Duquesne Light

Nuclear Division
P.O. Box 4
Shippingport, PA 15077-0004

Telephone (412) 393-6000

July 16, 1984

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
NUREG-0737, item II.B.3, Request for Additional Information

Gentlemen:

Your letter dated April 20, 1984 requested additional information on our Post-Accident Sampling System (PASS). Information in three areas was requested. On May 31, a conference call was conducted by members of our respective staffs to obtain clarification on the third of these three items. The results of that conference was documented in my letter of June 5, 1984 and a commitment was made to provide all information by July 15. This submittal provides the requested information for each of your three items.

Item 1 (II.B.3, criterion 2):

Provide a procedure to estimate core damage.

Response

Included as Attachment 1 is the Core Damage Assessment Procedure for Beaver Valley, Unit 1. This procedure utilizes the Westinghouse Owner's Group Post Accident Core Damage Assessment Methodology as adapted to Beaver Valley.

Item 2 (II.B.3, criterion 2)

Describe the accuracy of the chloride analysis.

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Response

NUREG-0737, item II.B.3, criterion 2(c) requested that there exist an onsite chemical analysis capability to provide, within a three-hour time frame, quantification of the chloride concentration in liquids. The accuracy of the PASS chloride analysis was contained in our response dated August 31, 1982 to criterion 10 of your letter dated June 30, 1982. The PASS will analyze a sample for chloride concentration at an accuracy of ±10% of full scale in accordance with the accuracy stated in our August 1982 submittal. In the event it becomes necessary, an undiluted sample may be grabbed and will be analyzed in accordance with our response to criterion 2(c) dated August 1982 and within the time period of four days as stated in our response to criterion 5 in the same submittal. Additional information on the accuracy of the chloride analysis is provided in the following response to Item 3.

Item 3 (II.B.3, criterion 10):

Provide information demonstrating applicability of procedures and instrumentation in the post-accident water chemistry and radiation environment.

Response

Clarification of this item was obtained during a conference call on May 31, 1984 at which time we were instructed to provide information on the accuracy of the PASS instruments when tested with the standard test matrix (STM) as to whether the instruments had met those accuracies to which we had committed. The testing of the PASS instruments with the STM did not include an induced radiation field of 10 E4 rads per gram of reactor coolant as originally requested in your letter of June 30, 1982. During the conference call it was stated that our PASS instruments were purchased with certification that they would function in a radiation field exceeding 10 E4 rads per gram of reactor coolant. The combination of the satisfactory test results using the STM, less the radiation field, with the instrument certifications, provide assurance that these instruments will function as designed. This method for evaluating instruments with respect to a radiation field was found acceptable to the NRC Chemical Engineering Branch subsequent to the referenced conference call.

The PASS instrument accuracies for the boron and chloride analyzers were confirmed to be within the stated accuracies of our August 1982 submittal. The following are the results of our tests using the STM.

	STM Nominal Concentration (ppm)	Actual Value (ppm)	PASS Reading (ppm)	Accuracy Required	Instrument Error (Avg.) (ppm)
Boron	2000	1978	1785.6[1]	±10% of Full Scale (±600 ppm)	-192.4
Chloride	10	13	15.2[2]	±10% of Full Scale (±10 ppm)	+ 2.2

1. Three reading average
2. Five reading average

The pH value of the STM was unknown, however, the pH reading obtained from the PASS agreed with the chemistry laboratory instrument. The following are the pH electrode test results.

PASS Reading	Chemistry Lab Reading	
4.77 pH	4.79 pH	STM without spray additive
7.22 pH	7.22 pH	STM with spray additive

The dissolved hydrogen and dissolved oxygen readings were unaffected by the STM. However, the dissolved oxygen reading obtained during the testing of the PASS agreed with the theoretical value of dissolved oxygen associated with an air saturated liquid sample at 25°C.

