D. C. COOK

POST ACCIDENT

CORE DAMAGE ASSESSMENT METHODOLOGY

August, 1984

NOTICE

The D. C. Cook Post Accident Core Damage Assessment Methodology Report consists of using the Westinghouse Owner's Group Revision 1 generic report and modifying it to include relevant D. C. Cook plant specific parameters. Where a change in the text of the generic report has been made to incorporate plant specific information, brackets, [], have been used to indicate the change.

In the generic report the last section consisted of a step-by-step example on the use of the core damage assessment methodology. In this report the example section is replaced with a procedure specific to D. C. Cook. Also included is an example of this procedure.

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1.0 INTRODUCTION AND PURPOSE

In March 1982 the NRC issued a "Post Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage" as a supplement to the post accident sampling criteria, of NUREG-0737[1]. The stated purpose of this guide was to aid utilities in preparation of a methodology for relating post accident core damage with measurements of radionuclide concentrations and other plant indicators. The primary interest of the NRC was, in the event of an accident, to have some means of realistically differentiating between four major fuel conditions: no damage, cladding failure, fuel overheating, and core melt. The methodology developed is intended to enable qualified personnel to provide an estimate of this damage. In order to comply with the NRC request for such a methodology, Westinghouse, under contract to the Westinghouse Owners Group (WOG), prepared the generic technical report[13].

This report is cognizant of NRC's initial intention. Additionally, the report reflects input by NRC and various representatives of the WOG provided during several meetings held on this subject during the past year.

[This report has been arranged to present the technical basis for the methodology (Section 1 through 5), and to provide a procedure based on this methodology (Appendix A).]

1.1 METHODOLOGY

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, obtained with the post accident sampling system. Greater release of fission products into the primary coolant can occur if insufficient cooling is supplied to the fuel elements. Those fission products contained in the fuel pellet - fuel cladding interstices are presumed to be completely released upon failure of cladding. Additional fission products from the fuel pellet are assumed to be released during overtemperature and fuel melt conditions. These radionuclide measurements,

together with auxiliary readings of core exit thermocouple temperatures, water level within the pressure vessel, containment radiation monitors, and hydrogen production are used to develop an estimate of the kind and extent of fuel damage.

2.0 TECHNICAL BASIS FOR CORE DAMAGE ASSESSMENT METHODOLOGY

2.1 CHARACTERISTIC FISSION PRODUCTS

Depending on the extent of core damage, characteristic fission products are expected to be released from the core. An evaluation was conducted to select the fission product isotopes which characterize a mechanism of release relative to the extent of core damage. Nuclides were selected to be associated with the core damage states of clad damage, fuel overheat, and fuel melt. The selection of nuclides for this methodology was based on half-life, energy, yield, release characteristics, quantity present in the core, and practicality of measurement using standard gamma spectrometry techniques.

The nuclides selected for this methodology have sufficient core inventories and radioactive half-lives to ensure that there will be sufficient activity for detection and analysis of the nuclides for some time following an accident. Most of the nuclides selected have half-lives which enable them to reach equilibrium quickly within the fuel cycle. The list of selected nuclides contains nuclides with half-lives of 1 day or less which are assumed to reach equilibrium in approximately 4 days. These nuclides are used to assess core damage for cores that have been operational in a given cycle for less than a month. For cores that have been operating for more than a month, the list contains nuclides with half-lives greater than 1 day which reach equilibrium at some time during the first month of operation depending on the half life of the nuclide. Both groups of nuclides are used to assess core damage for cores that have been operational in a given cycle for more than a month. Other factors considered during the selection process were the energy and yield of the nuclides along with the practicality of detecting and analyzing the nuclides.

Nuclides were chosen based on their release characteristics to be representative of the specific states of core damage. The Rogovin Report[2] noted that as the core progressed through the damage states certain nuclides associated with each damage state would be released. The volatility of the nuclides is the basis for the relationship between certain nuclides and a particular core damage state.

A list of the selected nuclides for this core damage assessment methodology is shown in Table 2-1.

2.2 CORE INVENTORIES

Implementation of the core damage assessment methodology requires an estimation of the fission product source inventory available for release. The fission product source inventory of the fuel pellet was calculated using the ORIGEN[3] computer code, based on a three-region equilibrium cycle core at end-of-life. The three regions were assumed to have operated for 300, 600, and 900 effective full power days, respectively. For use in this methodology the fission product inventory is assumed to be evenly distributed throughout the core. As such, the fission product inventory can be applicable to other equilibrium cores with different regional characteristics. The fuel pellet inventory of the selected fission products and some additional fission products of interest [for D. C. Cook Unit 1 and Unit 2 is shown in Table 2-2.]

2.3 POWER CORRECTION FOR CORE INVENTORIES

The source inventory shown in Table 2-2 presents inventories for an equilibrium, end-of-life core that has been operated at 100 percent power. For this methodology a source inventory at the time of an accident that accounts for the power history is needed. For those cases where the core has reached equilibrium, a ratio of the steady state power level to the rated power level is applied. Within the accuracy of this methodology, a period of four half-lives of a nuclide is sufficient to assume equilibrium for that nuclide. For nuclides with half-lives less than one day the power ratio based on the steady-state power level of the prior four days to reactor shutdown can be used to determine the inventory. To use a simple power ratio to determine the inventories of the isotopes with half-lives greater than 1 day, the core should have operated at a constant power for at least 30 days prior to reactor shutdown. The assumption is made that constant power exists when the power level does not vary more than ±10 percent of the rated power level from the time averaged value. For transient power histories where a steady state power condition has not been obtained, a power correction factor has been developed to calculate the source inventory at the time of the accident.

TABLE 2-1

SELECTED NUCLIDES FOR CORE DAMAGE ASSESSMENT

Core Damage State	Nuclide	Half-Life*	Predominant Gammas (Kev) Yield (%)*
Clad Failure	Kr-85m**	4.4 h	150(74), 305(13)
	Kr-87	76 m	403(84), 2570(35)
	Kr-88**	2.8 h	191(35), 850(23), 2400(35)
	Xe-131m	11.8 d	164(2)
	Xe-133	5.27 d	81(37)
	Xe-133m**	2.26 d	233 (14)
	Xe-135**	9.14 h	250(91)
	I-131	8.05 d	364(82)
	I-132	2.26 h	773(89), 955(22), 1400(14)
	I-133	20.3 h	530(90)
	I-135	6.68 h	1140(37), 1280(34), 1460(12), 1720(19)
	Rb-88	17.8 m	898(13), 1863(21)
Fuel Overheat	Cs-134	2 yr	605(98), 796(99)
	Cs-137	30 yr	662(85)
	Te-129	68.7 m	455(15)
	Te-132	77.7 h	230(90)
Fuel Melt	Sr-89	52.7 d	(beta emitter)
	Sr-90**	28 yr	(beta emitter)
	Ba-140	12.8 d	537(34)
	La-140	40.22 h	487(40), 815(19), 1596(96)
	La-142	92.5 m	650(48), 1910(9), 2410(15), 2550(11)
	Pr-144	17.27 m	695(1.5)

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* Values obtained from Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition.

** These nuclides are marginal with respect to selection criteria for candidate nuclides; they have been included on the possibility that they may be detected and thus utilized in a manner analogous to the candidate nuclides.

TABLE 2-2

FUEL PELLET INVENTORY*

Inventory, Curies

Nuclide	Unit 1 (3250 Mwt)	Unit 2 (3391 Mwt)
Kr 85m	2.0(7)**	2.1(7)
Kr 87	3.6(7)	3.8(7)
Kr 88	5.2(7)	5.4(7)
Xe 131m	5.7(5)	6.0(5)
Xe 133	1.8(8)	1.9(8)
Xe 133m	2.5(7)	2.7(7)
Xe 135	3.4(7)	3.5(7)
I 131	8.9(7)	9.3(7)
I 132	1.3(8)	1.3(8)
I 133	1.8(8)	1.9(8)
I 135	1.6(8)	1.7(8)
Rb 88	5.3(7)	5.5(7)
Cs 134	2.1(7)	2.2(7)
Cs 137	1.0(7)	1.0(7)
Te 129	3.0(7)	3.1(7)
Te 132	1.3(8)	1.3(8)
Sr 89	7.2(7)	7.5(7)
Sr 90	6.6(6)	6.8(6)
Ba 140	1.5(8)	1.6(8)
La 140	1.6(8)	1.7(8)
La 142	1.4(8)	1.4(8)
Pr 144	1.1(8)	1.1(8)

* Inventory based on ORIGEN run for equilibrium, end-of-life core.
** 1.2(7) = 1.2 x 10^7. This notation is used throughout this report.

There are a few selected nuclides with half-lives around one year or longer which in most instances do not reach equilibrium during the life of the core. For these few nuclides and within the accuracy of the methodology, a power correction factor which compares the effective full power days of the core to the total number of calendar days of cycle operation of the core is applied.

Due to the production characteristics of cesium-134, special consideration must be used to determine the power correction factor for Cs-134. This power correction factor can be obtained from Figure 2-1.

2.3.1 POWER CORRECTION FACTOR

A) Steady state power prior to shutdown.

1) Half-life of nuclide < 1 day

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

2) Half-life of nuclide > 1 day

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

3) Half life of nuclide ≈ 1 year

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

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

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

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

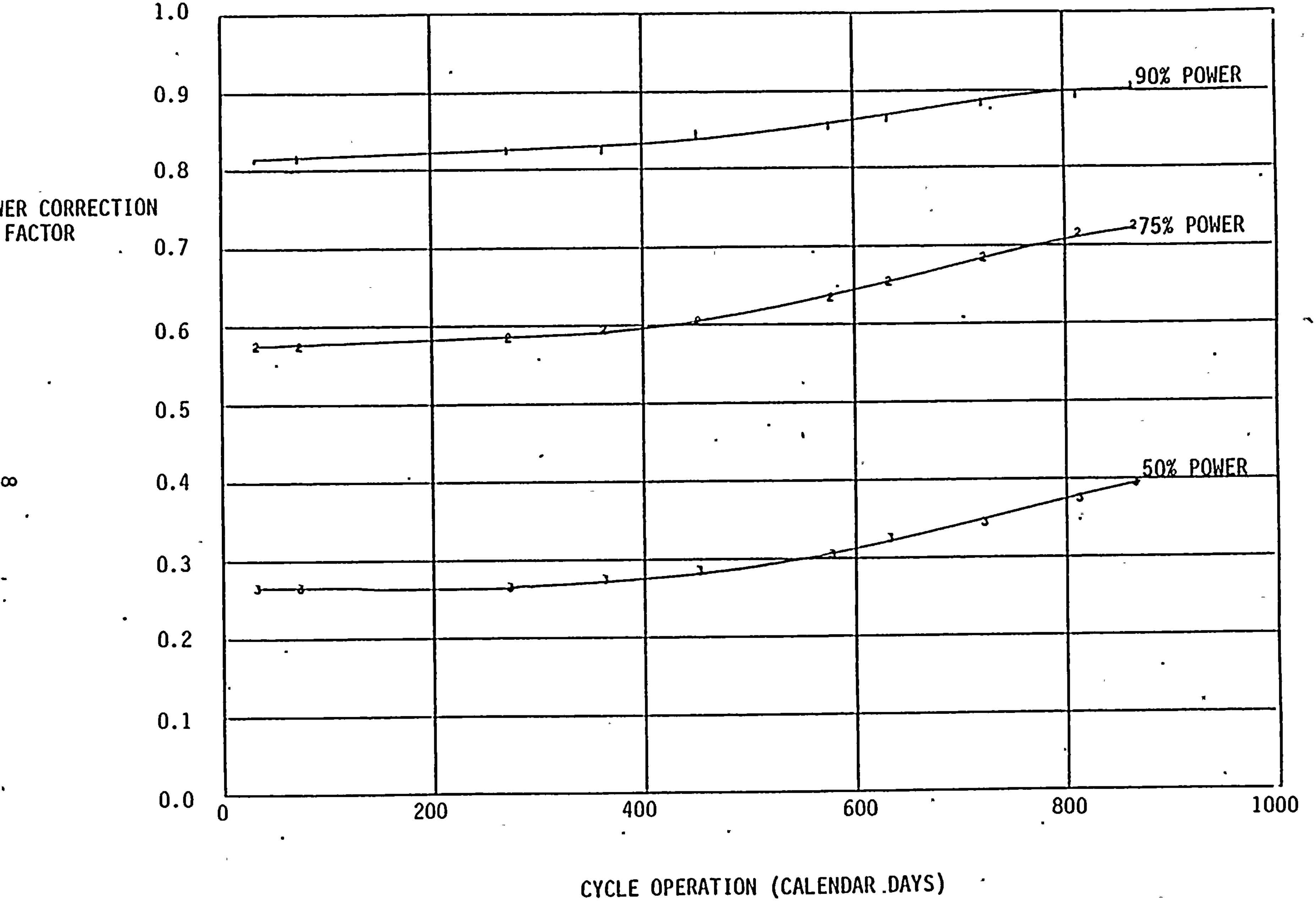

FIGURE 2-1 POWER CORRECTION FACTOR FOR CS-134 BASED ON AVERAGE POWER DURING OPERATION

$$\text{Power Correction Factor} = \frac{\Sigma_j\ P_j\ (1-e^{-\lambda_i t_j})\ e^{-\lambda_i t^\circ_j}}{RP\ (1-e^{-\lambda_i\ \Sigma t_j})}$$

where:

P_j	=	average power level (Mwt) during operating period t_j
RP	=	rate 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	=	time between end of period j and time of reactor shutdown in days.

If the total period of operation is greater than four half-lives of the nuclide being considered, the power correction is as follows. This is within the accuracy of this methodology.

$$\Sigma_j t_j \geq 4 \times \frac{0.693}{\lambda_i}$$

$$\text{Power Correction Factor} = \frac{\Sigma_j\ P_j\ (1-e^{-\lambda_i t_j})\ e^{-\lambda_i t^\circ_j}}{RP}$$

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.

$$\text{Power Correction Factor} = \frac{\text{EFPD}}{\text{total calendar days of cycle operation}}$$

C) For Cs-134 Figure 2-1 is used to determine the power correction factor. To use Figure 2-1, the average power during the entire operating period is required.

2.4 RELATIONSHIP OF CLAD DAMAGE WITH ACTIVITY

2.4.1 GAP INVENTORY

During operation, volatile fission products collect in the gap. These fission products are isotopes of the noble gases and iodine.

To determine the fission product inventory of the gap, the ANS 5.4[4] Standard formulae were used with the average temperature and burnup of the fuel rod. The average gap inventory for the entire core for this methodology was estimated by assuming the core is divided into three regions - a low burnup region, a middle burnup region, and a high burnup region. Using the ANS 5.4 Standard, the gap fraction and subsequent gap inventory were calculated for each region. Each region is assumed to represent one-third of the core. The total gap inventory was then calculated by summing the gap inventory of each region. For the purposes of this core damage assessment methodology, this gap inventory is assumed to be evenly distributed throughout the core. [Table 2-3 shows the calculated gap inventories for Unit 1 and Unit 2 of the noble gases and iodines. Table 2-3-1 shows the minimum and maximum gap inventories.] The minimum and maximum gap inventory were determined by assuming the entire core was operating at the low burnup condition and the high burnup conditions, respectively.

2.4.2 SPIKING PHENOMENA

Reactor coolant system pressure, temperature, and power transients may result in iodine spiking. (Cesium spiking may also occur but is not considered in this methodology.) Spiking is noted by an increase in reactor coolant iodine concentrations during some time period after the transient. In most cases, the iodine concentration would return to normal operating activity at a rate based on the system purification half-life. 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.

TABLE 2-3

GAP INVENTORY*

Gap Inventory, Curies

Nuclide	Unit 1 (3250 Mwt)	Unit 2 (3391 Mwt)
Kr 85m**	3.44(3)	3.59(3)
Kr 87	3.29(3)	3.43(3)
Kr 88**	7.26(3)	7.58(3)
Xe 131m	8.05(2)	8.41(2)
Xe 133	1.60(5)	1.67(5)
Xe 133m**	1.53(4)	1.60(4)
Xe 135**	8.17(3)	8.53(3)
I-131	2.58(5)	2.70(5)
I-132	4.15(4)	4.33(4)
I-133	1.75(5)	1.82(5)
I-135	8.92(4)	9.31(4)

* Total core inventory based on 3 region equilibrium core at end-of-life. Gap inventory based on ANS 5.4 Standard.

** Additional nuclides; no graphs provided.

TABLE 2-3-1

GAP INVENTORY MINIMUM AND MAXIMUM

Nuclide	Gap Inventory, Curies (Minimum - Maximum)** Unit 1 (3250 Mwt)	Unit 2 (3391 Mwt)
Kr 85m*	6.28(2)-8.71(3)	6.56(2)-9.09(3)
Kr 87	6.20(2)-8.39(3)	6.47(2)-8.76(3)
Kr 88*	1.29(3)-1.81(4)	1.35(3)-1.89(4)
Xe 131m	1.44(2)-2.01(3)	1.50(2)-2.10(3)
Xe 133	3.03(4)-4.10(5)	3.16(4)-4.28(5)
Xe 133m*	1.16(3)-1.61(4)	1.22(3)-1.68(4)
Xe 135*	3.74(3)-5.11(4)	3.90(3)-5.33(4)
I 131	4.90(4)-6.69(5)	5.12(4)-6.98(5)
I 132	7.78(3)-1.06(5)	8.12(3)-1.11(5)
I 133	3.21(4)-4.46(5)	3.35(4)-4.66(5)
I 135	1.62(4)-2.27(5)	1.69(4)-2.37(5)

* Additional nuclides; no graphs provided.

** Minimum values are based on the low burnup region (5,000 MWD/MTU).
Maximum values are based on the high burnup region (25,000 MWD/MTU).

For this methodology consideration of the spiking phenomena into the radionuclide analysis is limited to the I-131 information found in WCAP-9964[5]. WCAP 9964 presents releases in Curies of I-131 due to a transient which results in spiking based on the normal primary coolant activity of the nuclides. The WCAP gives an average release and 90 percent confidence interval. These values are presented in Table 2-4. The use of this data is demonstrated in Section 2.4.3.2.

2.4.3 ACTIVITY ASSOCIATED WITH CLAD DAMAGE

Clad damage is characterized by the release of the fission products which have accumulated in the gap during the operation of the plant. The cladding may rupture during an accident when heat transfer from the cladding to the primary coolant has been hindered and the cladding temperature increases. Cladding failure is anticipated in the temperature range of 1300 to 2000°F depending upon the conditions of the fission product gas and the primary system pressure. Clad damage can begin to occur in regions of high fuel rod peak clad temperature based on the radial and axial power distribution. As the accident progresses and is not mitigated, other regions of the core are expected to experience high temperatures and possibly clad failure. 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 the primary system water during the accident.

To determine an approximation of the extent of clad damage, the total activity of a fission product released is compared to the total source inventory of the fission product at reactor shutdown. Included in the measured quantity of the total activity released is a contribution from the normal operating activity of the nuclide. An adjustment should be made to the measured quantity of release to account for the normal operating activity. Direct correlations can then be developed which describe the relationship between the percentage of total source inventory released and the extent of clad damage for each nuclide. Figures 2-2 through 2-9 present the direct correlations for each nuclide in graphical form. The contribution of the normal operating activity

TABLE 2-4

EXPECTED IODINE SPIKE

Average, μCi/gm	I-131 Total Release, Curies
0.5 < SA* < 1.0	3400
0.1 < SA < 0.5	380
0.05 < SA < 0.1	200
0.01 < SA < 0.05	200
0.005 < SA < 0.01	100
0.001 < SA < 0.005	100
SA < 0.001	2
90/90 Upper Confidence Level, μCi/gm	
0.5 < SA < 1.0	6500
0.1 < SA < 0.5	950
0.05 < SA < 0.1	650
0.01 < SA < 0.05	650
0.005 < SA < 0.001	300
0.001 < SA < 0.005	300
SA < 0.001	10

* SA is the normal operating I-131 specific activity (μCi/gm) in the primary coolant.

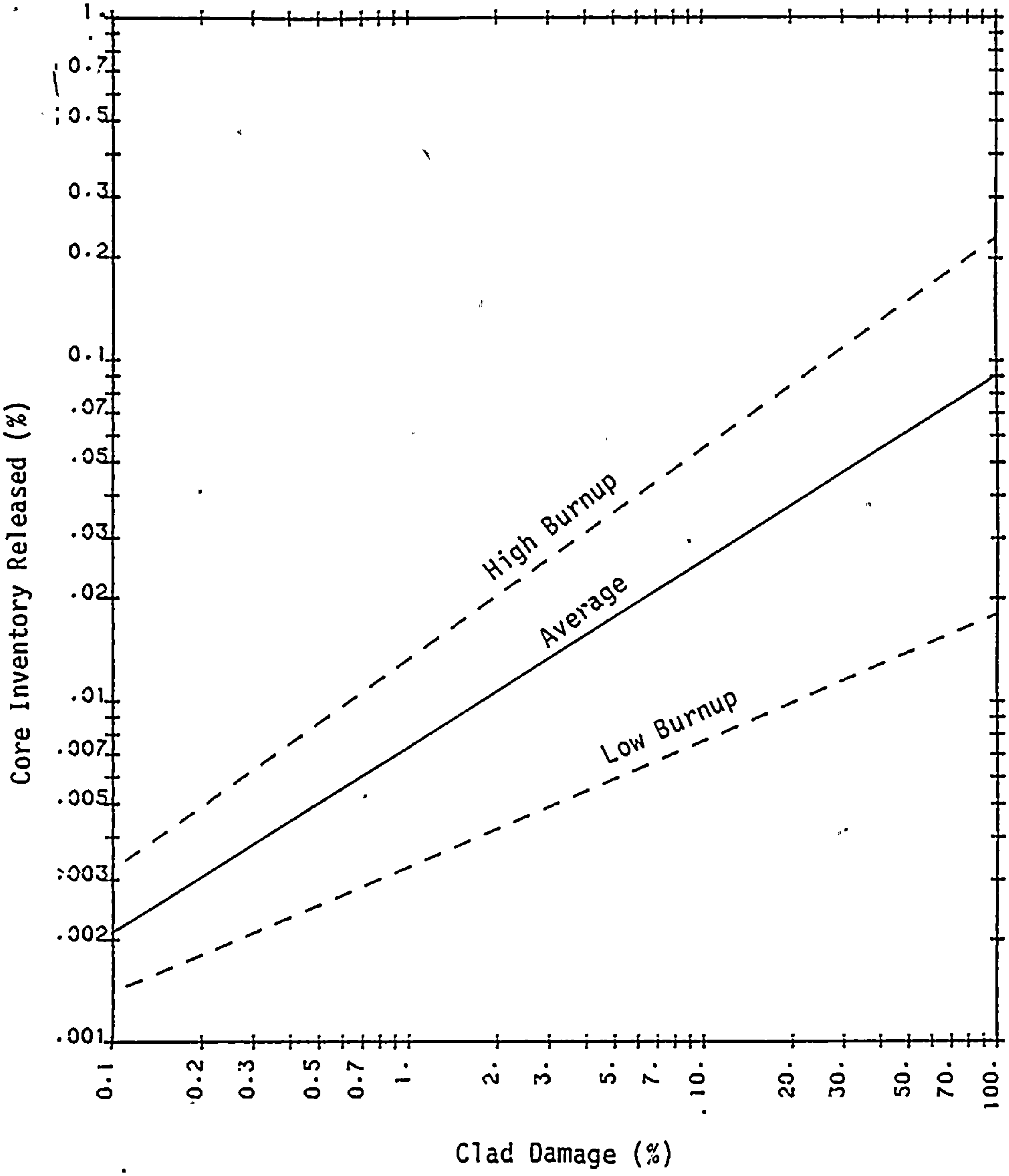

FIGURE 2-2 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF XE-133 .

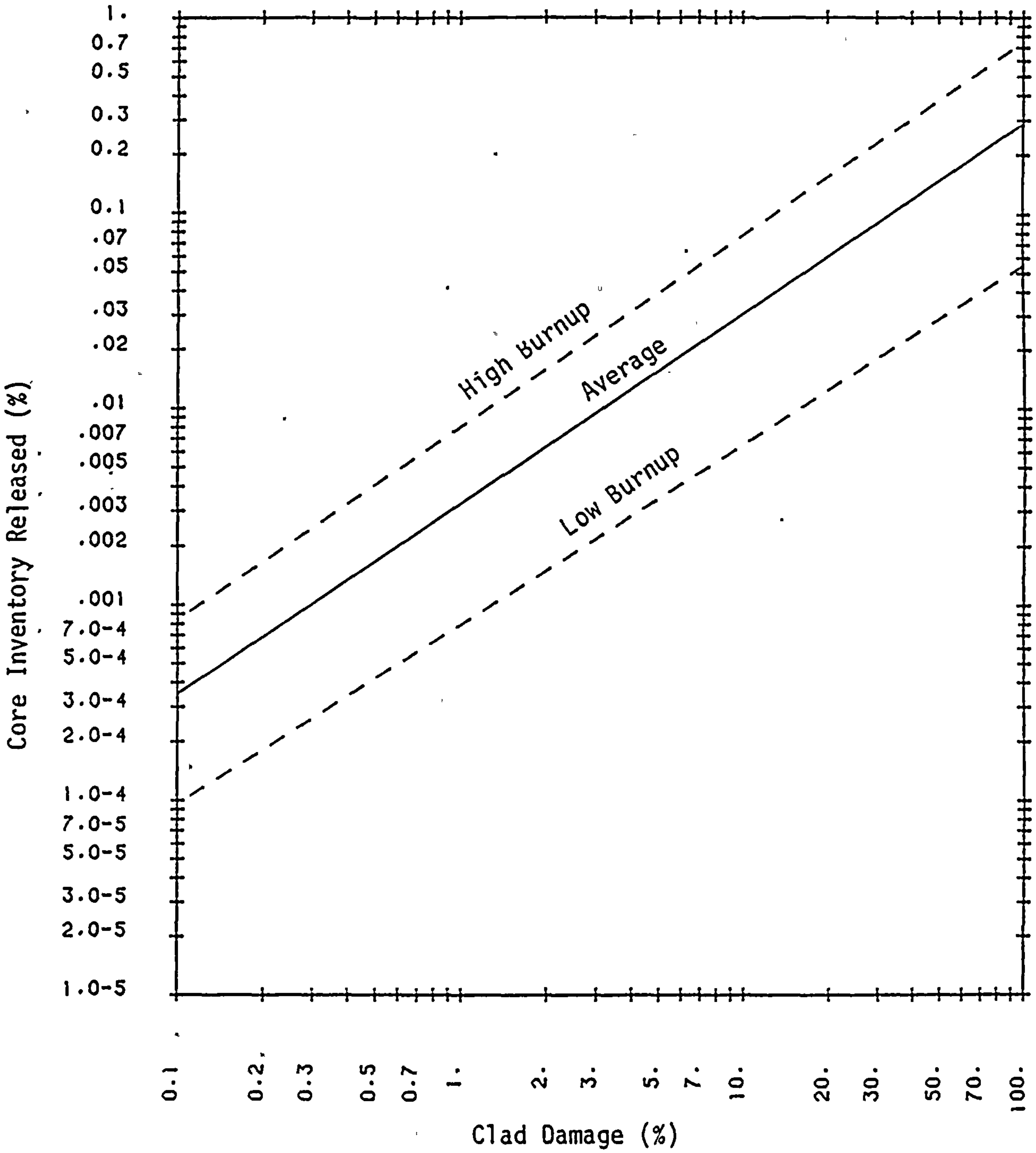

FIGURE 2-3 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-131

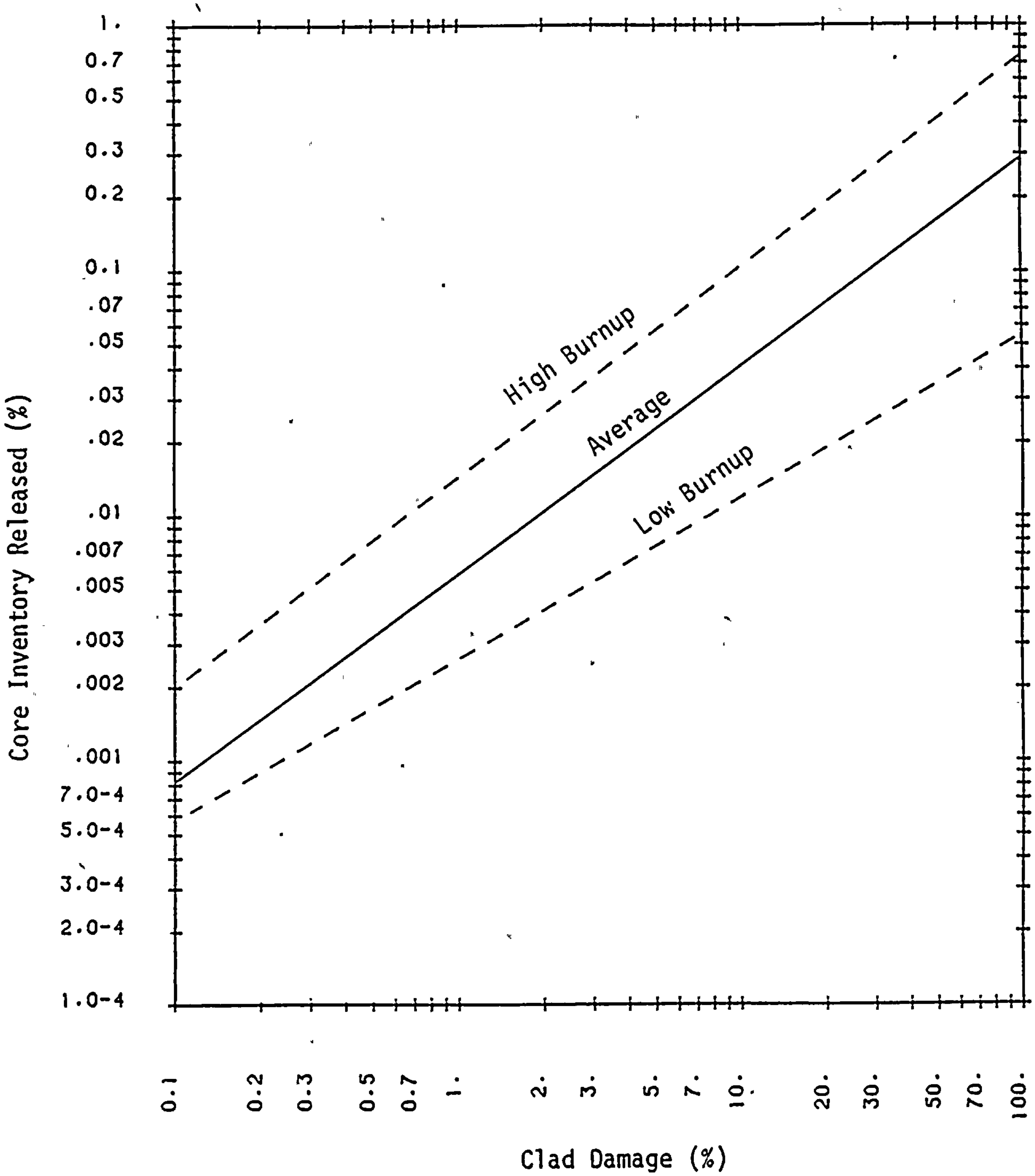

FIGURE 2-4 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-131 WITH SPIKING

