
11	______
12	______
13	______
14	______
15	______
16	______
17	______
18	______
19	______
20	______
21	______
22	______
23	______
24	______
25	______
26	______
27	______
28	______
29	______
30	______

2. Total number of days of operation D = ______

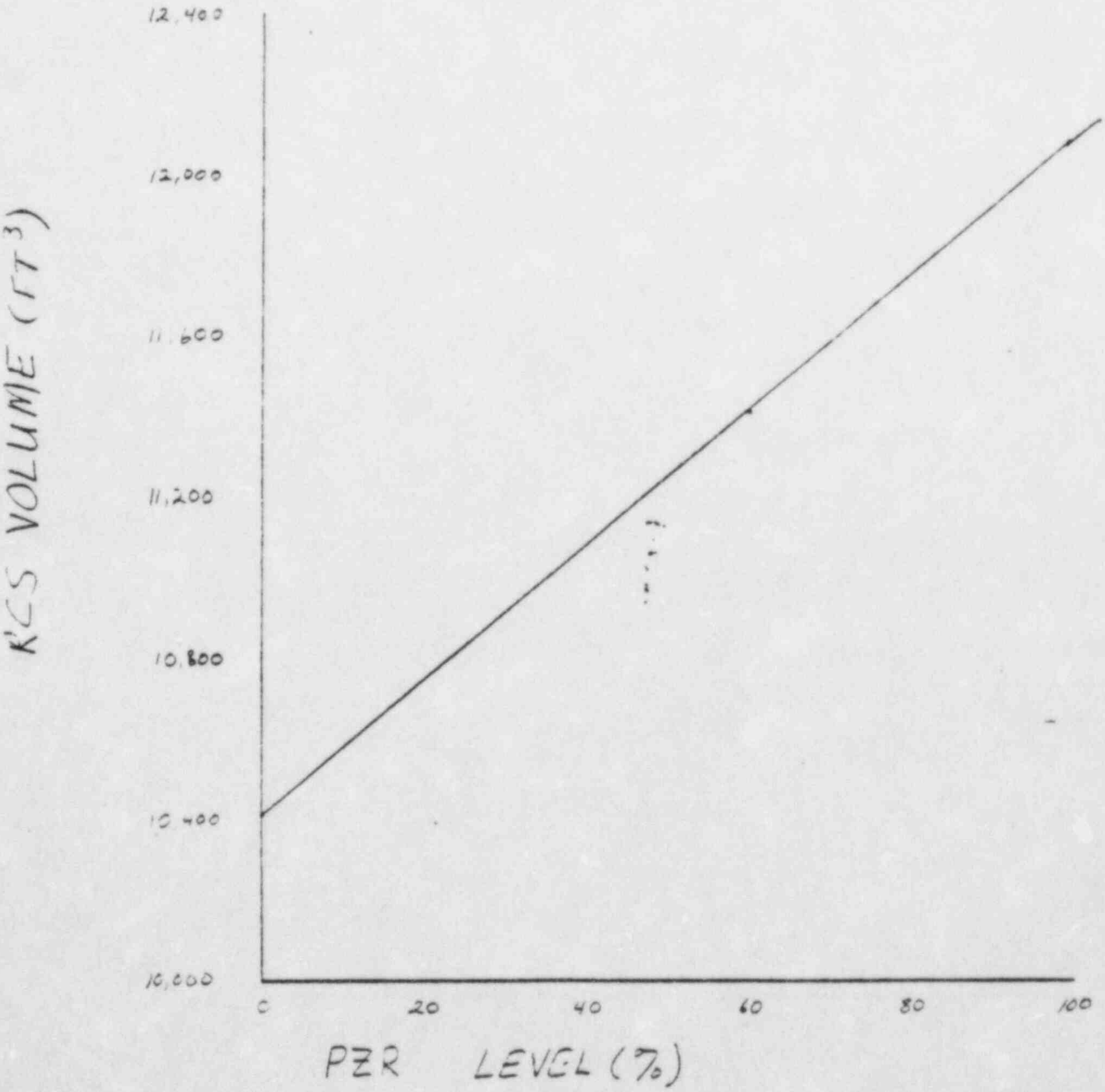
3. EFPD = ______

4. Average power during entire period of operation

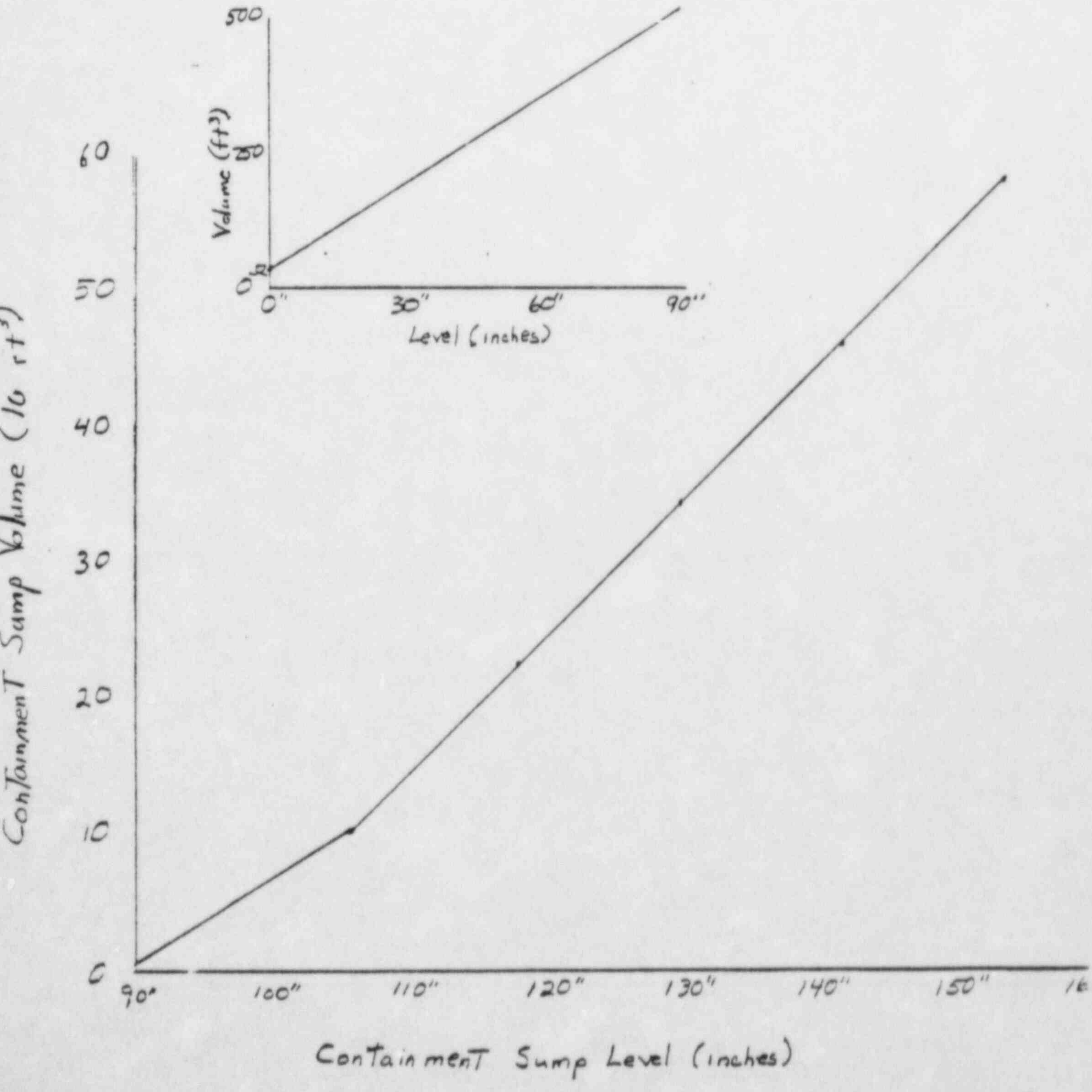
$$= \frac{EFPD}{D} \times 100\% =$$

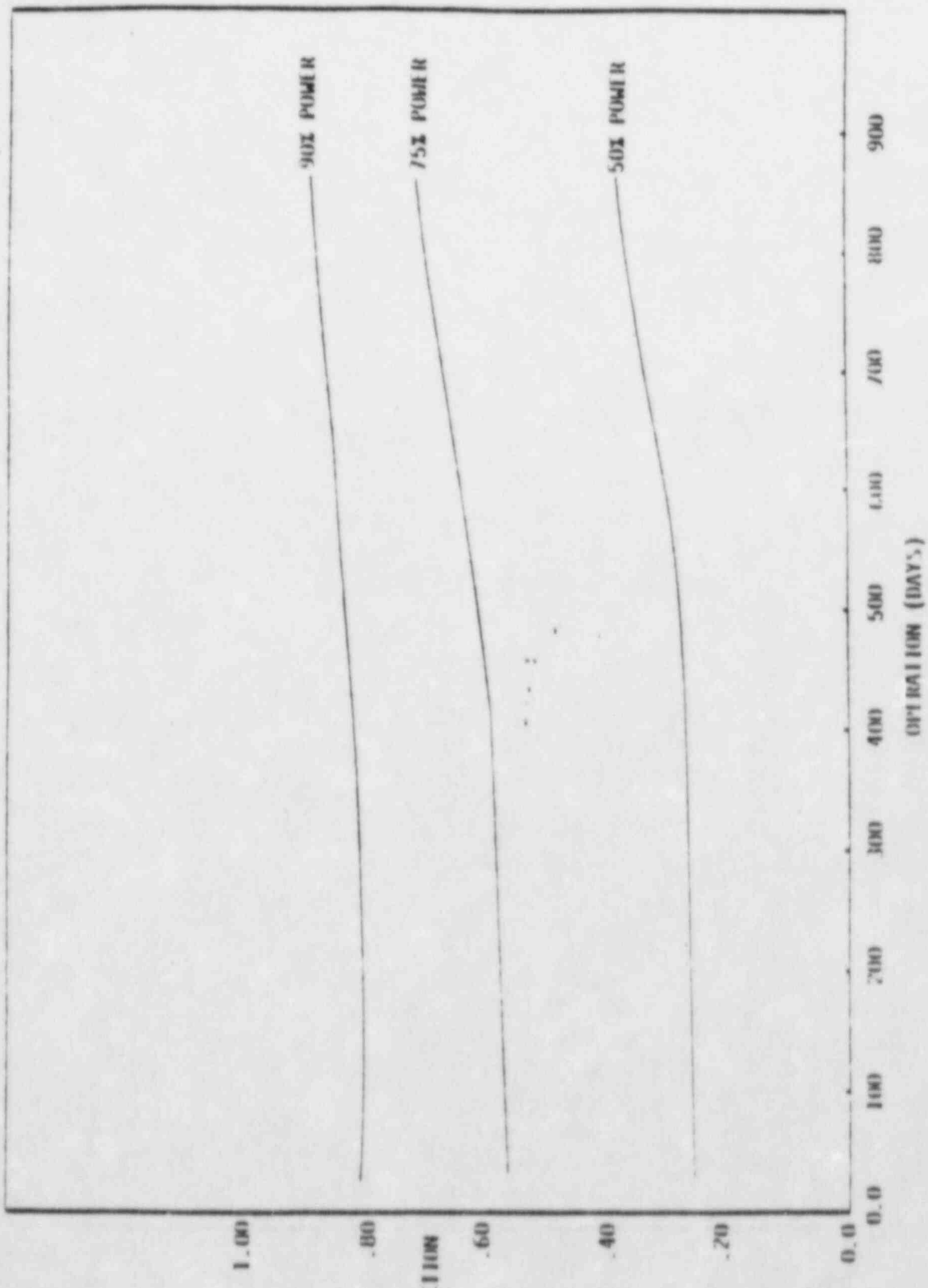
NUCLIDE	$T_{1/2}$	λ	INVENTORY	CORE DAMAGE STATE
Kr-87	76m	$.00912m^{-1}$	4.1(7)	Clad Failure
Rb-88	18m	$.0385\ m^{-1}$	6.1(7)	
Xe-131m	12d	$.0578\ d^{-1}$	6.5(5)	