	Theoretical Value	Actual Value	Accuracy Required	Actual Error
Dissolved Oxygen	8.4 ppm	8.2 ppm	±12 ppm (±6% of full scale)	-.2 ppm

The dissolved hydrogen instrument indicated there was no dissolved hydrogen present which was correct. The acceptance test for this instrument demonstrated instrument accuracies within ±6% of full scale in accordance with our previous submittal.

Our previous submittals on the PASS stated that the boron and chloride in-line analyzers were not installed due to equipment delivery problems. This was found acceptable as documented in the Order confirming licensee commitments on post-TMI related issues dated March 14, 1983. This condition no longer exists since the installation of the boron and chloride analyzers was recently completed.

Should you have any further questions on this issue, please contact me or members of my staff.

Very Truly Yours,

J. J. Carey
Vice President, Nuclear

Attachment

cc: Mr. W. M. Troskoski, Resident Inspector
U. S. Nuclear Regulatory Commission
Beaver Valley Power Station
Shippingport, PA 15077

U. S. Nuclear Regulatory Commission
c/o Document Management Branch
Washington, D. C. 20555

Attachment 1

Core Damage Assessment Procedure

for

Beaver Valley, Unit 1

Response to NRC Request for Additional Information dated April 20, 1984

POST ACCIDENT SAMPLING SYSTEM (continued)

Core Damage Assessment Procedure

Introduction

The following procedure would be used to estmate theextent of core damage after an accident at Beaver Valley. By using the Westinghouse Owner's Group Post Accident Core Damage Assessment Methodology adapted for the Beaver Valley Plant, this procedure will differentiate the amount of core damage over seven broad regions:

1) No core damage
2) 0-50% Clad Failure
3) 50-100% Clad Failure
4) 0-50% Fuel Pellet Overtemperature
5) 50-100% Fuel Pellet Overtemperature
6) 0-50% Fuel Melt
7) 50-100% Fuel Melt

The procedure used Isotopic data obtained from the Post Accident Sampling System, core exit thermocouple readings, RM-219 containment radiation monitor readings, and containment hydrogen readings to classify the amount of core damage. It should only be used by personnel familiar with the limitations of the Core Damage Assessment Methodology (CDAM), and will normally be performed by the Chemistry Coordinator during activation of the Emergency Preparedness Plan.

Procedure

1. Obtain a Post-Sample of the RCS, containment atmosphere, and containment sump (if applicable). Perform an isotopic analysis of each.

2. Correct each Isotopic valve for decay to time of reactor shutdown using the decay correction in Section 2.4.5.3 of the CDAM (p. 33). Note: Isotopes listed as daughter products in Table 2-7 of the CDAM (p. 34) must also be corrected for parent-daughter decay.

3. The decay corrected specific activity of each isotope obtained in step 2 should now be corrected for pressure and temperature as described in Section 2.4.5.2 of the CDAM (p. 29).

4. The adjusted specific activity of each isotope obtained in step 3 can now be used to determine the total activity of each isotope in the RCS, containment atmosphere, and the containment sump (if applicable). Multiply the adjusted specific activity of each isotope in each medium (RCS, containment atmosphere, or containment sump) by the mass in each medium as described in Section 2.4.5.2 of the CDAM (p. 29).

POST ACCIDENT SAMPLING SYSTEM (continued)

5. Determine the total activity of each isotope released by adding the activities of the isotope in the RCS, containment atmosphere and containment sump.

6. Calcuate the total activity of each isotope available in the core at time of plant shutdown by multiplying the fuel pellet inventory of each isotope in Table 2-2 of the CDAM (p. 6) by the power correction factor described in Section 2.3.1 of the CDAM (p. 7).

7. Calculate the percentage of each isotope released by dividing the values obtained of each isotope in step 5 by the valves obtained for that isotope in step 6, and multiplying that value times 100%.