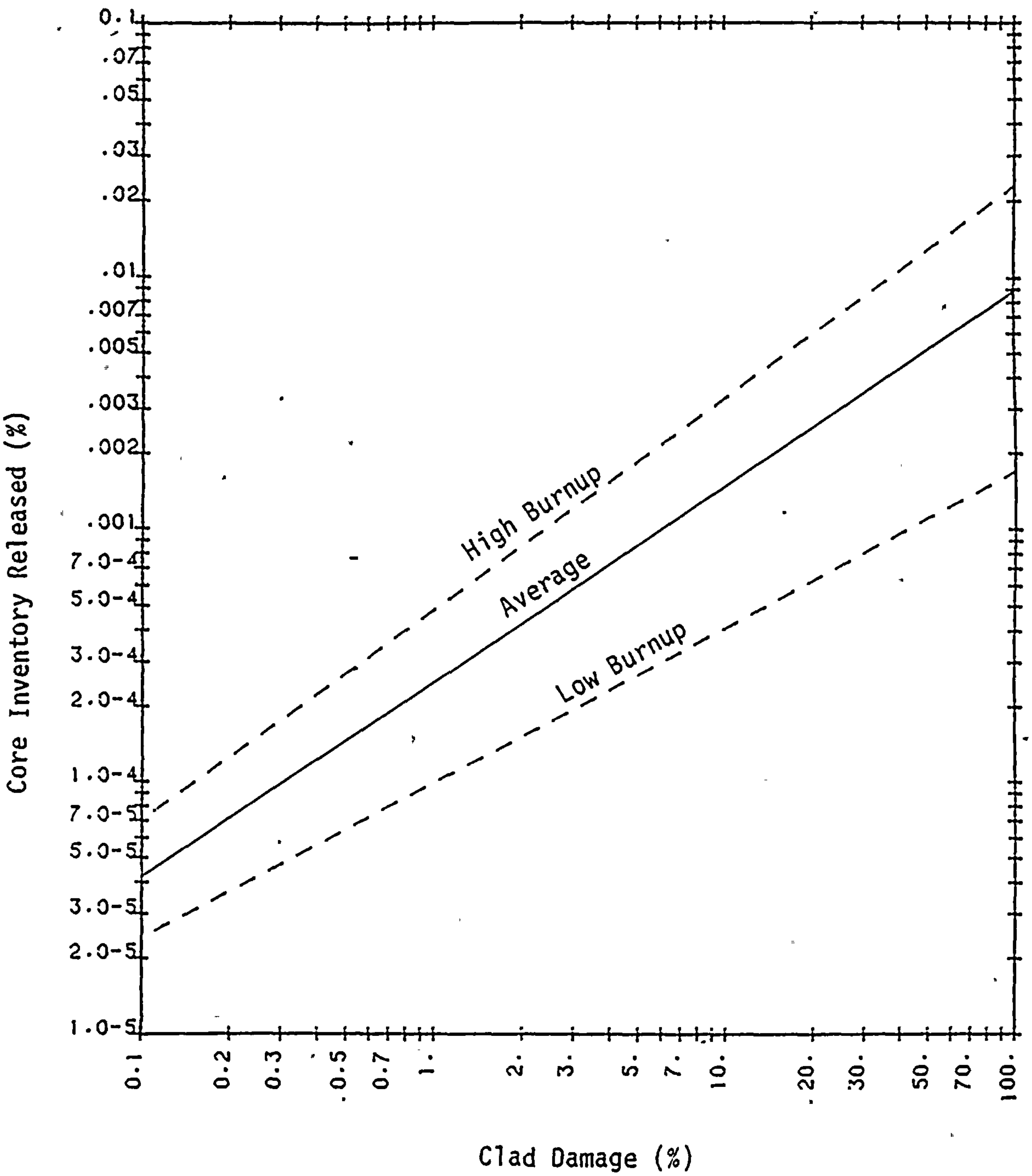

FIGURE 2-5 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF KR-87

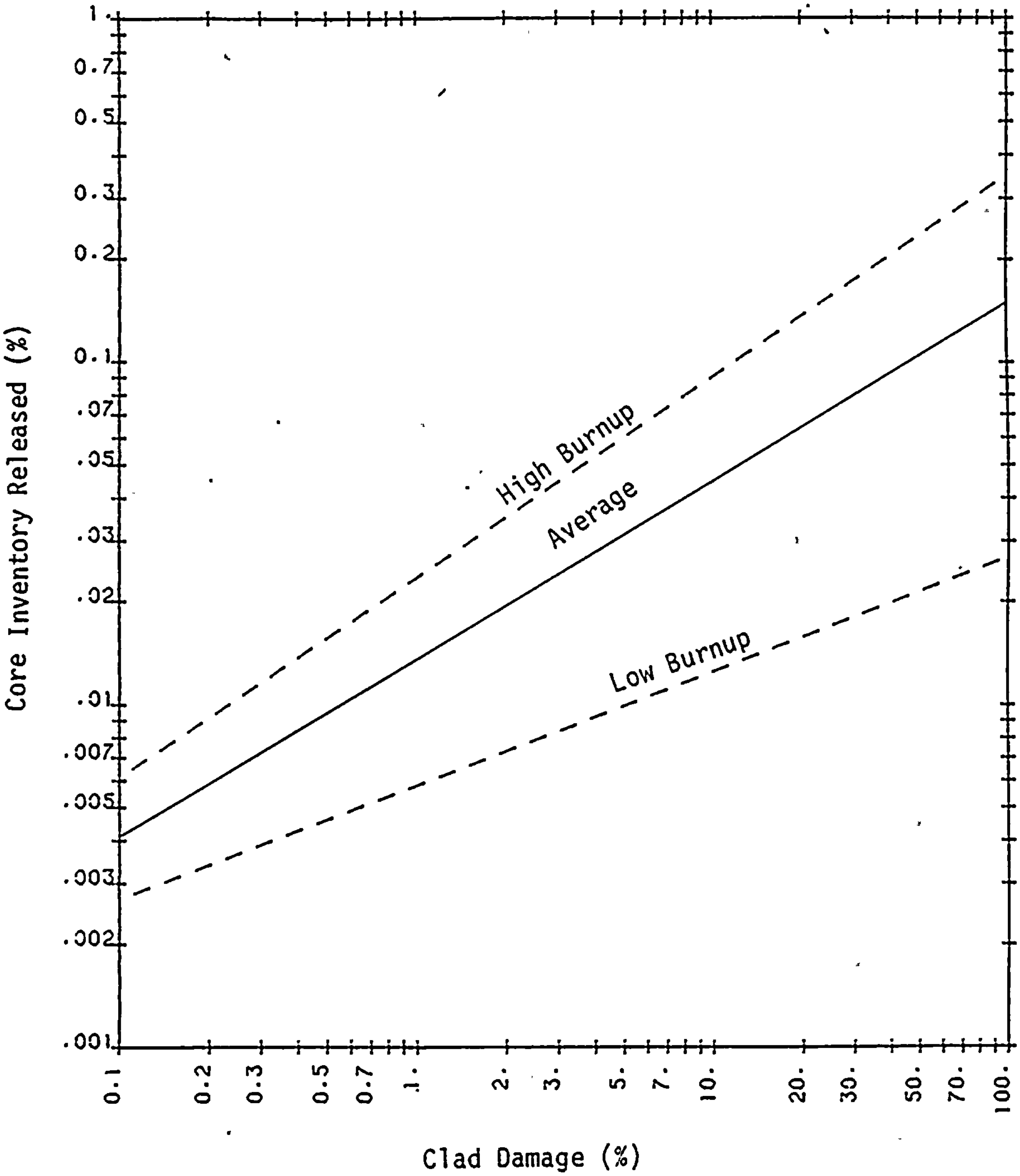

FIGURE 2-6 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF XE-131M

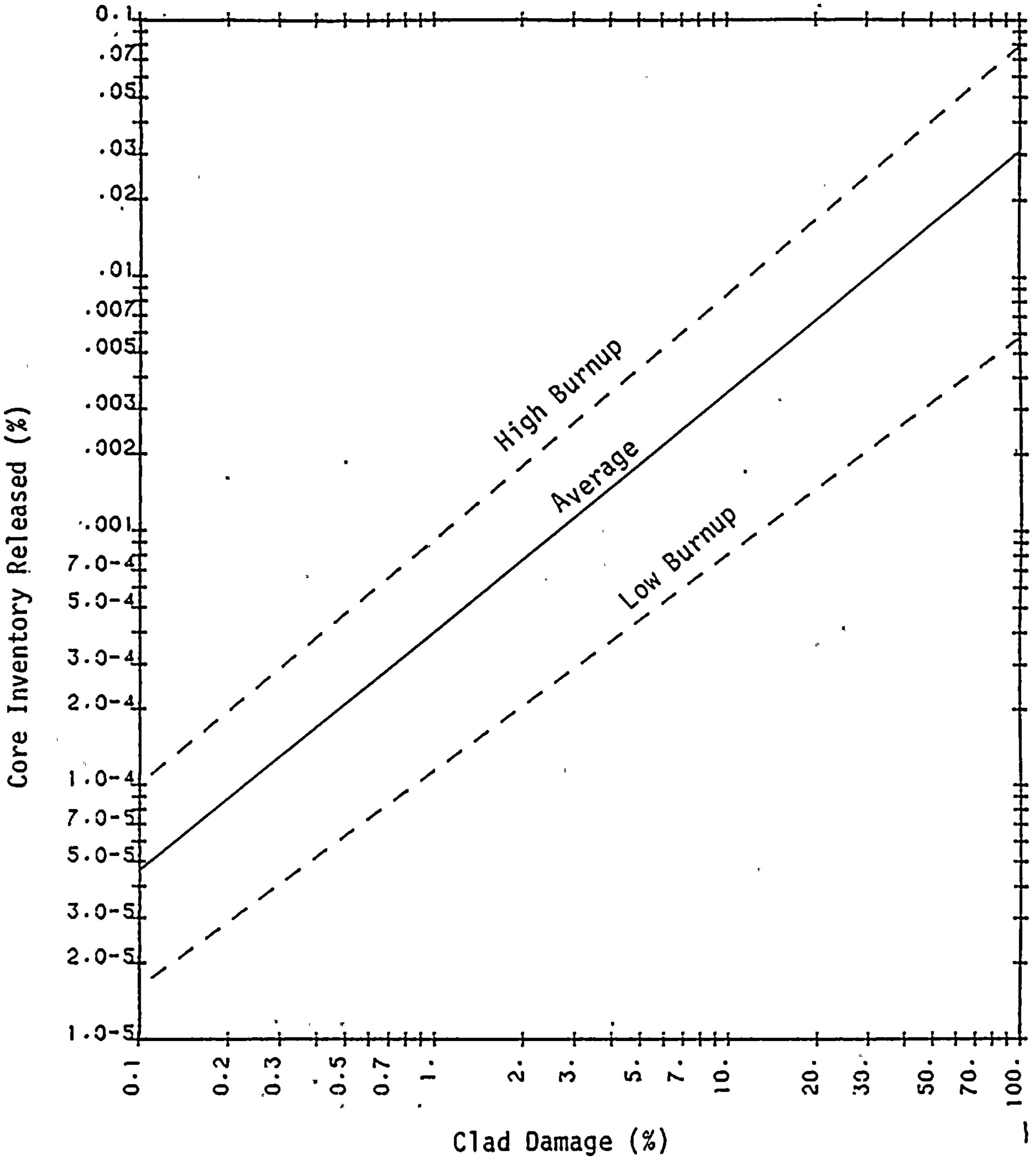

FIGURE 2-7 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-132

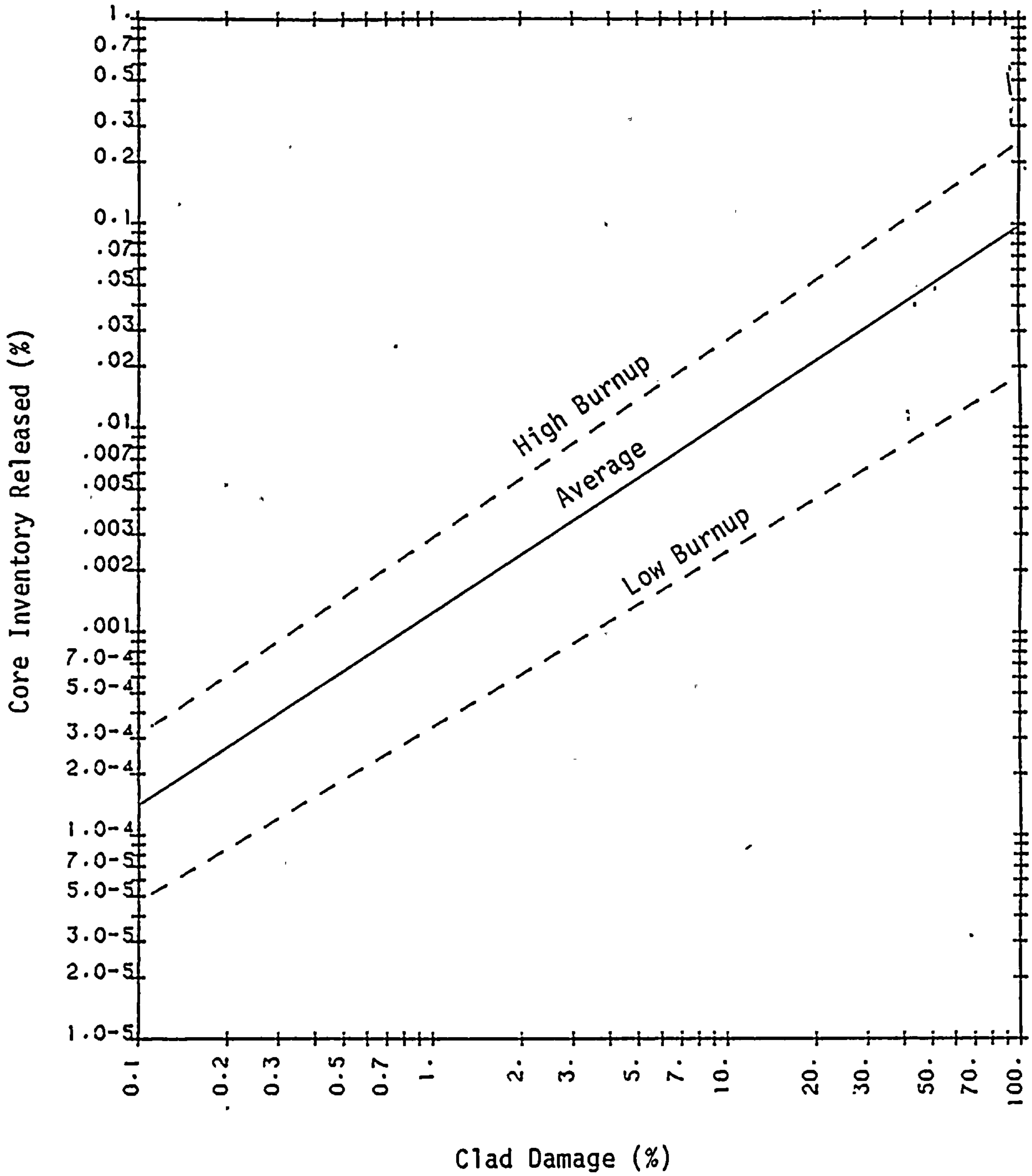

FIGURE 2-8 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-133

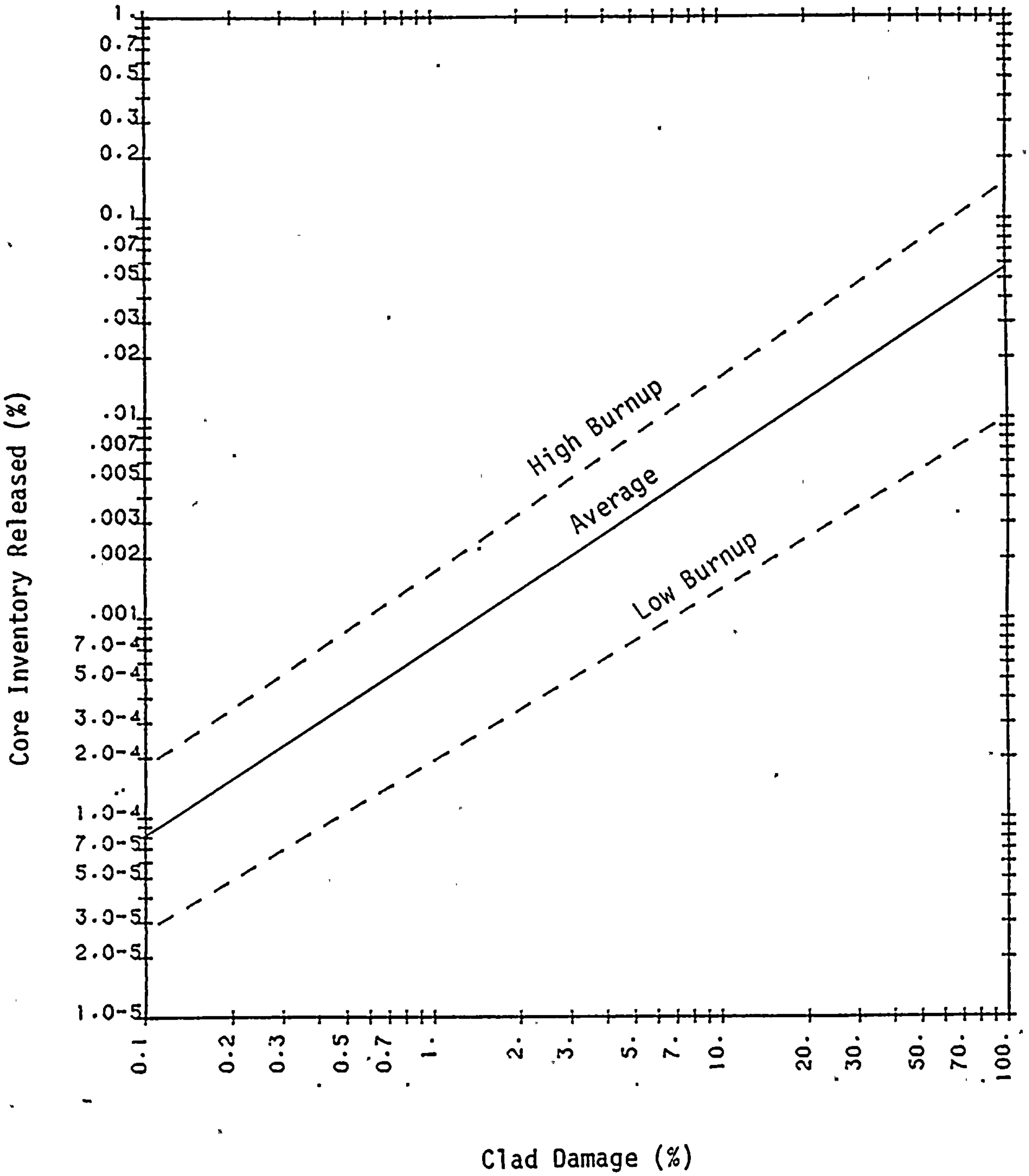

FIGURE 2-9 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-135

has been factored into the correlations shown in Figures 2-2 through 2-9. Examples of how to construct the correlations shown in Figures 2-2 through 2-4 are presented in the next two sections. Figures 2-5 through 2-9 were determined in the same fashion as described in the examples. It should be noted that not all of the fission products listed in Table 2-3 need to be analyzed but as many as possible should be analyzed to determine a reasonable approximation of clad damage.

2.4.3.1 Xe-133

A graphical representation can be developed which describes the linear relationship of the measured release percentage of Xe-133 to the extent of clad damage. Since the linear relationship is based on percentage of inventory released, the linear relationship applies to all Westinghouse standard plants. The Westinghouse 3-Loop plant is used as the base plant for developing the relation. The total source inventory of Xe-133 for a Westinghouse 3-Loop plant is 1.6×10^8 Curies[(13)]. For 100 percent clad damage, all of the gap inventory, which corresponds to 1.43×10^5 Curies[(13)] would be released. For 0.1 percent clad damage, 1.43×10^2 Curies would be released. These two values can be used to represent two points of the linear relationship between percentage of total inventory released and the extent of clad damage. However, the normal operating activity needs to be accounted into the relation. From Table 2-5 the normal operating activity of Xe-133 is 18 µCi/gm[(6)]. The average primary coolant mass of a 3-Loop plant is 1.78×10^8 grams. The total normal operating contribution to the total release of Xe-133 is 3200 Curies. Thus the adjusted releases are 3340 Curies and 1.46×10^5 Curies for 0.1 percent clad damage and 100 percent clad damage, respectively. This corresponds to 2.2×10^{-3} percent for 0.1 percent clad damage and 9.1×10^{-2} for 100 percent clad damage. This relation is shown in Figure 2-2.

Figure 2-2 also shows a minimum and a maximum relation which bound the best estimate line. The minimum and maximum lines were determined by bounding the fission product gap inventory. The minimum gap inventory was determined by assuming the entire core was operating at the low burnup condition used to calculate the average gap inventory as described in Section 2.4.1. The

TABLE 2-5

NORMAL OPERATING ACTIVITY*

Nuclide	Specific Activity in Reactor Coolant (μCi/gm)
Kr 85m	1.1 (-1)
Kr 87	6.0 (-2)
Kr 88	2.0 (-1)
Xe 131m	1.1 (-1)
Xe 133	1.8 (+1)
Xe 133m	2.2 (-1)
Xe 135	3.5 (-1)
I 131	2.7 (-1)
I 132	1.0 (-1)
I 133	3.8 (-1)
I 135	1.9 (-1)

* Values obtained from ANS 18.1

maximum gap inventory was determined by assuming the entire core was operating at the high burnup condition of Section 2.4.1. For the 3-Loop plant, the minimum gap inventory for Xe-133 is 2.71×10^4 Ci, and the maximum value is 3.67×10^5 Ci[13]. The normal operating activity is bounded by assuming a water mass of 1.23×10^8 grams (2-Loop plant) for the minimum value and 2.6 $\times 10^8$ grams (4-Loop plant) for the maximum value. The points of the minimum and maximum linear relations are calculated in the same manner as discussed above.

2.4.3.2 I-131

The gap inventory for a Westinghouse 3-Loop plant for I-131 is 2.31×10^5 Curies[13]. The minimum and maximum gap inventory for a 3-Loop plant for I-131 is 4.38×10^4 Ci and 5.98×10^5 Ci, respectively[13]. The source inventory of I-131 for a 3-Loop plant is 8.0×10^7 Curies[13]. The normal operating specific activity for I-131 from Table 2-5 is 0.27 μCi/gm. With a primary coolant mass of 1.78×10^8 gm for a standard 3-Loop plant, the normal operating activity of I-131 is 48 Curies. The points of the average, minimum, and maximum relations are calculated in the same manner as described in Section 2.4.3.1. Figure 2-3 shows the percentage of I-131 activity as a function of clad damage. The percentage release of I-131 calculated from the radionuclide analysis would be compared to Figure 2-3 to estimate the extent of clad damage.

For I-131, the possibility of iodine spiking should be considered when distinguishing between no clad damage and minor clad damage. The contribution of iodine spiking is discussed in Section 2.4.2 and is estimated to be as much as 950 Curies of I-131 released to primary system with an average release of 350 Curies based on a normal operating I-131 activity of 0.27 μCi per gram[6]. The linear relationships of Figure 2-3 are adjusted to account for the release due to iodine spiking by adding 950 Curies of I-131 to the maximum release and by adding 350 Curies of I-131 to the minimum and average release. Figure 2-4 shows the percentage of I-131 released with iodine spiking versus clad damage. Iodine spiking was not considered during the calculations of the correlations for the remaining iodines, I-132, I-133, and I-135, Figures 2-7 through 2-9, respectively.

2.4.4 GAP ACTIVITY RATIOS

Once equilibrium conditions are reached for the nuclides during operation, a fixed inventory of the nuclides exists within the fuel rod. For these nuclides which reach equilibrium, their relative ratios within the fuel pellet can be considered a constant.

Equilibrium conditions can also be considered to exist in the fuel rod gap. Under this condition the gap inventory of the nuclides is fixed. The distribution of the nuclides in the gap are 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. Since the relative diffusion rates of these nuclides under various operating conditions are approximately constant, the relative ratios of the nuclides in the gap are known.

In the presence of other indicators of a major release, the relative ratios of the nuclides can be compared with the relative ratios of the nuclides analyzed (corrected to shutdown) during an accident to determine the source of the fission product release. Table 2-6 presents the relative activity ratios for both the fuel pellet and the gap. The relative ratios for gap activities are significantly lower than the fuel pellet activity ratios. Measured relative ratios greater than gap activity ratios are indicative of more severe failures, e.g., fuel overheat.

2.4.5 ADJUSTMENTS TO DETERMINE ACTIVITY RELEASED

When analyzing a sample for the presence of nuclides, the isotopic concentration of the sample medium is expressed as the specific activity of the sample in either Curies per gram of liquid or Curies per cubic centimeter of atmosphere. The specific activity of the sample should then be adjusted to determine the total activity of that medium. The measured activity of the sample needs to be adjusted to account for the decay from the time the sample was analyzed to the time of reactor shutdown and adjusted to account for pressure and temperature difference of the sample relative to temperature and

TABLE 2-6

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	0.71
I-135	1.9	0.39

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

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

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* 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 <u>failure</u> conditions. Thus the ratios derived here are <u>not</u> related to fuel <u>defect</u> levels incurred during normal operation.

pressure conditions of the medium. Also the mass (liquid) or volume (gas) of the sample medium is required to calculate the isotopic activity of that medium. The following sections discuss the required adjustments.

2.4.5.1 DILUTION OF SAMPLE MEDIUM

The distribution of the total water inventory should be known to determine the water amount that is associated with each sample medium. If a sample is taken from the primary system, an approximation of the amount of water in the primary system is needed and a similar approximation is required for a sump sample. For the purposes of this methodology the water is assumed to be distributed within the primary system and the sump. However, consideration should be taken if a significant primary system to secondary system leak rate is noted as in the case of a steam generator tube rupture. The amount of water that is available for distribution is the initial amount of primary system water and the amount of water that has been discharged from the Refueling Water Storage Tank (RWST). Also, an adjustment must be made for water added via the containment spray systems, accumulators, chemical addition tanks, and ice condensers. To approximate the distribution of water, the monitoring systems of the reactor vessel, pressurizer, sump, and RWST can be employed. If not all of the monitoring systems are available, the monitoring systems which are working can be used by assuming that the total water inventory is distributed in the sump and the primary system with consideration given if a significant primary system to secondary system leak rate is noted. The approximate total activity of the liquid samples can then be calculated.

[The D. C. Cook Unit 1 and Unit 2 containments are each equipped with ice condensers. Each containment houses approximately 2.7×10^6 pounds of ice, which provides an additional source of water. The RWST can provide up to approximately 350,000 gallons of emergency core cooling water during an accident. The 4 accumulators are each equipped to provide approximately 950 ft^3 of water. The boron injection tank can supply 900 gallons of water.]

RCS activity (Curies) = Specific Activity (Ci/cc or Ci/gm) x
RCS water volume or mass (cc or gm).

Sump activity (Curies) = Specific Activity (Ci/cc or Ci/gm) x
Sump water volume or mass (cc or gm).

Total water activity = RCS activity + Sump activity +
Activity leaked to Secondary System + Activities from other sources (accumulators, ice condensers, spray additive tanks, etc.).

Note: The specific activities should be decay corrected to reactor shutdown, and the RCS amount should be corrected to account for temperature and pressure differences between sample and RCS.

The containment atmosphere activity can then be added to approximate the total activity released at time of accident.

Total Activity Released = Total Water Activity +
Containment Atmosphere Activity

2.4.5.2 PRESSURE AND TEMPERATURE ADJUSTMENT

The measurements for the containment atmosphere samples need to be adjusted if the pressure and temperature of the samples at the time of analysis are different than the conditions of containment atmosphere. The adjustments to the specific activity and the containment volume are as follows.

$$\text{Specific Activity (Atmosphere)} = \text{Specific Activity (Sample)} \times \frac{P_2}{P_1} \times \left(\frac{T_1 + 460}{T_2 + 460}\right)$$

where:

T_1, P_1 = measured sample temperature (°F) and pressure (psia)
T_2, P_2 = containment atmosphere temperature (°F) and pressure (psia).

$$\text{Corrected Containment Volume} = \text{Containment Free Volume (SCF)} \times \frac{P_3}{P_2} \left(\frac{T_2 + 460}{T_3 + 460}\right)$$

where:

T_2, P_2 = containment atmosphere temperature (°F) and pressure (psia)
T_3, P_3 = standard temperature (32°F) and pressure (14.7 psia).

[The above adjustments are based on molar volumes. For samples in which the atmosphere sample is drawn into a specified volume and the analysis is performed to this volume, no adjustments to either the sample specific activity or containment volume are required.]

For those plants with ice condensers, consideration should be given to account for a decrease in free volume due to the ice melting occupying a portion of the containment volume.

[Eventhough D. C. Cook is a plant with ice condensers, no adjustment is needed to the containment free volume due to the effect of the ice melting. The listed containment free volume (1.2×10^6 ft^3) takes into account the presence of solid ice. Since there is negligible difference between the densities of ice and water, no adjustment is required.]

The total activity released to the containment atmosphere is

Total Containment Activity = Specific Activity (Atmosphere) x
Corrected Containment Volume

where the specific activity (atmosphere) has been decay corrected to time of reactor shutdown.

The specific activity of the liquid samples requires no adjustment if the specific activity is reported on a per-gram basis (μCi/gm). If the specific activity is reported on a per-volume basis (μCi/cc), an adjustment is performed to convert the per-volume specific activity to a per-gram specific activity. The conversion is performed for consistency with later calculations. If the temperature of the sample is above 200°F, an adjustment is required to the conversion. In most cases the sample temperature will be

below 200°F and no adjustment is necessary. Figure 2-10 shows a relation of water density at some temperature relative to the water density at standard temperature and pressure.

The mass of the liquid medium (RCS or sump) can be calculated from the volume of the medium. If the medium (RCS or sump) temperature at time of sample is above 200°F, an adjustment is required to the conversion.

A. RCS or Sump temperature > 200°F

$$\text{RCS or sump mass (gm)} = \text{RCS or Sump Volume (ft}^3) \times \frac{\rho}{\rho_{STP}}(2) \times \rho_{STP} \times \frac{28.3 \times 10^3 \text{ cc}}{\text{ft}^3}$$

where:

$\frac{\rho}{\rho_{STP}}(2)$ = water density ratio at medium (RCS or sump) temperature, Figure 2-10

ρ_{STP} = water density at STP = 1.00 gm/cc.

B. RCS or sump temperature < 200°F

$$\text{RCS or Sump Mass (gm)} = \text{RCS of Sump Volume (ft}^3) \times \rho_{STP} \times \frac{28.3 \times 10^3 \text{ cc}}{\text{ft}^3}$$

where:

ρ_{STP} = water density at STP = 1.00 gm/cc.

The total activity of the RCS or sump is as follows.

RCS or Sump Activity = RCS or Sump Specific Activity (μCi/gm) x RCS or Sump Mass (gm)

where the specific activity has been decay corrected to time of shutdown.

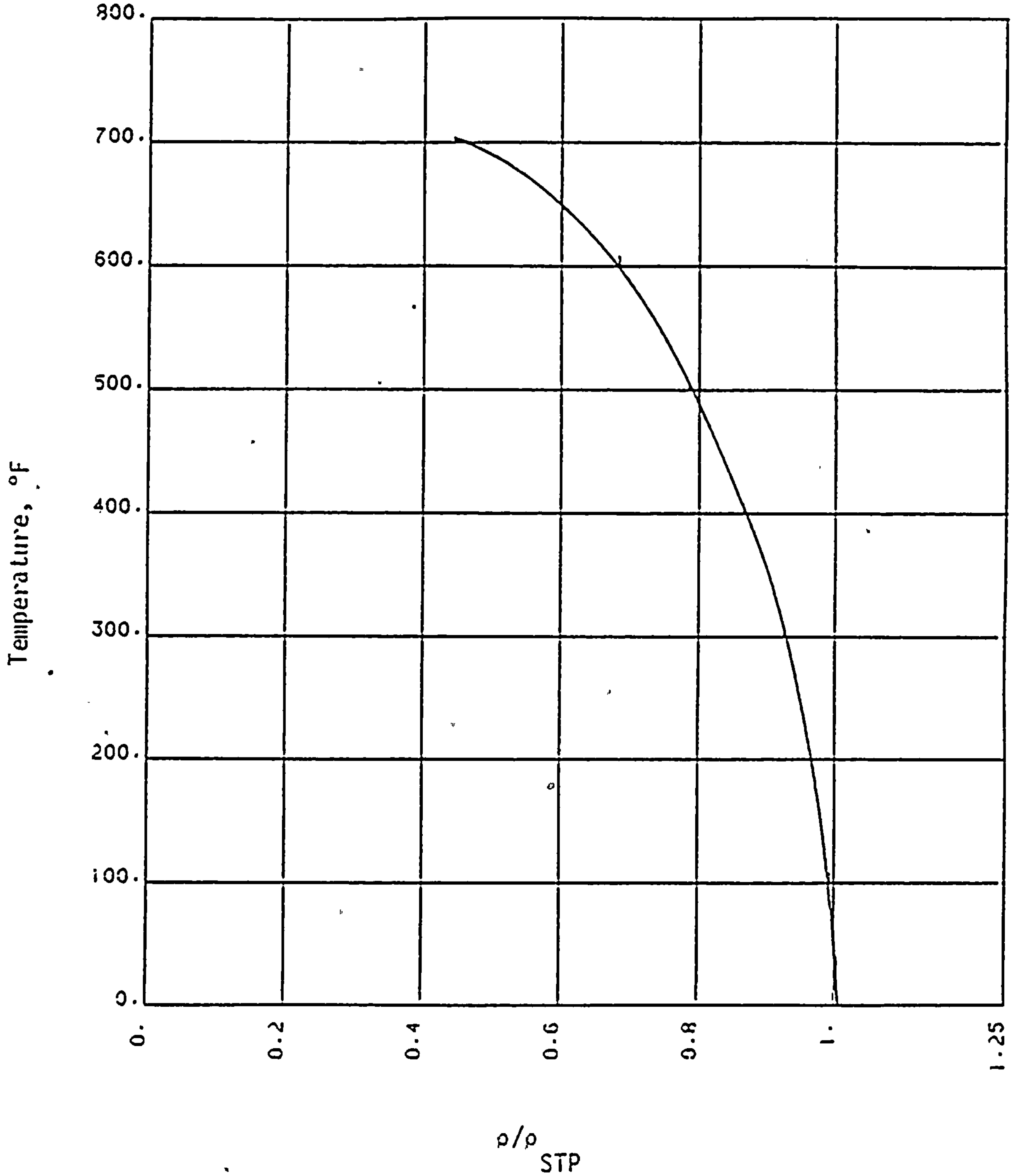


FIGURE 2-10 WATER DENSITY RATIO (TEMPERATURE VS. STP)

[The sump and containment water volume can be approximated from Figures 2-10A and 2-10B based on the readings of the water level indicators of the sump and containment. The reactor vessel level indication system can be used to approximate the RCS volume, as described by the following:

1. If the water level in the reactor vessel indicates the system is full, then the full reactor coolant system water volume is used. For Unit 1 and Unit 2 the RCS volume of each is approximately 11, 780 ft^3 at 570°F and 2250 psia.

2. If the water level in the reactor vessel is below the low end capability of the indicator, the RCS volume is unknown. In this case, the sump sample should be given primary concern.

3. If the reactor vessel level indication system is not working, then, by knowing the water sources available, the other monitors can be used to estimate the RCS volume. If it is known how much water is available (volumes of RWST, accumulators, boron injection tank, and original RCS volume), the volume of the sump and containment water is substracted from the available water volume to estimate the RCS volume. Also to be considered as a source of water is water from the melting ice. An assumption can be made that all the ice melts in approximately 3 to 5 hours after the start of an accident.]

2.4.5.3 DECAY CORRECTION

The specific activity of a sample is decay adjusted to time of reactor shutdown using the following equation.

$$\text{Specific activity at shutdown} = \frac{\text{Specific activity (measured)}}{e^{-\lambda_i t}},$$

where:

λ_i = radioactive decay constant, 1/sec

t = time period from reactor shutdown to time of sample analysis, sec.

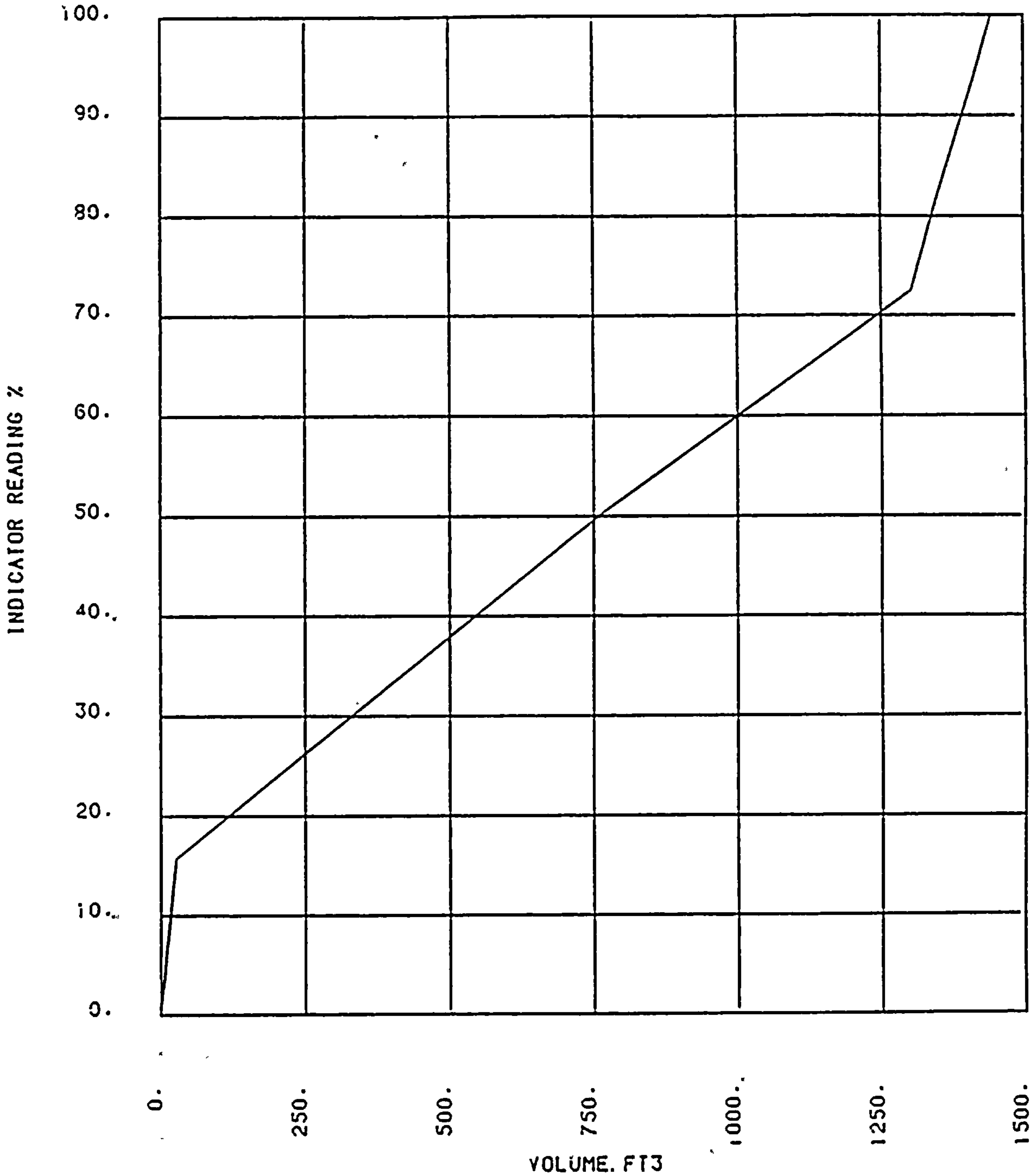


FIGURE 2-10A SUMP WATER VOLUME VERSUS SUMP LEVEL INDICATION

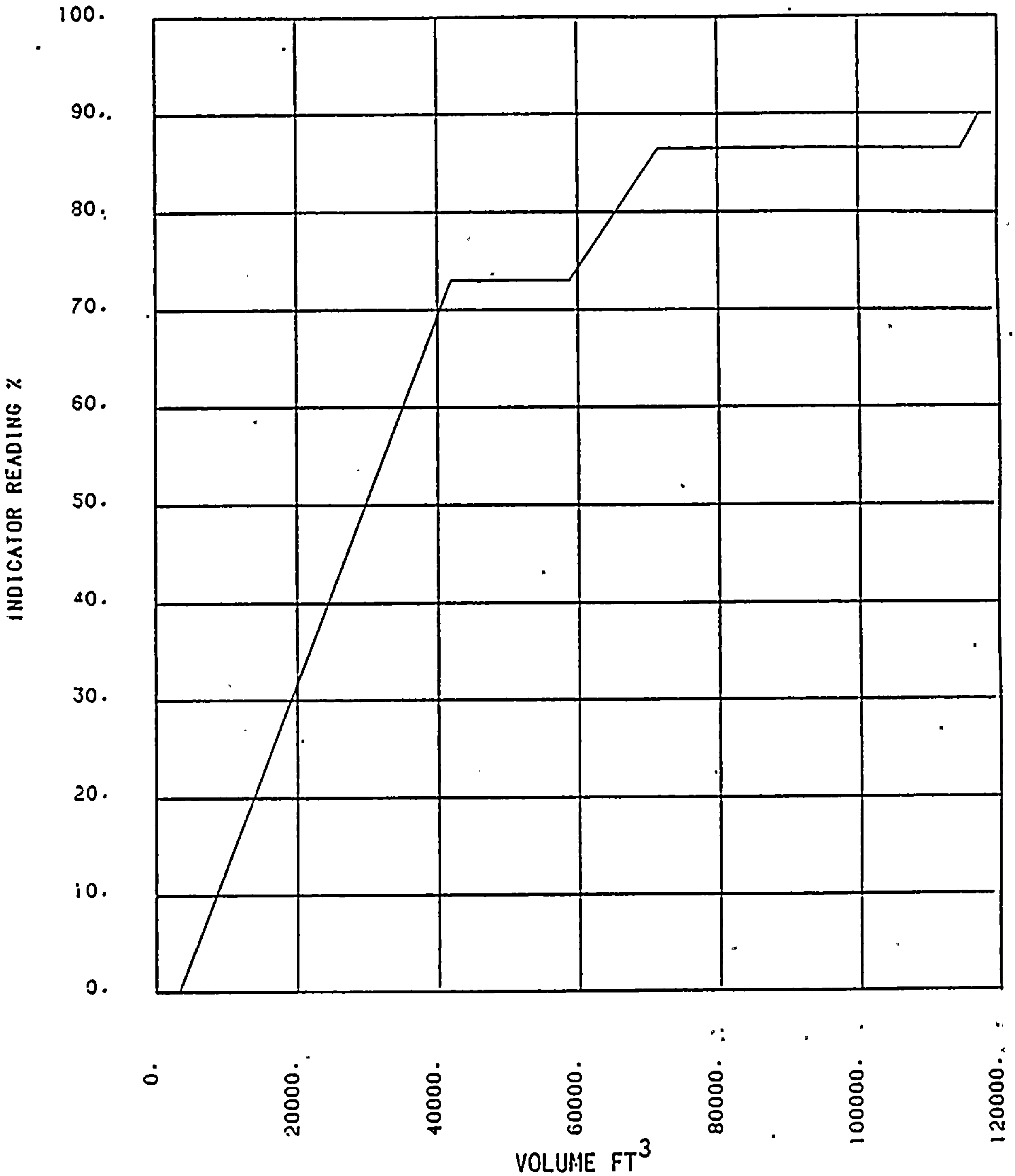


FIGURE 2-10B CONTAINMENT WATER VOLUME VERSUS CONTAINMENT LEVEL INDICATION

Since this correction may also be performed by some analytical equipment, care must be taken to avoid duplicate correction. Also, consideration must be given to account for precursor effect during the decay of the nuclide. For this methodology, only the parent-daughter relationships are considered. Table 2-7 lists the significant parent-daughter relationships associated with the methodology. The decay scheme of the parent-daughter relationship is described by the following equation.

$$Q_B = \frac{\lambda_B}{\lambda_B - \lambda_A} Q_A^o \left(e^{-\lambda_A t} - e^{-\lambda_B t}\right) + Q_B^o e^{-\lambda_B t}$$

where:

Q_A^o	=	activity (Ci) or specific activity (µCi/gm or µCi/cc) of the parent at shutdown
Q_B^o	=	activity (Ci) or specific activity (µCi/gm or µCi/cc) of the daughter at shutdown
Q_B	=	activity (Ci) or specific activity (µCi/gm or µCi/cc) of the daughter at time of sample
λ_A	=	decay constant of the parent, sec^{-1}
λ_B	=	decay constant of the daughter, sec^{-1}
t	=	time period from reactor shutdown to time of sample analysis, sec.

Since the activity of the daughter at sample time is due to the decay of the parent and the decay of the daughter initially released at shutdown, an estimation of the fraction of the measured activity at sample time due to only the decay of daughter is required. To use the above equation to determine this fraction, an assumption is made that the percentages of the source inventories of the parent and the daughter released at time of shutdown are

TABLE 2-7

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

equal (for the nuclides used here within a factor of 2). 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.

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.

$$Q_B(t) = K\frac{\lambda_B}{\lambda_B-\lambda_A} \quad Q_A^o \quad (e^{-\lambda_A t} - e^{-\lambda_B t}) + Q_B^o e^{-\lambda_B t}$$

where:

Q_A^o	=	100% source inventory (Ci) of parent, Table 2-2 or 2-8
Q_B^o	=	100% source inventory (Ci) of daughter, Table 2-2 or 2-8
$Q_B(t)$	=	hypothetical daughter activity (Ci) at sample time
K	=	if parent has 2 daughters, K is the branching factor, Table 2-7
λ_A	=	parent decay constant, sec^{-1}
λ_B	=	daughter decay constant, sec^{-1}
t	=	time period from shutdown to time of sample, sec.

2. Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time

$$Fr = \frac{Q_B^o \; e^{-\lambda_B t}}{Q_B\;(t)}$$

TABLE 2-8

SOURCE INVENTORY OF RELATED PARENT NUCLIDES

Nuclide	Unit 1 (3250 MWt)	Unit 2 (3391 MWt)
Xe-135m	3.8(7)	4.0(7)
Sb-129	2.9(7)	3.0(7)
Te-129m	7.3(6)	7.6(6)
Ba-142	1.5(8)	1.5(8)
Ce-144	1.0(8)	1.0(8)

3. Calculate the amount of the measured sample specific activity associated with the decay of the daughter that was released.

$$M_B = Fr \times \text{measured specific activity } (\mu Ci/gm \text{ or } \mu Ci/cc)$$

4. Decay correct the specific activity (M_B) to reactor shutdown.

$$M_B^o = \frac{M_B}{e^{-\lambda_B t}}$$

2.5 RELATIONSHIP OF FISSION PRODUCT RELEASE WITH OVERTEMPERATURE CONDITIONS

The current concept of the mechanisms for fission product release from UO_2 fuel under accident conditions has been summarized in 2 documents, draft NUREG-0956[7] and IDCOR Task 11.1[8]. These documents describe five principal release mechanisms; burst release, diffusional release of the pellet-to-cladding gap inventory, grain boundary release, diffusion from the UO_2 grains, and release from molten material. The release which occurs when the cladding fails, i.e., gap release, is utilized to quantify the extent of clad failure as discussed in Section 2.4. Table 2-9 presents the expected fuel damage state associated with fuel rod temperatures.

Fission product release associated with overtemperature fuel conditions arises initially from that portion of the noble gas, cesium and iodine inventories that was previously accumulated in grain boundaries. For high burnup rods, it is estimated that approximately 20 percent of the initial fuel rod inventory of noble gases, cesium, and halogens would be released. Release from lower burnup fuel would no doubt be less. Following the grain boundary release, additional diffusional release from UO_2 grains occurs. Estimates of the total release, including UO_2 diffusional release, vary from 20 to 40 percent of the noble gas, iodine and cesium inventories.

Additional information on the release of fission products during overtemperature conditions was obtained from the TMI accident[9]. In this instance current opinion is that although the core had been overheated, fuel melt had not occurred. Values of core inventory fraction of various fission products released during the accident are given in Table 2-10. These values,

TABLE 2-9

EXPECTED FUEL DAMAGE CORRELATION WITH FUEL ROD TEMPERATURE[8]

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.

TABLE 2-10

PERCENT ACTIVITY RELEASE FOR 100 PERCENT OVERTEMPERATURE CONDITIONS

Nuclide	Min.*	Max.*	Nominal**	Min.***	Max.***
Kr-85	40	70			
Xe-133	42	66	52.	40	70
I-131	41	55			
Cs-137	45	60			
Sr-90	0.08****				
Ba-140	0.1	0.2	0.15	0.08	0.2

* Release values based on TMI-2 measurements.

** Nominal value is simple average of all Kr, Xe, I, and Cs measurements.

*** Minimum and maximum values of all Kr, Xe, I and Cs measurements.

**** Only value available.

derived from radiochemical analysis of primary coolant, sump, and containment gas samples, provide much greater releases of the noble gases, halides, and cesiums, than is expected to be released solely from cladding failures. In addition, small amounts of the more refractory elements, barium-lanthanum, and strontium were released. In the particular case of TMI, the release mechanism, in addition to diffusional release from grain boundaries and UO_2 grains, is believed to arise from UO_2 grain growth in steam.

The relationship between extent of fuel damage and fission product release for several radioisotopes during overtemperature condition is depicted graphically in Figures 2-11 and 2-12. To construct the figures, the extent of fuel damage, expressed as a percentage of the core, is plotted as a linear function of the percentage of the source inventory released for various nuclides. The values used in constructing the graphs were obtained from Table 2-10. For example, if 100 percent of the core experienced overtemperatures, 52 percent of Xe-133 core inventory would be released. If 1 percent of the core experienced overtemperature, 0.52 percent of Xe-133 core inventory would be released. The assumption is also made that nuclides of any element, e.g., I-131 and I-133, have the same magnitude of release. In order to apply these figures to a particular plant, power, decay, and dilution corrections described earlier in this report must be applied to the concentrations of nuclides determined from analysis of radionuclide samples. The maximum and minimum estimates of release percentages are those given in Table 2-10 as the range of values: nominal values of release are simple averages of the miminum and maximum values.

2.6 RELATIONSHIP OF NUCLIDE RELEASE WITH CORE MELT CONDITIONS

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.

Values of the release of fission products during fuel melt conditions are derived from ex-pile experiments performed by various investigators.

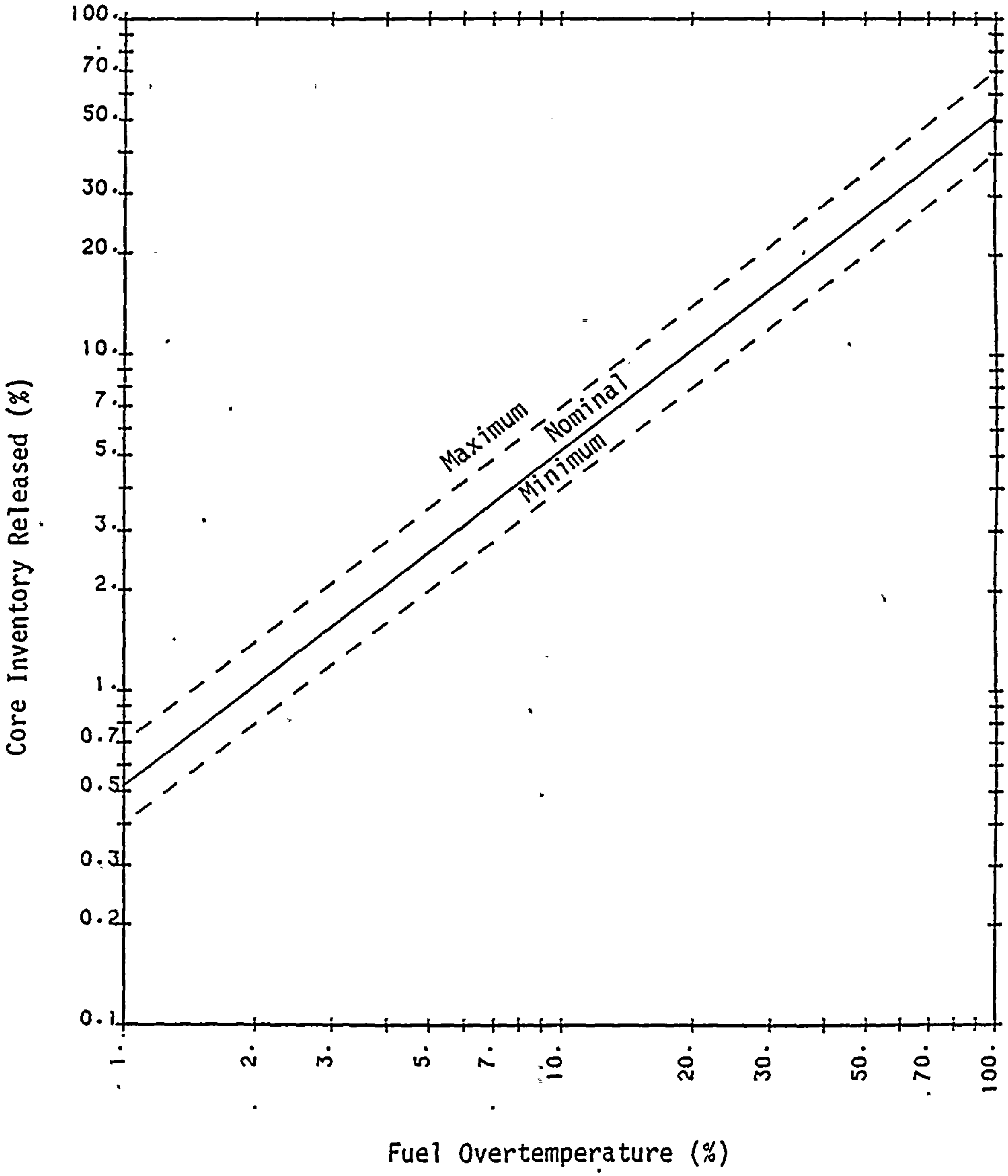


FIGURE 2-11 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF XE, KR, I, OR CS

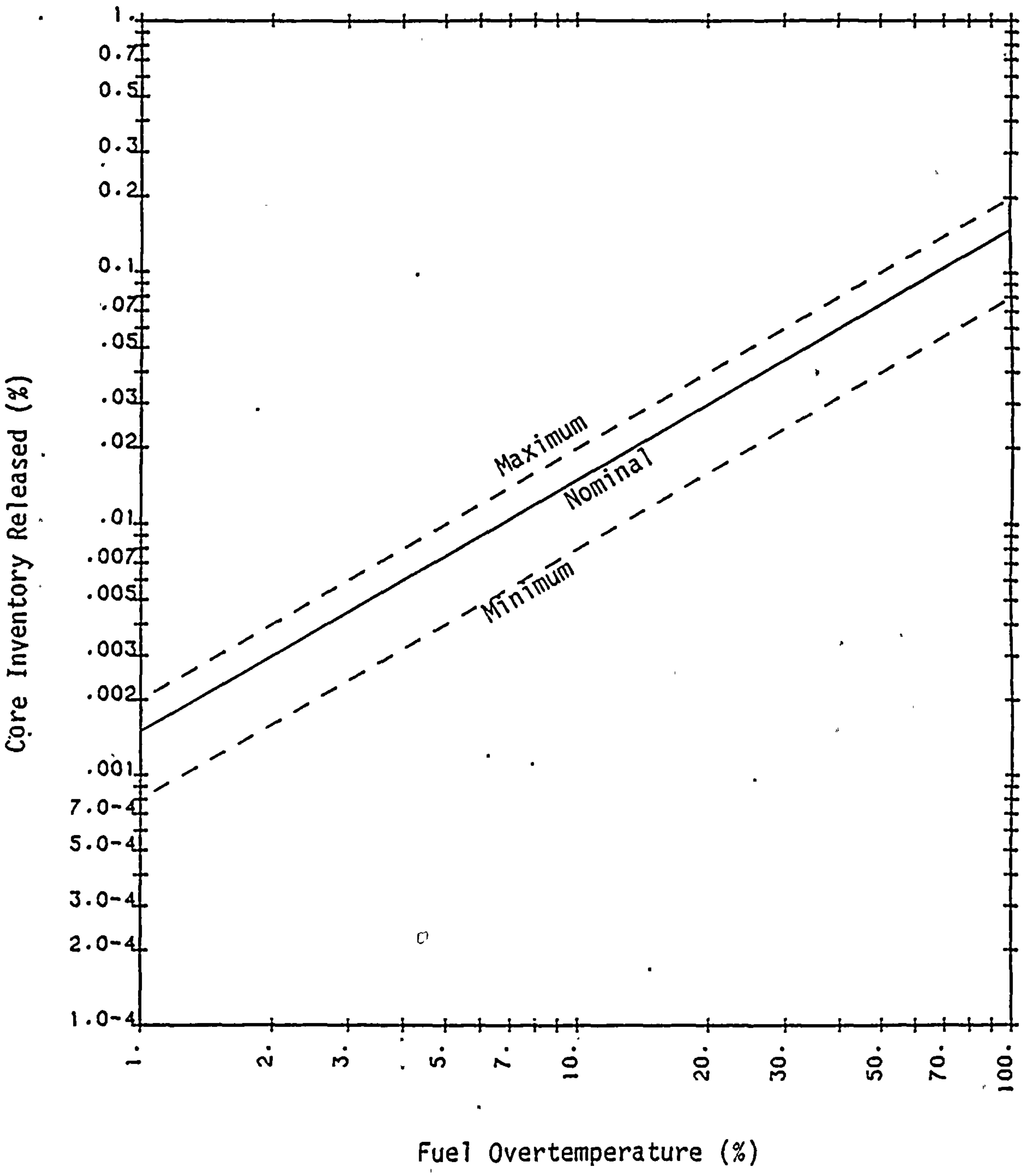


FIGURE 2-12 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF BA OR SR

These release measurements have been expressed as release rate coefficients for various temperature regimes. These release rate coefficients have been represented by a simple exponential equation in draft NUREG-0956. This equation has the form:

$K(T)$	=	Ae^{BT} where
$K(T)$	=	release rate coefficient
A & B	=	constants
T	=	temperature.

These release rate coefficients were utilized with core temperature profiles to develop fission product release estimates for various accident sequences for which core melt is postulated in draft NUREG-0956.

Fission product release percentages for three accident sequences which lead to 100 percent core melt are given in Table 2-11. The xenon, krypton, cesium, iodine, and tellurium elements have been arranged into a single group because of similarity in the expected magnitude of overtemperature release. The assumption is also made that nuclides of any element e.g., Iodine 131 and Iodine 133, have the same magnitude of release. The differences in the calculated releases of the various elements for the different accident sequences were used to determine minimum and maximum values of expected release; nominal values of release are simple averages of all release values within a group.

The percentage release of various nuclides has been correlated to percentage of core melt with the linear extrapolations shown in Figures 2-13 through 2-15.

2.7 SAMPLING LOCATIONS

A survey of a number of Westinghouse plants has indicated that the post accident sampling system locations for liquid and gaseous samples varies for each plant. To obtain the most accurate assessment of core damage, it is recommended to sample and analyze radionuclides from the reactor coolant system, the containment atmosphere, and the containment sump (if available). Other samples can be taken dependent on the plant's capabilities. The

TABLE 2-11

PERCENT ACTIVITY RELEASE FOR 100 PERCENT CORE MELT CONDITIONS

Species	Large* LOCA	Transient*	Small* LOCA	Nominal** Release	Min.*** Release	Max.*** Release
Xe	88.35	99.45	78.38			
Kr	88.35	99.45	78.38			
				87	70	99
I	88.23	99.44	78.09			
Cs	88.55	99.46	78.84			
Te	78.52	94.88	71.04			
Sr	10.44	28.17	14.80	24	10	44
Ba	19.66	43.87	24.08			
Pr	0.82	2.36	1.02	1.4	0.8	2.4

* Calculated releases for severe accident scenarios without emergency safeguard features, taken from draft NUREG-0956

** Nominal release are averages of Xe, Kr, I, Cs, and Te groups, or Sr and Ba groups.

*** Maximum and minimum releases represent extremes of the groups.

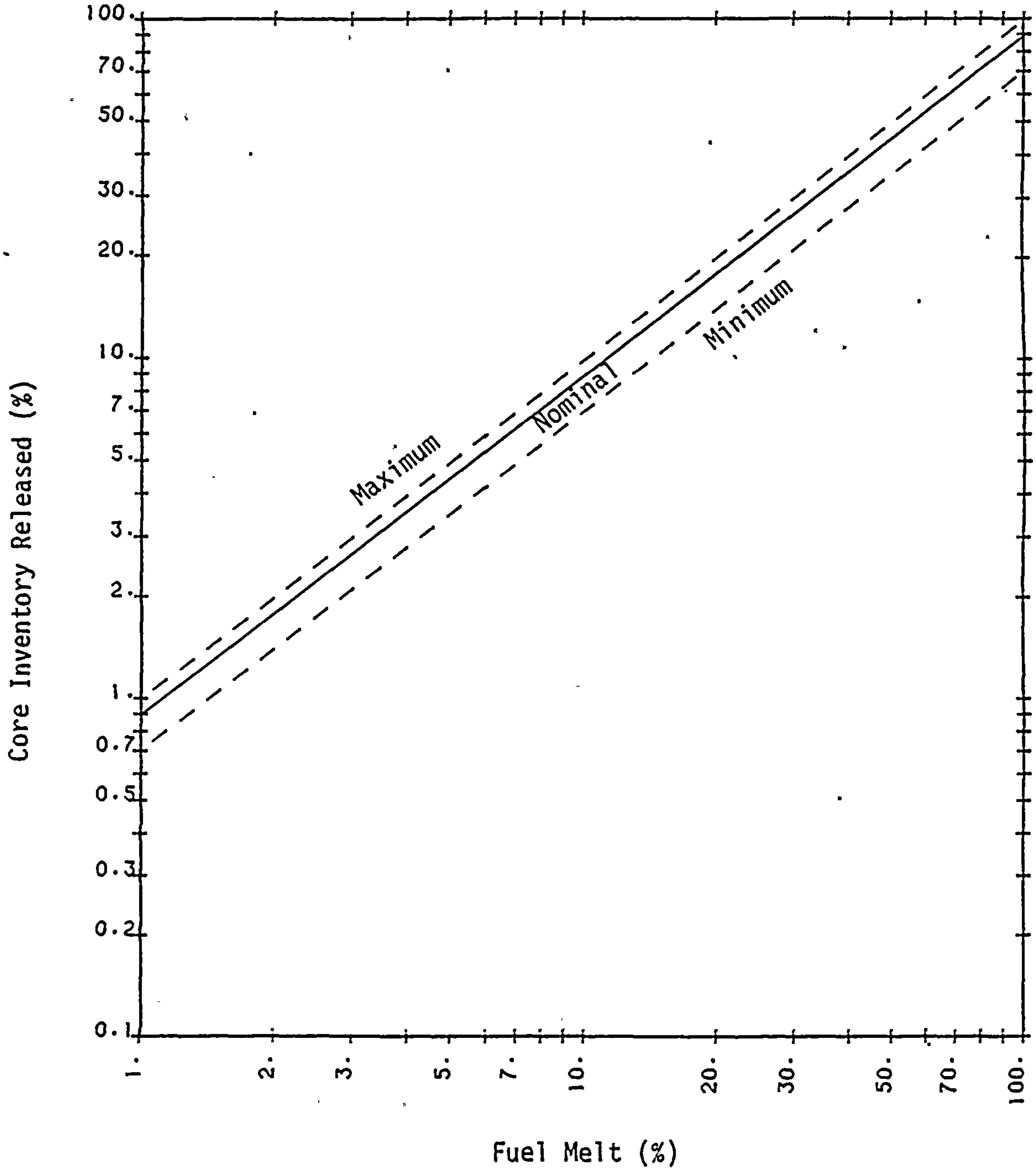


FIGURE 2-13 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF XE, KR, I, CS, OR TE

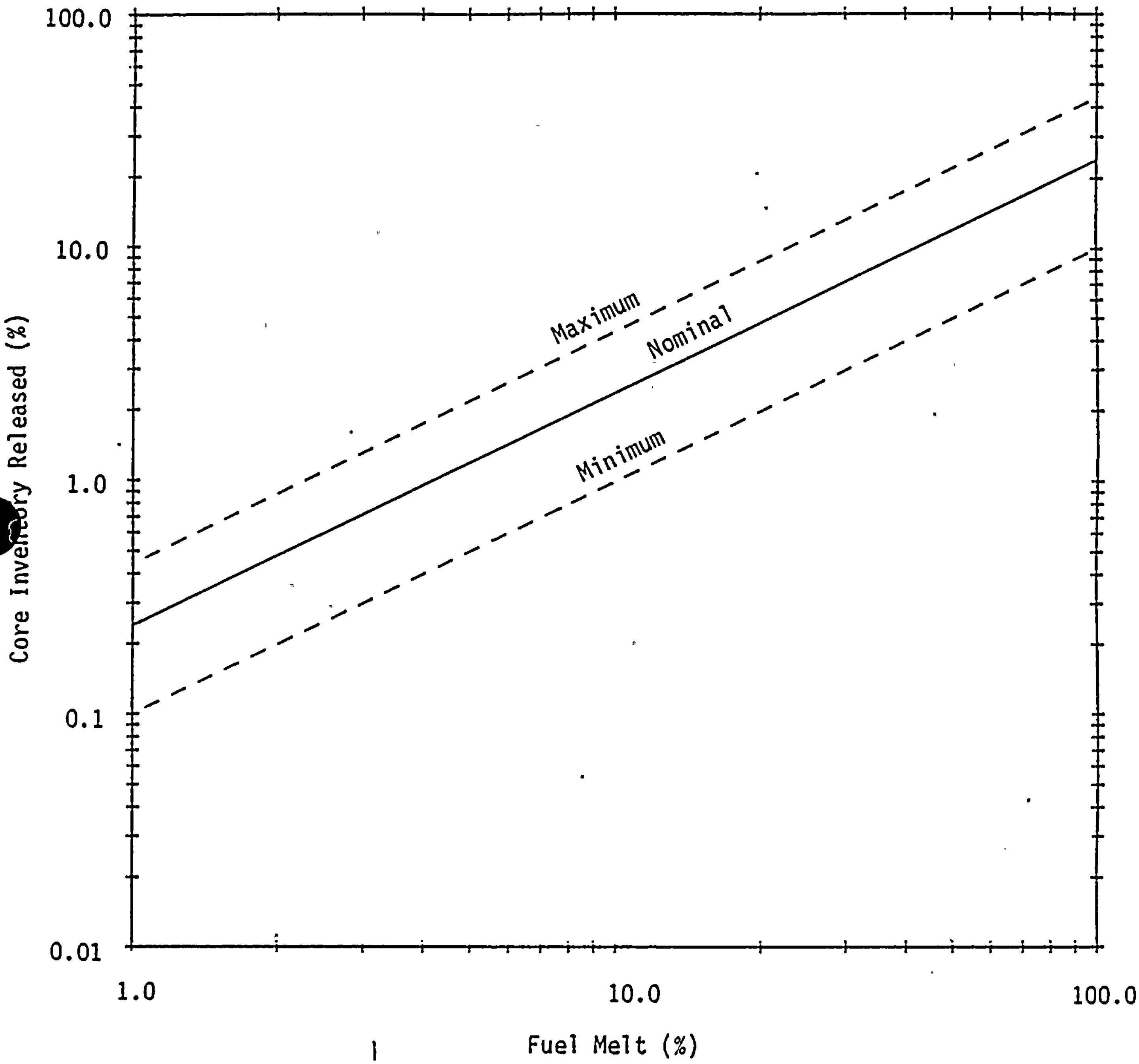


FIGURE 2-14 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF BA OR SR

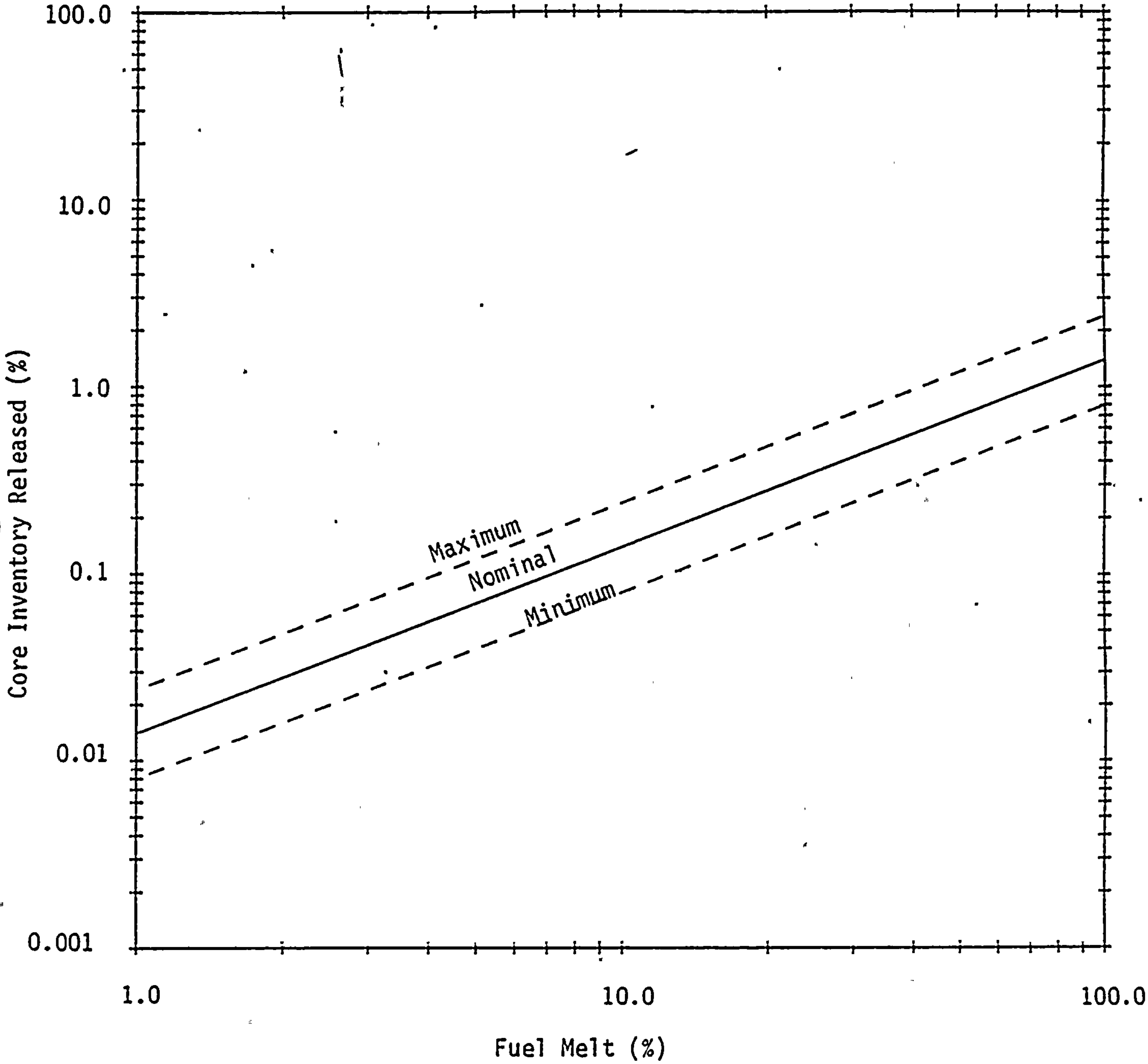


FIGURE 2-15 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF PR

specific sample locations to be used during the initial phases of an accident should be selected based on the type of accident in progress. If the type of accident scenario is unknown, known plant parameters (pressure, temperature, level indications, etc.) can be used as a basis to determine the prime sample locations. Consideration should be given to sampling secondary system if a significant leak from the primary system to secondary system is noted. Table 2-12 presents a list of the suggested sample locations for different accident scenarios based on the usefulness of the information derivable from the sample.

[D. C. Cook's PASS is equipped to obtain samples from hot loop 1 and 3, east and west RHR, containment sump, pressurizer steam space and containment air. Plant personnel will use Table 2-12 as a guide in determining sample locations, but final discretion is left up to the plant personnel.]

TABLE 2-12

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.

3.0 AUXILIARY INDICATORS

There are plant indicators monitored during an accident which by themselves cannot provide a useful estimate but can provide verification of the initial estimate of core damage based on the radionuclide analysis. These plant indicators include containment hydrogen concentration, core exit thermocouple temperatures, reactor vessel water level, and containment radiation level. When providing an estimate for core damage, these plant indicators, if available, should confirm the results of the radionuclide analysis. For example, if the core exit thermocouple readings and reactor vessel water level indicate a possibility of clad damage and the radionuclide concentrations indicate no clad damage, then a recheck of both indications may be performed or certain indications may be discounted based on engineering judgment.

3.1 CONTAINMENT HYDROGEN CONCENTRATION

An accident, in which the core is uncovered and the fuel rods are exposed to steam, may result in the reaction of the zirconium of the cladding with the steam which produces hydrogen. The hydrogen production characteristic of the zirconium water reaction is that for every mole of zirconium that reacts with water, two moles of hydrogen are produced. For this methodology it is assumed that all of the hydrogen that is produced is released to the containment atmosphere. The hydrogen dissolved in the primary system during normal operation is considered to contribute an insignificant amount of the total hydrogen released to the containment. [For Unit 1 and Unit 2, the release of the dissolved hydrogen and the hydrogen in the pressurizer gas space to the containment corresponds to a containment hydrogen concentration of 0.1 percent by volume, which can be considered insignificant within the accuracy of this report.] In the absence of hydrogen control measures, monitoring this containment hydrogen concentration during the accident can provide an indication of the extent of zirconium water reaction. The percentage of zirconium water reaction does not equal the percentage of clad damaged but it does provide a qualitative verification of the extent of clad damage estimated from the radionuclide analysis.

Figure 3-1 shows the relationship between the hydrogen concentration and the percentage of zirconium water reaction for Unit 1 and Unit 2. The relationship shown in Figure 3-1 does not account for any hydrogen depletion due to hydrogen recombiners and hydrogen ignitions. The recombiners that now exist are capable of dealing effectively with the relatively small amounts of hydrogen that result from radiolysis and corrosion following a design basis LOCA. However, they are incapable of handling the hydrogen produced in an extensive zirconium-steam reaction such as would result from severe core degradation. Current recombiners can process gas that is approximately 4 to 5 percent hydrogen or less[10]. Each recombiner unit can process an input flow in the range of 100 SCFM to 200 SCFM. 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. Uncontrolled ignition of hydrogen and deliberate ignition will hinder any quantitative use of hydrogen concentration as an auxiliary indicator. However, the oxygen amount depleted during the burn, if known, can be used to estimate the amount of hydrogen burned. If the oxygen amount depleted is not known, it can be assumed that for ignition of hydrogen to occur a minimal concentration of 4 percent hydrogen is needed. [Since Units 1 and 2 are ice condenser containments, deliberate ignition of the hydrogen is utilized to control the containment hydrogen concentration. As stated above, a minimal concentration of 4 percent hydrogen is needed.] This assumption can be used qualitatively to indicate that some percentage of zirconium has reacted, but it is difficult to determine the extent of the reaction.

Containment hydrogen concentrations can be obtained from the Post Accident Sampling System or the containment gas analyzers. [Figure 3-1 shows the relationship between the hydrogen concentration (percent volume) and the percentage of zirconium water reaction for Unit 1 and Unit 2. The hydrogen concentration shown is the result of the analysis of a dry containment sample. The curves were based on average containment volumes and the average initial zirconium mass of the fuel rods for each unit, which are shown in Table 3-1. Table 3-1 also presents the correlation between hydrogen concentration and percentage of zirconium water reaction.] To use the auxiliary indicator of hydrogen concentration, the assumptions were that all hydrogen from zirconium water reaction is released to containment, a well-mixed atmosphere, and ideal gas behavior in containment.

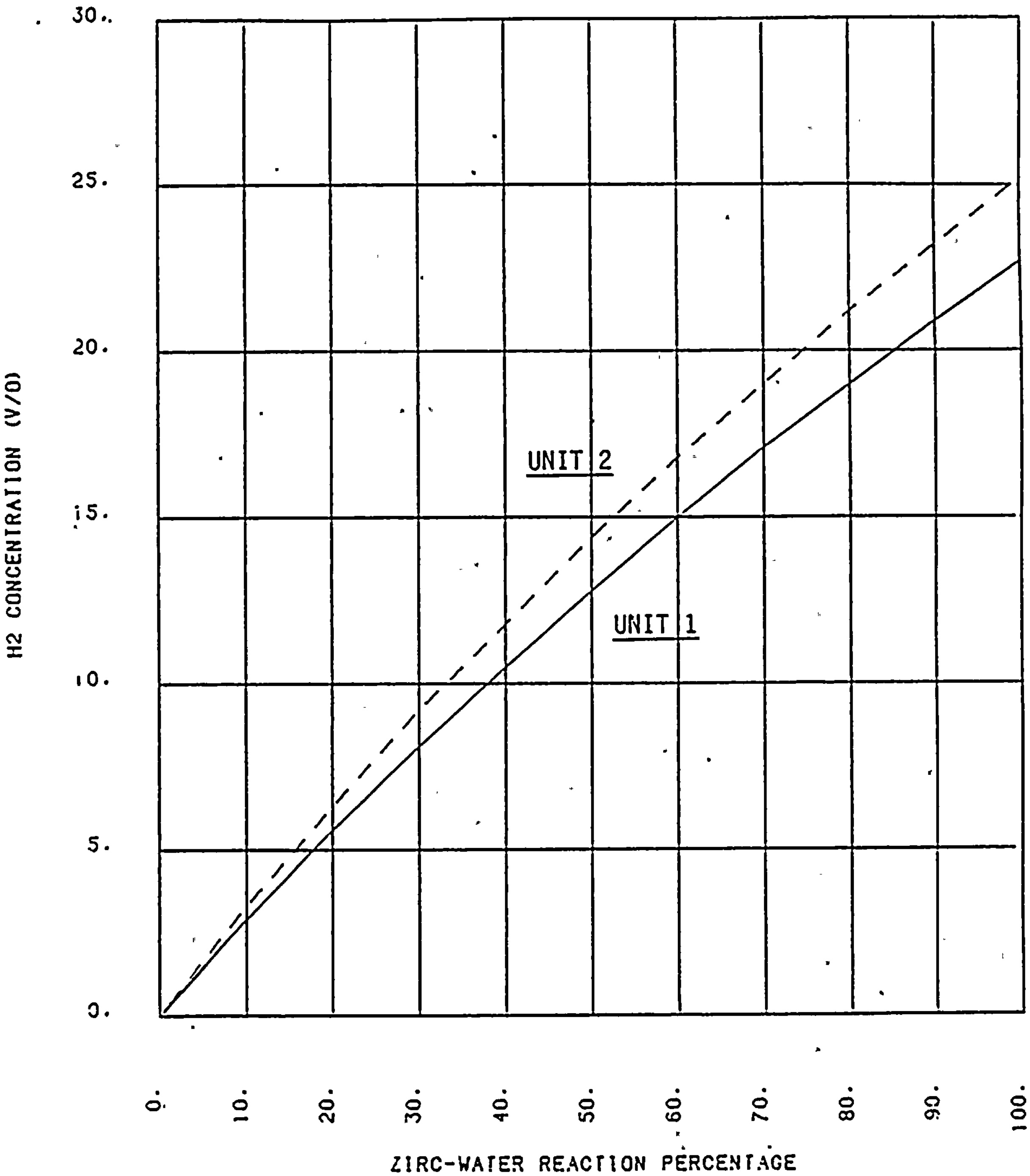


FIGURE 3-1 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM WATER REACTION

TABLE 3-1

CONTAINMENT VOLUME AND ZIRCONIUM MASS

Plant Type	Zirconium Mass (lbm)	Containment Volume (SCF)
Unit 1	44,547	1.2×10^6
Unit 2	50,913	1.2×10^6

Relationship between hydrogen concentration of a dry sample and fraction of zirconium water reaction is based on the following formula.

$$\% H_2 = \frac{(FZWR)(ZM)(H)}{(FZWR)(ZM)(H) + V} \times 100$$

where: FZWR = fraction of zirconium water reaction

ZM = total zirconium mass, lbm

H = conversion factor, 7.92 SCF of H_2 per pound of zirconium reacted

V = containment volume, SCF

3.2 CORE EXIT TEMPERATURES AND REACTOR VESSEL WATER LEVELS

Core exit thermocouples (CETCs) measure the temperature of the fluid at the core exit at various radial core locations [(Figure 3-2)]. The typical thermocouple system is qualified to read temperatures as high as 1650°F. This is the ability of the system to measure the fluid temperatures at the incore thermocouples locations and not core temperatures.

Most reactor vessel level indication systems (RVLIS) use differential pressure (d/p) measuring devices to measure vessel level or relative void content of the circulating primary coolant system fluid. The system is redundant and includes automatic compensation for potential temperature variations of the impulse lines. Essential information is displayed in the main control room in a form directly usable by the operator.

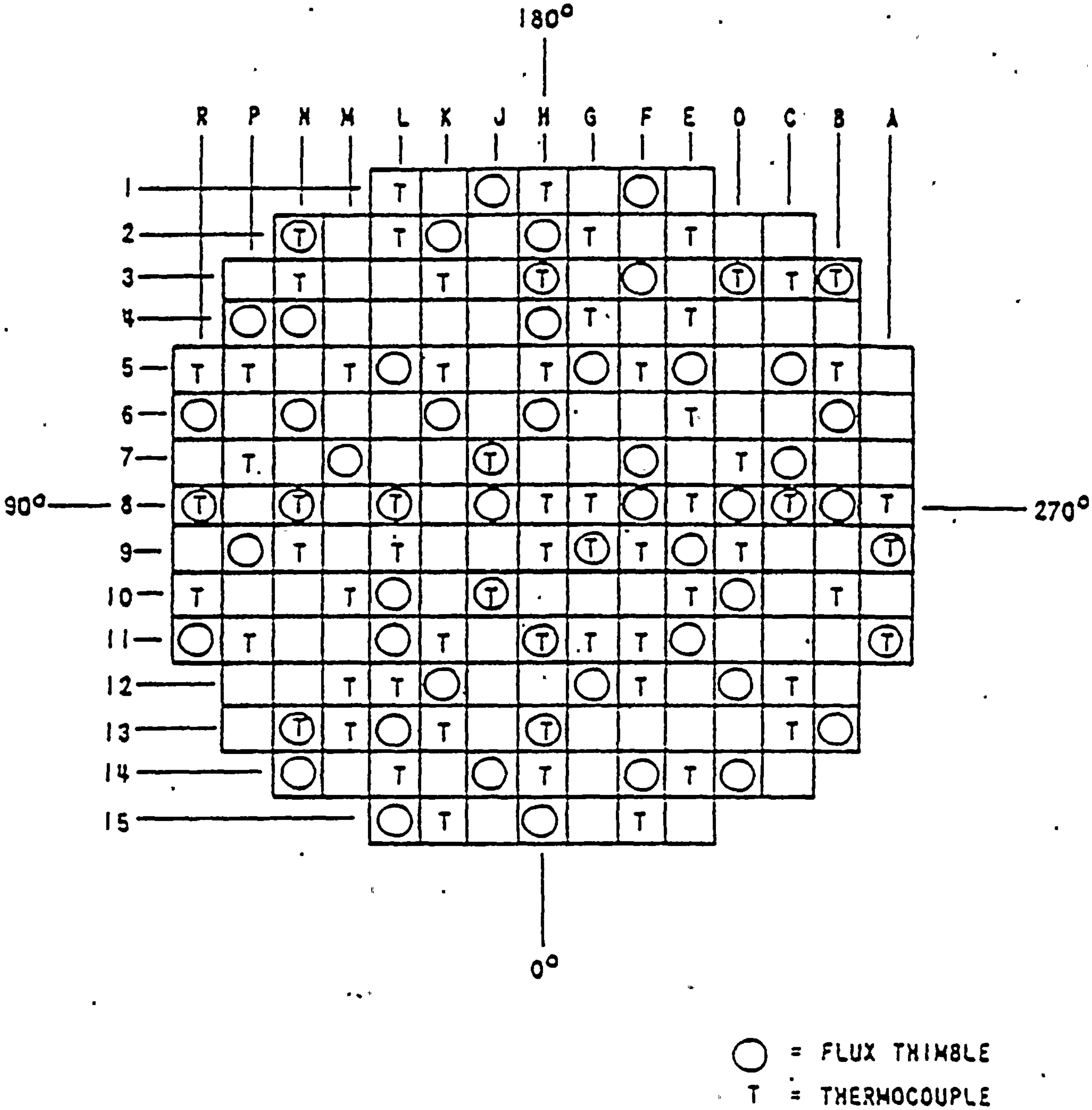
RVLIS and CETC readings can be used for verification of core damage estimates in the following ways[11].

- Due to the heat transfer mechanisms between the fuel rods, steam, and thermocouples, the highest clad temperature will be higher than the CETC readings. Therefore, if thermocouples read greater than 1300°F, clad failure may have occurred. 1300°F is the lower limit for cladding failures.