Xe-133	5.4d	$.128\ d^{-1}$	1.9(8)	
I-131	8d	$.0867\ d^{-1}$	9.8(7)	
I-132	2.3h	$.3014\ h^{-1}$	1.5(8)	
I-133	21h	$.033\ h^{-1}$	2.0(8)	
I-135	6.8h	$.102\ h^{-1}$	1.9(8)	
Cs-134	2y	$.3466y^{-1}$	1.3(7)	Fuel Overheat
Cs-137	30y	$.0231y^{-1}$	1.1(7)	
Te-129	68.7m	$.010\ m^{-1}$	3.4(7)	
Te-132	77.7h	$.009\ h^{-1}$	1.4(7)	
Sr-89	53d	$.0131d^{-1}$	8.2(7)	Fuel Melt
Ba-140	12.8d	$.0541d^{-1}$	1.7(8)	
La-140	40h	$.0173h^{-1}$	1.8(8)	
La-142	90m	$.0077m^{-1}$	1.6(8)	
Pr-144	17.3	$.0401m^{-1}$	1.2(8)	

RCS VOLUME

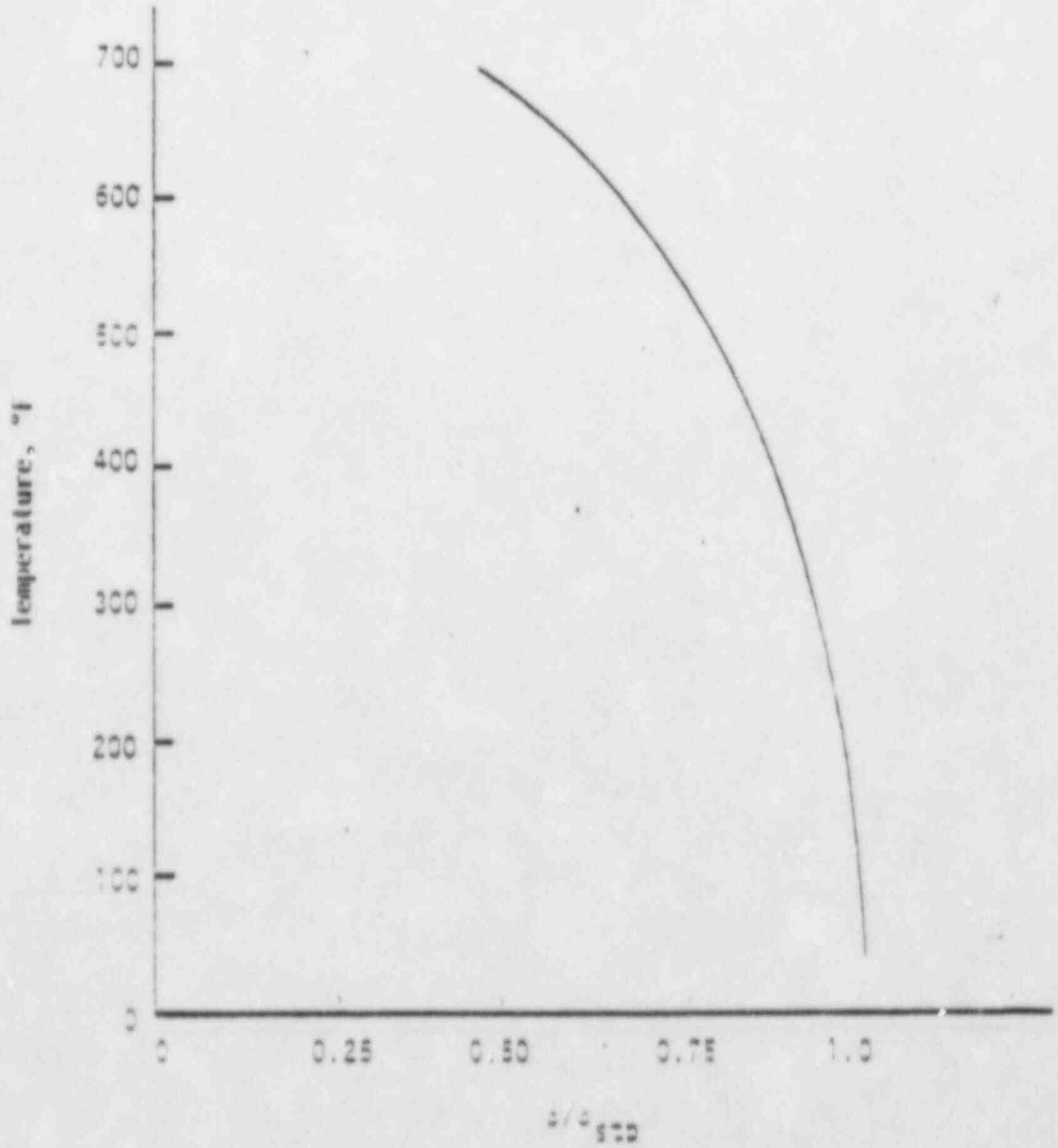

12,400
12,000
11,600
11,200
10,800
10,400
10,000
RCS VOLUME (FT^3)
0
20
40
60
80
100
PZR LEVEL (%)