8. The percent inventory released determined in step 7 can then be used to determine:

   A) % Clad Damage - Use Figures 2-2 through 2-9 of the CDAM (p. 15 through 22)

   B) %Fuel Overtemperature - Use Figures 2-11 and 2-12 of the CDAM (p. 41 and 42)

   C) %Fuel Melt - Use Figures 2-13 through 2-15 of the CDAM (p. 45 through 47)

9. Auxiliary indicators can now be used to confirm the valves calculated in step 8:

   a) Core exit thermocouple readings - (CADM Section 3.2, P. 57) temperatures greater than 1300°F indicate the likelihood of clad damage. Use Table 4-1 of the CADM (p. 63).

   B) Containment Radiation Monitor RM-219 - (CADM Section 3.3, P. 58). Calcuate the normalized gamma exposure rate as follows:

$$\text{R/hr-MWt} = \frac{\text{RM-219 reading}}{2652} \times .9$$

   Use this value with Figure 3-3 of the CDAM (P. 61) to calculate % Noble Gas Release. In general:

   Less than 0.3% Noble Gas Release - Indicative of Clad Failures

   Between 0.3% and 52% Noble Gas Release - Indicative of Fuel Overtemperature Conditions

   More than 52% Noble Gas Release - Indicative of some core melt

   C) Containment Hydroden - (CADM Section 3.1, P. 51) using the Containment Hydrogen Monitors, obtain the % hydrogen present in

POST ACCIDENT SAMPLING SYSTEM (continued)

containment. Use Table 4-1 of the CDAM (p. 63) to classify the amount of core damage.

10. Using the values obtained in step 8, as well as the values obtained in step 9, use Table 4-1 of the CDAM (p. 63) to diferentiate between the various core damage states. As many indicators as possible should be used. Because of overlapping values of release and potential simultaneous conditions of clad damage, overtemperature, and/or core melt, considerable judgement must be applied in making a final determination.

Reference: Westinghouse Owners Group Post Core Damage Assessment Methodology (Modified for BVPS) 1984.

WESTINGHOUSE OWNER'S GROUP

POST ACCIDENT

CORE DAMAGE ASSESSMENT METHODOLOGY

(CDAM)

MODIFIED FOR BVPS

TABLE OF CONTENTS

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 enable utilities to prepare a procedure for relating post accident core damage with measurements of radionuclide concentrations. 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 failures, 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 following generic technical report.

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 step-by-step methodology by way of an example, which can be made applicable to various sizes and types of Westinghouse pressurized water reactors (Section 6).

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 thermocouples, water level within the pressure vessel, containment radiation monitors and hydrogen production are used to develop indicators of the various categories 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 spectrometry techniques.

The nuclides selected for this methodology have sufficient core inventories and radioactive half-lives which 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 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 $\pm$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	
	Sr-90**	28 yr	
	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 2-2

FUEL PELLET INVENTORY*

Inventory, Curies

Nuclide	3-Loop (2652 Mwt)
Kr 85m***	1.6(7)
Kr 87	3.0(7)
Kr 88***	4.2(7)
Xe 131m	4.7(7)
Xe 133	1.4(8)
Xe 133m***	2.1(7)
Xe 135***	2.7(7)
I 131	7.3(7)
I 132	1.1(8)
I 133	1.4(8)
I 135	1.3(8)
Rb 88	4.2(7)
Cs 134	1.7(7)
Cs 137	7.9(6)
Te 129	2.4(7)
Te 132	1.1(7)
Sr 89	5.9(7)
Sr 90***	5.4(6)
Ba 140	1.3(8)
La 140	1.3(8)
La 142	1.1(8)
Pr 144	9.1(7)

* Inventory based on ORIGEN run for equilibrium, end-of-life core.
** Westinghouse Standard Plants.
*** Additional nuclides.

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 for isotopes with less than 1 day half-life and greater than 1 day half-life.

1) Half-life < 1 day

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

2) Half-life > 1 day

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

3) Half life ≈ 1 years

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

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.