- If any RCPs are running, the CETCs will be good indicators of clad temperatures and no core damage should occur since the forced flow of the steam-water mixture will adequately cool the core.

  If RCPs are not running, the following apply.

- No generalized core damage can occur if the core has not uncovered. So if RVLIS full range indicates that the collapsed liquid level has never been below the top of the core and no CETC has indicated temperatures corresponding to superheated steam at the corresponding RCS pressure, then no generalized core damage has occurred.



Distribution of Thermocouples and Flux Thimbles for Unit 1 and Unit 2

Figure 3-2

- If RVLIS indicates less than 3.5 ft. collapsed liquid level in the core or CETCs indicate superheated steam temperatures, then the core has uncovered and core damage may have occurred depending on the time after reactor trip, length and depth of uncovery. Best estimate small break (1 to 4 inches) analyses and the Three Mile Island (TMI)[12] accident data indicate that about 20 minutes after the core uncovers clad temperatures start to reach 1200°F and 10 minutes later they can be as high as 2200°F. These times will shorten as the break size increases due to the core uncovering faster and to a greater depth.

- If the RVLIS indication is between 3.5 ft collapsed liquid level in the core and the top of the core, then the CETCs should be monitored for superheated steam temperatures to determine if the core has uncovered.

As many thermocouples as possible should be used for evaluation of the core temperature conditions. The Emergency Response Guidelines[11] recommend that a minimum of one thermocouple near the center of the core and one in each quadrant be monitored at identified high power assemblies. Caution should be taken if a thermocouple reads greater than 1650°F or is reading considerably different than neighboring CETCs. This may indicate that the thermocouple has failed. Caution should also be used when looking at CETCs near the vessel walls because reflux cooling from the hot legs may cool the fluid in this area. CETCs can also be used as an indicator of hot areas in the core and may be used to determine radial location of possible local core damage.

Therefore, core exit thermocouples and RVLIS are generally regarded as reliable indicators of RCS conditions that may cause core damage. They can predict the time of core uncovery to within a few minutes by monitoring the core exit thermocouples for superheat after RVLIS indicates collapsed liquid level at the top of the core. The onset and extent of fuel damage after core uncovery depend on the heat generation in the fuel and the rapidity and duration of uncovery. However, if the core has not uncovered, no generalized fuel damage has occurred. Core exit thermocouples reading 1300°F or larger indicate the likelihood of clad damage.

3.3 CONTAINMENT RADIATION MONITORS AND CORE DAMAGE

Post accident radiation monitors in nuclear plants can be used to estimate the xenon and krypton concentrations in the containment.

An analysis has been made to correlate these monitor readings in R/hr to estimate gaseous radioactivity concentrations. For this analysis the following assumptions were made:

1. Radiogases released from the fuel are all released to containment.

2. Accidents were considered in which 100% of the noble gases, 52% of noble gases, and 0.3% of the noble gases were released to the containment.

3. Halogens and other fission products are considered not to be significant contributors to the containment monitor readings.

A relation can be developed which describes the gamma ray exposure rate of a detector with time, based on the amount of noble gases released. The exposure rate reading of a detector is dependent on plant specific parameters: the operating power of the core, the efficiency of the monitor, and the volume seen by the monitor. The plant specific response of the detector as a function of time following the accident can be calculated from the instantaneous gamma ray source strengths due to noble gas release, Table 3-2, and the plant characteristics of the detector. The gamma ray source strengths presented in Table 3-2 are based on 100 percent release of the noble gases. To determine the exposure rate of the detector based on 52 percent and 0.3 percent noble gas release, 52 percent and 0.3 percent, respectively, of the gamma ray source strength are used.

Alternately, the energy rates in Mev/watt-sec given in Table 3-2 can be expressed in terms of an instaneous flux by assuming the energy is absorbed in a cm^3 of air. These energy rate values, in Mev/watt-sec-cm^3, when divided by discrete values of Mev/photon and the gamma absorption coefficient for air, μ, considered as a constant (3.5×10^{-5} cm^{-1}), provide values of the photon flux, photons/watt-cm^2-sec, as shown in Table 3-2A. The discrete values of Mev/photon were obtained by using the average values of the energy groups, Mev/gamma, from Table 3-2.

TABLE 3-2

INSTANTANEOUS GAMMA RAY SOURCE STRENGTHS DUE TO A 100 PERCENT RELEASE OF NOBLE GASES AT VARIOUS TIMES FOLLOWING AN ACCIDENT

Energy Group	Source Strength at Time After Release (Mev/watt-sec)				
Mev/gamma	0 Hours	0.5 Hours	1 Hour	2 Hours	8 Hours
0.20 - 0.40	1.2×10^9	3.0×10^8	2.6×10^8	2.4×10^8	2.0×10^8
0.40 - 0.90	1.5×10^9	3.4×10^8	2.6×10^8	1.9×10^8	5.9×10^7
0.90 - 1.35	1.3×10^9	9.4×10^7	6.7×10^7	4.7×10^7	9.8×10^6
1.35 - 1.80	1.8×10^9	3.4×10^8	2.1×10^8	1.4×10^7	2.9×10^7
1.80 - 2.20	1.4×10^9	5.4×10^8	3.6×10^8	2.4×10^8	5.2×10^7
2.20 - 2.60	1.3×10^9	8.5×10^8	7.1×10^8	5.3×10^8	1.1×10^8
2.60 - 3.00	4.0×10^8	6.6×10^6	5.1×10^6	3.5×10^6	5.0×10^5
3.00 - 4.00	3.5×10^8	6.3×10^5	4.5×10^6	2.6×10^6	9.7×10^4
4.00 - 5.00	3.1×10^7	4.4×10^4	3.6×10^2	0	0
5.00 - 6.00	0	0	0	0	0

Mev/gamma	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	1.3×10^8	3.0×10^7	1.5×10^6	0	0
0.40 - 0.90	1.1×10^7	1.5×10^4	1.5×10^4	1.5×10^4	1.4×10^4
0.90 - 1.35	1.8×10^5	0	0	0	0
1.35 - 1.80	5.5×10^5	0	0	0	0
1.80 - 2.20	9.9×10^5	0	0	0	0
2.20 - 2.60	2.0×10^6	0	0	0	0
2.60 - 3.00	8.5×10^3	0	0	0	0
3.00 - 4.00	0	0	0	0	0
4.00 - 5.00	0	0	0	0	0
5.00 - 6.00	0	0	0	0	0

TABLE 3-2A

INSTANTANEOUS GAMMA RAY FLUXES DUE TO 100% RELEASE OF NOBLE GASES AT VARIOUS TIMES FOLLOWING AN ACCIDENT

Energy Group	Photon Flux at Time After Release (photons/cm^2-watt-sec)				
Mev/gamma	0 Hours	0.5 Hours	1 Hour	2 Hours	8 Hours
0.3	1.1×10^{14}	2.7×10^{13}	2.4×10^{13}	2.2×10^{13}	1.8×10^{13}
0.65	1.0×10^{14}	2.3×10^{13}	1.7×10^{13}	1.3×10^{13}	3.9×10^{12}
1.13	3.3×10^{13}	2.4×10^{12}	1.7×10^{12}	1.2×10^{12}	2.5×10^{11}
1.58	3.3×10^{13}	6.2×10^{12}	3.8×10^{12}	2.5×10^{11}	5.3×10^{11}
2.0	2.0×10^{13}	7.7×10^{12}	5.1×10^{12}	3.4×10^{12}	7.4×10^{11}
2.4	1.5×10^{13}	1.0×10^{13}	8.4×10^{12}	6.3×10^{12}	1.3×10^{12}
2.8	4.1×10^{12}	6.7×10^{10}	5.2×10^{10}	3.6×10^{10}	5.1×10^{9}
3.5	2.9×10^{12}	5.3×10^{9}	3.8×10^{10}	2.2×10^{10}	8.1×10^{8}
4.5	1.9×10^{11}	2.8×10^{8}	2.3×10^{6}	0	0

Mev/gamma	1 Day	1 Week	1 Month	6 Months	1 Year
0.3	1.2×10^{13}	2.7×10^{12}	1.4×10^{11}	0	0
0.65	7.3×10^{11}	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}
1.13	4.5×10^{9}	0	0	0	0
1.58	1.0×10^{10}	0	0	0	0
2.0	1.4×10^{10}	0	0	0	0
2.4	2.4×10^{10}	0	0	0	0
2.8	8.7×10^{7}	0	0	0	0
3.5	0	0	0	0	0
4.5	0	0	0	0	0

In general, values below 0.3% releases are indicative of clad failures, values between 0.3% and 52% release are in the fuel pellet overtemperature regions, while values between 52% release and 100% release are in the core melt regime. To represent the release of the normal operating noble gas activity in the primary coolant as obtained from ANS 18.1[6], $1.0 \times 10^{-3}\%$ of the gamma ray source strength is used. In actual practice it must be recognized that there is overlap between the regimes because of the nature in which core heating occurs. The hottest portion of the core is in the center due to flux distribution and hence greater fission product inventory. Additionally heat transfer is greater at the core periphery due to proximity of pressure vessel walls. Thus conditions could exist where there is some molten fuel in the center of the core and overtemperature conditions elsewhere. Similar conditions can occur which lead to overtemperature in the central portions of the core, and clad damage elsewhere. Thus, estimation of extent of core damage with containment radiation readings must be used in a confirmatory sense -- as backup to other measurements of fission product release and other indicators such as pressure vessel water levels and core exit thermocouples.

[Figure 3-3 presents the relationship of the reading (R/hr) of Unit 1 and Unit 2 high range containment area radiation monitors as a function of time following reactor shutdown. Each unit has two high range monitors with one monitor mounted approximately 7 feet above the operating floor between loop 2 and loop 3 steam generator doghouses and the other monitor mounted in the lower compartment on the outside containment wall.]

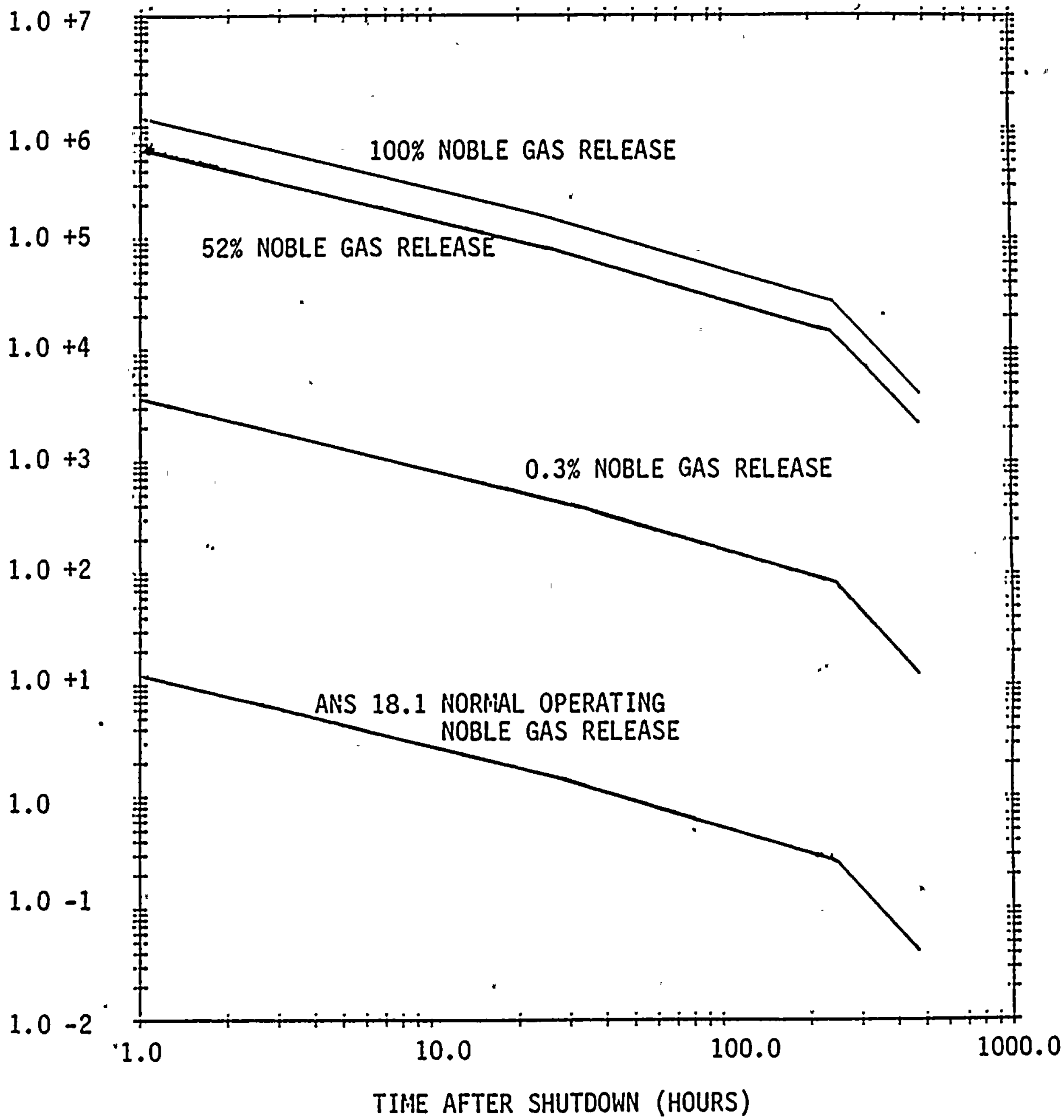


FIGURE 3-3 PERCENT NOBLE GASES IN CONTAINMENT
FOR UNIT 1 AND UNIT 2

4.0 GENERALIZED CORE DAMAGE ASSESSMENT APPROACH

Selected results of various analyses of fission product release, core exit thermocouple readings, pressure vessel water level, containment radiogas monitor readings and hydrogen monitor readings have been summarized in [Table 4-1.] The intent of the summary is to provide a quick look at various criteria intended to define core damage over the broad ranges of:

No Core Damage

0-50%	clad failure
50-100%	clad failure
0-50%	fuel pellet overtemperature
50-100%	fuel pellet overtemperature
0-50%	fuel melt
50-100%	fuel melt

Although this table is intended for generic applicability to most Westinghouse pressurized water reactors, except where noted, various prior calculations are required to ascertain percentage release fractions, power, and containment volume corrections. These corrections are given within the prior text of this technical basis report.

The user should use as many indicators 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/or core melt, considerable judgement needs to be applied.

TABLE 4-1
CHARACTERISTICS OF CATEGORIES OF FUEL DAMAGE*

Core Damage Category	Core Damage Indicator: Percent and Type of Fission Products Released	Fission Product Ratio	Containment Radiogas Monitor (R/hr) 10 hrs after shutdown**	Core Exit Thermocouples Readings (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)*** & Plant Type
No clad damage	Kr-87 $< 1\times10^{-3}$ Xe-133 $< 1\times10^{-3}$ I-131 $< 1\times10^{-3}$ I-133 $< 1\times10^{-3}$	Not 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 - 660	750 - 1300	Core uncovery	0 - 13
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	660 to 1325	1300 - 1650	Core uncovery	13 - 24
0-50% fuel pellet overtemperature	Xe-Kr,Cs,I 1 - 20 Sr-Ba 0 - 0.1	Kr-87 = 0.22 I-133 = 2.1	1325 to 1.7(5)	> 1650	Core uncovery	13 - 24
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	1.7(5) to 3.4(5)	> 1650	Core uncovery	13 - 24
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	3.4(5) to 5.8(5)	> 1650	Core uncovery	13 - 24
50-100% fuel melt	Xe,Kr,Cs,I,Te > 70 Sr,Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	5.8(5)	> 1650	Core uncovery	13 - 24

* This table is intended to supplement the methodology outlined in this report and should not be used without referring to this report and without considerable engineering judgement.
** Values should be revised per times other than 10 hours.
*** Ignitors may obviate these values.
**** $\frac{Kr\text{-}87}{Xe\text{-}133}$, $\frac{I\text{-}133}{I\text{-}131}$

5.0 LIMITATIONS

The emphasis of this methodology is on radiochemical analysis of appropriate liquid and gaseous samples. The assumption has been made that appropriate post-accident systems are in place and functional and that representative samples are obtained. Of particular concern, in the area of representative sampling, is the potential for plateout in the sample lines. In order to preclude such plateout, it is assumed that proper attention to heat tracing of the sample lines and maintenance of sufficient purge velocities is inherent in the sampling system design.

Having obtained a representive sample, radiochemical analysis via gamma spectrometry are used to calculate the specific activity of various fission products released.

Radiochemical analyses of fission products under normal plant operating conditions are accurate to ±10 percent. Radiochemical analyses of post accident samples which may be much more concentrated, and contain unfamiliar nuclides, and which must be performed expeditiously may have an error band of 20 to 50 percent.

Having obtained specific activity analysis, the calculation of total release requires knowledge of the total water volume from which the samples were taken. Care must thus be exercised in accounting for volumes of any water added via ECCS and spray systems, accumulators, chemical addition tanks, and melting ice of ice condenser plants. Additionally estimates of total sump water volumes have to be determined with data from sump level indicators. Such estimates of water volume are probably accurate to ±10 percent.

The specific activity also requires a correction to adjust for the decay of the nuclide in which the measured specific activity is decay corrected to time of reactor shutdown. For some nuclides, precursor effects must be considered in the decay correction calculations. The precursor effect is limited to parent-daughter relationships for this methodology. A major assumption is made that the release percentages of the parent and daughter are equal. For overtemperature and melt releases, this assumption is consistent with the technical basis presented in Sections 2.5 and 2.6, but the gap releases could be different by as much as a factor of 2.

The models used for estimation of fission product release from the gap activity are based on the ANS 5.4 standard. Background material for this report indicate the model, though empirical, is believed to have an accuracy of 20-25 percent. In our application of these models to core wide conditions, the core has arbitrarily been divided into three regions of low, intermediate, and high burnup. This representation predicted nominal values of release with maximum and minimum values that approach ±100 percent of the nominal value. Therefore these estimates of core damage should only be considered accurate to a factor of 2.

The models employed for estimates of release at higher temperature have not been completely verified by experiment. Additionally, calculations of expected core temperatures for severe accident conditions are still being refined. These uncertainties are exacerbated by the manner in which various accident scenarios leading to core melt have been combined to produce fission product release predictions for the core melt condition. Consideration of the melt release estimates shown in Table 2-11 for the refractory nuclides indicate a range of approximately ± 70 percent.

From these considerations it is clear that the combined uncertainties are such that core damage estimates using this methodology are sufficient only to establish major categories of fuel damage. This categorization, and confirmation of subcategorization will require extensive additional analysis for some several days past the accident date.

7.0 REFERENCES

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11. "Westinghouse Owners Group Transmittal of Volume III for High Pressure of Emergency Response Guidelines," O. D. Kingsley, Jr. to D. G. Eisenhut, Letter No. OG83, Section FR-C.1, January 1983.

12. Analysis of the Three Mile Island Accident and Alternative Sequences, Prepared for NRC by Battelle, Columbus Laboratories, NUREG/CR-1219, January 1980.

[13. Westinghouse Owner's Group Post Accident Core Damage Assessment Methodology, Revision 1, March, 1984.]

INDIANA & MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT

UNIT 1 AND UNIT 2

POST ACCIDENT CORE DAMAGE ASSESSMENT

1.0 OBJECTIVE

1.1 The purpose of this procedure is to provide a method to classify and estimate the extent of core damage through measurement of fission products released to the coolant and containment atmosphere together with auxiliary measurements of core exit thermocouple temperature, water level within the pressure vessel, containment radiation monitors, and containment atmosphere hydrogen monitors.

2.0 REFERENCES

2.1 Westinghouse Owner Group Post Accident Core Damage Assessment Methodology, Revision 1, March 1984.

3.0 RESPONSIBILITIES

3.1 The Plant Evaluation Team in the Technical Support Center will be responsible for core damage assessment based on radionuclide analysis and auxiliary measurements.

4.0 APPLICABILITY

4.1 Any plant condition in which the operator would suspect a loss of reactor core cooling or reactor core cooling can no longer be maintained.

4.2 Any plant condition in which the operator would suspect failed fuel, and an estimate of the amount of failed fuel is required.

5.0 INSTRUCTIONS

5.1 Nuclide Sampling

5.1.1 Request samples of reactor coolant, containment atmosphere, and containment sump as indicated in Table 2. Table 1 lists the selected nuclides for core damage assessment.

5.1.2 Analyze the selected samples for isotopic specific activity with no decay correction applied to sample activities.

5.1.3 Complete Table 3A, RCS Activity Worksheet, if sample was available as follows:

5.1.3.1 Record elapsed time from reactor shutdown to sample count.

5.1.3.2 Record specific activities of nuclides in Ci/gm.

5.1.3.3 Determine and record decay correction factor using Table.4, Decay Correction Factor With Parent-Daughter Effect.

5.1.3.4 Determine and record the corrected specific activity by multiplying the measured specific activity by the decay correction factor.

5.1.4 Complete Table 3B, Containment Sump Activity Worksheet, if sample was available, as follows:

5.1.4.1 Record elapse time from reactor shutdown to sample count.

5.1.4.2 Record specific activities of nuclides.

5.1.4.3 Determine and record decay correction factor using Table 4, Decay Correction Factor With Parent-Daughter Effect.

5.1.4.4 Determine and record the corrected specific activity by multiplying the measured specific activity by the decay correction factor.

5.1.5 Complete Table 3C, Containment Atmosphere Activity Worksheet as follows:

5.1.5.1 Record elapse time from reactor shutdown to sample count.

5.1.5.2 Record specific activities of nuclides.

5.1.5.3 Determine and record decay correction factor using Table 4, Decay Correction Factor With Parent-Daughter Effect.

5.1.5.4 Determine and record the corrected specific activity by multiplying the measured specific activity by the decay correction factor.

5.2 Liquid Mass

5.2.1 Estimate the total liquid mass by completing Table 5, Estimate of Total Liquid Mass Worksheet.

5.2.2 If both a RCS sample and a containment sump sample was obtained, an estimate of the RCS water mass and containment water mass is needed. Use Table 6, Estimate of RCS Water Mass

and Containment Water Mass Worksheet to estimate the distribution of the water. Record the RCS mass in Table 3A and the containment mass in Table 3B.

5.2.3 If only one of the liquid samples (RCS or containment sump) was obtained, use the total liquid mass calculated in 5.2.1 as the water mass associated with that sample. Record water in either Table 3A (RCS) or Table 3B (containment sump).

5.3 Containment Volume

5.3.1 Since the containment atmosphere sample is collected at the containment building pressure and the sample volume is not corrected to standard conditions, no adjustment factor is needed to the known containment volume. The known containment volume (3.5×10^{10} cc) is recorded in Table 3C.

5.4 Total Activity Released

5.4.1 RCS

5.4.1.1 Calculate total activity of each nuclide released to the RCS by multiplying the decay corrected specific activity by the RCS mass. Record in Table 3A.

5.4.2 Containment Sump

5.4.2.1 Calculate total activity of each nuclide released to the containment water by multiplying the decay corrected specific activity by the containment water mass. Record in Table 3B.

5.4.3 Containment Atmosphere

5.4.3.1 Calculate total activity of each nuclide released to the containment atmosphere by multiplying the decay corrected specific activity by the containment volume. Record in Table 3C.

5.4.4 Total Activity Released of Each Nuclide

5.4.4.1 Record in Table 7, Total Release Activity/Percent Released, the activity of each nuclide of each sample location.

5.4.4.2 Sum the activities of each nuclide of each sample to determine total activity released of each nuclide. Record in Table 7.

5.5 Total Core Inventory

5.5.1 Power History

5.5.1.1 Record in Table 8, Power Correction Factor, the plant power history during the 30 days prior to shutdown.

5.5.2 Power Correction Factor

5.5.2.1 If power history indicates steady state power level during the 30 days or 4 days (depending on the nuclide) prior to shutdown, use the steady state power correction equation shown in Table 8 to determine power correction factor (PCF). Record in Table 7.

5.5.2.2 If power history indicates fluctuating power levels during the 30 days prior to shutdown, use the transient power correction equation shown in Table 8 to determine power correction factor (PCF). Record in Table 7.

5.5.2.3 To determine the power correction factor for Cs-134 first determine the average power during the entire operating period during the cycle prior to shutdown. Use this average power and Figure 4 to estimate power correction factor. Record in Table 7.

5.5.3 Adjusted Core Inventory

5.5.3.1 Determine and record in Table 7 the adjusted core inventory for each nuclide by multiplying the equilibrium full-power inventory (listed in Table 7) by the power correction factor.

5.6 Estimation of Percent Fuel Damage

5.6.1 Determine the percentage of the corrected core inventory released of each nuclide by dividing the total activity released by the corrected core inventory. Record in Table 7.

5.6.2 Using the appropriate core damage graphs, Figures 5 through 17, determine the percent clad failure, fuel overtemperature, and fuel melt as a function of the nuclide release percentage. Use the curve that best represents core burnup. Record the percentages of clad damage, fuel overtemperature, and fuel melt in Table 10, Core Damage Assessment Evaluation Sheet.

<u>Note</u>: Iodine spiking should be considered for cases where the assessment is between no fuel damage and minor clad failure. If percent clad failure is not in agreement with values obtained from other nuclides, spiking may have occurred. Refer to Figure 8 if this is the case.

5.7 Nuclide Activity Ratios

5.7.1 Determine the activity ratios for noble gases and iodines by completing Table 11, Nuclide Activity Ratios.

5.7.2 Compare the calculated activity ratios with the gap activity ratios and fuel pellet ratios listed in Table 11. Calculated activity ratios less than gap activity ratios are indicative of clad failures. Calculated activity ratios greater than gap activity ratios are indicative of more severe failures (fuel overheat and fuel melt).

5.7.3 Record in Table 10 the calculated core damage state.

5.8 Auxiliary Indicators

5.8.1 Determine from reactor vessel level instrumentation or other sources if at any time the core became uncovered. No uncovery is indicative of no fuel damage, and core uncovery is indicative of all core damage states. Record uncovery history in Table 10.

5.8.2 Obtain core exit thermocouple readings and compare these values with those listed in Table 12. Based on Table 12, Characteristics of Categories of Fuel Damage, record temperature in Table 10 under appropriate core damage state.

5.8.3 Obtain containment hydrogen concentration. Compare hydrogen concentration hydrogen concentration under appropriate core damage state.

5.8.4 Use hydrogen concentration with Figure 18 to determine extent of zirconium-water reaction. Record percentage of zirconium water reaction in Table 10.

<u>Note</u>: If ignitors have been activated or a burn has been indicated, quantitative use of the hydrogen concentration is limited. It can be assumed that for ignition of hydrogen to occur a minimal concentration of 4 percent hydrogen is needed. This assumption can

be used qualitatively to indicate that some percentage of zirconium has reacted, but it is difficult to determine extent of the reaction.

5.8.5 Obtain the containment high range area radiation monitor readings and the time after shutdown the readings were obtained. Compare the readings with Figure 19 to estimate the corresponding extent of core damage. Record the monitor reading in Table 10 under the appropriate core damage state.

5.9 Core Damage Assessment

5.9.1 Perform the final core damage assessment by evaluating the data in Table 10. It is unlikely that complete agreement between the indicators will result in the same estimate of core damage. The evaluation should be the best estimate based on all parameters, their interrelationship, and engineering judgment.

The user should use as many indicators 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/or core melt, considerable judgement needs to be applied.

TABLE 1

SELECTED NUCLIDES FOR CORE DAMAGE ASSESSMENT

Core Damage State	Nuclide	Half-Life*	Predominant Gammas (Kev) Yield (%)*
Clad Failure	Kr-85m**	4.4 h	150(74), 305(13)
	Kr-87	76 m	403(84), 2570(35)
	Kr-88**	2.8 h	191(35), 850(23), 2400(35)
	Xe-131m	11.8 d	164(2)
	Xe-133	5.27 d	81(37)
	Xe-133m**	2.26 d	233 (14)
	Xe-135**	9.14 h	250(91)
	I-131	8.05 d	364(82)
	I-132	2.26 h	773(89), 955(22), 1400(14)
	I-133	20.3 h	530(90)
	I-135	6.68 h	1140(37), 1280(34), 1460(12), 1720(19)
	Rb-88	17.8 m	898(13), 1863(21)
Fuel Overheat	Cs-134	2 yr	605(98), 796(99)
	Cs-137	30 yr	662(85)
	Te-129	68.7 m	455(15)
	Te-132	77.7 h	230(90)
Fuel Melt	Sr-89	52.7 d	(beta emitter)
	Sr-90**	28 yr	(beta emitter)
	Ba-140	12.8 d	537(34)
	La-140	40.22 h	487(40), 815(19), 1596(96)
	La-142	92.5 m	650(48), 1910(9), 2410(15), 2550(11)
	Pr-144	17.27 m	695(1.5)

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* Values obtained from Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition.

** These nuclides are marginal with respect to selection criteria for candidate nuclides; they have been included on the possibility that they may be detected and thus utilized in a manner analogous to the candidate nuclides.

TABLE 2

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.

TABLE 3A

RCS ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/gms	Decay Correction Factor	Corrected Specific Activity Ci/gm	RCS Mass gms	RCS Activity Ci
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131						
I 132						
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137						
Te 129						
Te 132						
Ba 140						
La 140						
La 142						
Pr 144						

TABLE 3B

CONTAINMENT SUMP ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/gms	Decay Correction Factor	Corrected Specific Activity Ci/gm	Containment Water Mass gms	Containment Water Activity Ci
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131						
I 132						
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137						
Te 129						
Te 132						
Ba 140						
La 140						
La 142						
Pr 144						

TABLE 3C

CONTAINMENT ATMOSPHERE ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/cc	Decay Correction Factor	Corrected Specific Activity Ci/cc	Containment Volume cc	Containment Activity Ci
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131						
I 132						
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137						
Te 129						
Te 132						
Ba 140						
La 140						
La 142						
Pr 144						

TABLE 4

DECAY CORRECTION FACTOR*

WITH PARENT-DAUGHTER EFFECT

Nuclide	Correction Factor
Kr 85m	$e^{0.158t}$
Kr 87	$e^{0.547t}$
Kr 88	$e^{0.248t}$
Xe 131m	$1/-2.66e^{(-3.59E-3)t} + 3.66e^{(-2.45E-3)t}$
Xe 133	$1/-0.187e^{(-3.41E-2)t} - 0.10e^{(-5.48E-3)t} + 1.287e^{(-1.28E-2)t}$
Xe 133m	$1/-0.10e^{(-3.41E-2)t} + 1.11e^{(-1.28E-2)t}$
Xe 135	$1/-9.14e^{(-1.04E-1)t} -0.033e^{(-2.67)t} + 10.17e^{(-7.58E-2)t}$
I 131	$e^{(3.59E-3)t}$
I 132	$1/1.03e^{(-8.92E-3)t} -0.03e^{(-3.07E-1)t}$
I 133	$e^{(3.41E-2)t}$
I 135	$e^{0.104t}$
Rb 88	$1/1.10e^{(-0.248)t} -0.10e^{(-2.34)t}$
Cs 134	1.0
Cs 134	1.0
Te 129	$1/1.09e^{(-0.161)t} + 0.167e^{(-8.47E-4)t} - 0.257e^{(0.605)t}$
Te 132	$e^{(8.92E-3)t}$
Ba 140	$e^{(2.26E-3)t}$
La 140	$1/1.08e^{(-2.26E-3)t} - 0.08e^{(-1.72E-2)t}$
La 142	$1/-0.145e^{(-3.78)t} + 1.145e^{(-0.450)t}$
Pr 144	$1/0.909e^{(-1.02E-4)t} + 0.091e^{(-2.41)t}$

*Time, t, is the number of hours between shutdown and time of sample count.

TABLE 5

ESTIMATE OF TOTAL LIQUID MASS

1. Estimate the volume added for the following:

Tank	Estimated Volume Added	Maximum Volume Added (gallons)
a. Refueling Water Storage Tank	__________	372,250
b. Accumulator A	__________	7,263
c. Accumulator B	__________	7,263
d. Accumulator C	__________	7,263
e. Accumulator D	__________	7,263
f. Boron Injection Tank	__________	900
g. Spray Additive Tank	__________	4,000
h. Other source __________	__________	
	__________ Total	
i. Melted Ice	Estimated Mass Added	Maximum Mass Added (lbm)
	__________	2.7×10^6

2. Convert estimated volume added from gallons to grams.

Added volume:

________________, gallons x 3785 gms/gal = ________________ gms

3. Convert ice melted mass from lbm to grains

________________, lbm x 454 grams/lbm = ______________ gms

4. The average Reactor Coolant System Mass is 2.40×10^8 gms.

5. Determine the Total Liquid Mass as follows:

Mass added ______________ gms + melted ice mass ________________ gms
+ RCS mass 2.4×10^8 gms = __________________ gms

TABLE 6

ESTIMATE OF RCS WATER MASS* AND CONTAINMENT WATER MASS

AVERAGE OPERATING RCS VOLUME = 11,780 ft^3

1. Record the reactor vessel level, pressurizer level, and RCS temperature at time when sample was taken.

   Reactor vessel level = ________________%
   Pressurizer level = ________________%
   RCS temperature = ________________°F

2. Determine RCS volume at time of sample by estimating from level indications the percentage of water in the RCS.

   ____________ ft^3 x ____________% ÷ 100 = ____________ ft^3

3. Determine RCS specific gravity from Figure 1.

   RCS specific gravity =

4. Determine RCS mass as follows:

   $$\text{RCS volume } (ft^3) \times \text{specific gravity} \times \frac{1.0g}{cc} \times \frac{28.3 \times 10^3 cc}{ft^3}$$

   $$\_\_\_\_\_\_ \ ft^3 \times \_\_\_\_\_\_ \times \frac{1.0g}{cc} \times \frac{28.3 \times 10^3 cc}{ft^3} = \_\_\_\_\_\_\_\_ g$$

5. Record the Containment Sump level indication and the containment level indication.

   Containment Sump Level = ________________%
   Containment Level = ________________%

TABLE 6 (Continued)

ESTIMATE OF RCS WATER MASS* AND CONTAINMENT WATER MASS

AVERAGE OPERATING RCS VOLUME = 11,780 ft^3

6. Determine containment water volume from Figures 2 and 3 using the levels from Step 5.

   Note: If sump level indicates sump is full use Figure 3.

   Containment Water Volume = ______________ ft^3

7. Determine containment water specific gravity from Figure 1.

   Containment water specific activity = ____________

8. Determine containment water mass as follows:

$$\text{Containment water volume} \times \text{specific gravity} \times \frac{1.0\ \text{gm}}{\text{cc}} \times \frac{28.3 \times 10^3\ \text{cc}}{\text{ft}^3}$$

$$\_\_\_\_\_\_\_\_\ \text{ft}^3 \times \_\_\_\_\_\_\_\_\_\_ \times \frac{1.0\ \text{gm}}{\text{cc}} \times \frac{28.3 \times 10^3\ \text{cc}}{\text{ft}^3} = \_\_\_\_\_\_\ \text{gms}$$

*If a reactor vessel level indication is not available or is consider inaccurate based on engineering judgments subtract the estimated containment water mass from the estimated total water mass (Table 5) to determine RCS water mass.

Total Water Mass __________gms - containment water mass ___________ gms
= RCS mass ___________gms

TABLE 7A

TOTAL RELEASE ACTIVITY/PERCENT RELEASED - UNIT 1

Nuclide	RCS Activity Ci	Containment Sump Activity Ci	Containment Atmosphere Activity Ci	Total Activity Ci	Equilibrium Core Inventory* Ci	Power Correction Factor	Corrected Core Inventory Ci	Release Percentage* %
Kr 85m					2.0 (7)			
KR 87					3.6 (7)			
Kr 88					5.2 (7)			
Xe 131m					5.7 (5)			
Xe 133					1.8 (8)			
Xe 133m					2.5 (7)			
Xe 135					3.4 (7)			
I 131					8.9 (7)			
I 132					1.3 (8)			
I 133					1.8 (8)			
I 135					1.6 (8)			
Rb 88					5.3 (7)			
Cs 134					2.1 (7)			
Cs 137					1.0 (7)			
Te 129					3.0 (7)			
Te 132					1.3 (8)			
Ba 140					1.5 (8)			
La 140					1.6 (8)			
La 142					1.4 (8)			
Pr 144					1.1 (8)			

* 2.0 (7) = 2.0×10^7. This notation is used throughout the procedure.

**Release Percentage = $\frac{\text{Total Activity}}{\text{Corrected Core Inventory}} \times 100$

TABLE 7B

TOTAL RELEASE ACTIVITY/PERCENT RELEASED - UNIT 2

Nuclide	RCS Activity Ci	Containment Sump Activity Ci	Containment Atmosphere Activity Ci	Total Activity Ci	Equilibrium Core Inventory* Ci	Power Correction Factor	Corrected Core Inventory Ci	Release Percentage* %
Kr 85m					2.1 (7)			
Kr 87					3.8 (7)			
Kr 88					5.4 (7)			
Xe 131m					6.0 (5)			
Xe 133					1.9 (8)			
Xe 133m					2.7 (7)			
Xe 135					3.5 (7)			
I 131					9.3 (7)			
I 132					1.3 (8)			
I 133					1.9 (8)			
I 135					1.7 (8)			
Rb 88					5.5 (7)			
Cs 134					2.2 (7)			
Cs 137					1.0 (7)			
Te 129					3.1 (7)			
Te 132					1.3 (8)			
Ba 140					1.6 (8)			
La 140					1.7 (8)			
La 142					1.4 (8)			
Pr 144					1.1 (8)			

$$\text{**Release Percentage} = \frac{\text{Total Activity}}{\text{Corrected Core Inventory}} \times 100$$

TABLE 8

POWER HISTORY OF 30 DAYS PRIOR TO SHUTDOWN

Interval j	Average Power Level* P_j	Operating Period at P_j t_j, hours	Period Between end of t_j and Reactor Shutdown t_j, hours

Power Correction Factor (PCF)**

Steady-State Power Condition PCF	Transient Power Condition PCF
I. Half-Life of Nuclide < 1 Day	
$\frac{\text{Average Power Level (MWt) for prior 4 days}}{\text{Rated Power Level (Mwt)}}$	$\frac{\sum_j P_j (1 - e^{-\lambda_j t_j})\, e^{-\lambda_i t_j^\circ}}{\text{Rated Power Level (MWt)}}$
II. Half-Life of Nuclide > 1 Day	
$\frac{\text{Average Power Level (MWt) for prior 30 days}}{\text{Rated Power Level (Mwt)}}$	$\frac{\sum_j P_j (1 - e^{-\lambda_j t_j})\, e^{-\lambda_i t_j^\circ}}{\text{Rated Power Level (MWt)}}$
III. Half-Life of Nuclide $\simeq$ 1 Year	
$\frac{\text{Average Power Level (MWt) for prior 1 year}}{\text{Rated Power Level (MWt)}}$	$\frac{\text{Effective Full Power Days (EFPD)}}{\text{Total Calendar Days of Cycle Operation}}$

* Average Power Level is defined as the power level at which the power level does not vary more than ±10 percent of the rated power level from the time averaged value.