CONTAINMENT SUMP VOLUME





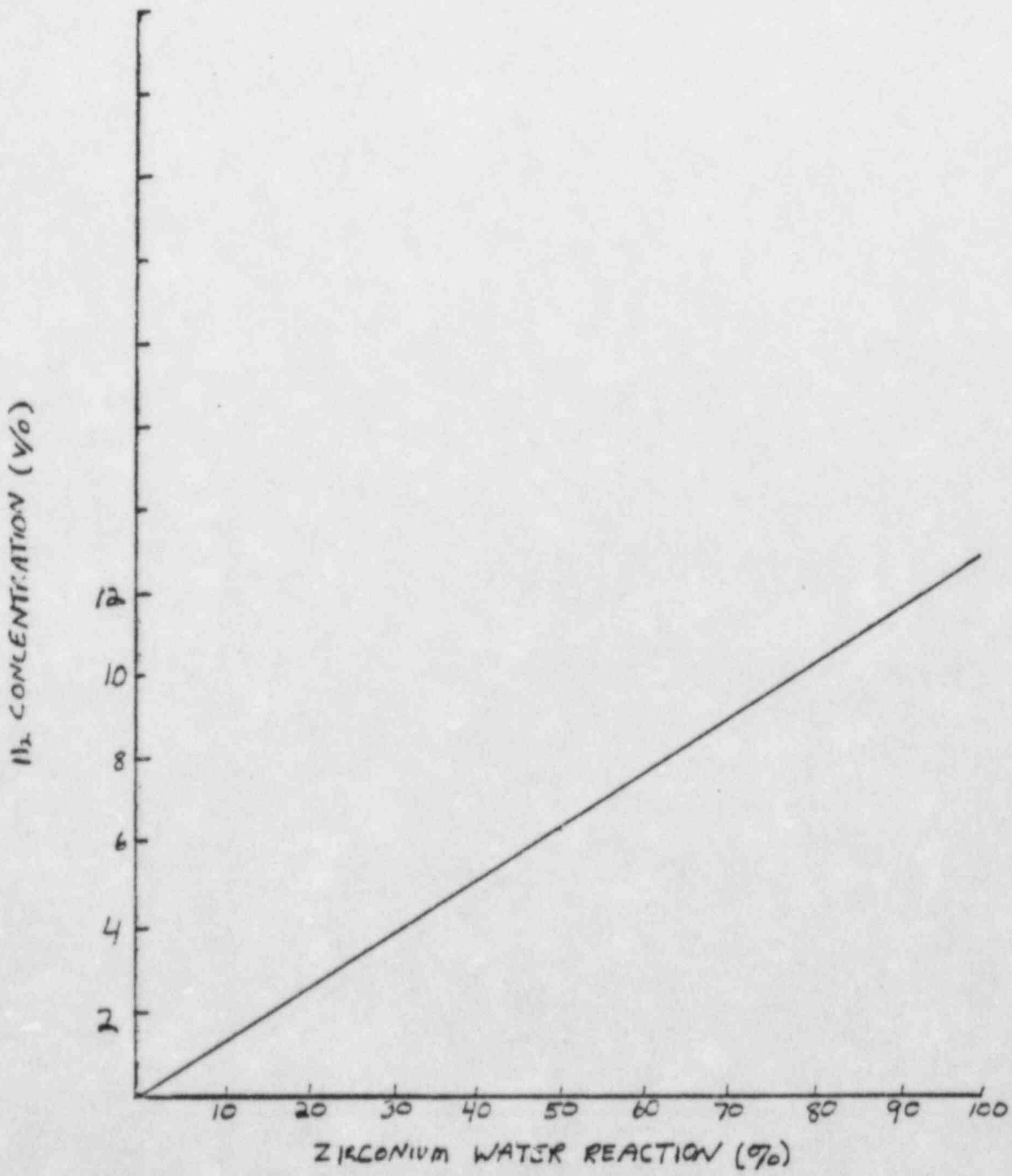
POWER CORRECTION FACTOR FOR CS-134

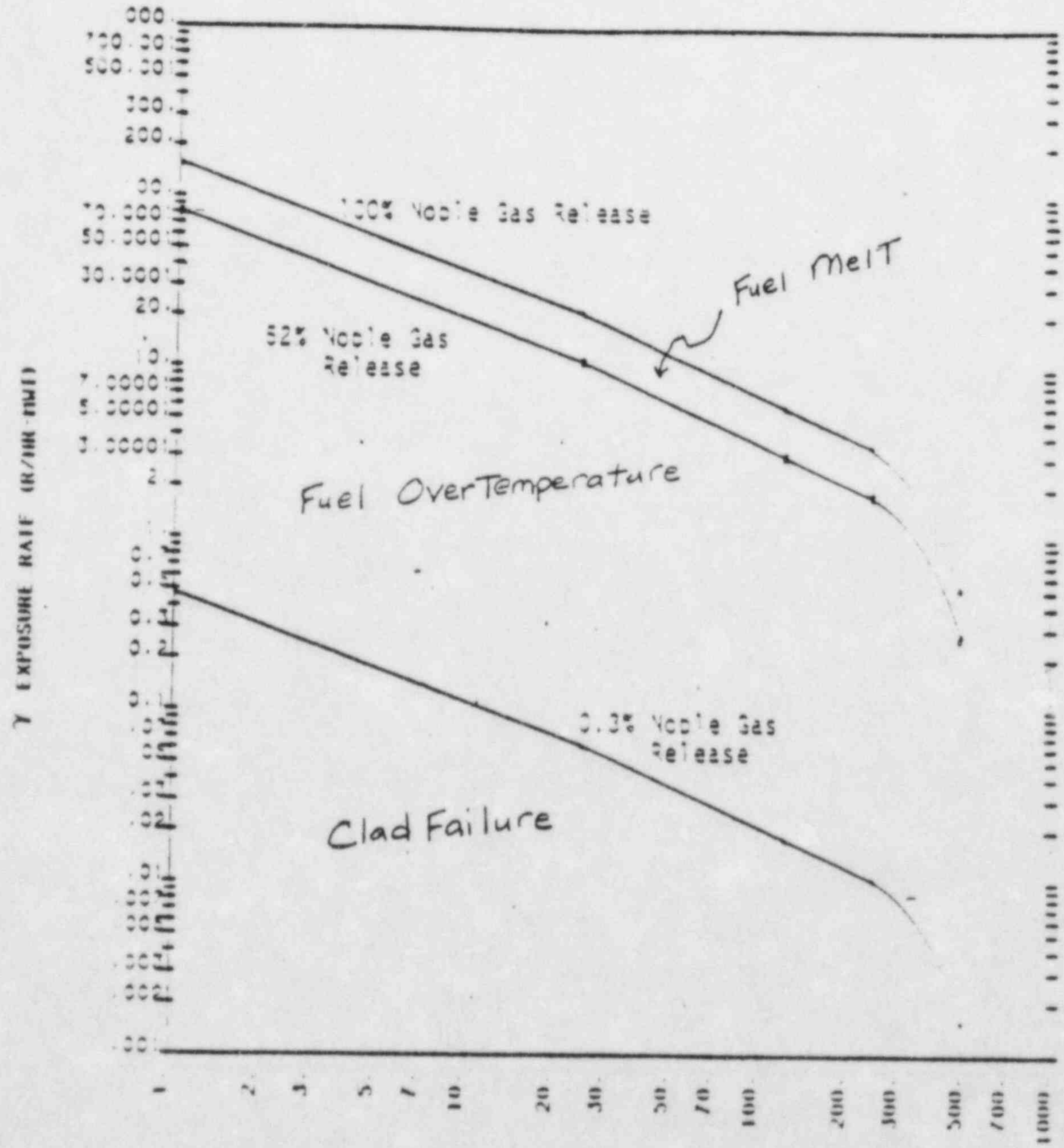


WATER DENSITY RATIO (TEMPERATURE VS. STP)

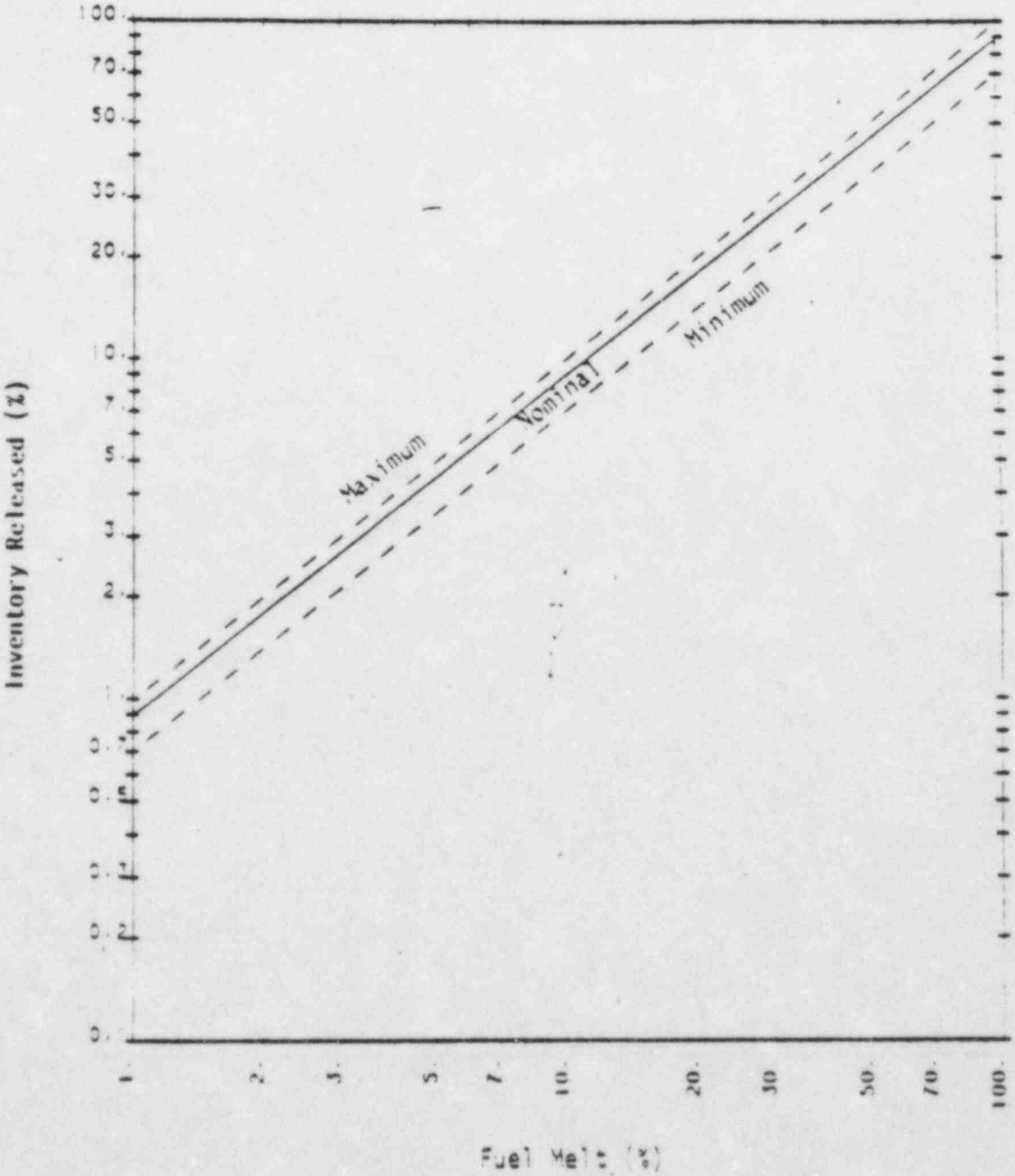
CORE DAMAGE	FISSION PRODUCT RATIO	CORE EXIT THERMOCOUPLE READINGS (°F)	CORE UNCOVERY	H_2 Monitor (VOL % H_2)
No Clad Damage	Kr-87=0.022 I-133=0.71	--	None	Negligible
0-50% Clad Damage	Kr-87=0.022 I-133=0.71	750-1300	50% 5-30 min	0-6.5
50-100% Clad Damage	Kr-87=0.022 I-133=0.71	1300-1650	100% 5-30 min	6.5-13
0-50% Overtemperature	Kr-87=0.22 I-133=2.1	>1650	50% 45-75 min	--
50-100% Overtemperature	Kr-87=0.22 I-133=2.1	>1650	100% 45-75 min	--
0-50% Fuel Melt	Kr-87=0.22 I-133=2.1	>1650	50% >75 min	--
50-100% Fuel Melt	Kr-87=0.22 I-133=2.1	>1650	100% >75 min	--