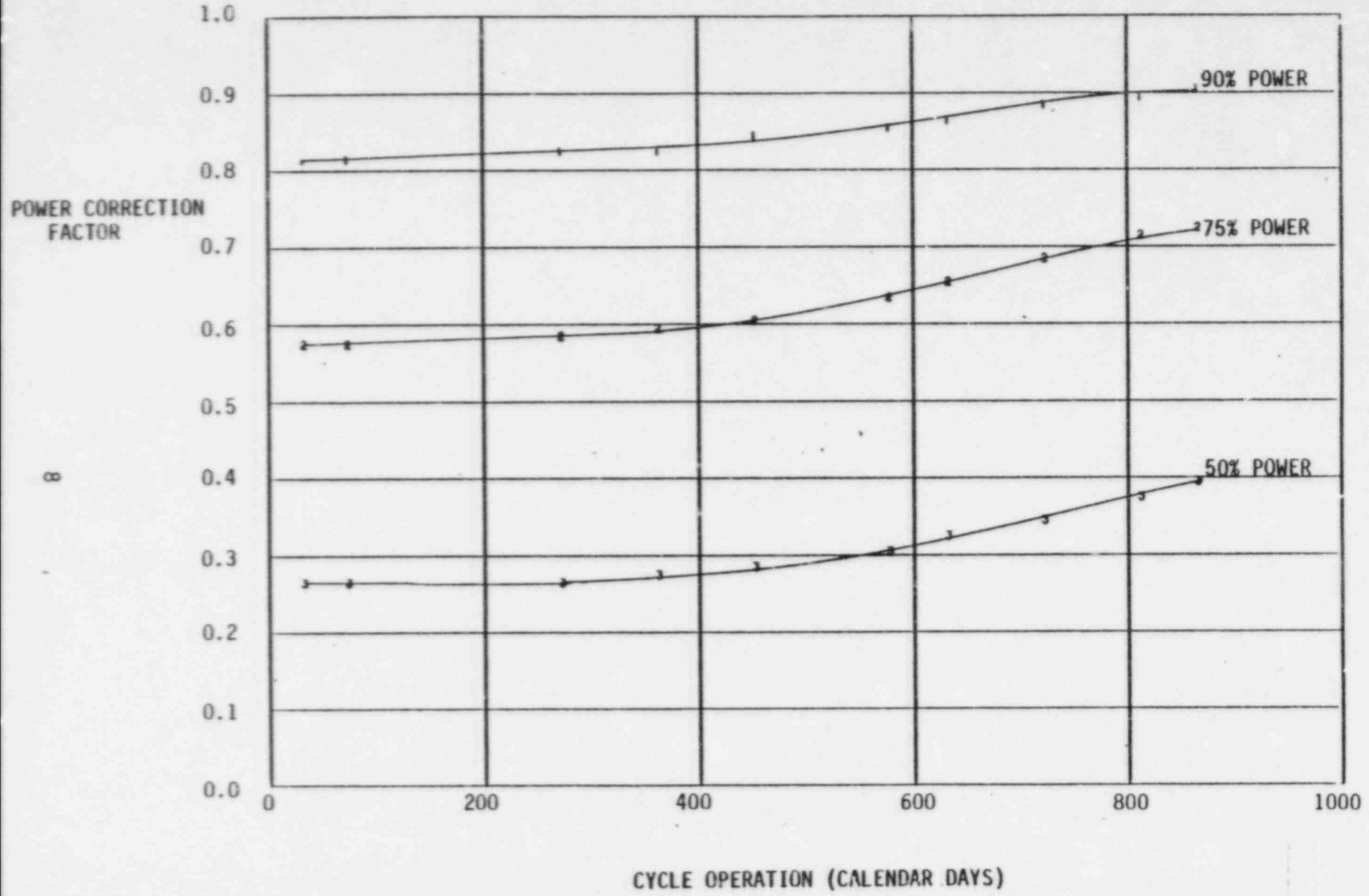

FIGURE 2-1 POWER CORRECTION FACTOR FOR CS-134 BASED ON AVERAGE POWER DURING OPERATION

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

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 cycle operating period is required.