**λ_i = decay constant in hours^{-1} of each nuclide. λ_i of each nuclide is listed in Table 9.

TABLE 9

DECAY CONSTANTS (λ_i) OF EACH NUCLIDE

Nuclide	Half-Life	λ_i, hours^{-1}
Kr 85m	4.4 h	0.158
Kr 87	76 m	0.547
Kr 88	2.8 h	0.248
Xe 131m	11.8d	2.45(-3)
Xe 133	5.27d	5.48(-3)
Xe 133m	2.26d	1.28(-2)
Xe 135	9.14h	7.58(-2)
I 131	8.05d	3.59(-3)
I 132	2.26h	0.307
I 133	20.3 h	3.41(-2)
I 135	6.68 h	0.104
Rb 88	17.8 m	2.34
Cs 134	2 yr	3.96(-5)
Cs 137	30 yr	2.64(-6)
Te 129	68.6 m	0.605
Te 132	77.7 h	8.92(-3)
Ba 140	12.8 d	2.26(-3)
La 140	40.22 h	1.72(-2)
La 142	92.5 m	0.450
Pr 144	17.27 m	2.41

TABLE 10

CORE DAMAGE ASSESSMENT EVALUATION SHEET

Indicator	Percent Clad Damage		Percent Overtemperature		Percent Fuel Melt	
	$\leq$ 50%	$\geq$ 50%	$\leq$ 50%	$\geq$ 50%	$\leq$ 50%	$\geq$ 50%
Radionuclide Analysis						
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131						
I 132						
I 133						
I 135						
Cs 134						
Cs 137						
Te 129						
Te 132						
Ba 140						
La 140						
La 142						
Pr 144						
Ratios						
Kr 85m/Xe 133						
Kr 87/Xe 133						
Kr 88/Xe 133						
Xe 131m/Xe 133						

TABLE 10 (Continued)

CORE DAMAGE ASSESSMENT EVALUATION SHEET

Indicator	Percent Clad Damage		Percent Overtemperature		Percent Fuel Melt	
	≤ 50%	≥ 50%	≤ 50%	≥ 50%	≤ 50%	≥ 50%
Ratio (Con't)						
Xe 133m/Xe 133						
Xe 135/Xe 133						
I 132/I 131						
I 133/I 131						
I 135/I 131						
Auxiliary Indicators						
Core Uncovered						
Core Exit Temp °F						
Containment H_2%						
Zirc - Water Reaction %						
Ignitors On?						
High Range Containment Monitor Reading R/hr						

TABLE 11

NUCLIDE ACTIVITY RATIOS

Nuclide	Gap Activity Ratio	Fuel Pellet Activity Ratio	Calculated Activity Ratio*
Kr 85m	0.022	0.11	
Kr 87	0.022	0.22	
Kr 88	0.045	0.29	
Xe 131m	0.004	0.004	
Xe 133	1.0	1.0	
Xe 133m	0.096	0.14	
Xe 135	0.051	0.19	
I 131	1.0	1.0	
I 132	0.17	1.5	
I 133	0.71	2.1	
I 135	0.39	1.9	

$$\text{*Noble Gas Ratio} = \frac{\text{Noble Gas Nuclide Released (Ci)}}{\text{Xe-133 Released (Ci)}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Nuclide Released (Ci)}}{\text{I-131 Released (Ci)}}$$

TABLE 12
CHARACTERISTICS OF CATEGORIES OF FUEL DAMAGE*

Core Damage Category	Core Damage Indicator: Percent and Type of Fission Products Released	Fission Product Ratio	Containment Radiogas Monitor (R/hr) 10 hrs after shutdown**	Core Exit Thermocouples Readings (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)*** & Plant Type
No clad damage	Kr-87 $< 1 \times 10^{-3}$ Xe-133 $< 1 \times 10^{-3}$ I-131 $< 1 \times 10^{-3}$ I-133 $< 1 \times 10^{-3}$	Not 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 - 660	750 - 1300	Core uncovery	0 - 13
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	660 to 1325	1300 - 1650	Core uncovery	13 - 24
0-50% fuel pellet overtemperature	Xe-Kr,Cs,I 1 - 20 Sr-Ba 0 - 0.1	Kr-87 = 0.22 I-133 = 2.1	1325 to 1.7(5)	> 1650	Core uncovery	13 - 24
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	1.7(5) to 3.4(5)	> 1650	Core uncovery	13 - 24
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	3.4(5) to 5.8(5)	> 1650	Core uncovery	13 - 24
50-100% fuel melt	Xe,Kr,Cs,I,Te > 70 Sr,Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	5.8(5)	> 1650	Core uncovery	13 - 24

* This table is intended to supplement the methodology outlined in this report and should not be used without referring to this report and without considerable engineering judgement.
** Values should be revised per times other than 10 hours.
*** Ignitors may obviate these values.
**** $\frac{\text{Kr-87}}{\text{Xe-133}}$, $\frac{\text{I-133}}{\text{I-131}}$

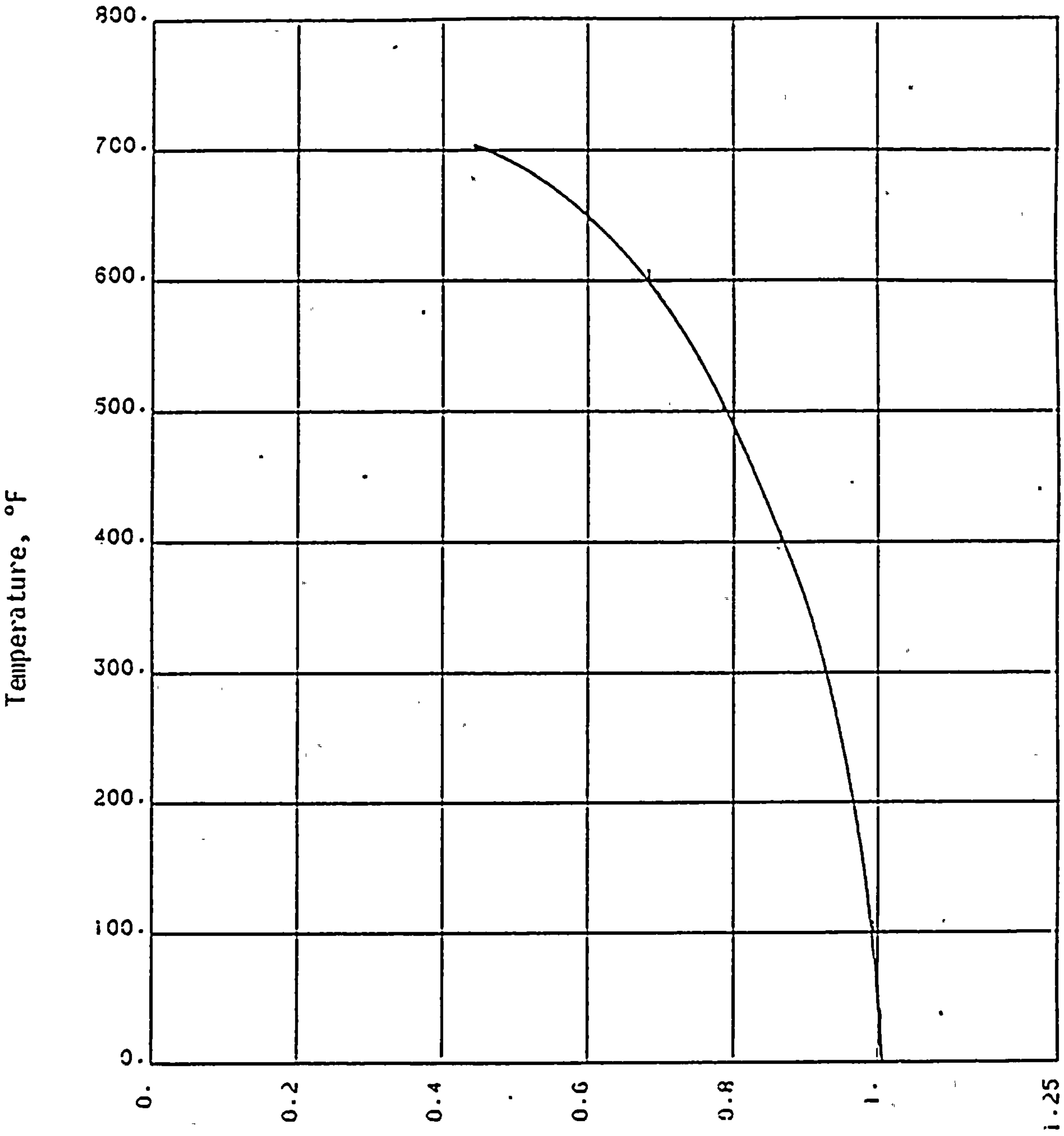

FIGURE 1 WATER DENSITY RATIO (TEMPERATURE VS. STP)