H_2 CONCENTRATION VS. ZIRCONIUM-WATER REACTION

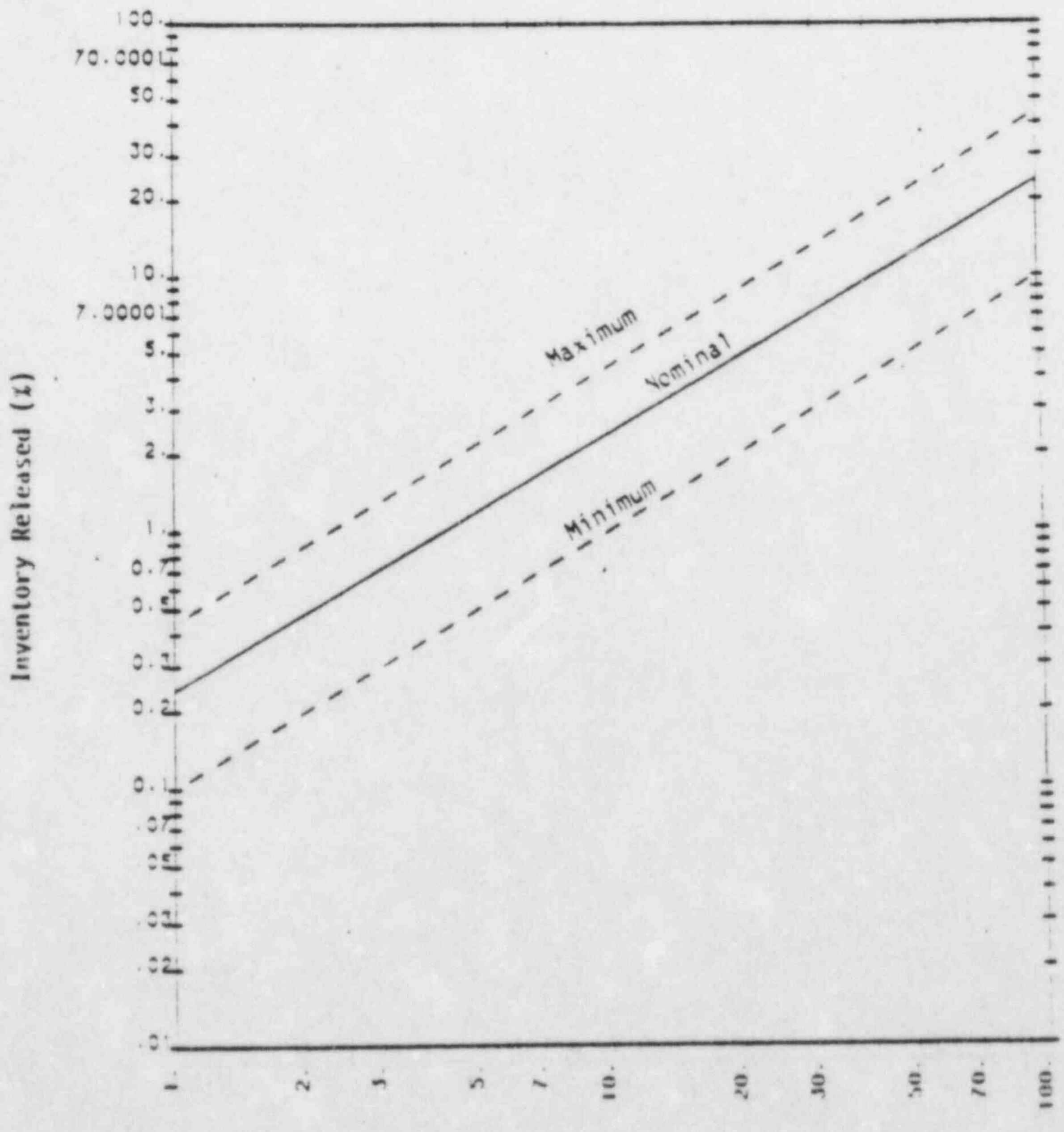




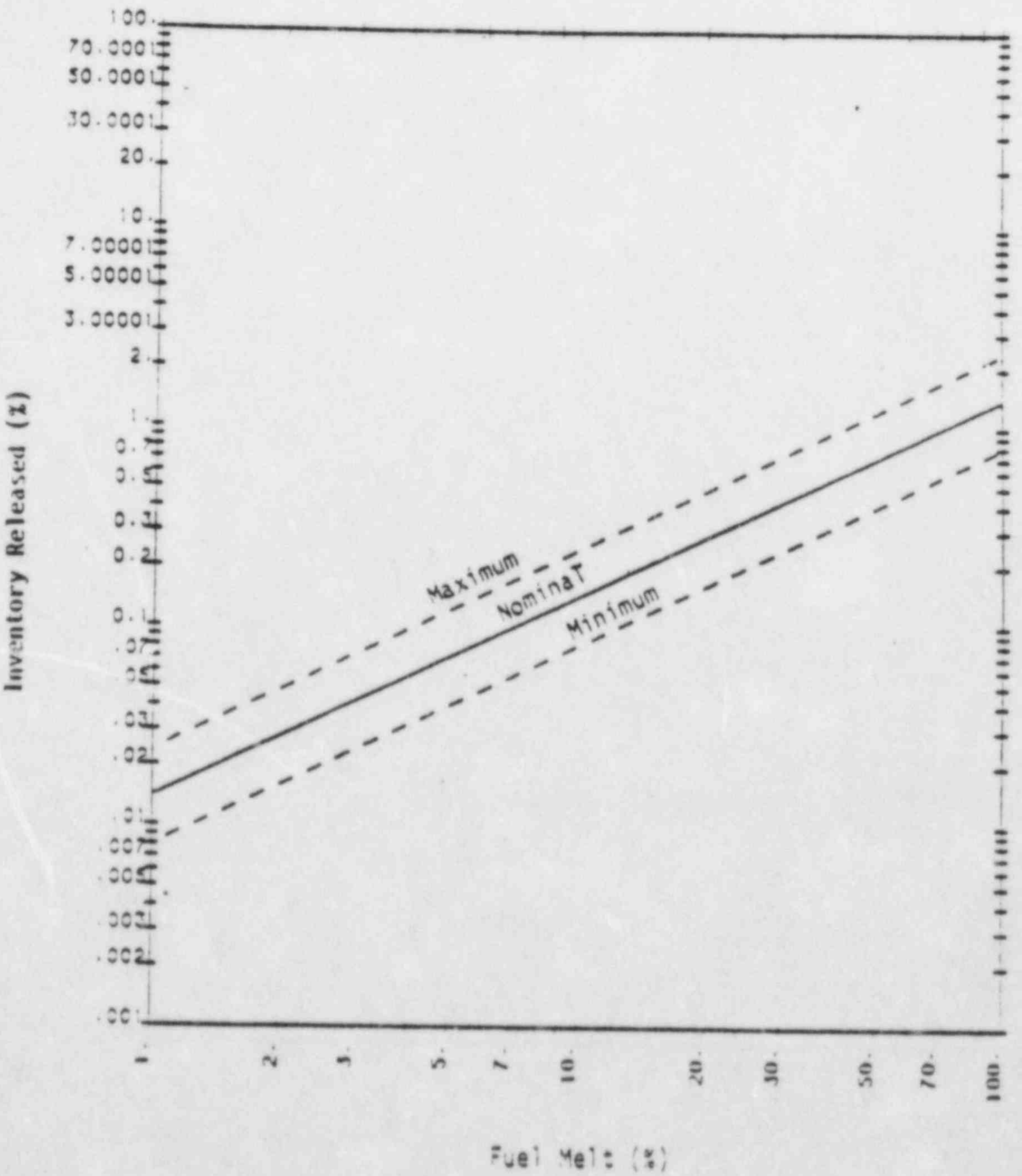
PERCENT NOBLE GASES IN CONTAINMENT



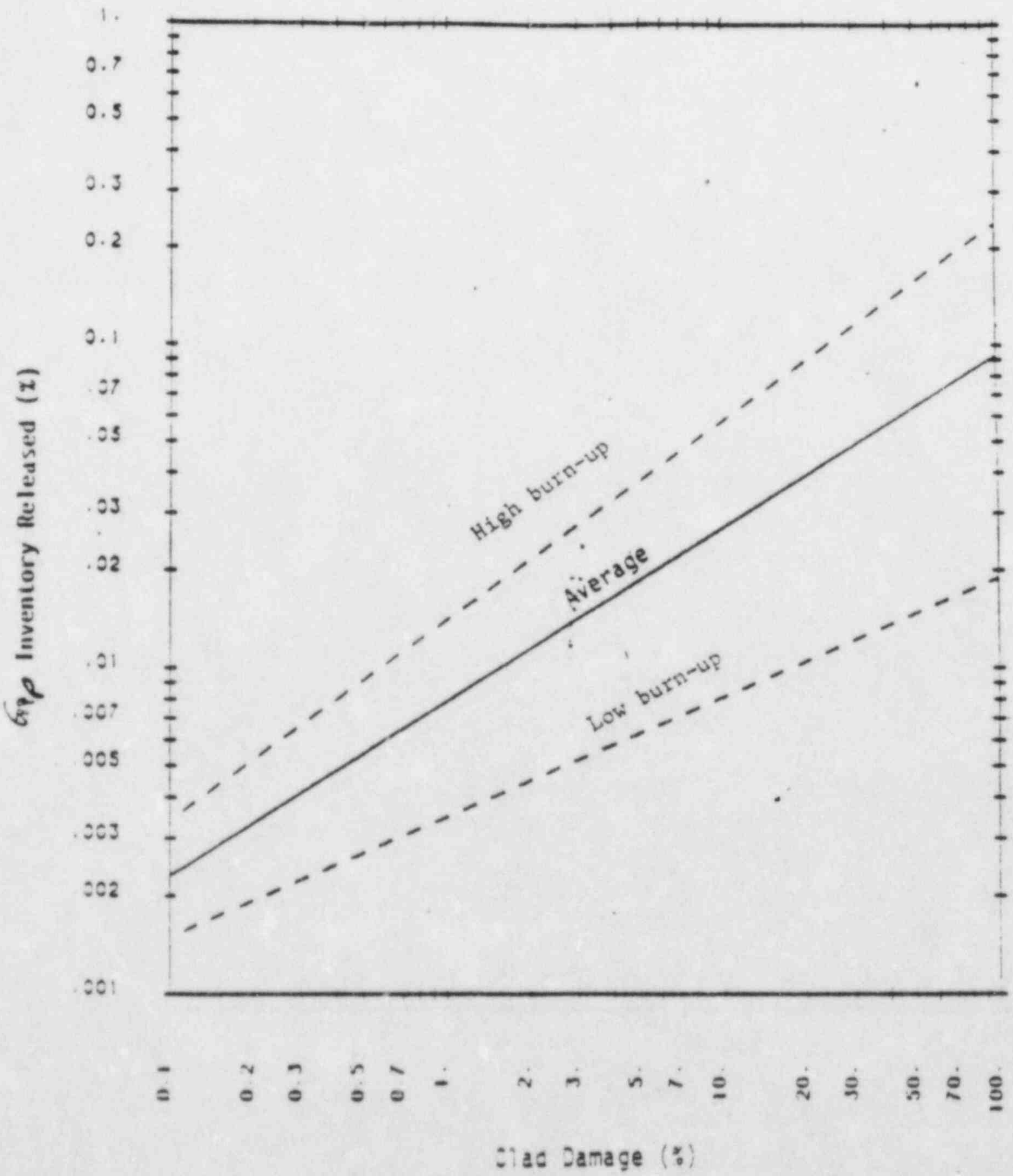
RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF XE, KR, I, CS, TE



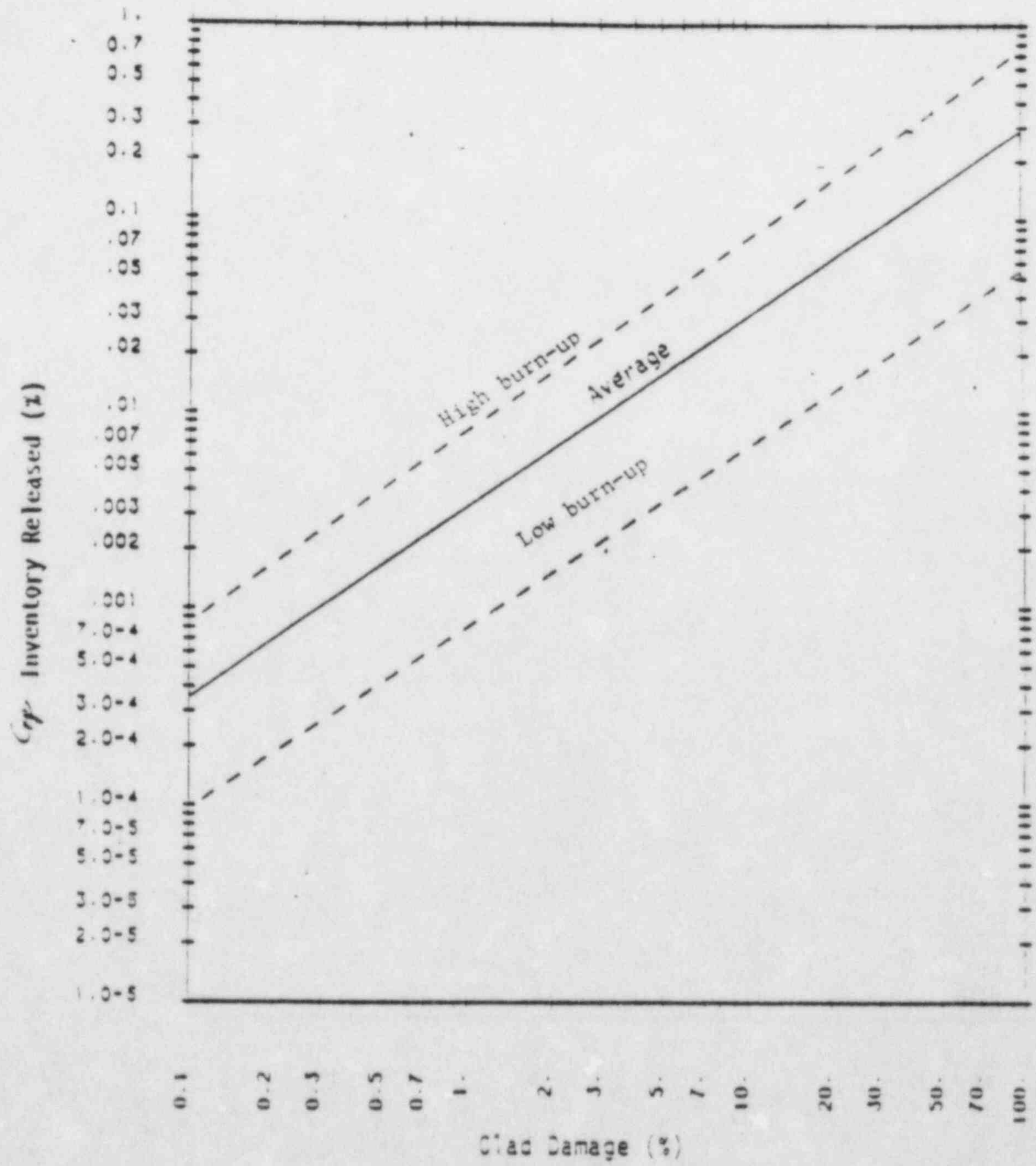
RELATIONSHIP OF % FUEL MELT WITH % INVENTORY
RELEASED OF BA, SR



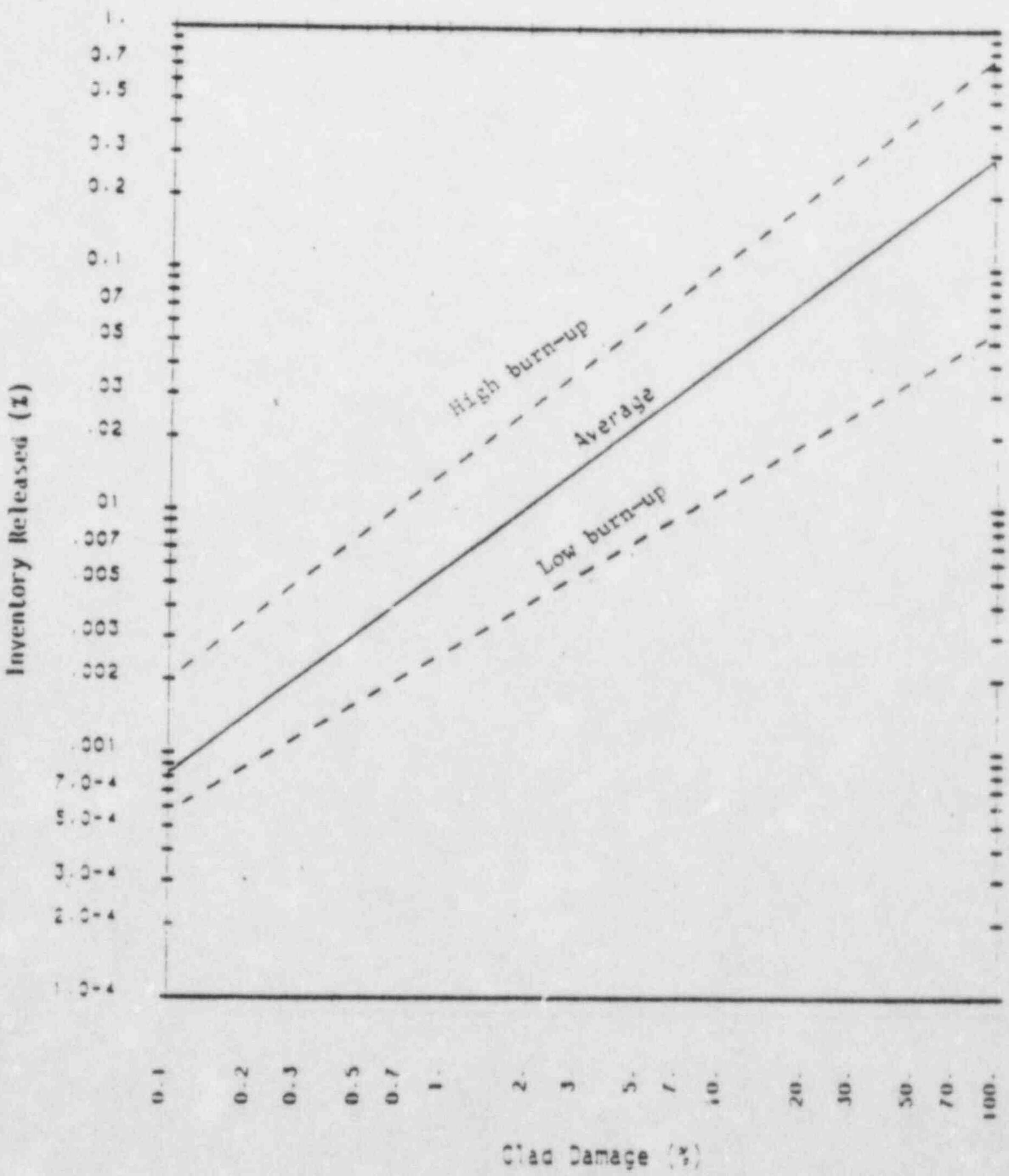
RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF PR