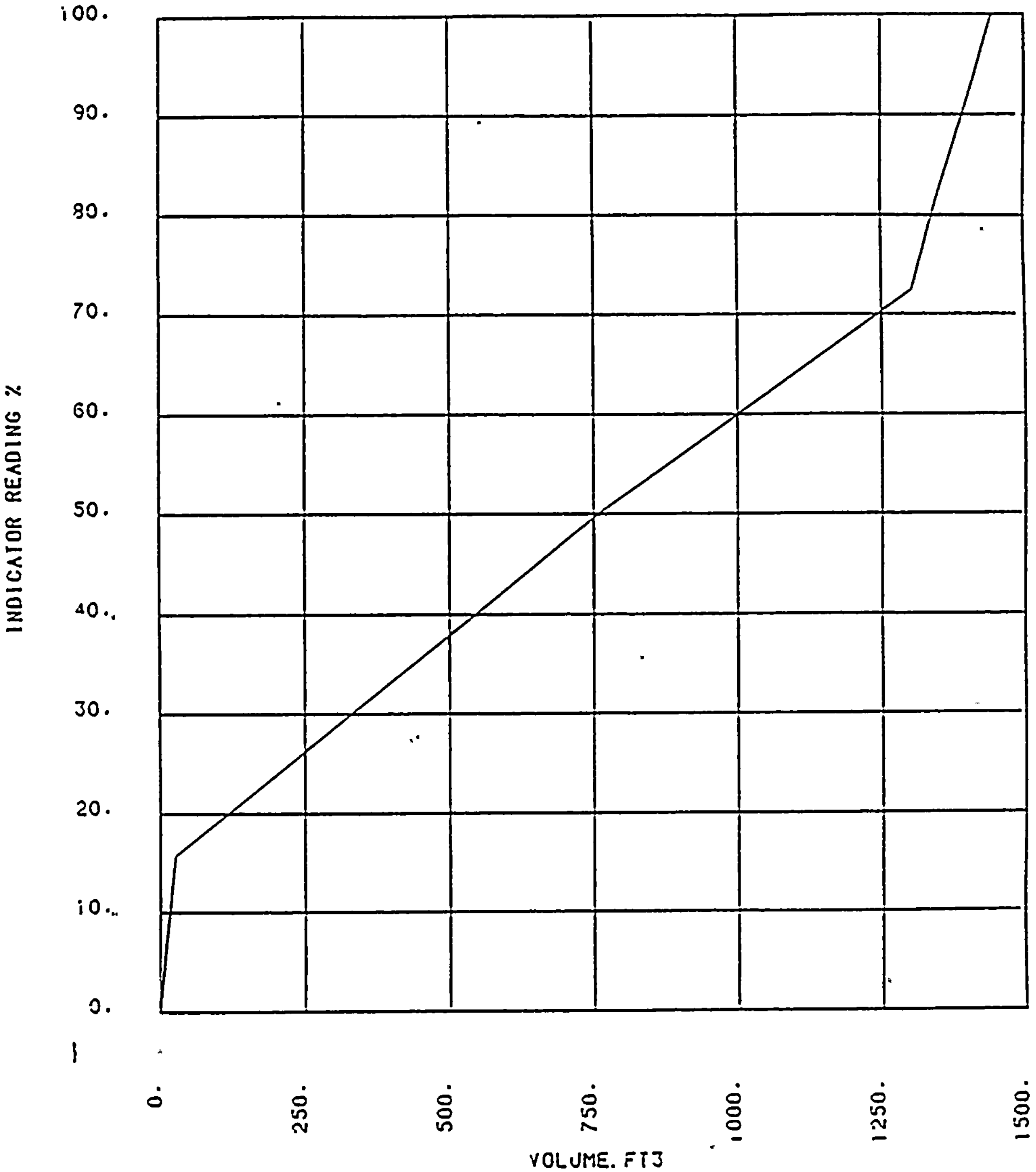

FIGURE 2 SUMP WATER VOLUME VERSUS SUMP LEVEL INDICATION

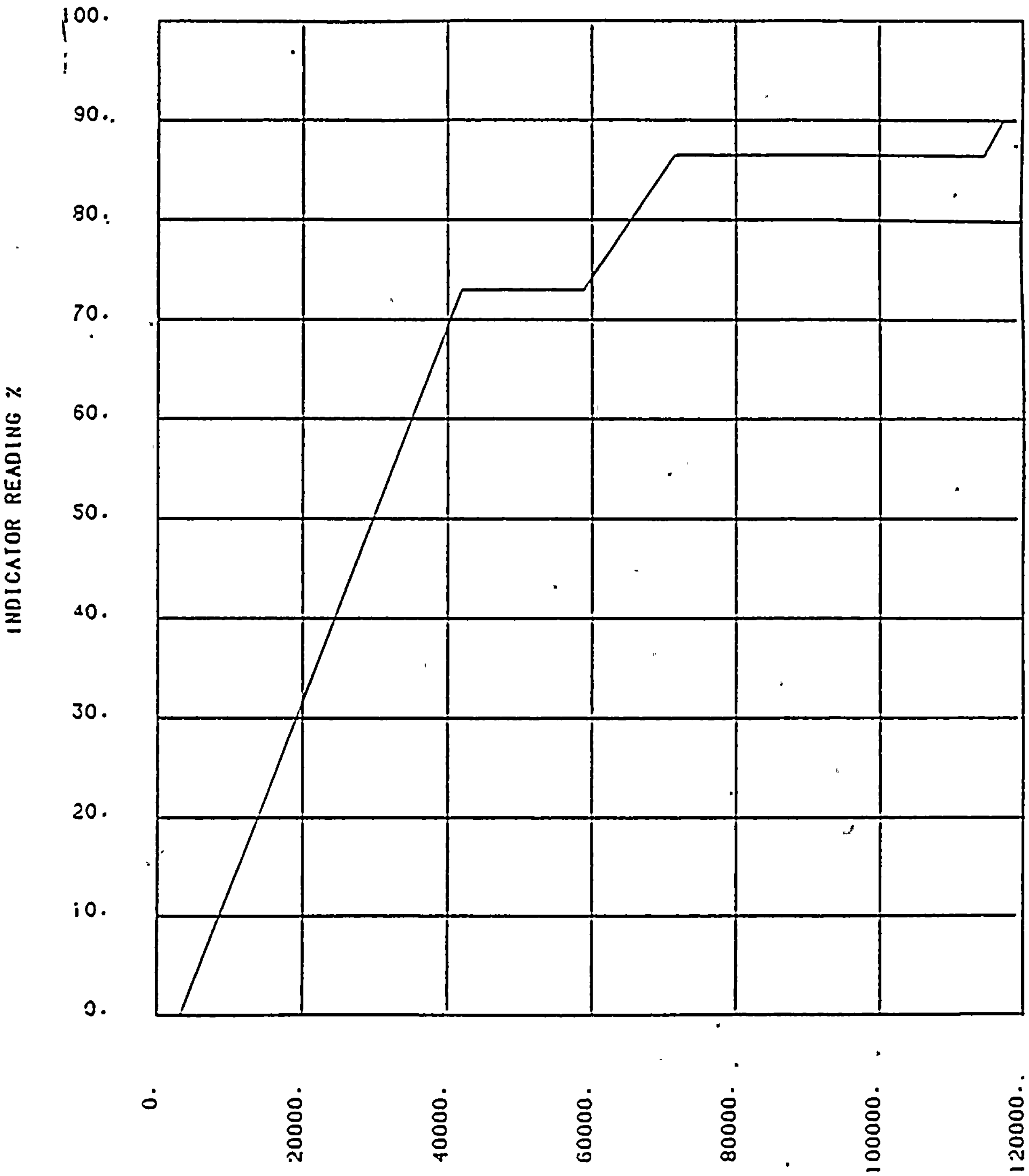

FIGURE 3 CONTAINMENT WATER VOLUME VERSUS CONTAINMENT LEVEL INDICATION

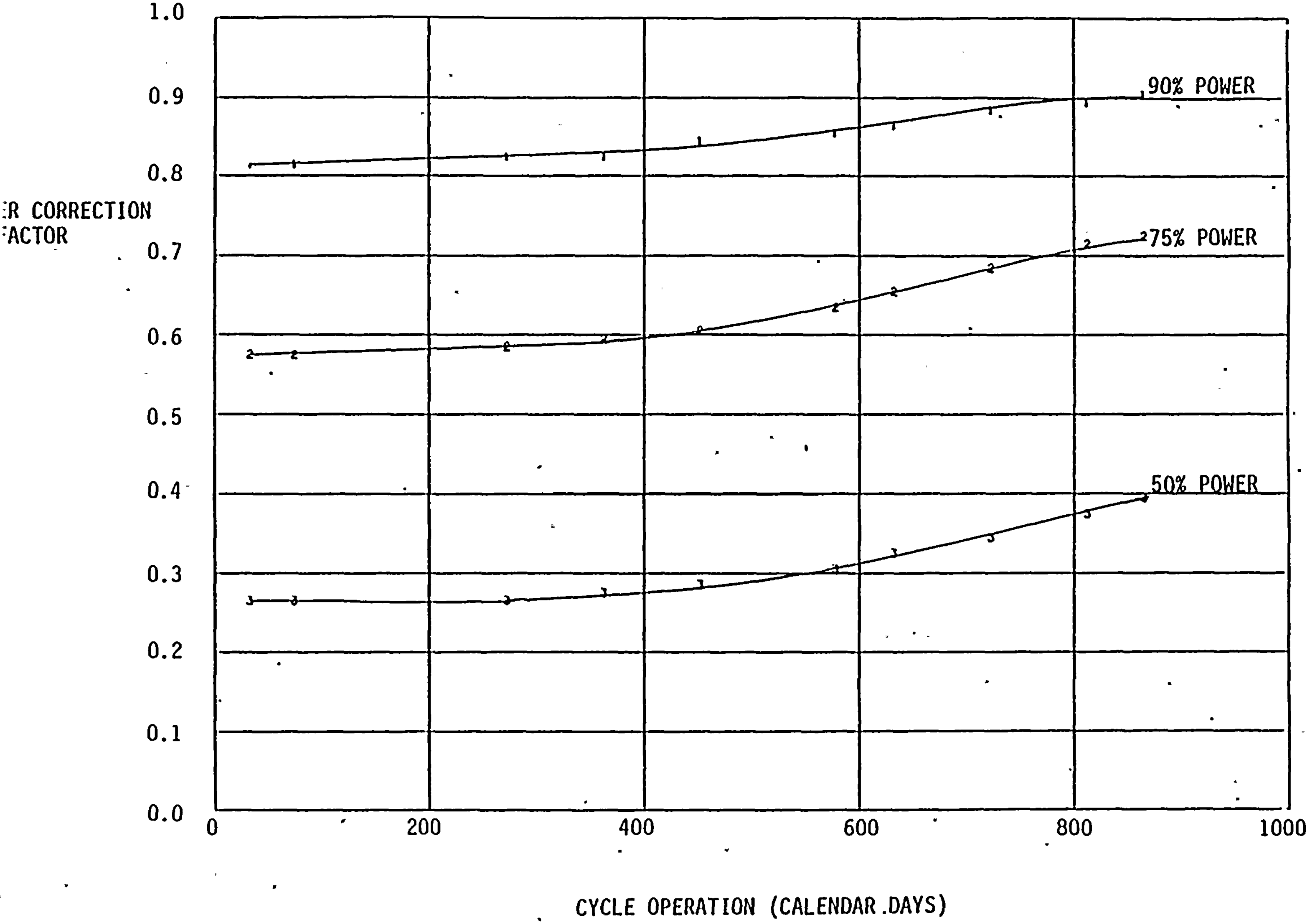

FIGURE 4 POWER CORRECTION FACTOR FOR CS-134 BASED ON AVERAGE POWER DURING OPERATION

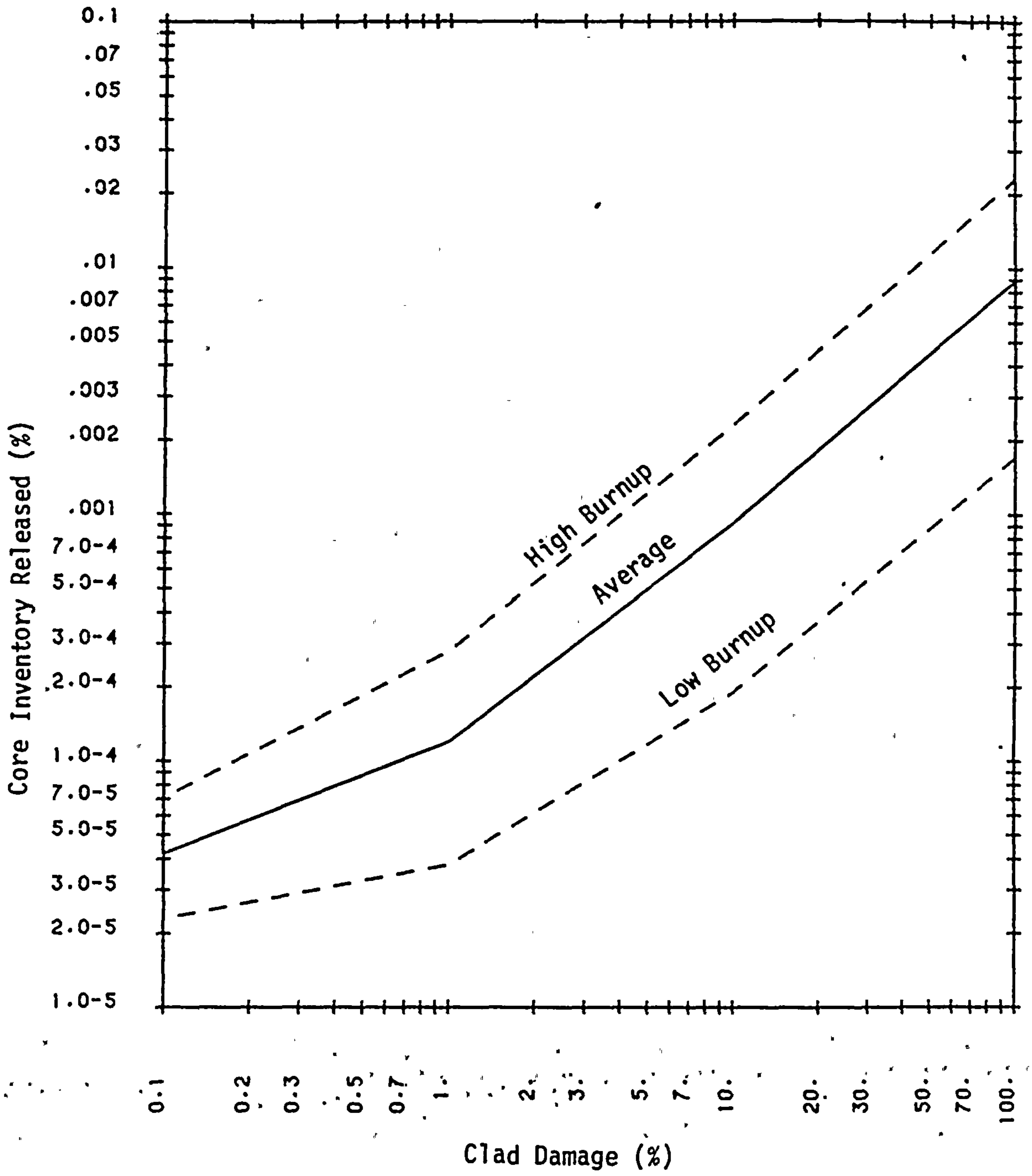

FIGURE 5 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF KR-87

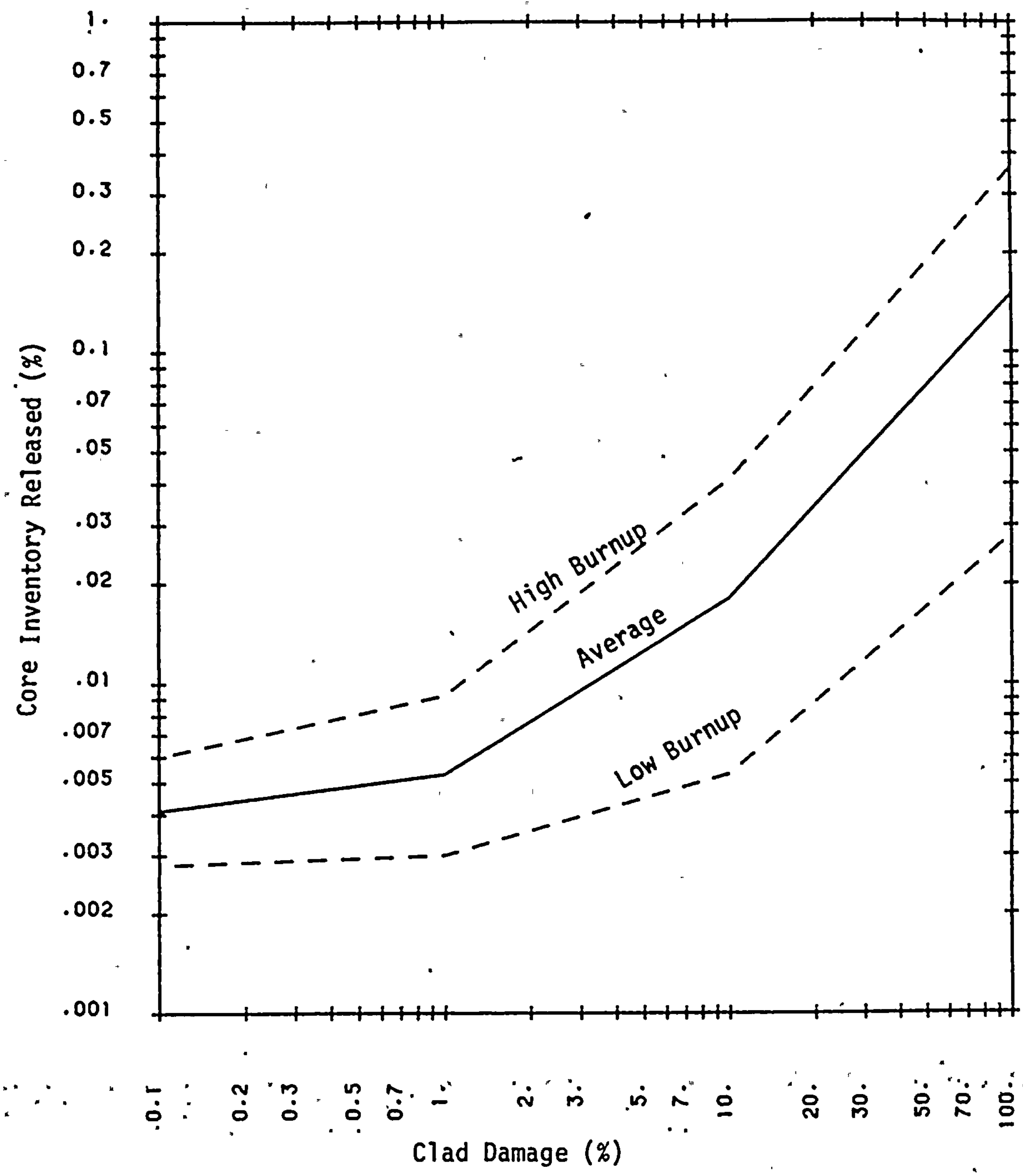

FIGURE 6 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF XE-131M

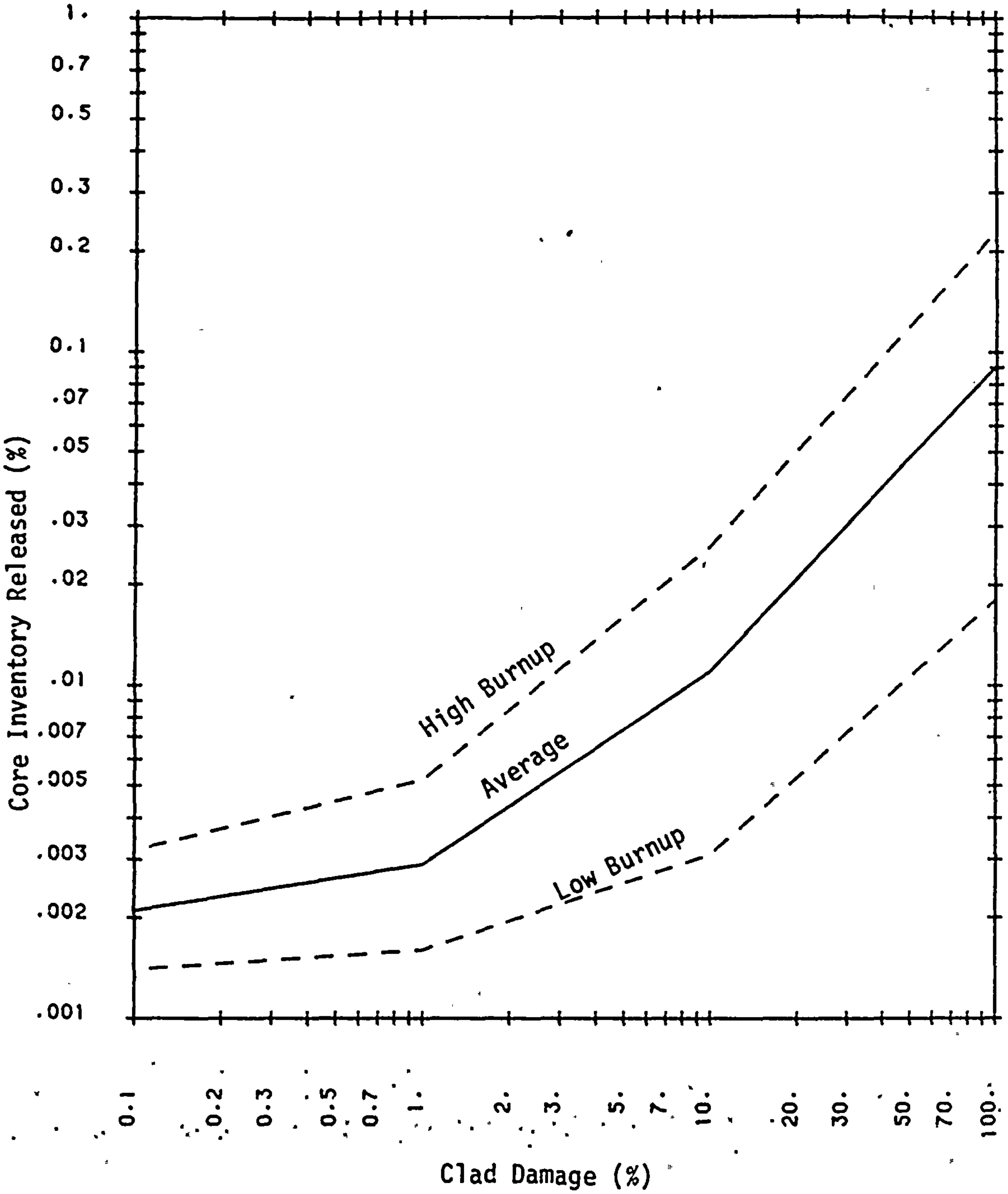

FIGURE 7 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF XE-133

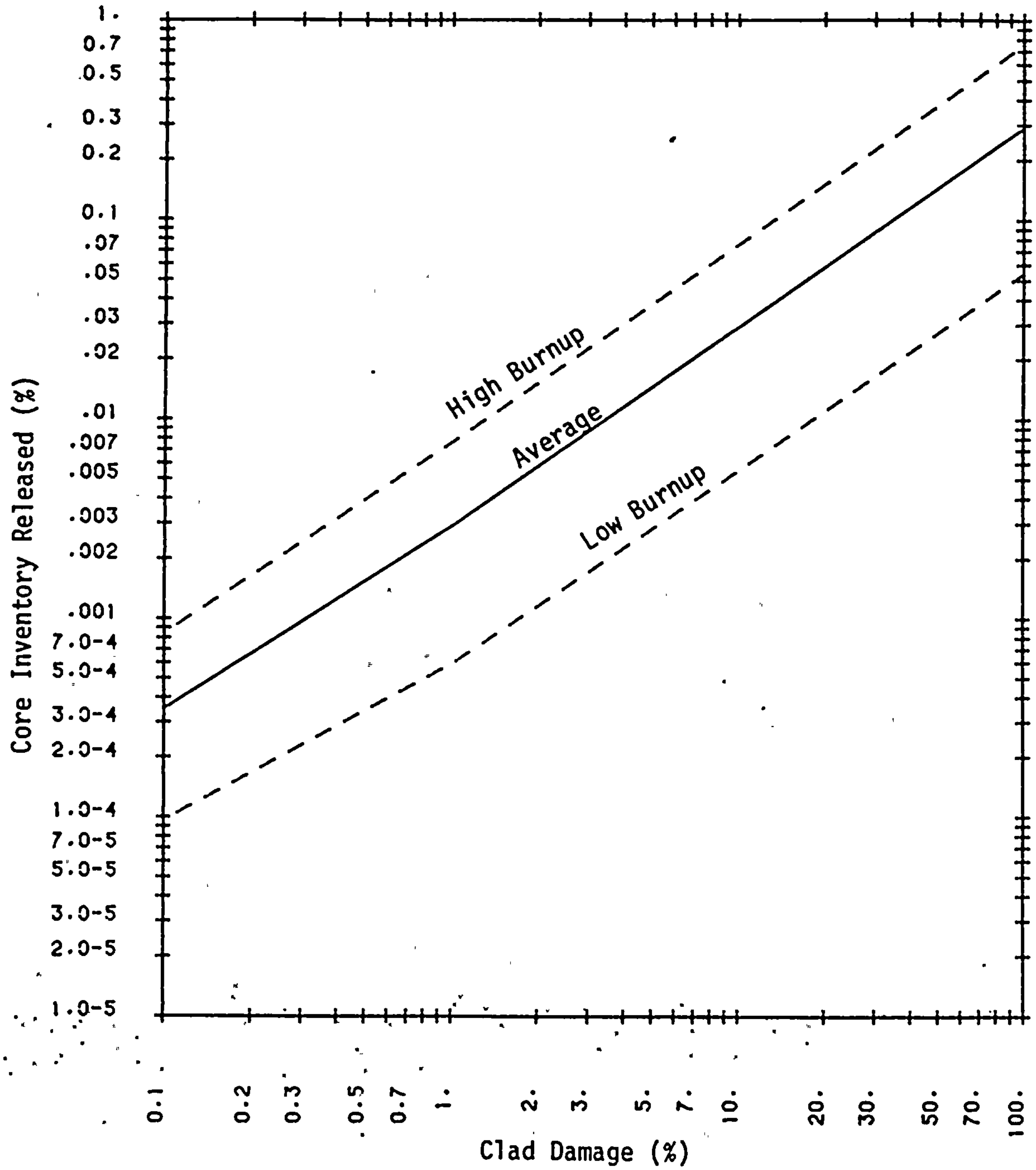

FIGURE 8 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-131

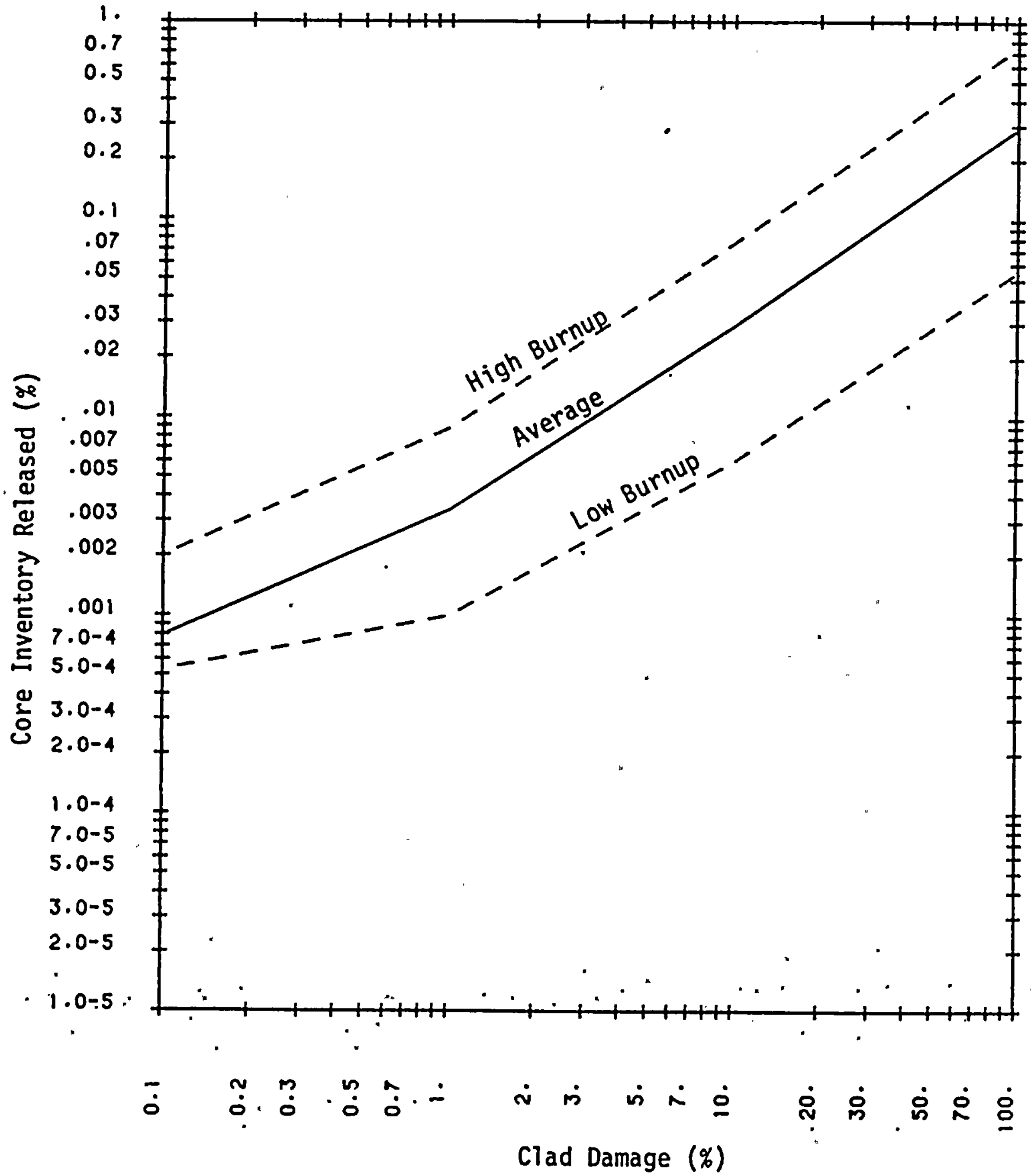

FIGURE 9 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-131 WITH SPIKING

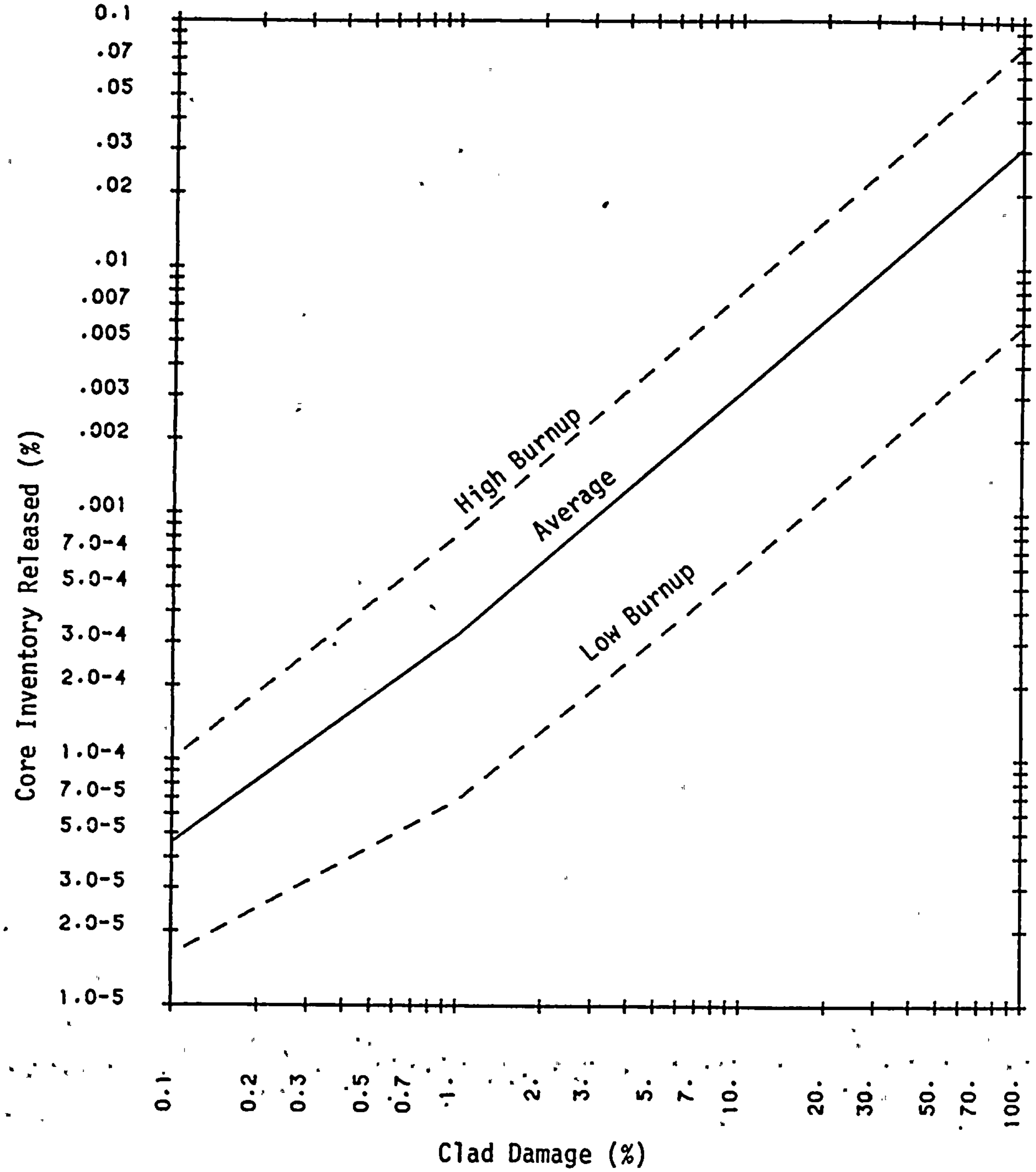

FIGURE 10 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-132

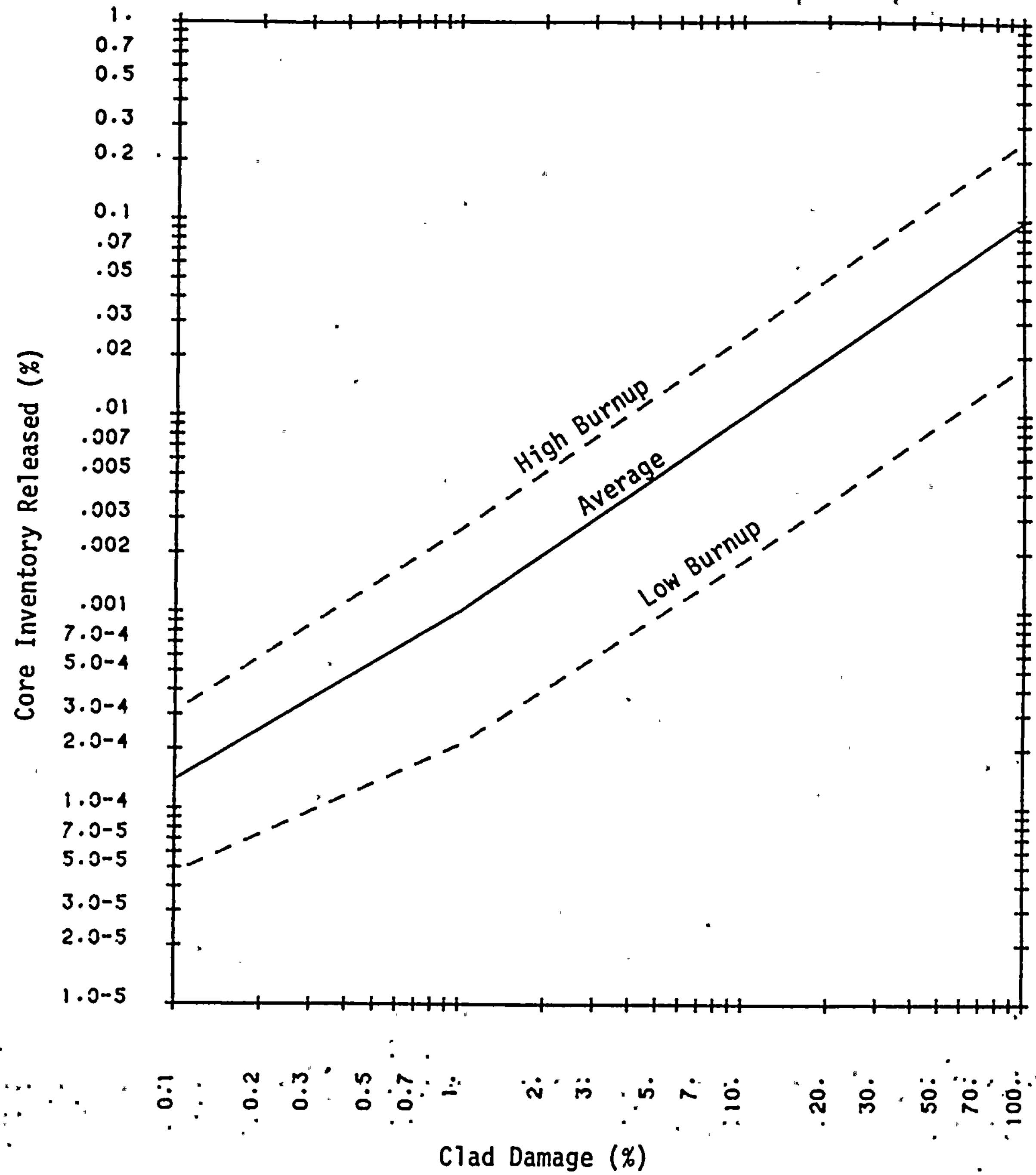

FIGURE 11 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-133

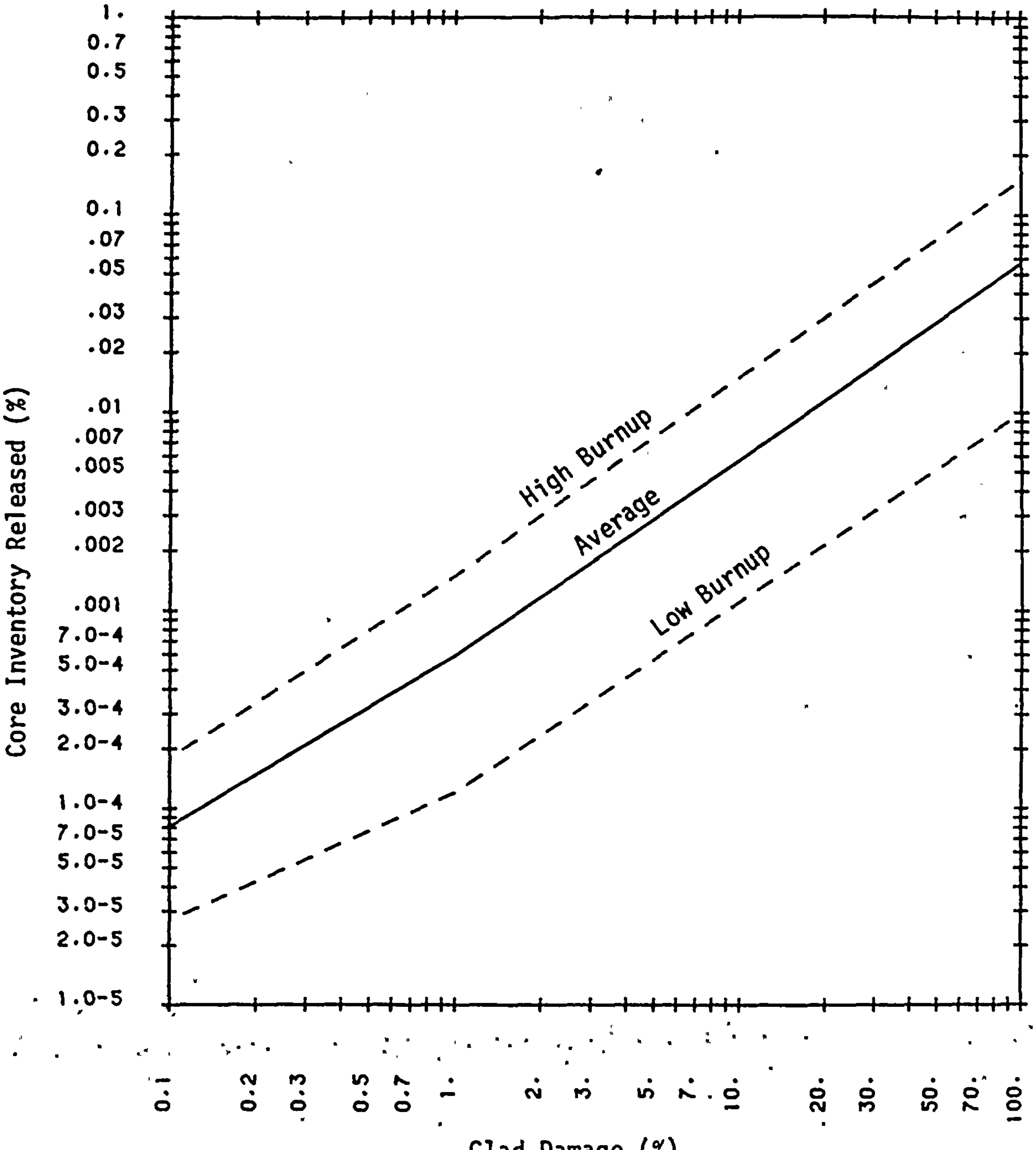

FIGURE 12 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-135

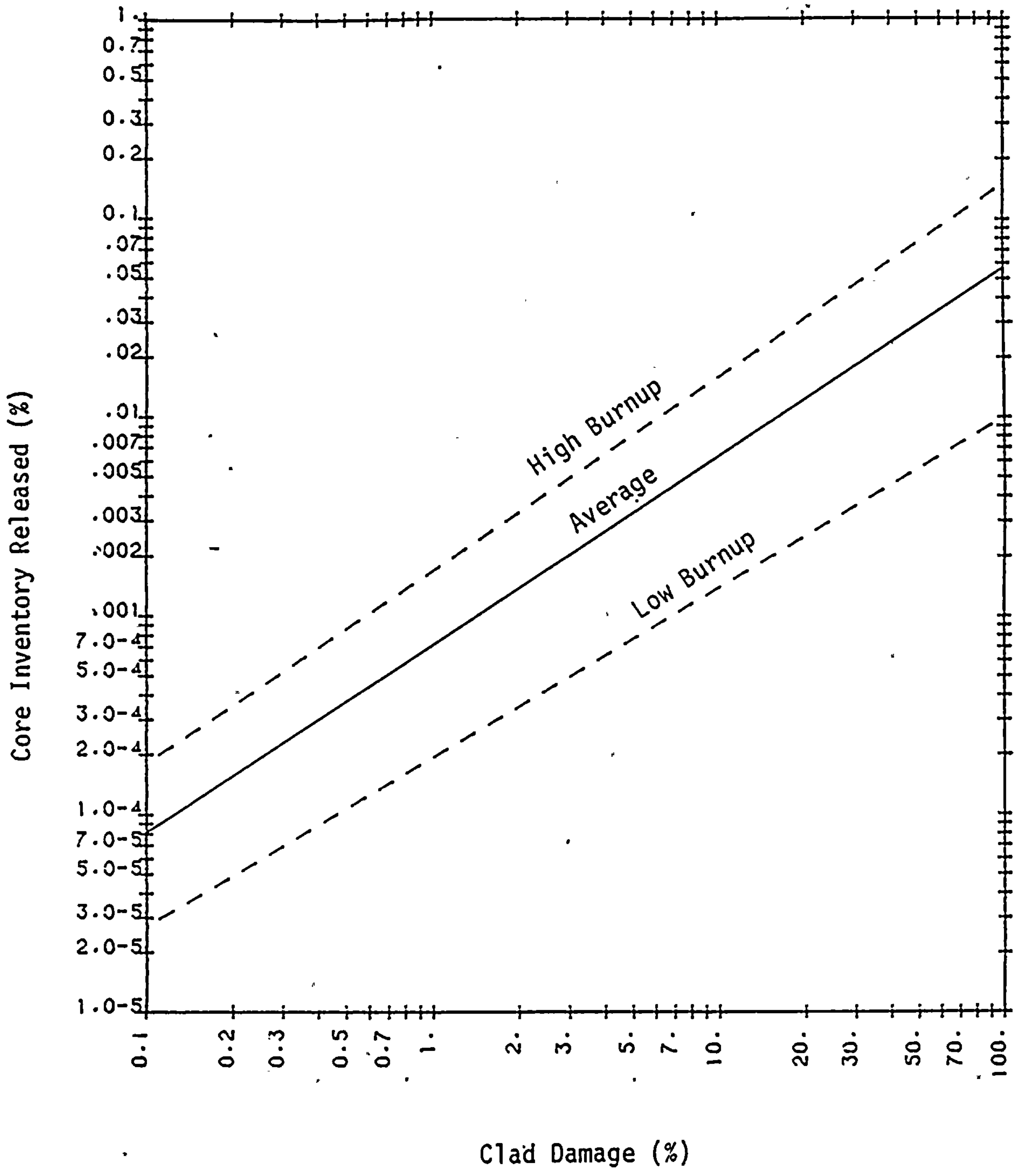

FIGURE 12 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-135

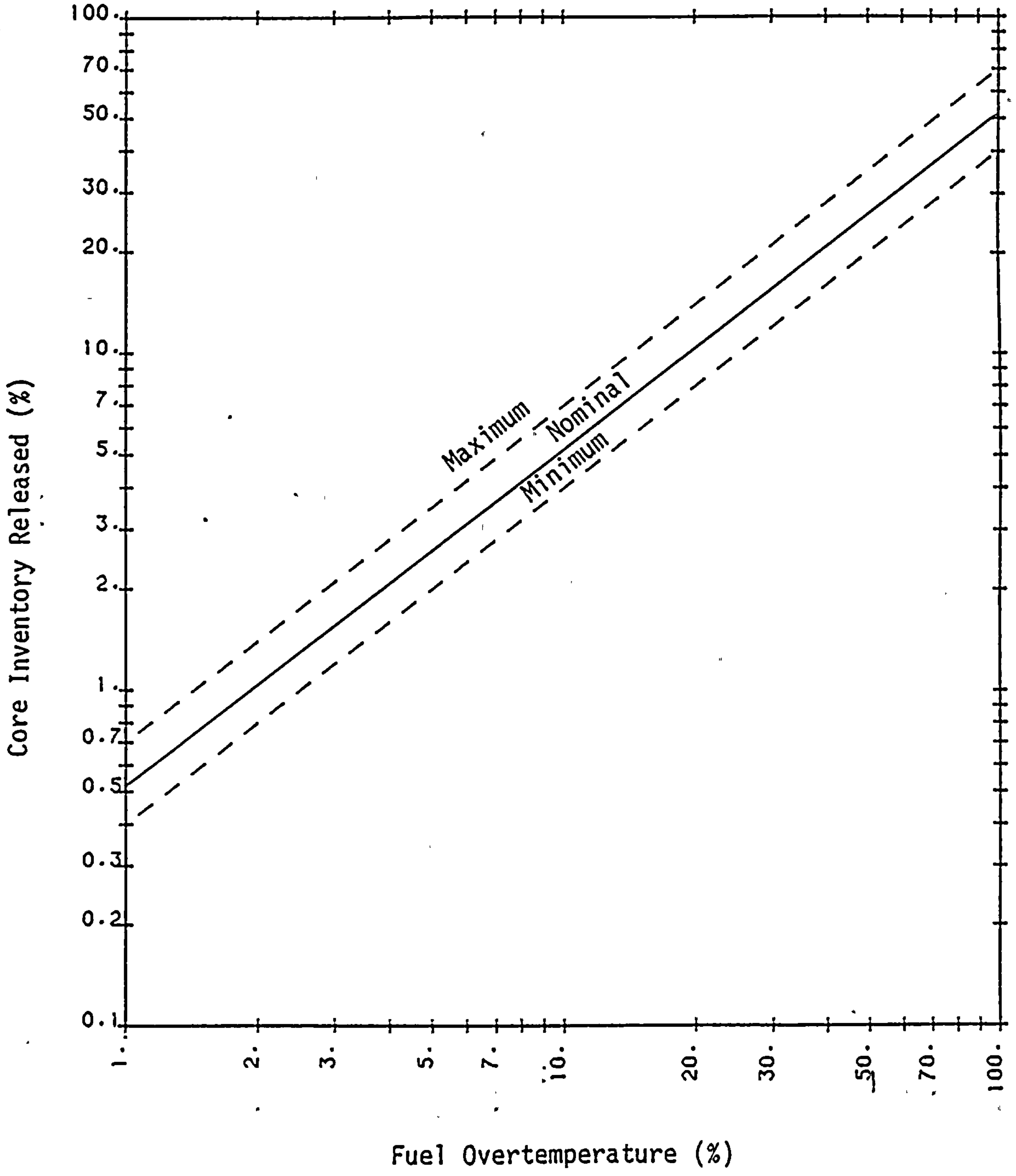

FIGURE 13 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF XE, KR, I, OR CS

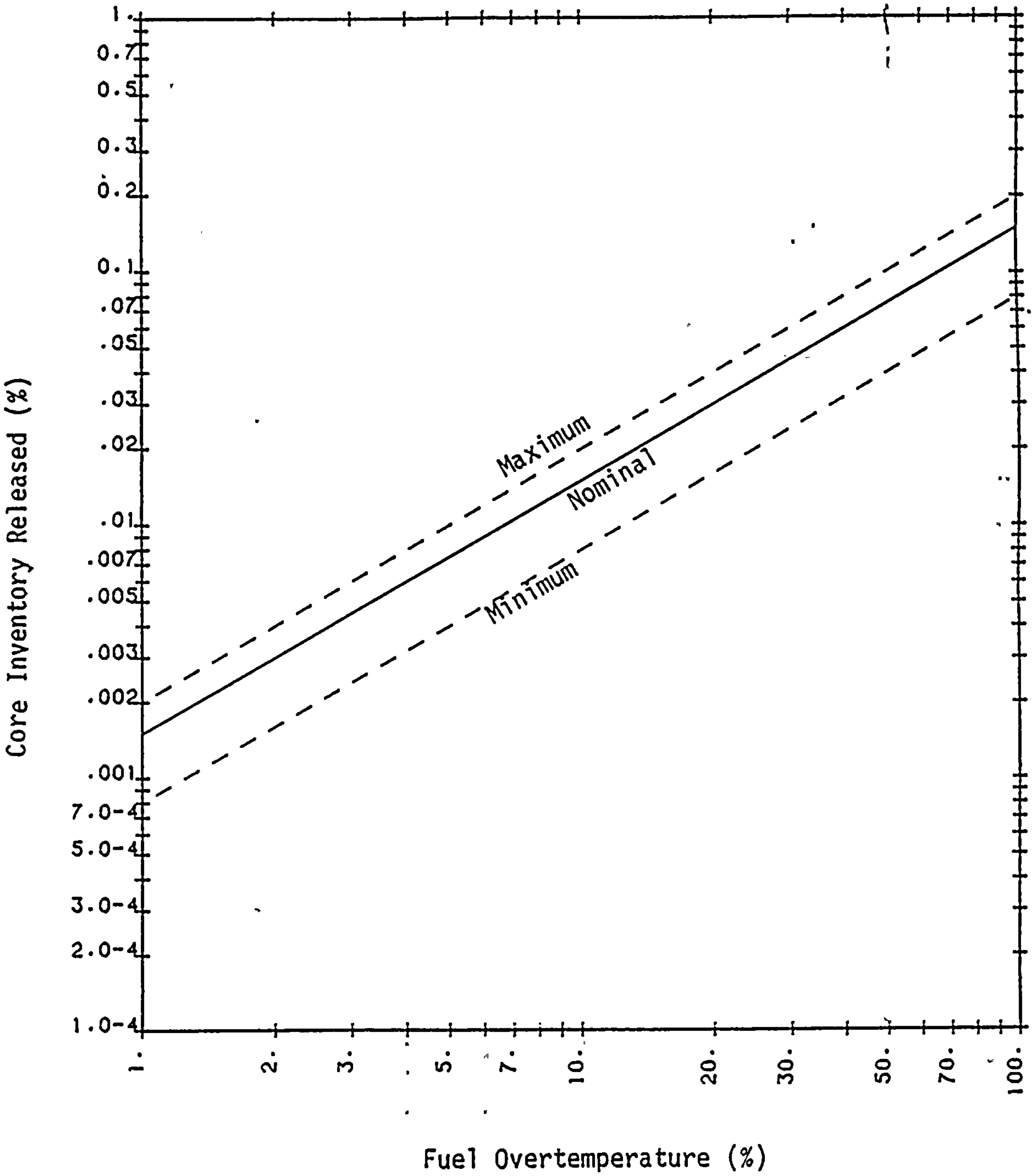

FIGURE 14 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF BA OR SR

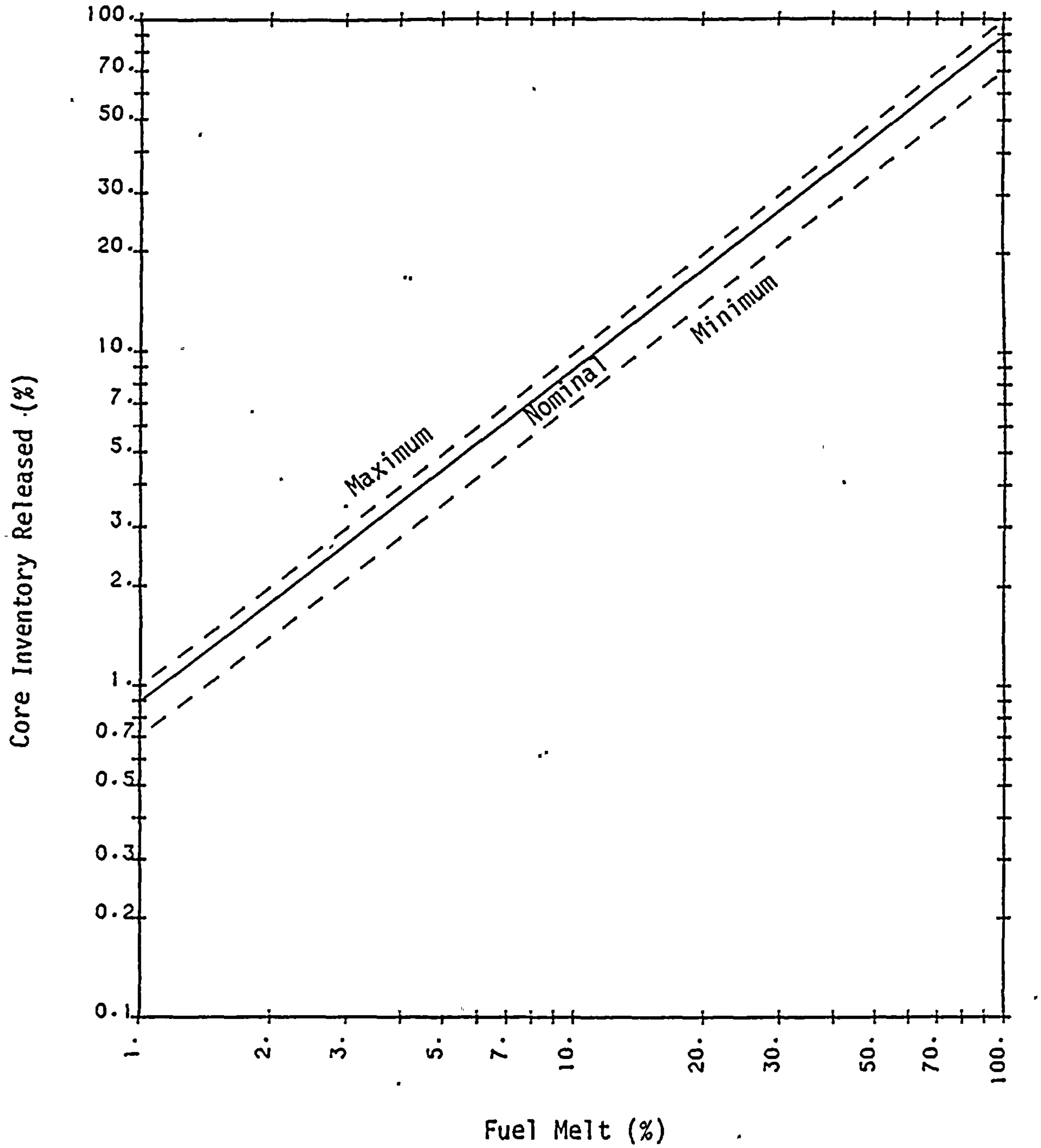

FIGURE 15 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF XE, KR, I, CS, OR TE

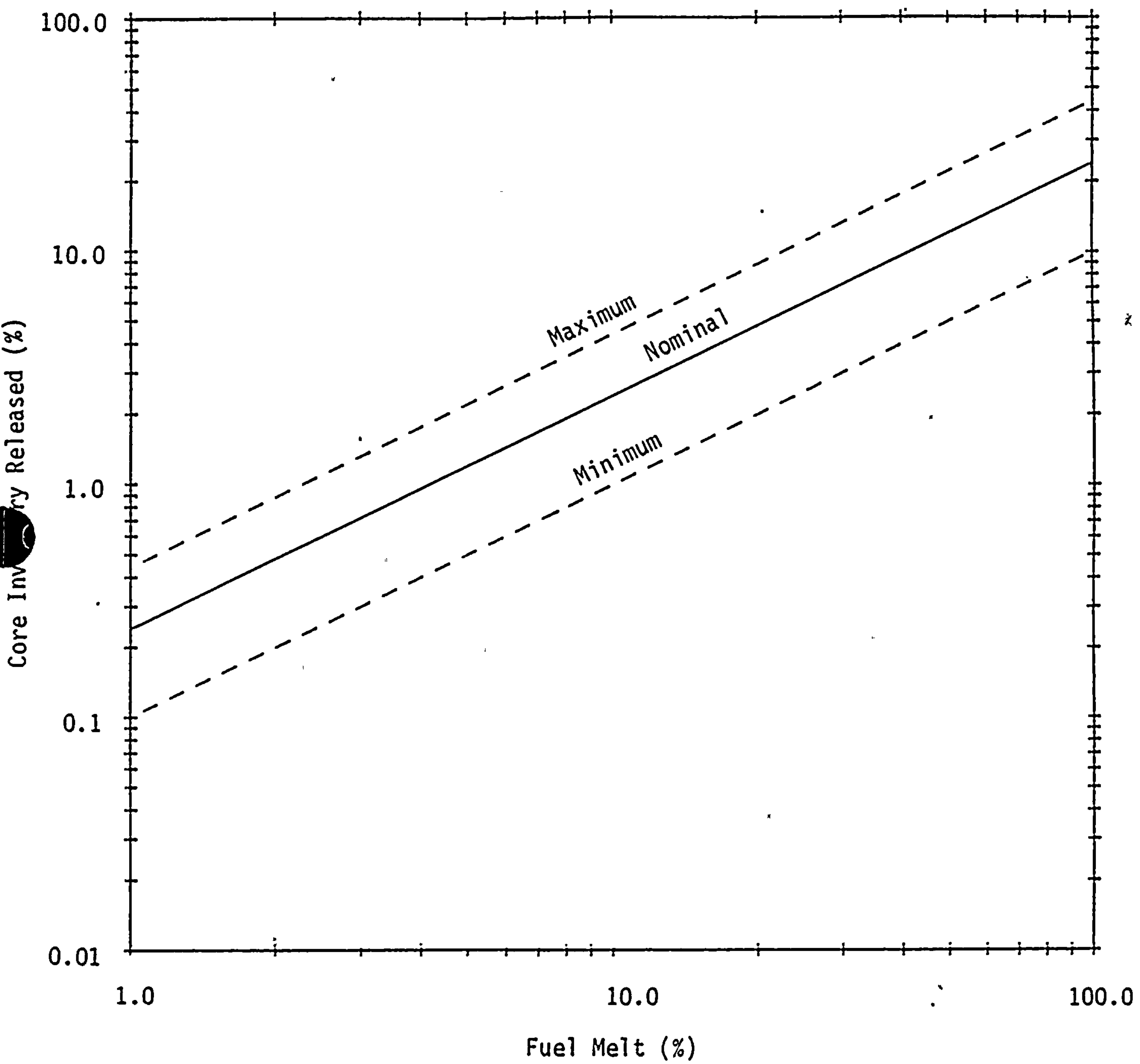

FIGURE 16 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF BA OR SR

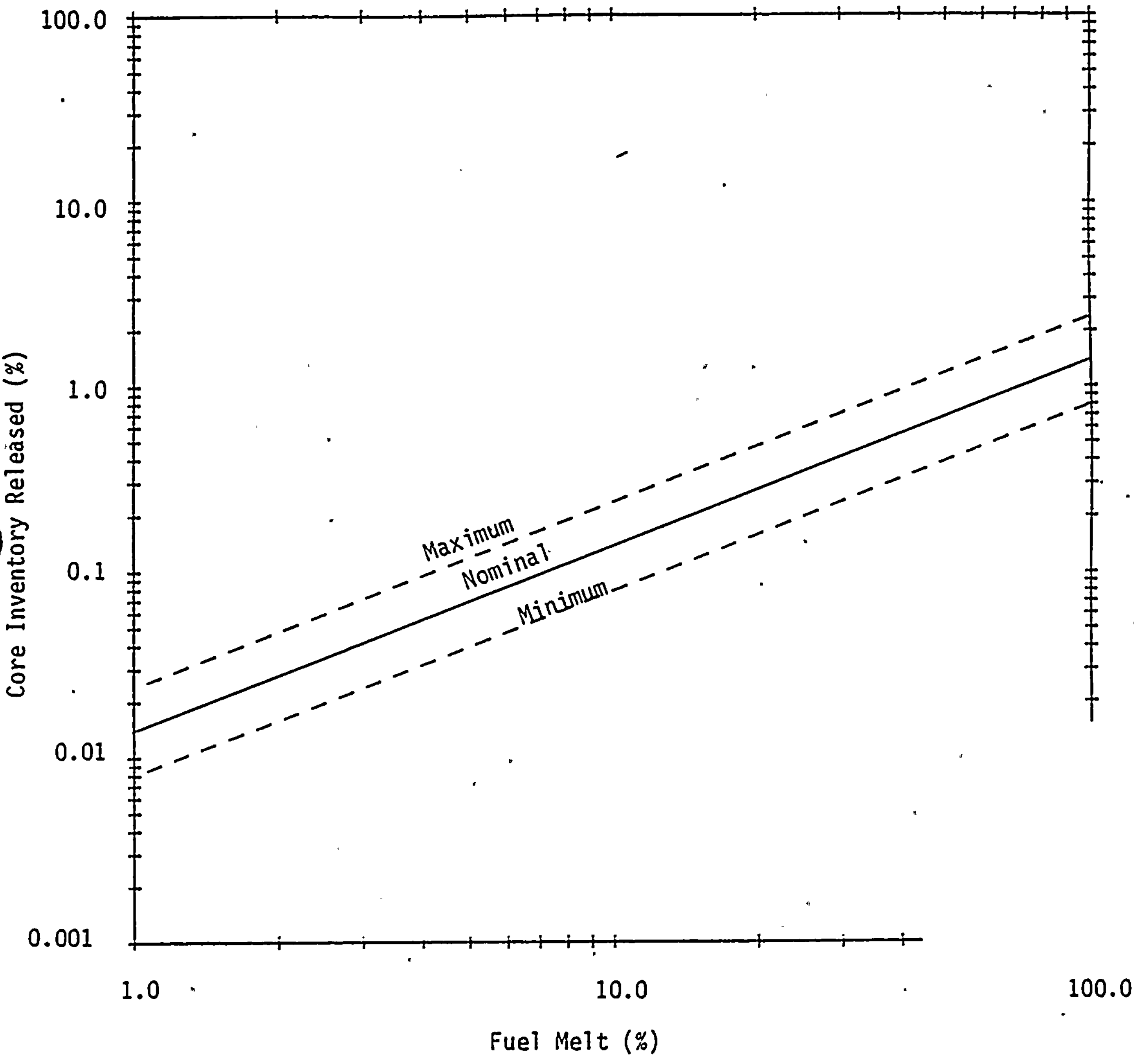

FIGURE 17 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF PR

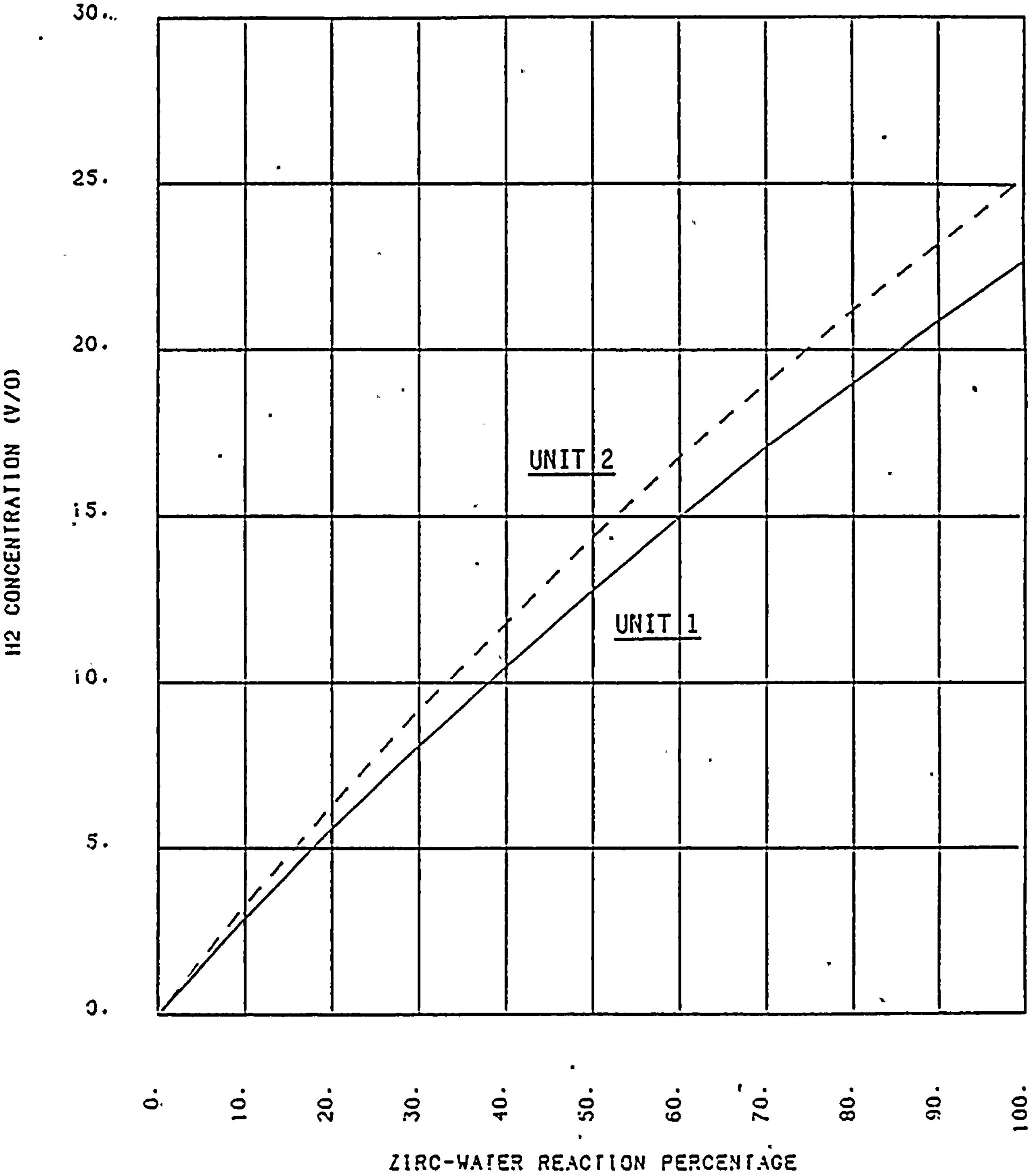

FIGURE 18 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM WATER REACTION

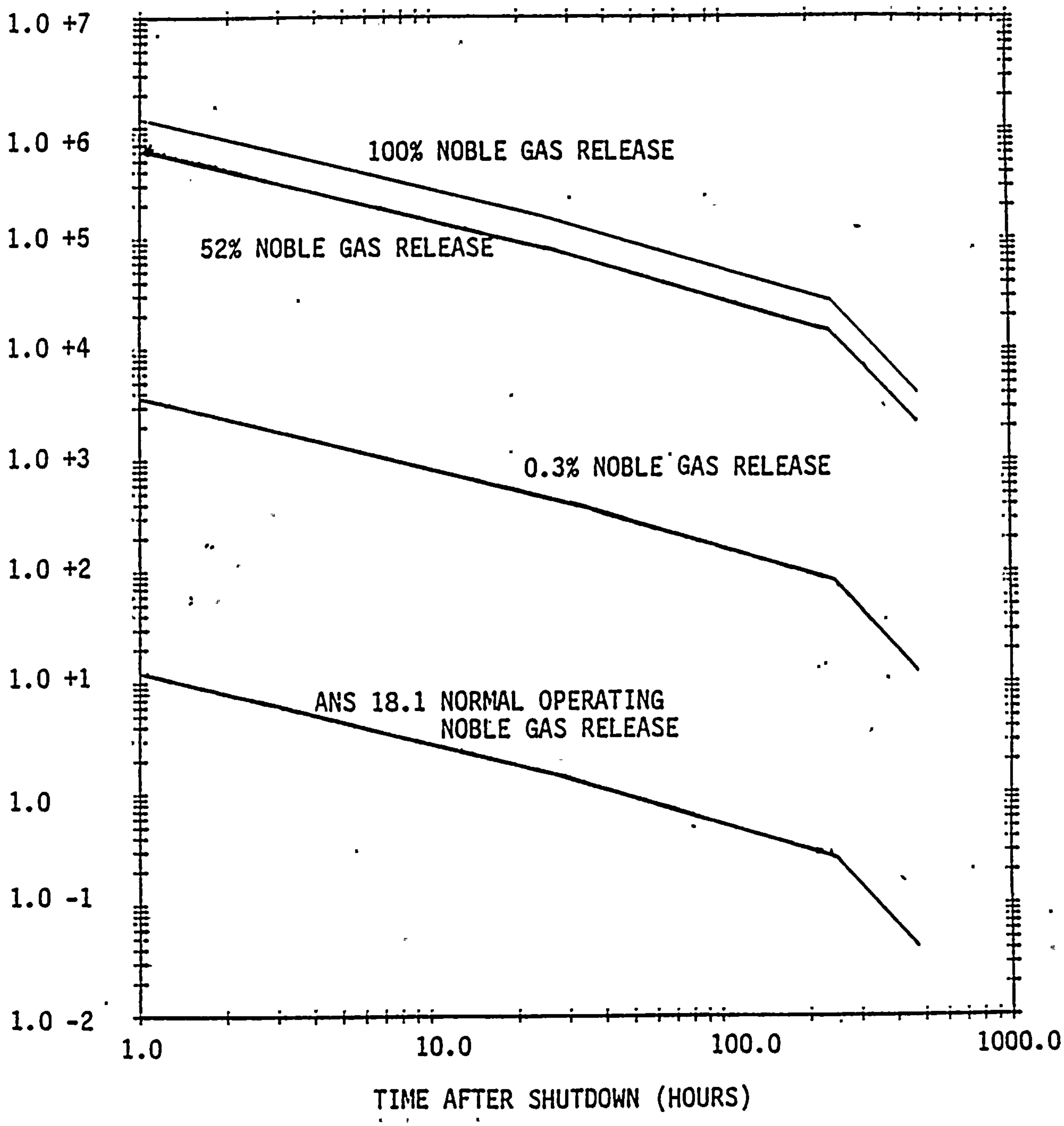

FIGURE 19 PERCENT NOBLE GASES IN CONTAINMENT FOR UNIT 1 AND UNIT 2

APPENDIX B

EXAMPLE OF CORE DAMAGE ASSESSMENT

The following example is presented to illustrate the use of this procedure.