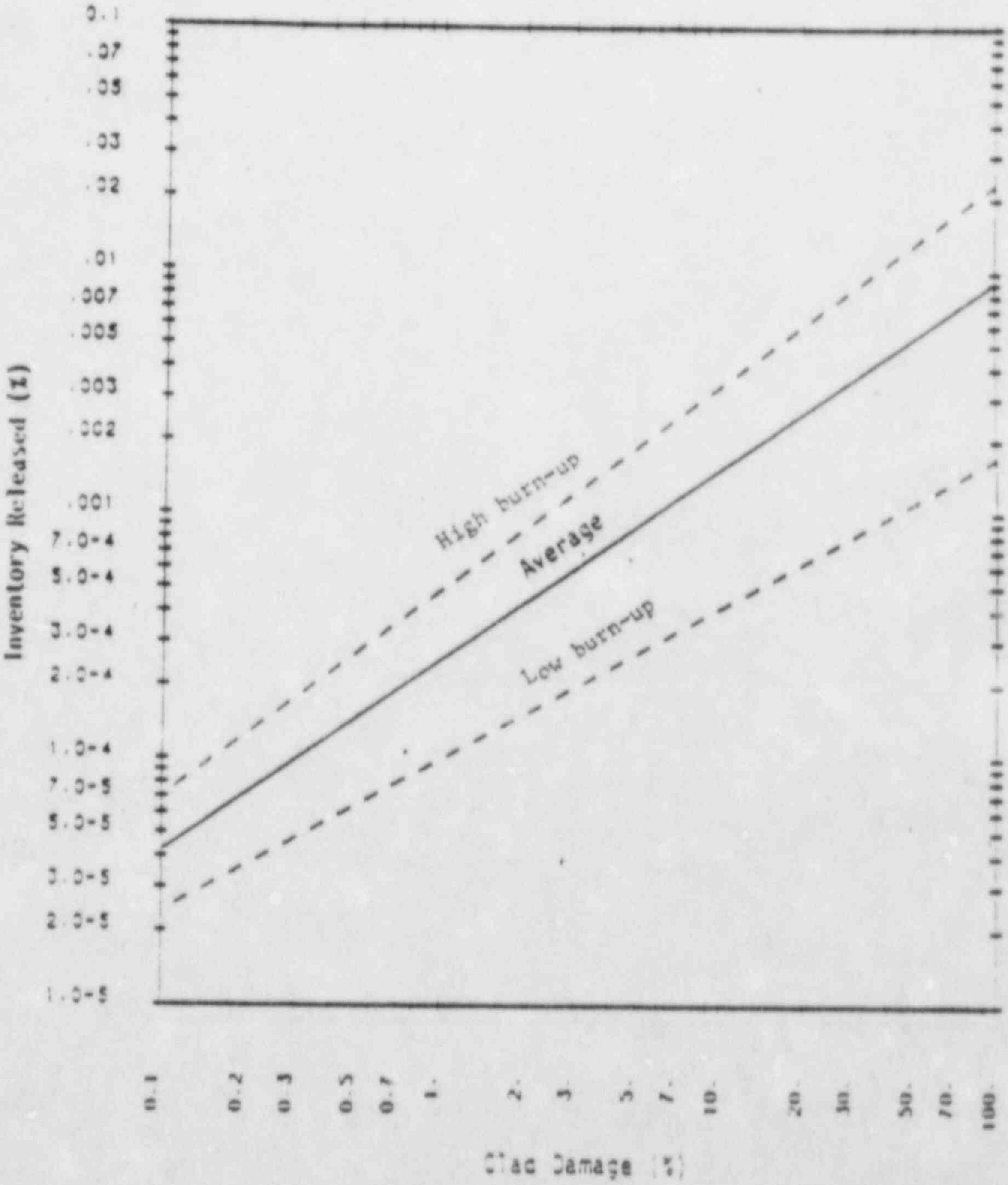
RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY
RELEASED OF XE-133
Xe



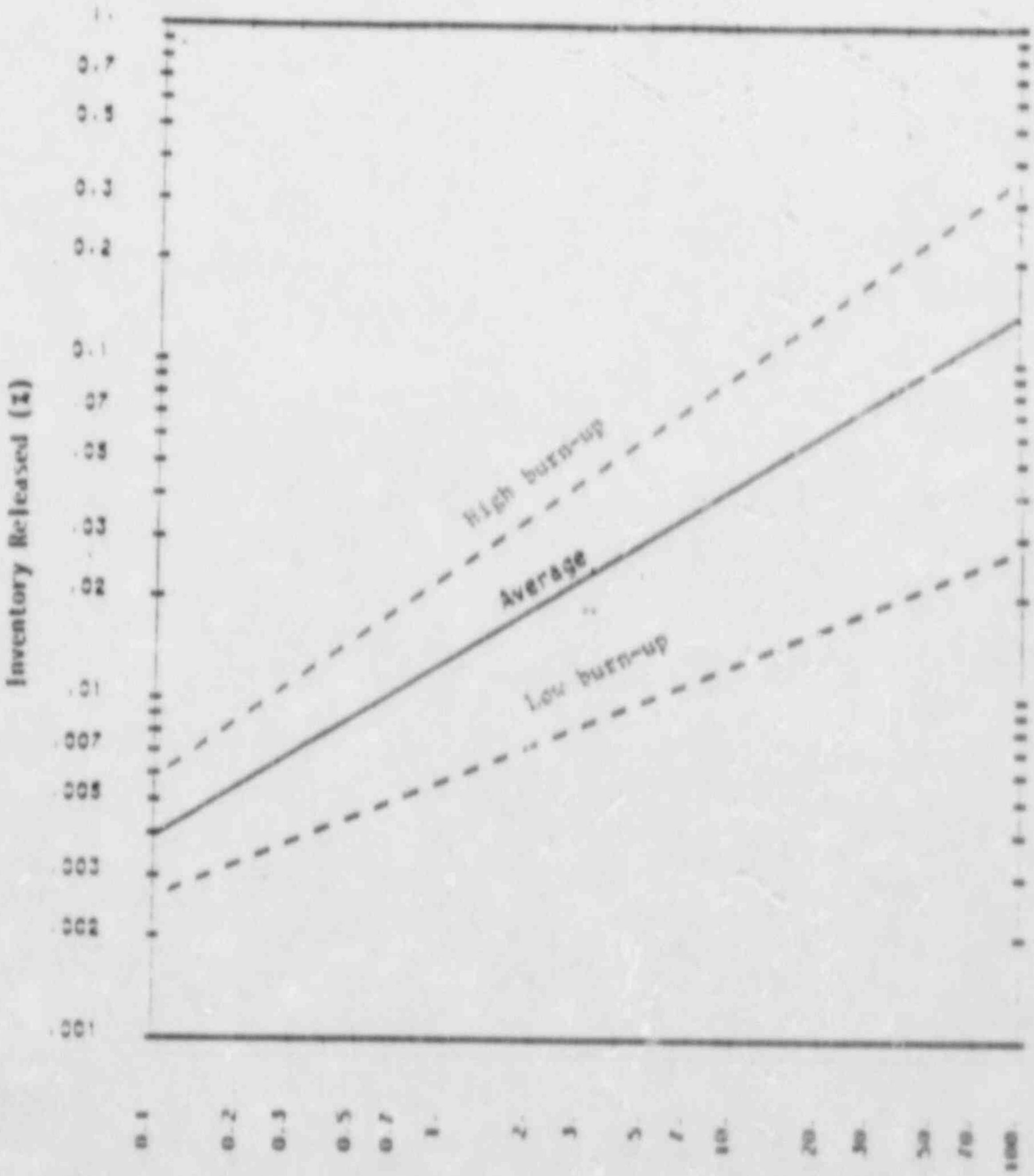
RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-131



RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY
RELEASED OF I-131 WITH SPIKING



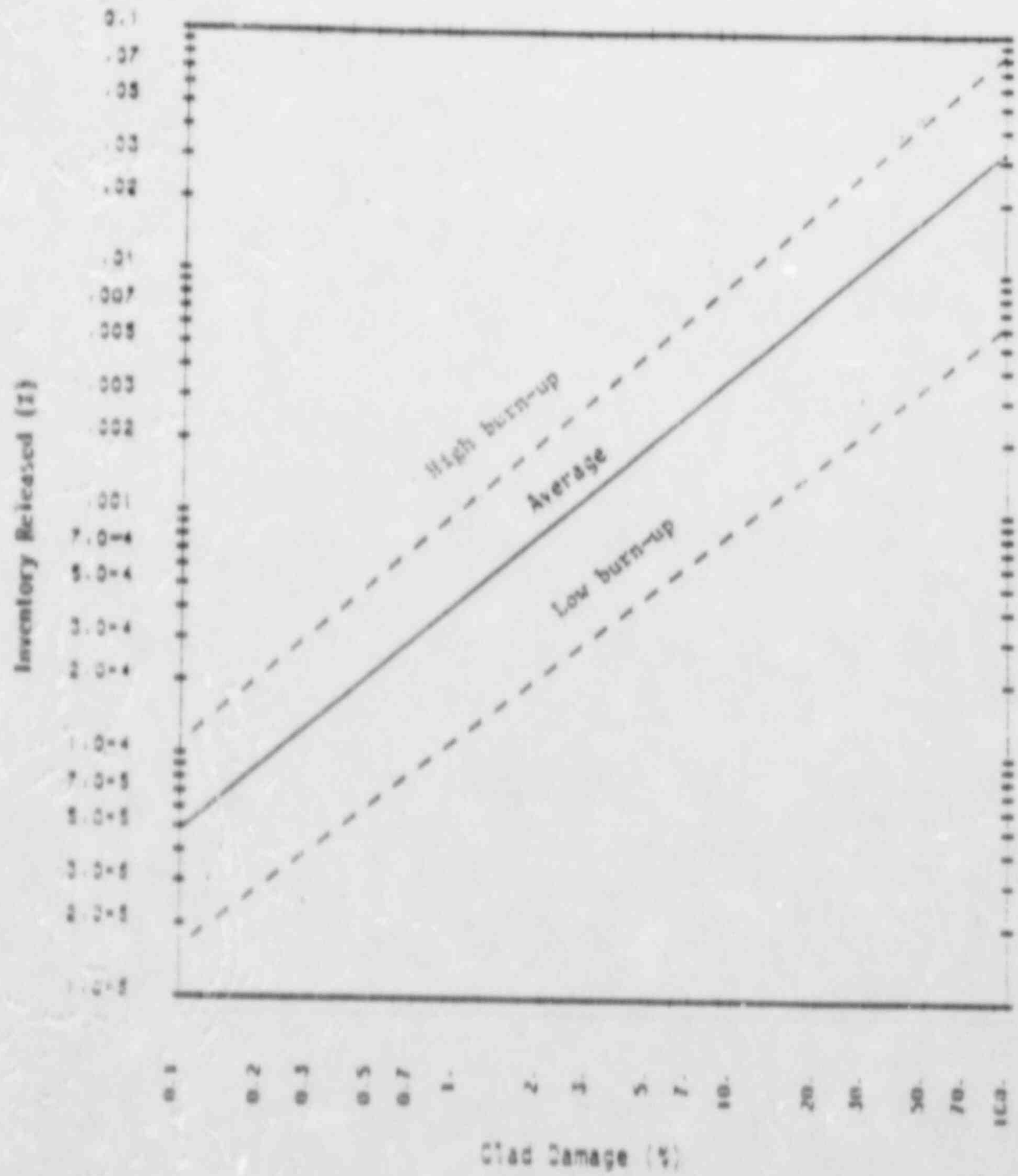
RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF Kr-87



RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF Xe-131m

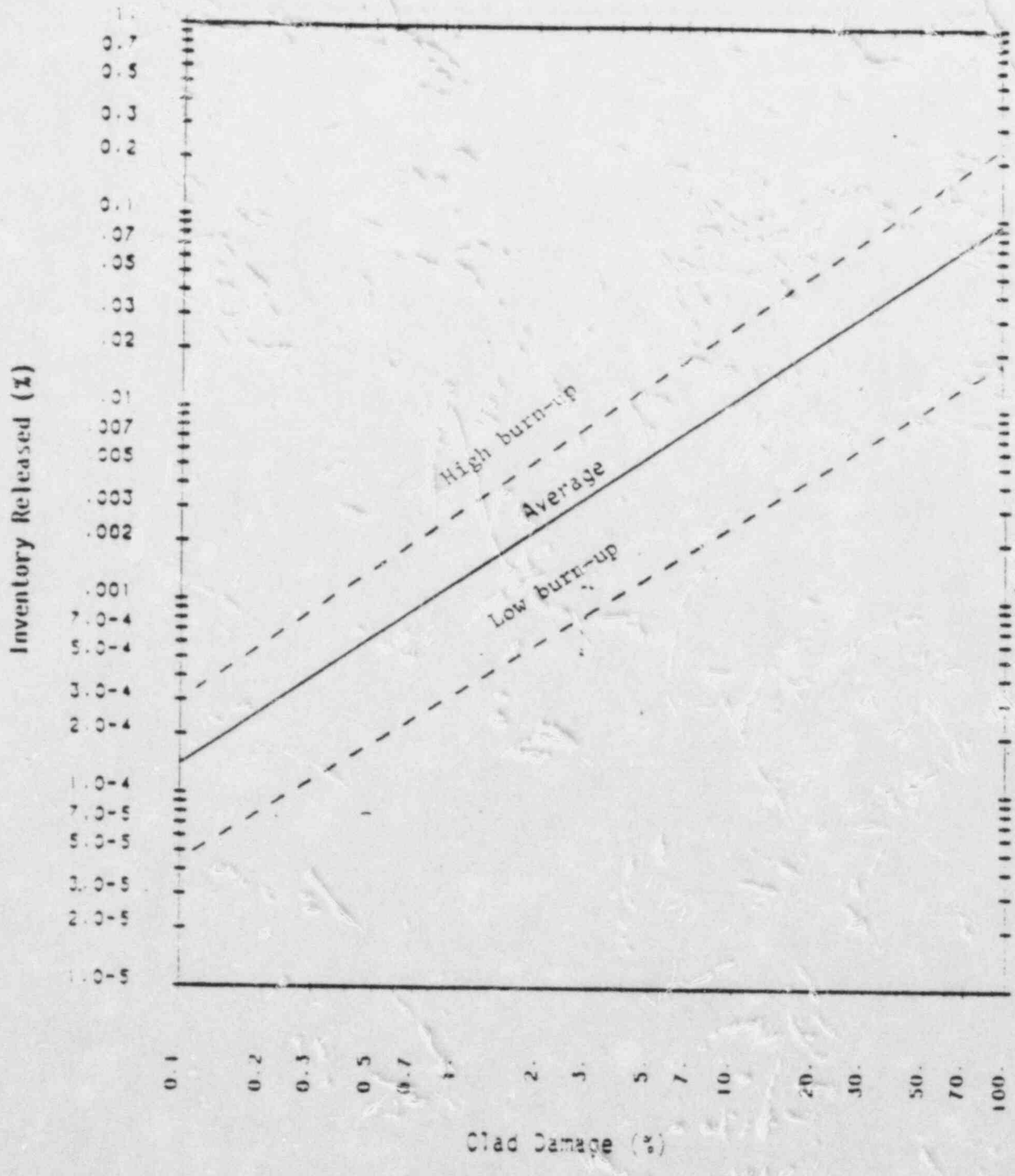
Proced. No. EDP-ZZ-00005

Rev. 0

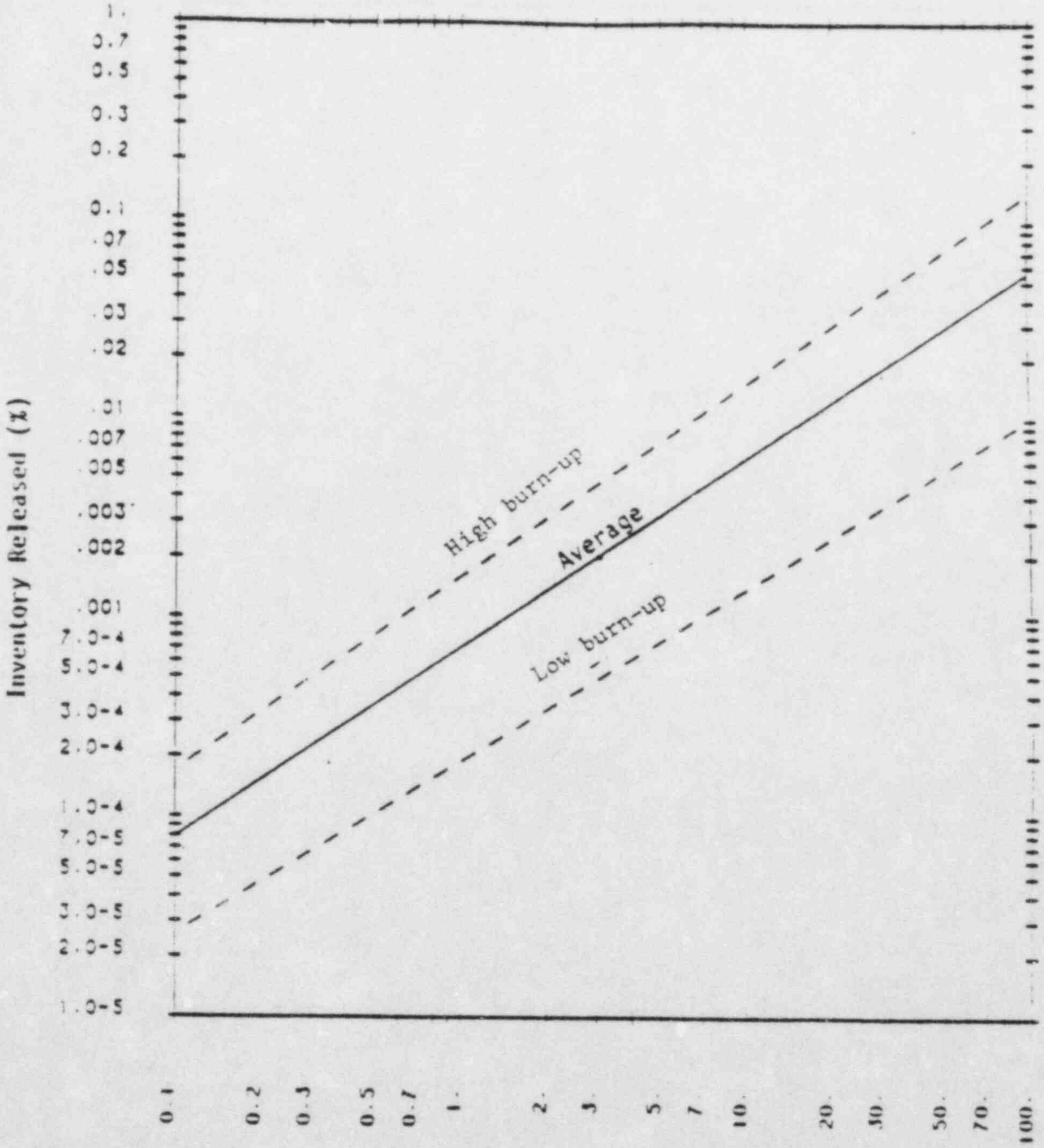


RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-132

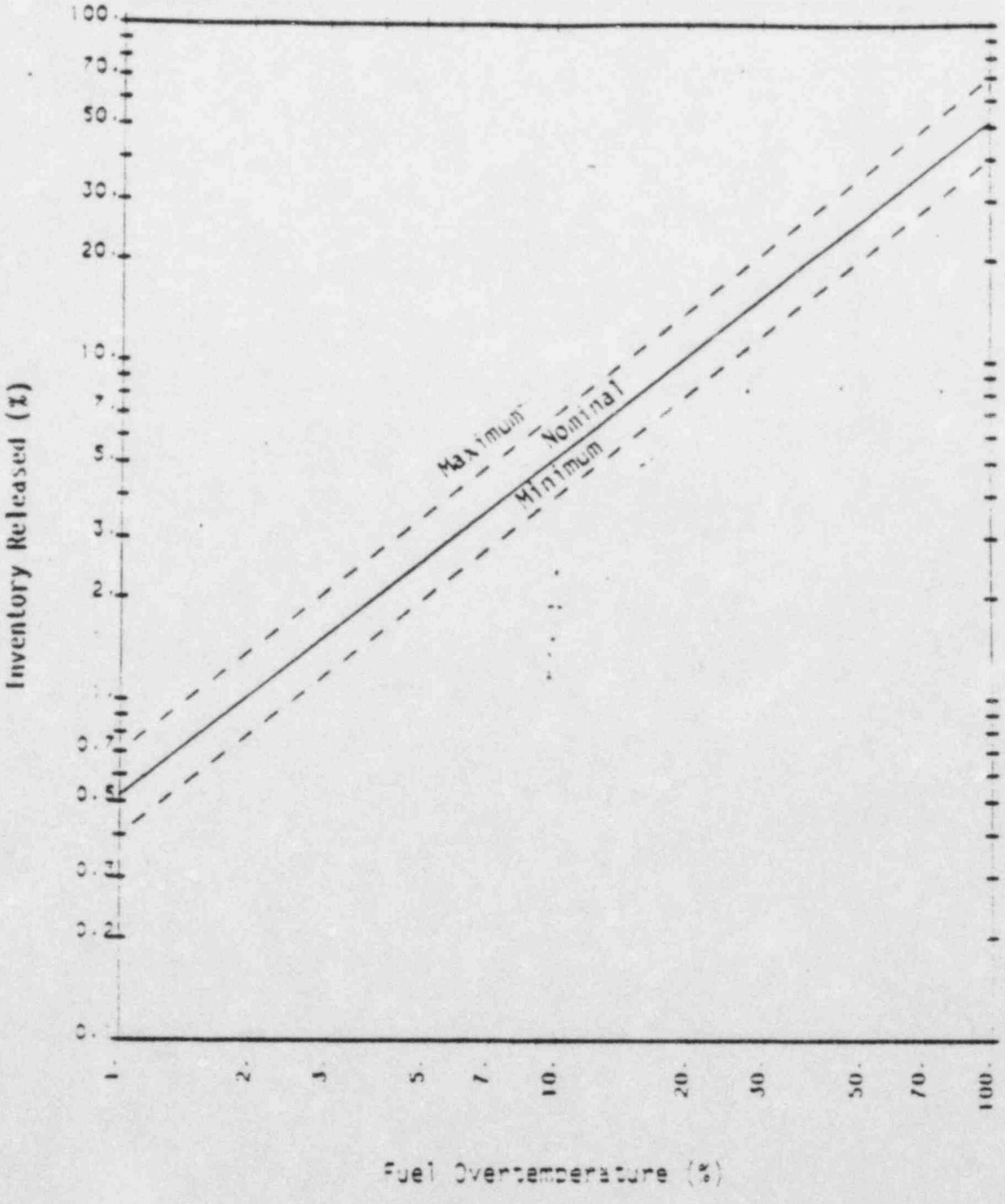
ATTACHMENT 19 Page 1 of 1



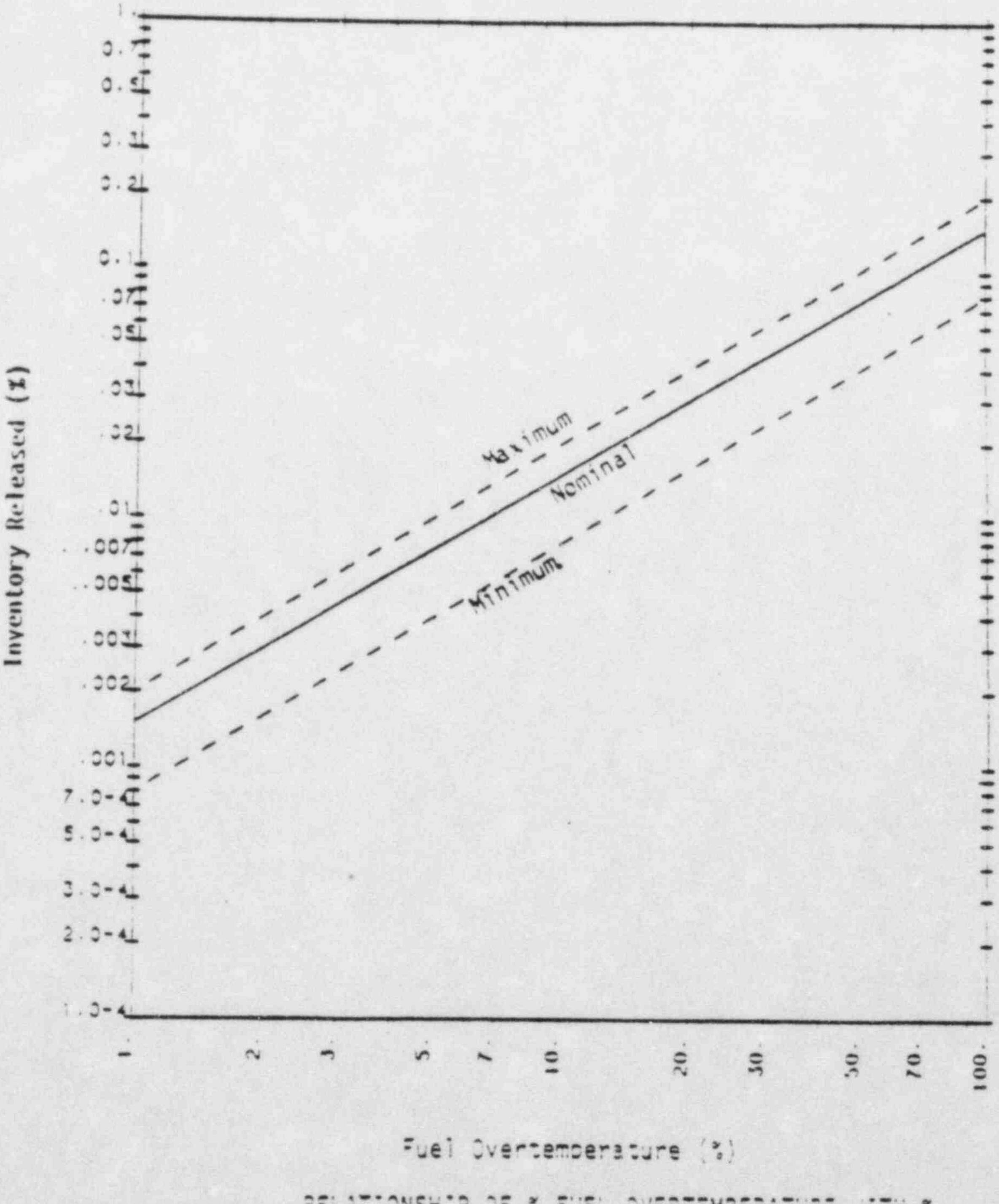
RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-133



RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-135



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



RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH %
INVENTORY RELEASED OF BA, SR

Enclosure 2

The ranges and manufacturer's accuracies for the SNUPPS Post-Accident Sampling System (PASS) are provided below.

LIQUID SAMPLES

ANALYSIS	RANGE	ACCURACY
Gross Radioactivity, Gamma Spectrum	10^{-3} μCi/cc to 10Ci/cc, Isotopic Analysis	Better than a factor of two
Boron Content	0-6500 ppm	±2% at 6500 ppm ±5% at 500 ppm ±40% at 50 ppm
Chloride Content	0.1 to 20 ppm	±10% to 1 ppm, ±0.15 below 1 ppm
Conductivity	0.1 to 1000 μmhos	±1% of full scale
pH	0-14	±0.1
Dissolved Oxygen	0-20 ppm	±1% of full scale
Dissolved Hydrogen	0-3000 cc/kg	±5% of measured value

GASEOUS SAMPLES

ANALYSIS	RANGE	ACCURACY
Gross Radioactivity Gamma Spectrum	10^{-7} μCi/cc to 10^{5} μCi/cc Isotopic Analysis	Better than a factor of two
Hydrogen	0-10 Volume Percent	±2.5% full scale
Oxygen	0-30 wt %	±1% full scale

The SNUPPS PASS was tested during the factory acceptance tests with two chemical matrixes identified below.

PASS TEST MATRIXES

Analysis	Matrix #1	Matrix #2
Boron	1200 ppm	600 ppm
Chloride	10 ppm	5 ppm
pH	4.9	4.4
Conductivity	33 umhos	17 umhos
DO_2	8 ppm	8 ppm
DH_2	5 ppm	5 ppm

The results of the factory acceptance tests meet the accuracy requirements of NUREG 0737, Item II.B.3, with the exception of the boronmeter. At the time of the factory acceptance testing, the boronmeter was not fully calibrated and results outside the accuracy criterion resulted. After final calibration, a test matrix was run in the field and the results of the boron analysis were within the accuracy criterion. Note, the boronmeter employed in the SNUPPS PASS is a fission counter (neutron absorption) which will not be affected by chemical interferences.