SIMULATED ACCIDENT SCENARIO

For this example, Unit 1 has experienced an accident where the plant's monitoring systems indicated that safety injection had initiated and a significant amount of water had accumulated in the containment. Samples were available from the primary coolant (hot leg), the containment sump, and the containment atmosphere.

NUCLIDE SAMPLING

Samples were counted 6 hours after reactor shutdown. The results of the sample counts are presented in Tables 3A, 3B, and 3C.

All sample activities reported represent the activity of the sample at the time of analysis and have not undergone a decay correction back to time of shutdown. The decay correction factors are determined from Table 4 and recorded in Tables 3A, 3B, and 3C. The corrected sample activities are then determined by multiplying the sample activity by the correction factor. The corrected sample activities are recorded in Tables 3A, 3B, and 3C.

LIQUID MASS

Table 5 was completed to determine total liquid mass available for distribution in the RCS and containment. All 4 accumulators had discharged, the RWST had supplied 350,000 gallons, and the boron injection tank (900 gallons) had depleted. Also, it is assumed that all of the ice had melted supplying 2.7×10^6 lbm of water. A total water mass of 2.91×10^9 gram was calculated.

At the time of sampling, the RCS temperature was 350°F, and the containment water temperatture was 150°F. The reactor vessel level indication system was not functioning properly at time of sampling, and no indication was able to be

recorded. As such, the containment water was then determined. The containment sump level indicated the sump was full while the containment level indicated an 87% height. Referring to Figure 3, 87 percent corresponds to a range of possible volumes for the containment. A containment water volume of 98,000 ft^3 was then estimated by taking the average of the range; 98,000 ft^3 of containment water at 150°F corresponds to 2.77×10^9 grams.

Subtracting this from the total water mass, a RCS water mass of 1.4×10^8 grams was determined. The RCS and containment water masses were recorded in Table 3A and 3B, respectively.

TOTAL ACTIVITY RELEASED

The total activity released of each nuclide for each sample location was then calculated by multiplying the corrected sample activity by the water mass or containment volume and recorded in Tables 3A, 3B, and 3C.

These values were again recorded in Table 7A. The total activity of each nuclide was calculated by summing the activity for each sample location and was recorded in Table 7A.

TOTAL CORE INVENTORY

The power history for the 30 days prior to reactor shutdown was recorded in Table 8. The power correction factors for Kr-87 and I-132 were determined by the steady-state power correction equation for nuclide with half-lives less than 1 day. The power correction factors for Xe-133, I-131, and Ba-140 were determined by the transient power correction factor for nuclides with half-lives greater than 1 day. For Cs-137, the transient power correction factor utilizing effective full power days of operation during the cycle was used. In this example, the core had operated for 240 effective full power days during the 400 days of cycle operation. The power correction factor for Cs-137 is

$$\frac{240 \text{ EFPD}}{400 \text{ Days}} = 0.6.$$

The power correction factors were recorded in Table 7A.

The total corrected inventory was then calculated by multiplying the equilibrium core inventory (listed in Table 7A) by the power correction factor. The total corrected core inventory was recorded in Table 7A.

ESTIMATION OF PERCENT FUEL DAMAGE

Completing Table 7A, the percentage of corrected core inventory released of each nuclide was calculated from the corrected activity released and the corrected core inventory. The percent released for each nuclide was used with the appropriate graphs of Figures 4 through 16 to determine the category and estimate of core damage. Estimates were entered in Table 10 under the appropriate categories.

NUCLIDE ACTIVITY RATIOS

Table 11 was completed to determine the nuclide activity ratios. The ratios were compared to the gap and fuel pellet activity ratios listed in Table 11 and then recorded in Table 10 under the appropriate categories.

AUXILIARY INDICATORS

It was determined that the core had uncovered for approximately 30 minutes during the accident. The core exit thermocouple readings reached 1750°F. These values were compared with Table 12 and recorded in Table 10 under the appropriate categories.

The containment hydrogen monitor indicated a 4% hydrogen concentration, but the ignitors had initiated and some hydrogen burning had taken place.

The high range containment area monitor indicated a reading of 2.5E4 R/hr at 6 hours after the shutdown. Comparing 2.5E4 R/hr with Figure 18 and Table 12, this value was recorded in Table 10 under the appropriate categories.

CORE DAMAGE ASSESSMENT

All data collected in Table 10 was evaluated to estimate the extent of core damage.

The nuclides analyzed for this assessment were Kr-87, Xe-133, I-131, I-132, Cs-137, and Ba-140. The noble gases, iodine, and cesium are released during all stages of core damage with Ba-140 being a characteristic fission product of fuel overtemperature and fuel melt. Based on the Ba-140 data, the damage had progressed to approximately 20% fuel overtemperature and minor fuel melt (< 1%). The noble gas and iodine data indicated greater than 100 percent clad damage had occurred. However, it is recognized that in actuality there is an overlap between the regimes of core damage states. The release due to overtemperature dominated the release due to clad damage, and it is estimated that a large amount (>50%) clad damage had occurred.

The auxiliary indicators supported the radionuclide analysis. The fact that the core uncovered and the core exit thermocouples reached around 1750°F are indicative that fuel overtemperature had occurred. The hydrogen concentration of 4% was inconclusive due to the ignitors forcing some hydrogen burns. However, the fact that there was a significant amount of hydrogen produced for burning to occur supports the assessment that the core experienced clad damage and fuel overtemperature. The high range containment area monitor readings of 3.5E4 supports the less than 50% fuel overtemperature damage state.

Thus, for this example, the final fuel damage assessment is greater than 50% clad failure, less than 50% fuel overtemperature, and the possibility of some very minor fuel melting (< 1%).

TABLE 1

SELECTED NUCLIDES FOR CORE DAMAGE ASSESSMENT

Core Damage State	Nuclide	Half-Life*	Predominant Gammas (Kev) Yield (%)*
Clad Failure	Kr-85m**	4.4 h	150(74), 305(13)
	Kr-87	76 m	403(84), 2570(35)
	Kr-88**	2.8 h	191(35), 850(23), 2400(35)
	Xe-131m	11.8 d	164(2)
	Xe-133	5.27 d	81(37)
	Xe-133m**	2.26 d	233 (14)
	Xe-135**	9.14 h	250(91)
	I-131	8.05 d	364(82)
	I-132	2.26 h	773(89), 955(22), 1400(14)
	I-133	20.3 h	530(90)
	I-135	6.68 h	1140(37), 1280(34), 1460(12), 1720(19)
	Rb-88	17.8 m	898(13), 1863(21)
Fuel Overheat	Cs-134	2 yr	605(98), 796(99)
	Cs-137	30 yr	662(85)
	Te-129	68.7 m	455(15)
	Te-132	77.7 h	230(90)
Fuel Melt	Sr-89	52.7 d	(beta emitter)
	Sr-90**	28 yr	(beta emitter)
	Ba-140	12.8 d	537(34)
	La-140	40.22 h	487(40), 815(19), 1596(96)
	La-142	92.5 m	650(48), 1910(9), 2410(15), 2550(11)
	Pr-144	17.27 m	695(1.5)

* Values obtained from Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition.

** These nuclides are marginal with respect to selection criteria for candidate nuclides; they have been included on the possibility that they may be detected and thus utilized in a manner analogous to the candidate nuclides.

TABLE 2

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.

TABLE 3A

RCS ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/gms	Decay Correction Factor	Corrected Specific Activity Ci/gm	RCS Mass gms	RCS Activity Ci
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131	6	1.1(-2)	1.02	1.1(-2)	1.4(8)	1.6(6)
I 132	6	1.7(-2)	1.03	1.8(-2)	1.4(8)	2.5(6)
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137	6	1.0(-3)	1.	1.0(-3)	1.4(8)	1.4(5)
Te 129						
Te 132						
Ba 140	6	5.0(-5)	1.01	5.0(-5)	1.4(8)	7.0(3)
La 140						
La 142						
Pr 144						

TABLE 38

CONTAINMENT SUMP ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/gms	Decay Correction Factor	Corrected Specific Activity Ci/gm	Containment Water Mass gms	Containment Water Activity Ci
Kr 85m						
Kr 87						
Kr 88						
Xe 131m						
Xe 133						
Xe 133m						
Xe 135						
I 131	6	1.3(-3)	1.02	1.3(-3)	2.77(9)	3.6(6)
I 132	6	1.9(-3)	1.03	2.0(-3)	2.77(9)	5.6(6)
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137	6	1.2(-4)	1.	1.2(-4)	2.77(9)	3.3(5)
Te 129						
Te 132						
Ba 140	6	6.0(-6)	1.01	6.1(-6)	2.77(9)	1.7(4)
La 140						
La 142						
Pr 144						

TABLE 3C

CONTAINMENT ATMOSPHERE ACTIVITY WORKSHEET

Nuclide	Elapse Time Shutdown to Sample Count t, hours	Measured Specific Activity Ci/cc	Decay Correction Factor	Corrected Specific Activity Ci/cc	Containment Volume cc	Containment Activity Ci
Kr 85m						
Kr 87	6	2.4(-6)	26.6	6.3(-5)	3.5(11)	2.2(6)
Kr 88						
Xe 131m						
Xe 133	6	2.7(-4)	1.06	2.9(-4)	3.5(10)	1.0(7)
Xe 133m						
Xe 135						
I 131						
I 132						
I 133						
I 135						
Rb 88						
Cs 134						
Cs 137						
Te 129						
Te 132						
Ba 140						
La 140						
La 142						
Pr 144						

TABLE 4

DECAY CORRECTION FACTOR*

WITH PARENT-DAUGHTER EFFECT

Nuclide	Correction Factor
Kr 85m	$e^{0.158t}$
Kr 87	$e^{0.547t}$
Kr 88	$e^{0.248t}$
Xe 131m	$1/-2.66e^{(-3.59E-3)t} + 3.66e^{(-2.45E-3)t}$
Xe 133	$1/-0.187e^{(-3.41E-2)t} - 0.10e^{(-5.48E-3)t} + 1.287e^{(-1.28E-2)t}$
Xe 133m	$1/-0.10e^{(-3.41E-2)t} + 1.11e^{(-1.28E-2)t}$
Xe 135	$1/-9.14e^{(-1.04E-1)t} -0.033e^{(-2.67)t} + 10.17e^{(-7.58E-2)t}$
I 131	$e^{(3.59E-3)t}$
I 132	$1/1.03e^{(-8.92E-3)t} -0.03e^{(-3.07E-1)t}$
I 133	$e^{(3.41E-2)t}$
I 135	$e^{0.104t}$
Rb 88	$1/1.10e^{(-0.248)t} -0.10e^{(-2.34)t}$
Cs 134	1.0
Cs 134	1.0
Te 129	$1/1.09e^{(-0.161)t} + 0.167e^{(-8.47E-4)t} - 0.257e^{(0.605)t}$
Te 132	$e^{(8.92E-3)t}$
Ba 140	$e^{(2.26E-3)t}$
La 140	$1/1.08e^{(-2.26E-3)t} - 0.08e^{(-1.72E-2)t}$
La 142	$1/-0.145e^{(-3.78)t} + 1.145e^{(-0.450)t}$
Pr 144	$1/0.909e^{(-1.02E-4)t} + 0.091e^{(-2.41)t}$

*Time, t, is the number of hours between shutdown and time of sample count.

TABLE 5

ESTIMATE OF TOTAL LIQUID MASS

1. Estimate the volume added for the following:

Tank	Estimated Volume Added	Maximum Volume Added (gallons)
a. Refueling Water Storage Tank	350,000	372,250
b. Accumulator A	7,263	7,263
c. Accumulator B	7,263	7,263
d. Accumulator C	7,263	7,263
e. Accumulator D	7,263	7,263
f. Boron Injection Tank	900	900
g. Spray Additive Tank	—	4,000
h. Other source ____________	—	
	379,952	Total
i. Melted Ice	Estimated Mass Added	Maximum Mass Added (lbm)
	2.7×10^6	2.7×10^6

2. Convert estimated volume added from gallons to grams.

Added volume:

379,952, gallons x 3785 gms/gal = 1.44×10^9 gms

3. Convert ice melted mass from lbm to grains

2.7×10^6, lbm x 454 grams/lbm = 1.23×10^9 gms

4. The average Reactor Coolant System Mass is 2.40×10^8 gms.

5. Determine the Total Liquid Mass as follows:

Mass added 1.44×10^9 gms + melted ice mass 1.23×10^9 gms + RCS mass 2.4×10^8 gms = 2.91×10^9 gms

TABLE 6

ESTIMATE OF RCS WATER MASS* AND CONTAINMENT WATER MASS

AVERAGE OPERATING RCS VOLUME = 11,780 ft^3

1. Record the reactor vessel level, pressurizer level, and RCS temperature at time when sample was taken.

Level indication system not working
GO TO STEP 5

Reactor vessel level = ___—___%
Pressurizer level = ___—___%
RCS temperature = ___350___°F

2. Determine RCS volume at time of sample by estimating from level indications the percentage of water in the RCS.

$$\_\_\_\_\_\_ \text{ ft}^3 \times \_\_\_\_\_\_ \% \div 100 = \_\_\_\_\_\_ \text{ ft}^3$$

3. Determine RCS specific gravity from Figure 1.

RCS specific gravity =

4. Determine RCS mass as follows:

$$\text{RCS volume (ft}^3) \times \text{specific gravity} \times \frac{1.0\text{g}}{\text{cc}} \times \frac{28.3 \times 10^3\text{cc}}{\text{ft}^3}$$

$$\_\_\_\_ \text{ ft}^3 \times \_\_\_\_ \times \frac{1.0\text{g}}{\text{cc}} \times \frac{28.3 \times 10^3\text{cc}}{\text{ft}^3} = \_\_\_\_\_\_ \text{g}$$

5. Record the Containment Sump level indication and the containment level indication.

Containment Sump Level = ___100___%
Containment Level = ___87___%

TABLE 6 (Continued)

ESTIMATE OF RCS WATER MASS* AND CONTAINMENT WATER MASS

AVERAGE OPERATING RCS VOLUME = 11,780 ft^3

6. Determine containment water volume from Figures 2 and 3 using the levels from Step 5.

Note: If sump level indicates sump is full use Figure 3.

Containment Water Volume = 98,000 ft^3

7. Determine containment water specific gravity from Figure 1.

Containment water specific activity = 1.0 @ 150°F

8. Determine containment water mass as follows:

$$\text{Containment water volume} \times \text{specific gravity} \times \frac{1.0\ \text{gm}}{\text{cc}} \times \frac{28.3 \times 10^3\ \text{cc}}{\text{ft}^3}$$

$$98000\ \text{ft}^3 \times 1.0 \times \frac{1.0\ \text{gm}}{\text{cc}} \times \frac{28.3 \times 10^3\ \text{cc}}{\text{ft}^3} = 2.77 \times 10^9\ \text{gms}$$

*If a reactor vessel level indication is not available or is consider inaccurate based on engineering judgments subtract the estimated containment water mass from the estimated total water mass (Table 5) to determine RCS water mass.

Total Water Mass 2.91×10^9 gms - containment water mass 2.77×10^9 gms = RCS mass 1.4×10^8 gms

T[illegible]A

TOTAL RELEASE ACTIVITY/PERCENT RELEASED - UNIT 1

Nuclide	RCS Activity Ci	Containment Sump Activity Ci	Containment Atmosphere Activity Ci	Total Activity Ci	Equilibrium Core Inventory* Ci	Power Correction Factor	Corrected Core Inventory Ci	Release Percentage* %
Kr 85m					2.0 (7)			
KR 87			2.2(6)	2.2(6)	3.6 (7)	0.75	2.7(7)	8.1
Kr 88					5.2 (7)			
Xe 131m					5.7 (5)			
Xe 133			1.0(7)	1.0(7)	1.8 (8)	0.68	1.2(8)	8.3
Xe 133m					2.5 (7)			
Xe 135					3.4 (7)			
I 131	1.6(6)	3.6(6)		5.2(6)	8.9 (7)	0.68	6.1(7)	8.5
I 132	2.5(6)	5.6(6)		8.1(6)	1.3 (8)	0.75	9.8(7)	8.3
I 133					1.8 (8)			
I 135					1.6 (8)			
Rb 88					5.3 (7)			
Cs 134					2.1 (7)			
Cs 137	1.4(5)	3.3(5)		4.7(5)	1.0 (7)	0.6	6.0(6)	7.8
Te 129					3.0 (7)			
Te 132					1.3 (8)			
Ba 140	7.0(3)	1.7(4)		2.4(4)	1.5 (8)	0.65	9.8(7)	2.4(-2)
La 140					1.6 (8)			
La 142					1.4 (8)			
Pr 144					1.1 (8)			

* 2.0 (7) = 2.0×10^7. This notation is used throughout the procedure.

$$\text{**Release Percentage} = \frac{\text{Total Activity}}{\text{Corrected Core Inventory}} \times 100$$

TABLE 7B

TOTAL RELEASE ACTIVITY/PERCENT RELEASED - UNIT 2

Nuclide	RCS Activity Ci	Containment Sump Activity Ci	Containment Atmosphere Activity Ci	Total Activity Ci	Equilibrium Core Inventory* Ci	Power Correction Factor	Corrected Core Inventory Ci	Release Percentage* %
Kr 85m					2.1 (7)			
Kr 87					3.8 (7)			
Kr 88					5.4 (7)			
Xe 131m					6.0 (5)			
Xe 133					1.9 (8)			
Xe 133m					2.7 (7)			
Xe 135					3.5 (7)			
I 131					9.3 (7)			
I 132					1.3 (8)			
I 133					1.9 (8)			
I 135					1.7 (8)			
Rb 88					5.5 (7)			
Cs 134					2.2 (7)			
Cs 137					1.0 (7)			
Te 129					3.1 (7)			
Te 132					1.3 (8)			
Ba 140					1.6 (8)			
La 140					1.7 (8)			
La 142					1.4 (8)			
Pr 144					1.1 (8)			

$$\text{**Release Percentage} = \frac{\text{Total Activity}}{\text{Corrected Core Inventory}} \times 100$$

TABLE 8

POWER HISTORY OF 30 DAYS PRIOR TO SHUTDOWN

Interval j	Average Power Level* P_j	Operating Period at P_j t_j, hours	Period Between end of t_j and Reactor Shutdown t_j, hours
1	2437	20 days x 24 = 480	25 days x 24 = 600
2	3250	10 x 24 = 240	15 x 24 = 360
3	1625	10 x 24 = 240	5 x 24 = 120
4	2437	5 x 24 = 120.	0

Power Correction Factor (PCF)**

Steady-State Power Condition PCF

Transient Power Condition PCF

I. Half-Life of Nuclide < 1 Day

$$\frac{\text{Average Power Level (MWt) for prior 4 days}}{\text{Rated Power Level (Mwt)}}$$

$$\frac{\Sigma_j P_j \left(1 - e^{-\lambda_j t_j}\right) e^{-\lambda_i t_j^\circ}}{\text{Rated Power Level (MWt)}}$$

II. Half-Life of Nuclide > 1 Day

$$\frac{\text{Average Power Level (MWt) for prior 30 days}}{\text{Rated Power Level (Mwt)}}$$

$$\frac{\Sigma_j P_j \left(1 - e^{-\lambda_j t_j}\right) e^{-\lambda_i t_j^\circ}}{\text{Rated Power Level (MWt)}}$$

III. Half-Life of Nuclide $\simeq$ 1 Year

$$\frac{\text{Average Power Level (MWt) for prior 1 year}}{\text{Rated Power Level (MWt)}}$$

$$\frac{\text{Effective Full Power Days (EFPD)}}{\text{Total Calendar Days of Cycle Operation}}$$

* Average Power Level is defined as the power level at which the power level does not vary more than $\pm$10 percent of the rated power level from the time averaged value.

**λ_i = decay constant in hours^{-1} of each nuclide. λ_i of each nuclide is listed in

TABLE 9

DECAY CONSTANTS (λ_i) OF EACH NUCLIDE

Nuclide	Half-Life	λ_i, hours^{-1}
Kr 85m	4.4 h	0.158
Kr 87	76 m	0.547
Kr 88	2.8 h	0.248
Xe 131m	11.8d	2.45(-3)
Xe 133	5.27d	5.48(-3)
Xe 133m	2.26d	1.28(-2)
Xe 135	9.14h	7.58(-2)
I 131	8.05d	3.59(-3)
I 132	2.26h	0.307
I 133	20.3 h	3.41(-2)
I 135	6.68 h	0.104
Rb 88	17.8 m	2.34
Cs 134	2 yr	3.96(-5)
Cs 137	30 yr	2.64(-6)
Te 129	68.6 m	0.605
Te 132	77.7 h	8.92(-3)
Ba 140	12.8 d	2.26(-3)
La 140	40.22 h	1.72(-2)
La 142	92.5 m	0.450
Pr 144	17.27 m	2.41

TABLE 10

CORE DAMAGE ASSESSMENT EVALUATION SHEET

Indicator	Percent Clad Damage < 50%	Percent Clad Damage > 50%	Percent Overtemperature < 50%	Percent Overtemperature > 50%	Percent Fuel Melt < 50%	Percent Fuel Melt > 50%
<u>Radionuclide Analysis</u>						
Kr 85m						
Kr 87		OFF SCALE	~20		~10	
Kr 88						
Xe 131m						
Xe 133		OFF SCALE	~20		~10	
Xe 133m						
Xe 135						
I 131		OFF SCALE	~20		~10	
I 132		OFF SCALE	-20		~10	
I 133						
I 135						
Cs 134						
Cs 137			~15		~8	
Te 129						
Te 132						
Ba 140			~20		< 1	
La 140						
La 142						
Pr 144						
<u>Ratios</u>						
Kr 85m/Xe 133						
Kr 87/Xe 133			← 0.22 →			
Kr 88/Xe 133						
Xe 131m/Xe 133						

TABLE 10 (Continued)

CORE DAMAGE ASSESSMENT EVALUATION SHEET

Indicator	Percent Clad Damage < 50%	Percent Clad Damage > 50%	Percent Overtemperature < 50%	Percent Overtemperature > 50%	Percent Fuel Melt < 50%	Percent Fuel Melt > 50%
Ratio (Con't)						
Xe 133m/Xe 133						
Xe 135/Xe 133						
I 132/I 131			1.56			
I 133/I 131						
I 135/I 131						
Auxiliary Indicators						
Core Uncovered	YES					
Core Exit Temp °F				1750		
Containment H_2%	4%					
Zirc - Water Reaction %	—					
Ignitors On?		YES				
High Range Containment Monitor Reading R/hr			3.5E4			

TABLE 11

NUCLIDE ACTIVITY RATIOS

Nuclide	Gap Activity Ratio	Fuel Pellet Activity Ratio	Calculated Activity Ratio*
Kr 85m	0.022	0.11	
Kr 87	0.022	0.22	0.22
Kr 88	0.045	0.29	
Xe 131m	0.004	0.004	
Xe 133	1.0	1.0	1.0
Xe 133m	0.096	0.14	
Xe 135	0.051	0.19	
I 131	1.0	1.0	1.0
I 132	0.17	1.5	1.56
I 133	0.71	2.1	
I 135	0.39	1.9	

$$\text{*Noble Gas Ratio} = \frac{\text{Noble Gas Nuclide Released (Ci)}}{\text{Xe-133 Released (Ci)}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Nuclide Released (Ci)}}{\text{I-131 Released (Ci)}}$$

TABLE 12
CHARACTERISTICS OF CATEGORIES OF FUEL DAMAGE*

Core Damage Category	Core Damage Indicator: Percent and Type of Fission Products Released	Fission Product Ratio	Containment Radiogas Monitor (R/hr) **	Core Exit Thermocouples Readings (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)*** & Plant Type
No clad damage	Kr-87 < $1x10^{-3}$ Xe-133 < $1x10^{-3}$ I-131 < $1x10^{-3}$ I-133 < $1x10^{-3}$	Not 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 - 1E2	750 - 1300	Core uncovery	0 - 13
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	1.E2 - 1.5E3	1300 - 1650	Core uncovery	13 - 24
0-50% fuel pellet overtemperature	Xe-Kr,Cs,I 1 - 20 Sr-Ba 0 - 0.1	Kr-87 = 0.22 I-133 = 2.1	1.5E3 - 4.0E4	> 1650	Core uncovery	13 - 24
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	4.0E4 - 2.5E5	> 1650	Core uncovery	13 - 24
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	2.5E5 - 3.5E5	> 1650	Core uncovery	13 - 24
50-100% fuel melt	Xe,Kr,Cs,I,Te > 70 Sr,Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	>3.5E5	> 1650	Core uncovery	13 - 24

* This table is intended to supplement the methodology outlined in this report and should not be used without referring to this report and without considerable engineering judgement.

** From Fig. 19 at 6 hours after shutdown.

*** Ignitors may obviate these values.

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

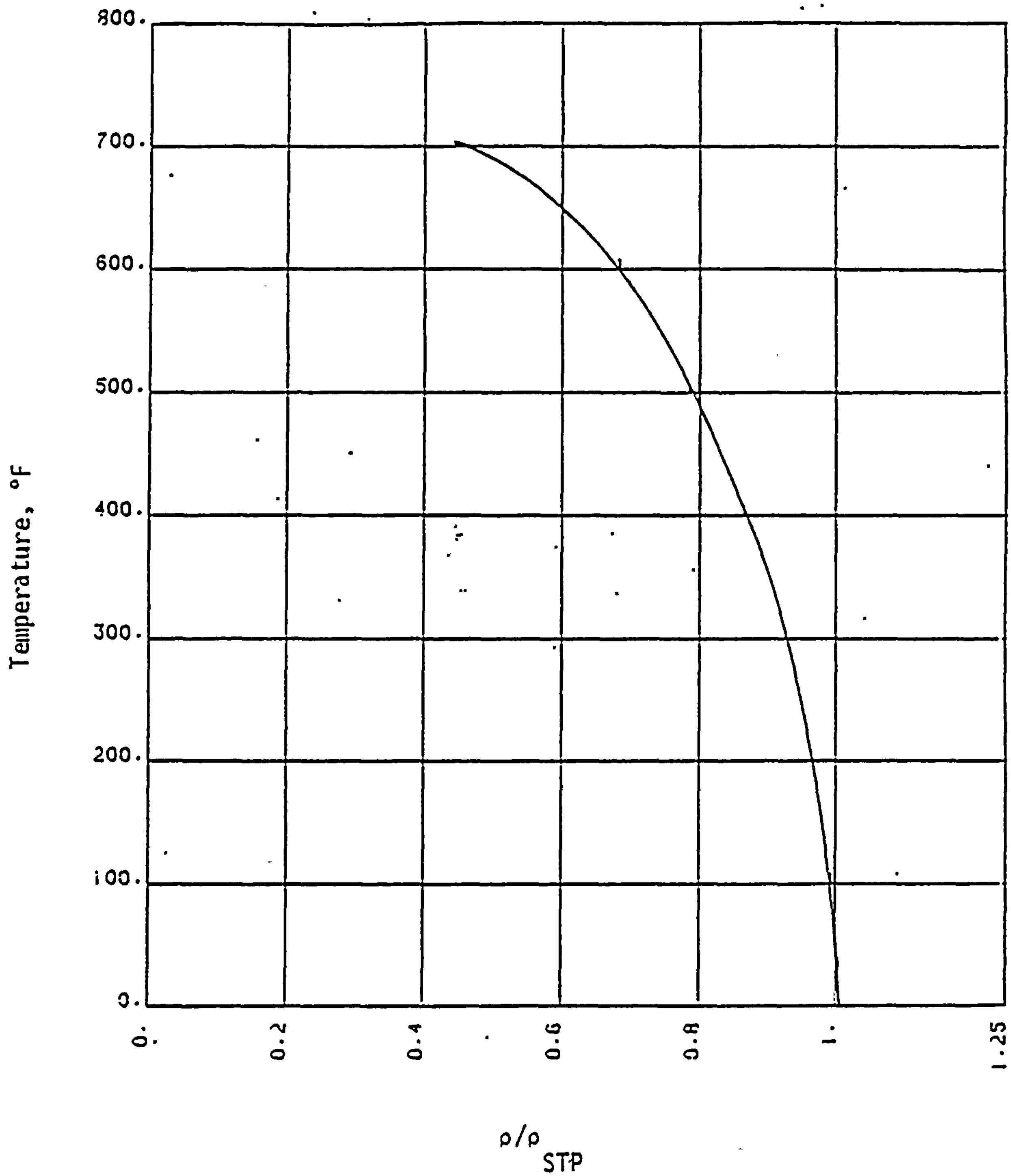

FIGURE 1 WATER DENSITY RATIO (TEMPERATURE VS. STP)