In addition to the above matrix tests, the Orion Chloride Analyzer was extensively tested by Orion with the standard NRC test matrix with results meeting the accuracies listed in the above table. The remaining inline analyzers are off the shelf analyzers widely used in the chemical and power industries. Their wide acceptance coupled with successful completion of the above tests assures their ability to perform their particular analysis.

The design of the PASS is such that radiation sensitive components are removed from the sample panel where feasible and located in a lower radiation environment. Sample lines and holdup volumes were minimized to reduce the activity in the panel. Detailed dose calculations were performed for the sample panel using the high activity sources postulated post accident. As a result of these calculations worst case integrated doses were calculated for the electrical components in the sample panel. In all cases, the total dose to any electrical component was below its damage threshold. All wetted materials in the system were also reviewed in light of the high activity and care was taken to select only those materials suitable for this service.

As a result of the above mentioned testing and analysis the SNUPPS PASS has demonstrated its ability to adequately provide an accurate analysis of liquid and gaseous samples post accident.

The SNUPPS PASS was designed for use during both normal plant operation and following postulated accidents. The procedures for sample analysis in a post-accident environment are identical to those used to obtain a PASS sample during normal operation with the exception of the need for additional emphasis on health physics requirements resulting from increased post-accident radiation levels. As discussed above, the PASS instrument accuracies will be maintained within required limits in the post-accident environment. Procedures for sampling in the post-accident environment, including obtaining grab samples, are practiced in conjunction with periodic emergency planning drills. The PASS sampling procedures were evaluated and found acceptable during the March 21, 1984 emergency planning drill at Callaway Plant. Operability of the PASS is assured by performance of system operational checks and system functional checks on a periodic basis. The operational checks will verify the ability to analyze routine samples. The functional checks will verify the ability to analyze known sample concentrations.