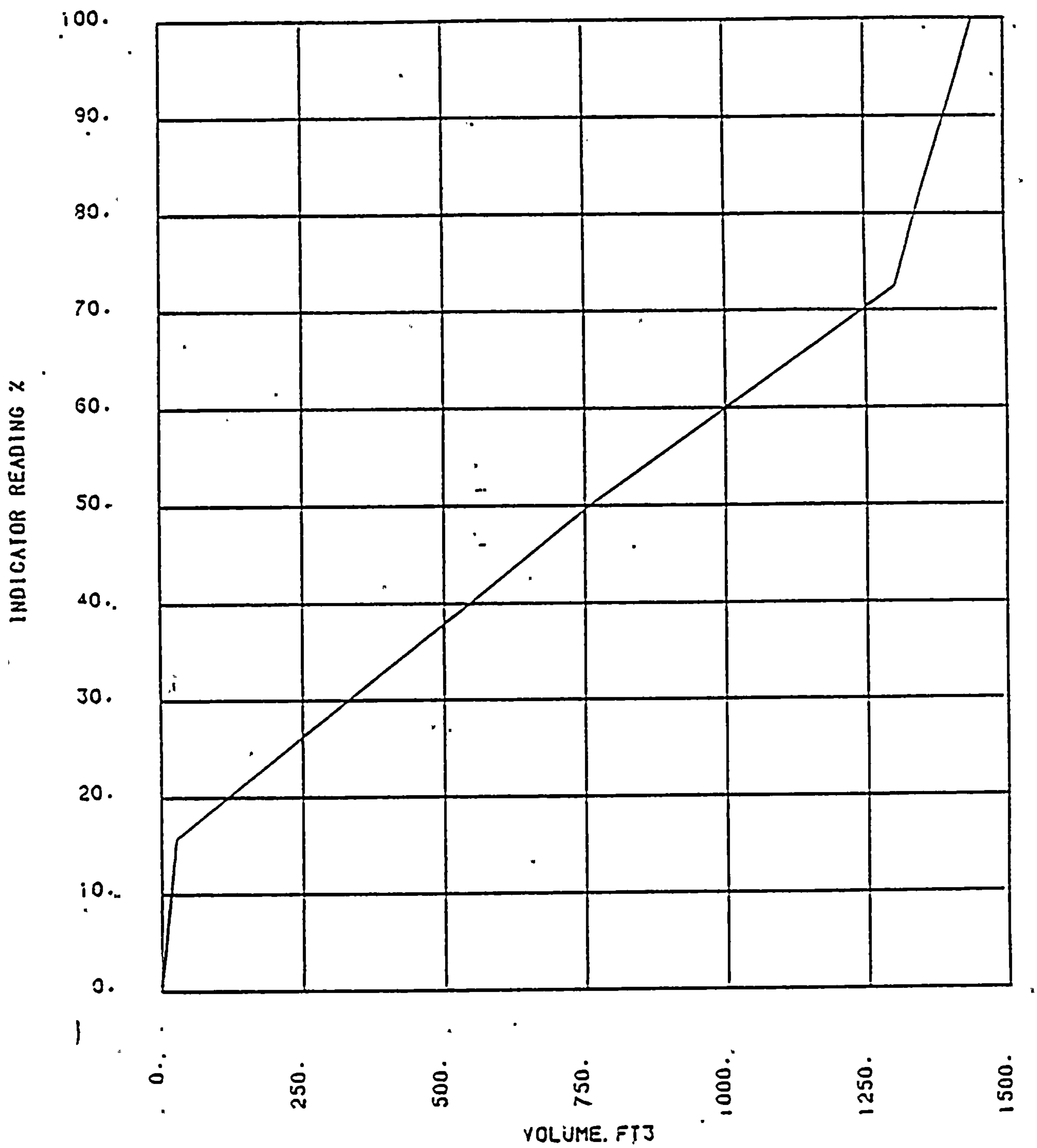

FIGURE 2 SUMP WATER VOLUME VERSUS SUMP LEVEL INDICATION

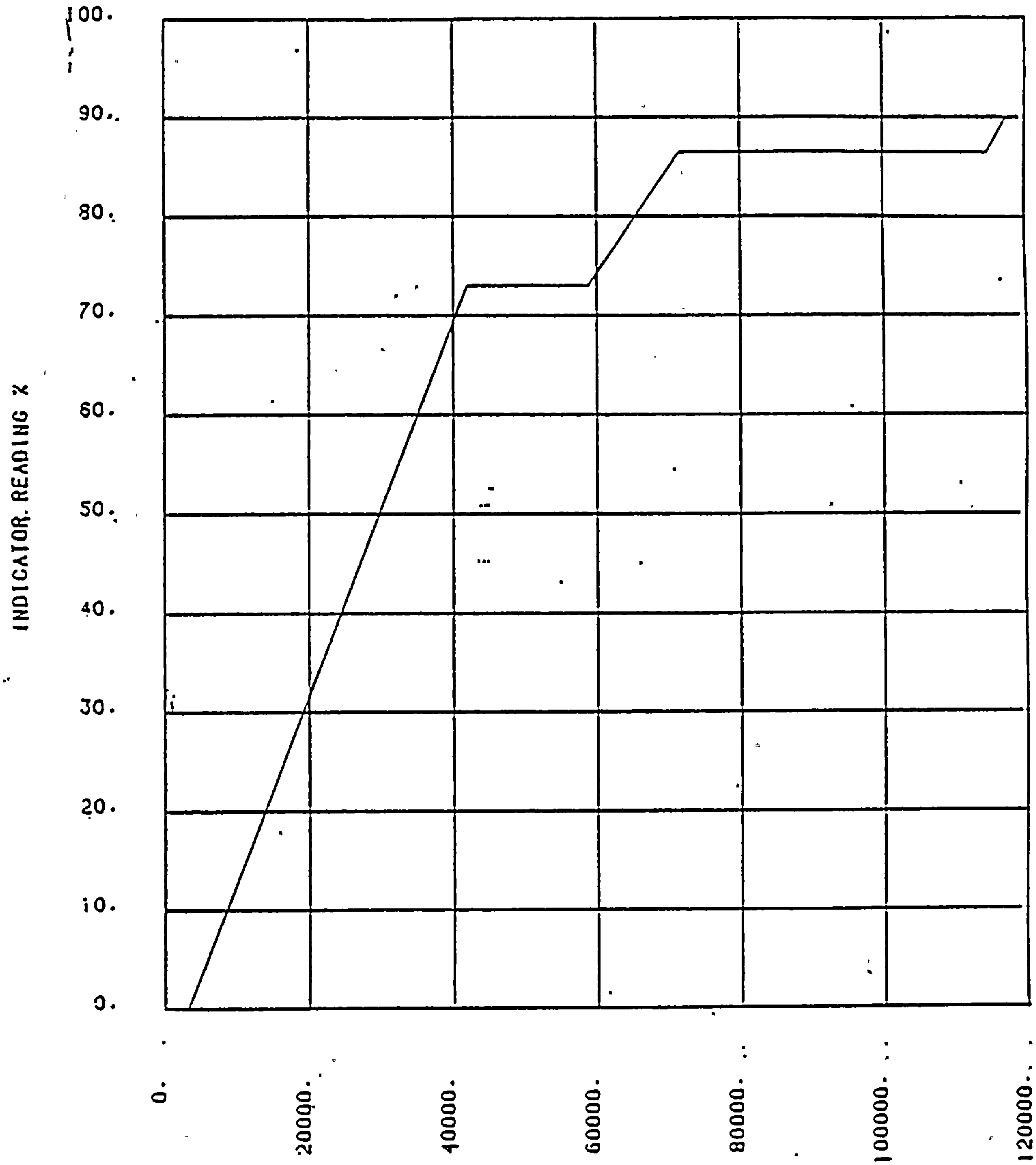

FIGURE 3 CONTAINMENT WATER VOLUME VERSUS CONTAINMENT LEVEL INDICATION

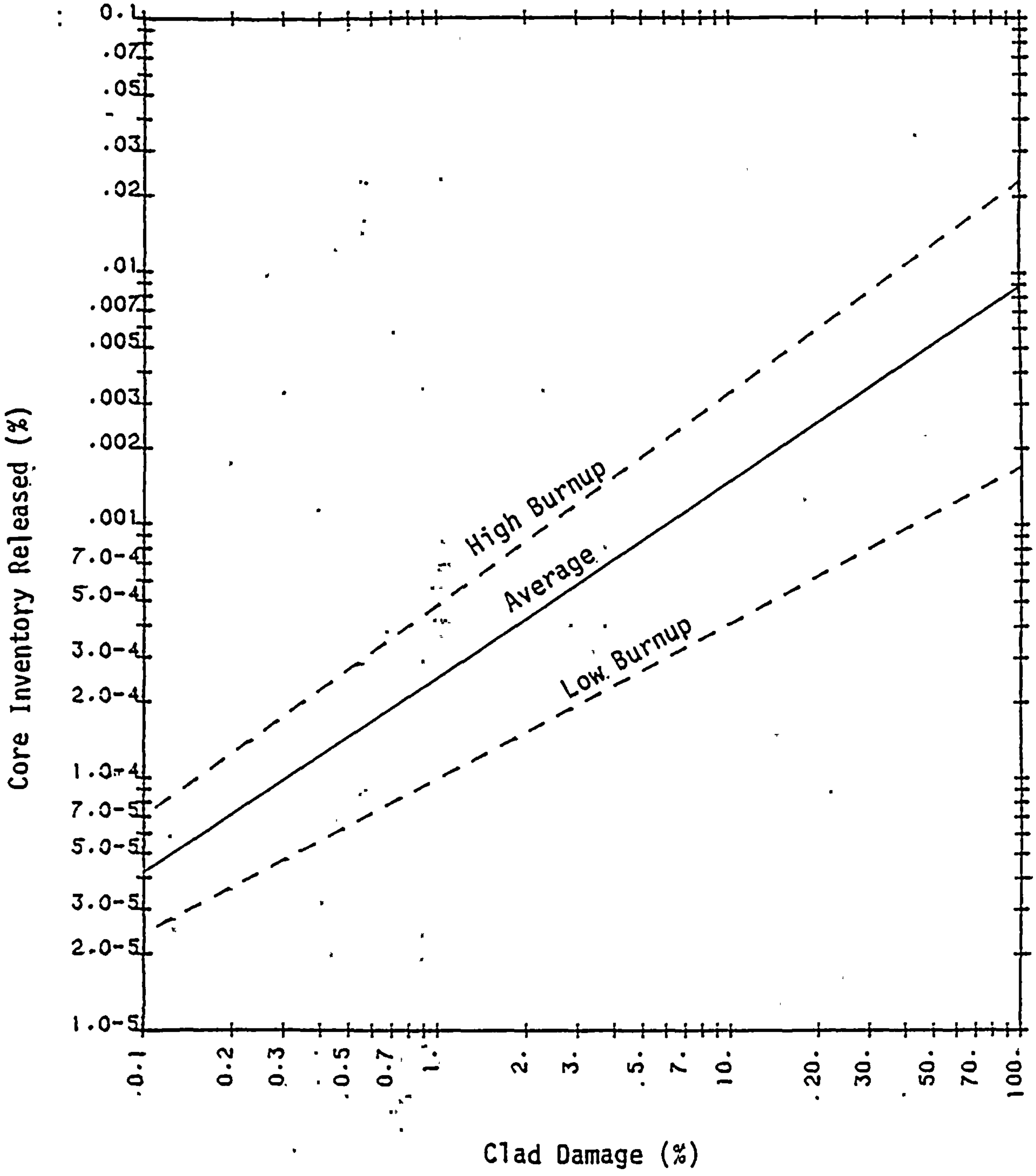

FIGURE 5 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF KR-87

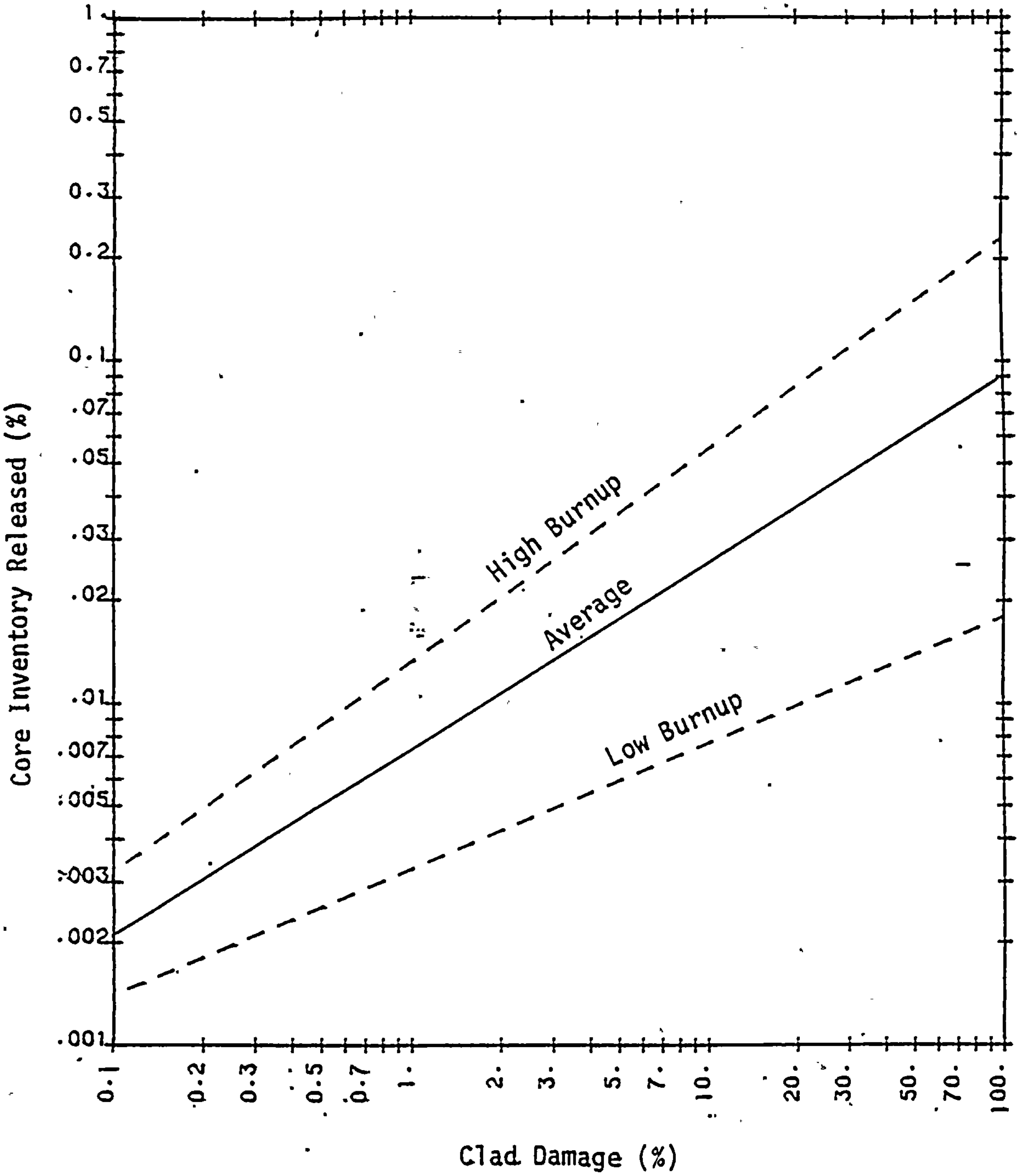

FIGURE 7 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF XE-133

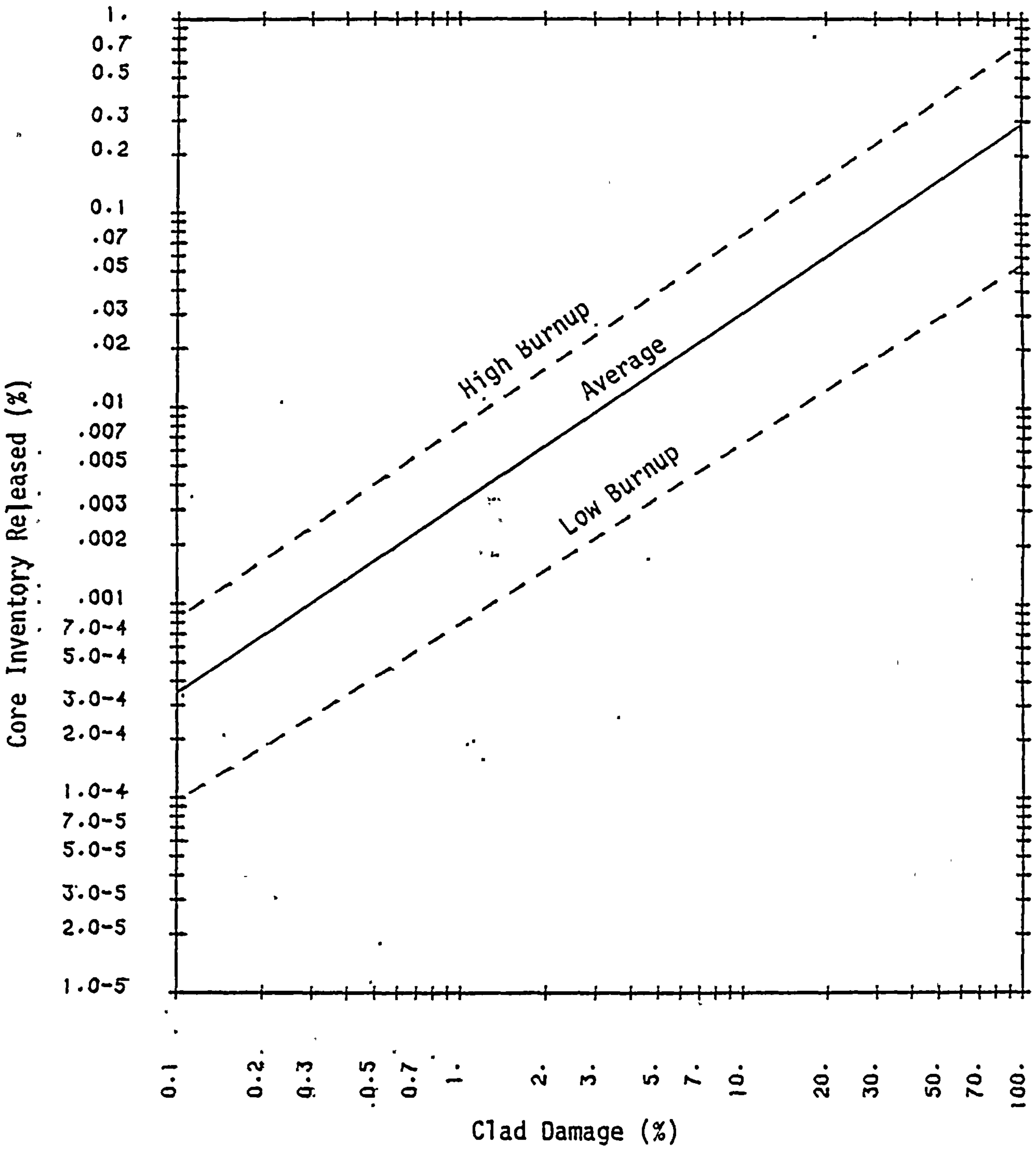

FIGURE 8 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-131

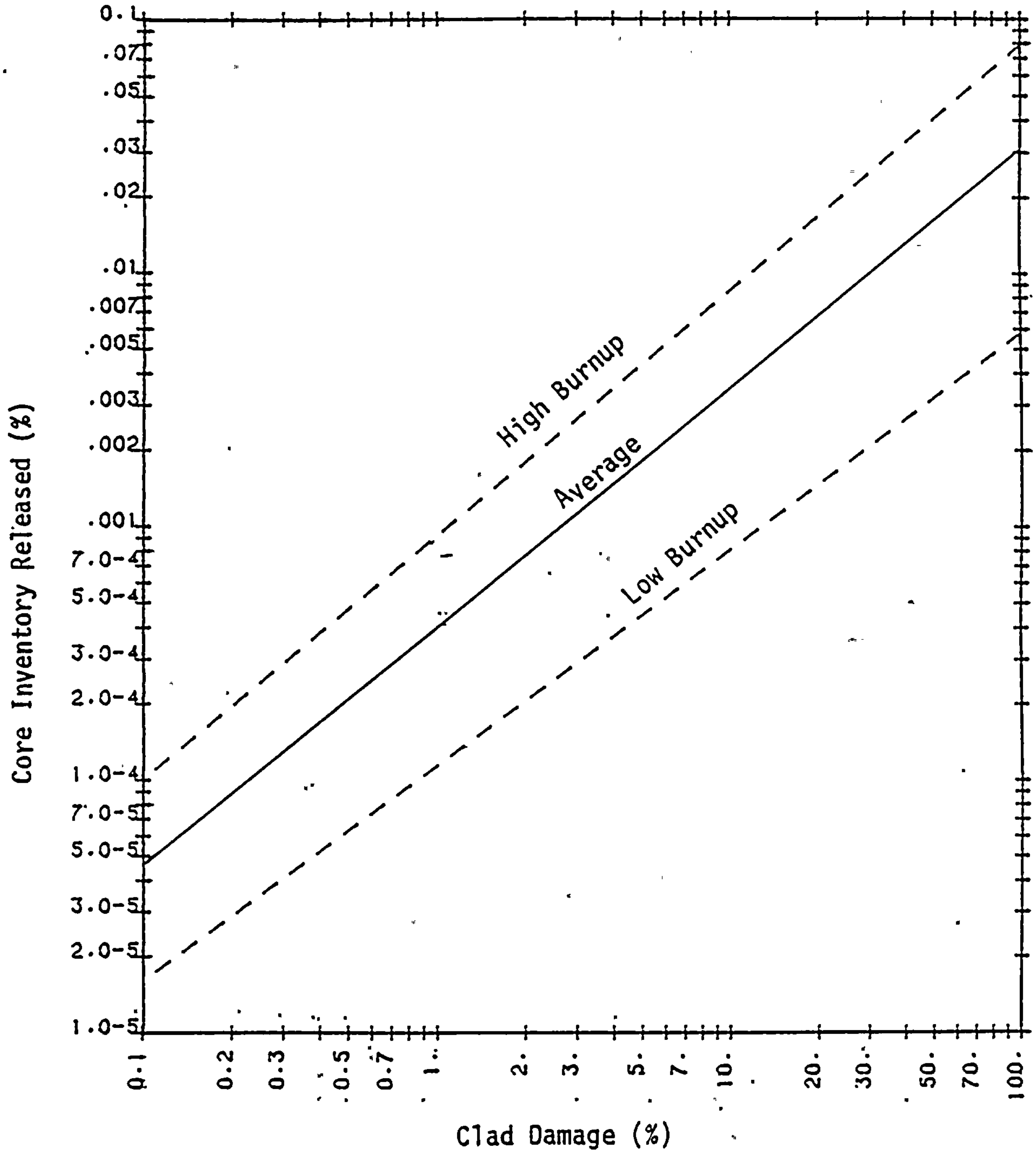

FIGURE 10 RELATIONSHIP OF % CLAD DAMAGE WITH % CORE INVENTORY RELEASED OF I-132

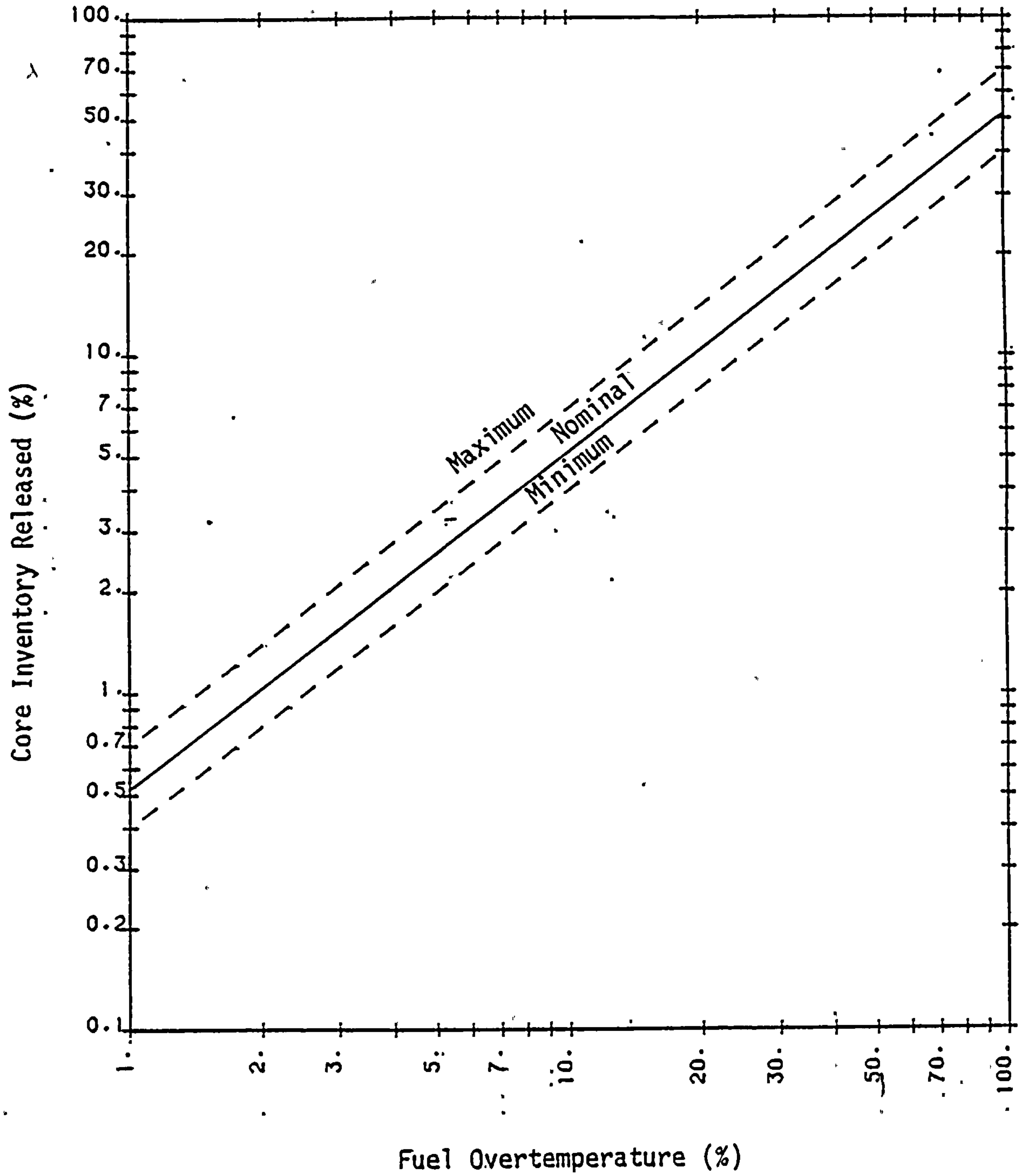

FIGURE I3 ; RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF XE, KR, I, OR CS

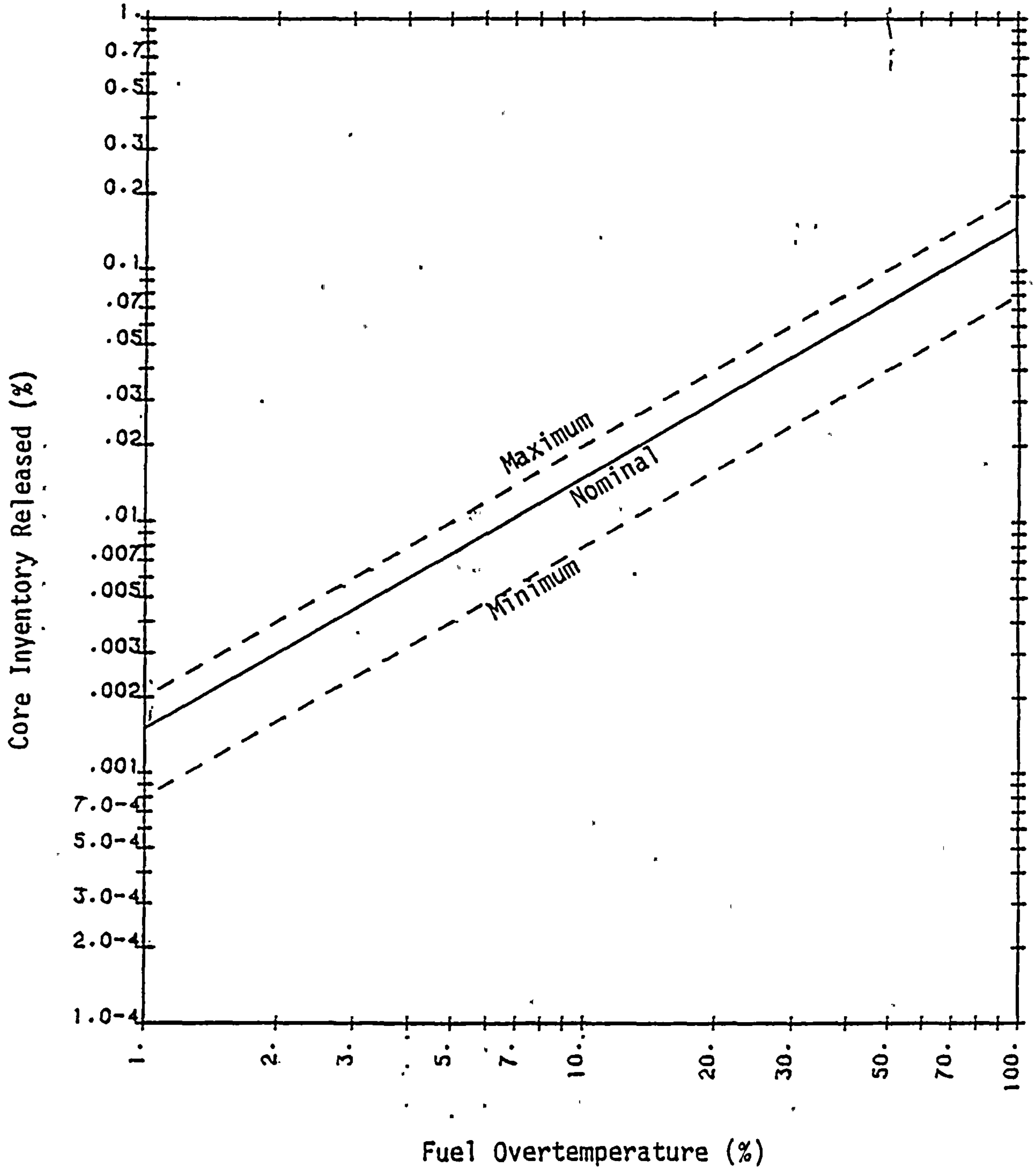

FIGURE 14 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF BA OR SR

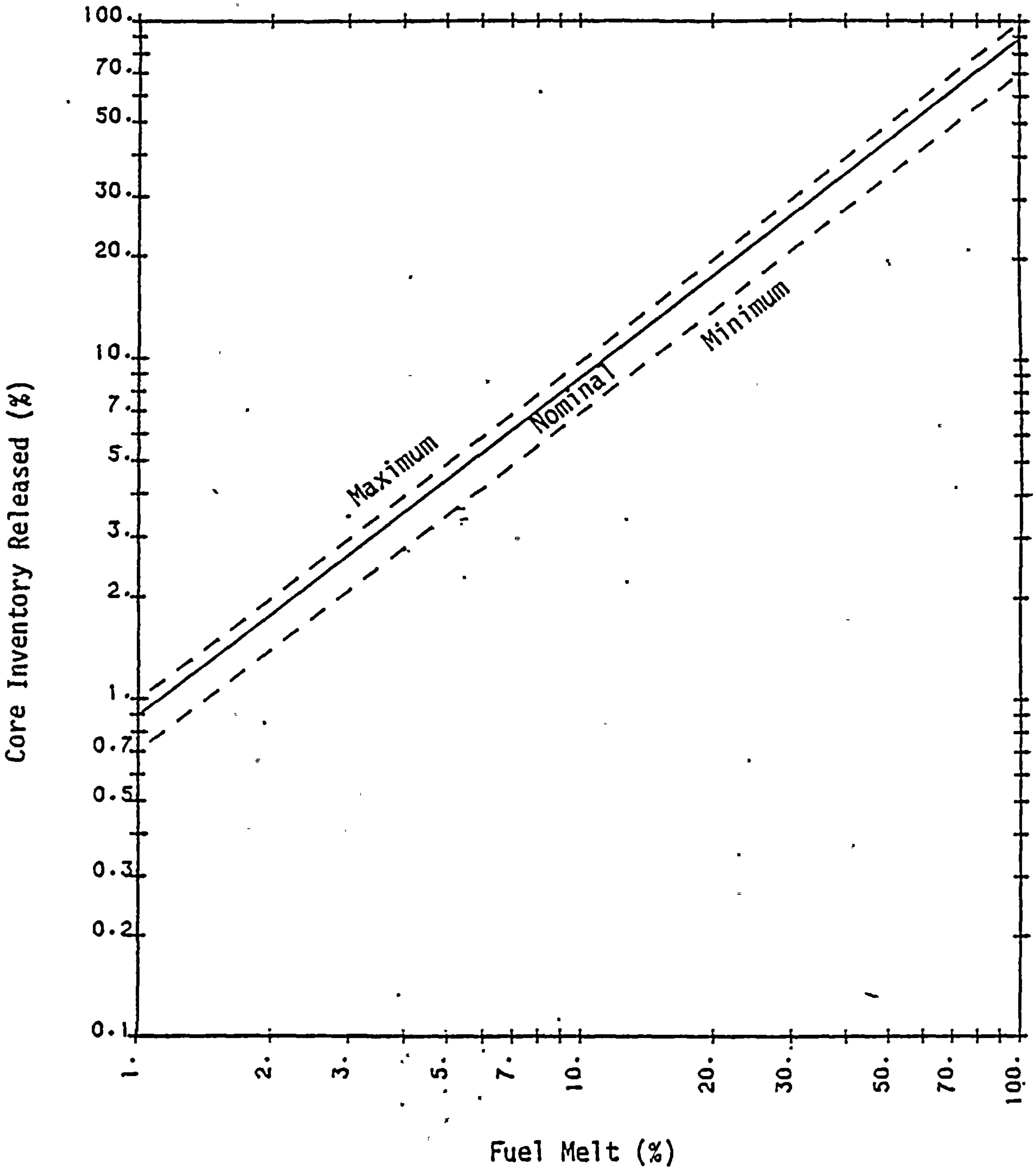

FIGURE 15 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF XE, KR, I, CS, OR TE

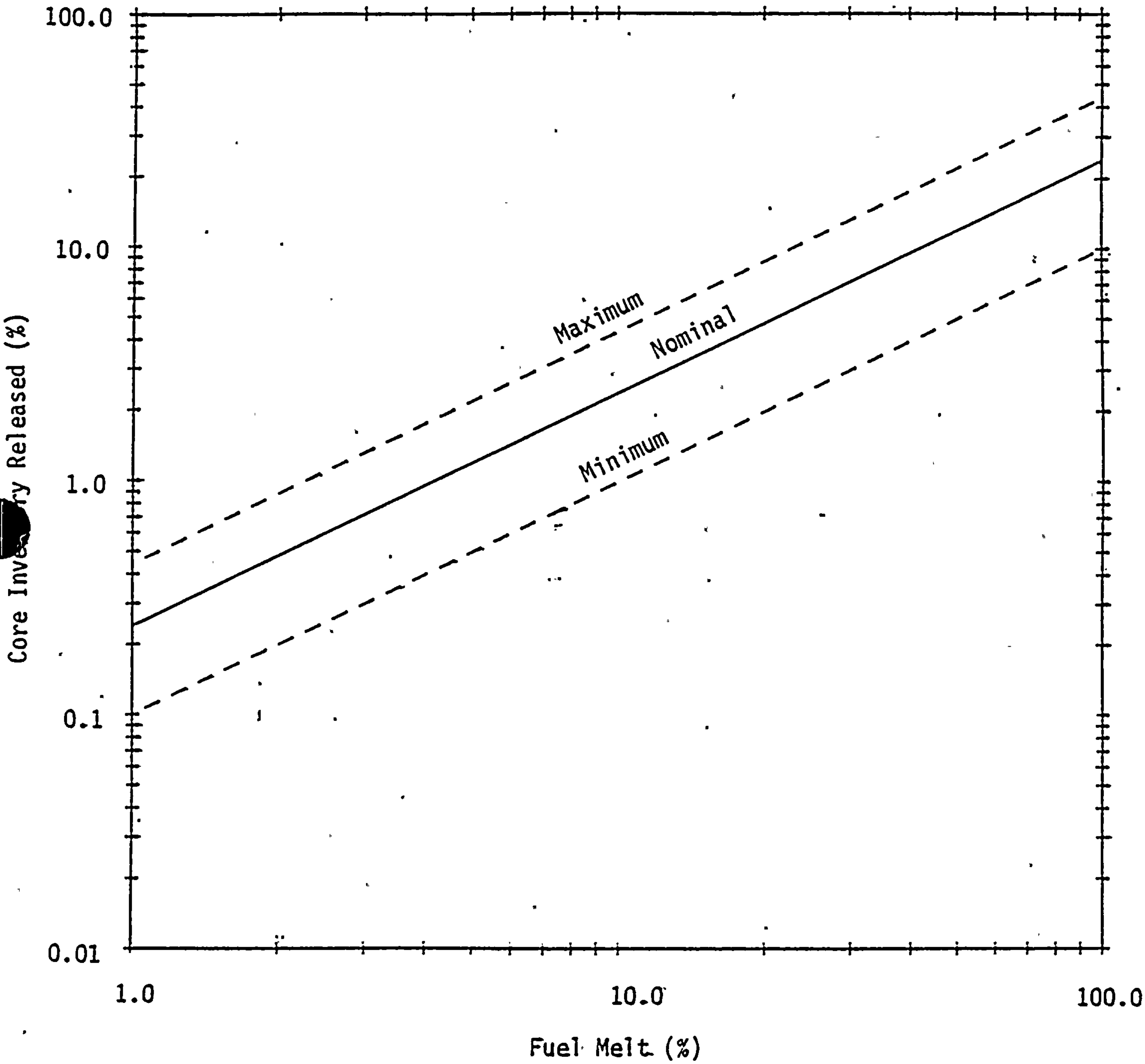

FIGURE 16 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF BA OR SR

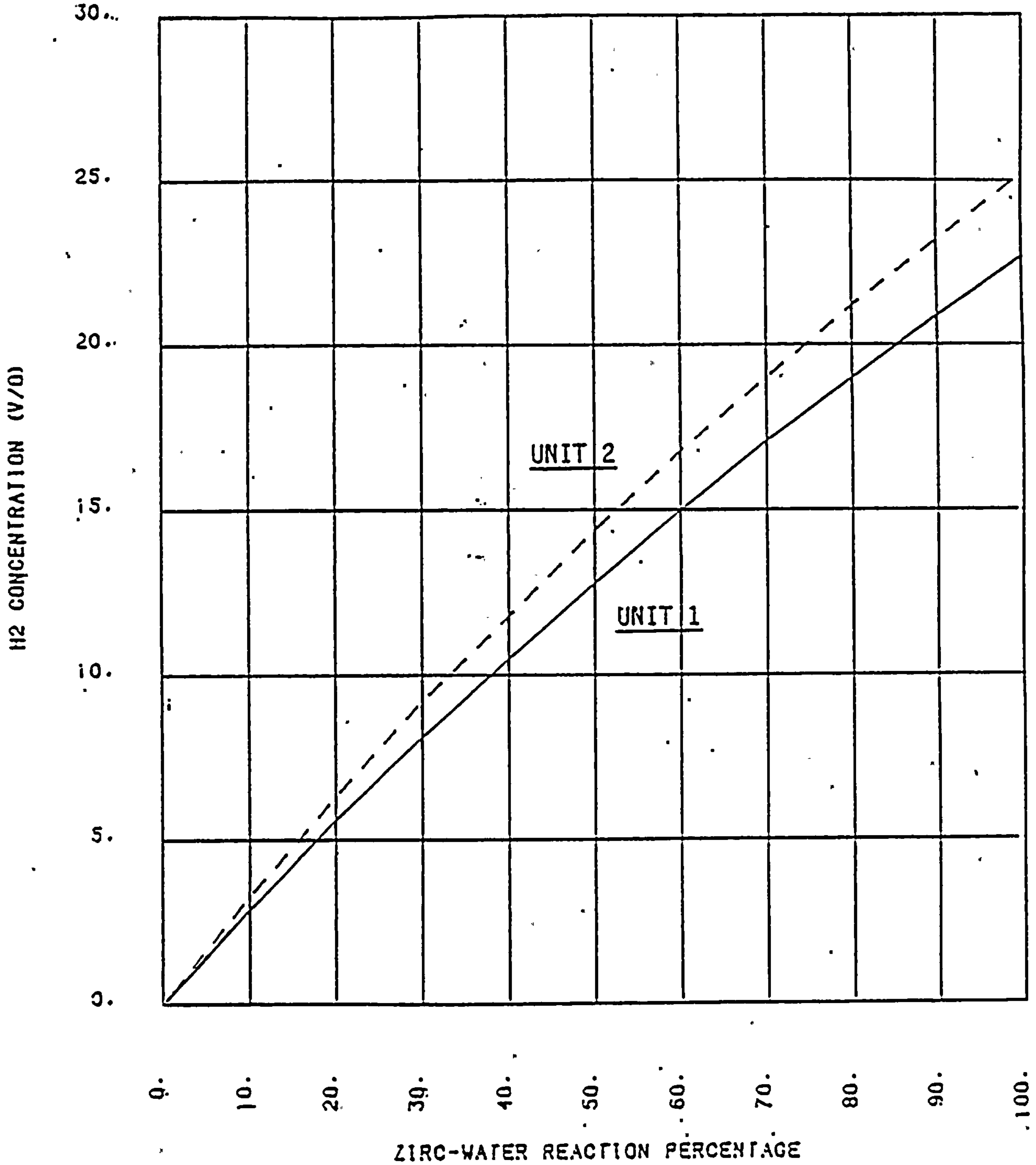

FIGURE 18 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM WATER REACTION

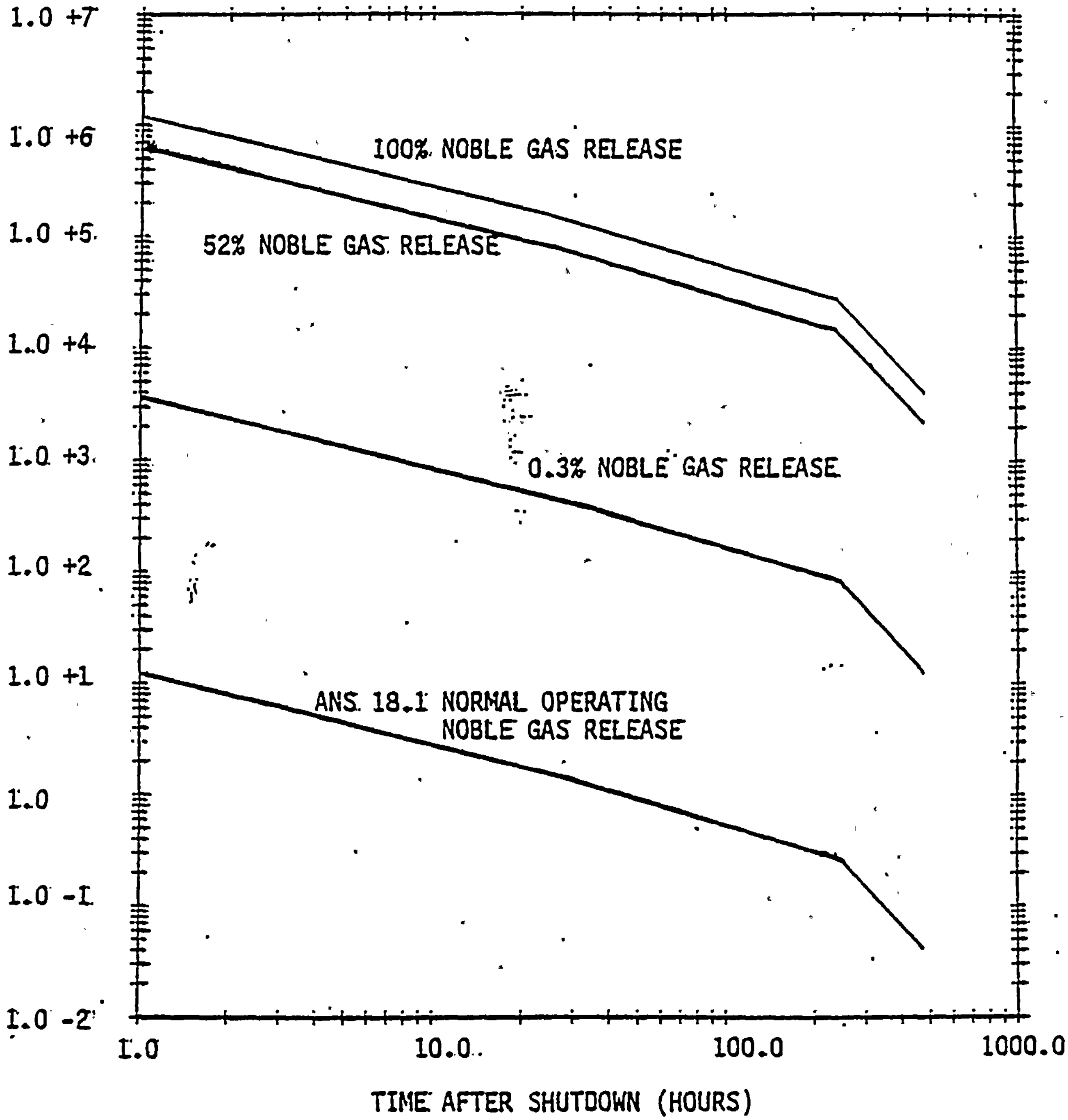

FIGURE 19 PERCENT NOBLE GASES IN CONTAINMENT FOR UNIT 1 AND UNIT 2