2.4 RELATIONSHIP OF CLAD DAMAGE WITH ACTIVITY

2.4.1 GAP INVENTORY

Since the temperature of the center of the fuel rod is significantly higher than the cladding surface during power operation, a temperature gradient exists across the fuel rod. The volatile fission products migrate along the temperature gradient through the fuel pellet to the fuel rod gap. These fission products, which accumulate in the gap, are isotopes of the noble gases, iodines and cesium elements.

To determine the fission product inventory of the gas gap, the ANS 5.4[4] Standard was used based on 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 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 had occurred.

TABLE 2-3

GAP INVENTORY*

Gap Inventory, Curies

Nuclide	2-Loop (1961 Mwt)	3-Loop (2900 Mwt)	4-Loop (3565 Mwt)	4-Loop (4100 Mwt)
Kr 85m**	2.10(3)	3.08(3)	3.78(3)	4.40(3)
Kr 87	2.00(3)	2.93(3)	3.61(3)	4.20(3)
Kr 88**	4.44(3)	6.49(3)	7.98(3)	9.28(3)
Xe 131m	4.92(2)	7.20(2)	8.85(2)	1.03(3)
Xe 133	9.80(4)	1.43(5)	1.76(5)	2.05(5)
Xe 133m**	9.35(3)	1.37(4)	1.68(4)	1.96(4)
Xe 135**	4.99(3)	7.30(3)	8.98(3)	1.04(4)
I-131	1.58(5)	2.31(5)	2.84(5)	3.30(5)
I-132	2.54(4)	3.71(4)	4.56(4)	5.30(4)
I-133	1.07(5)	1.56(5)	1.92(5)	2.23(5)
I-135	5.44(4)	7.97(4)	9.80(4)	1.14(5)

---

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

** Additional nuclides

TABLE 2-3-1

GAP INVENTORY MINIMUM AND MAXIMUM

Nuclide	2-Loop (1961 Mwt)	3-Loop (2900 Mwt)	4-Loop (3565 Mwt)	4-Loop (4100 Mwt)
	Gap Inventory, Curies (Minimum - Maximum)**			
Kr 85m*	3.84(2)-5.32(3)	5.62(2)-7.79(3)	6.90(2)-9.57(3)	8.03(2)-1.11(4)
Kr 87	3.79(2)-5.13(3)	5.54(2)-7.50(3)	6.81(2)-9.22(3)	7.93(2)-1.07(4)
Kr 88*	7.92(2)-1.11(4)	1.16(3)-1.62(4)	1.42(3)-1.99(4)	1.66(3)-2.32(4)
Xe 131m	8.79(1)-1.23(3)	1.29(2)-1.80(3)	1.58(2)-2.21(3)	1.84(2)-2.57(3)
Xe 133	1.85(4)-2.51(5)	2.71(4)-3.67(5)	3.33(4)-4.51(5)	3.88(4)-5.24(5)
Xe 133m*	7.13(2)-9.82(3)	1.04(3)-1.44(4)	1.28(3)-1.77(4)	1.49(3)-2.06(4)
Xe 135*	2.29(3)-3.12(4)	3.35(3)-4.56(4)	4.11(3)-5.61(4)	4.79(3)-6.53(4)
I 131	3.00(4)-4.09(5)	4.38(4)-5.98(5)	5.39(4)-7.35(5)	6.27(4)-8.55(5)
I 132	4.75(3)-6.50(4)	6.96(3)-9.51(4)	8.55(3)-1.17(5)	9.55(3)-1.36(5)
I 133	1.96(4)-2.72(5)	2.87(4)-3.99(5)	3.53(4)-4.90(5)	4.10(4)-5.70(5)
I 135	9.90(3)-1.39(5)	1.45(4)-2.03(5)	1.78(4)-2.49(5)	2.07(4)-2.90(5)

* Additional nuclides

** 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.

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

RECOMMENDED IODINE SPIKE

Average	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	
0.5 < SA < 1.0	6500
0.1 < SA < 0.5	950
0.05 < SA < 0.1	650
0.01 < SA < 0.5	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.

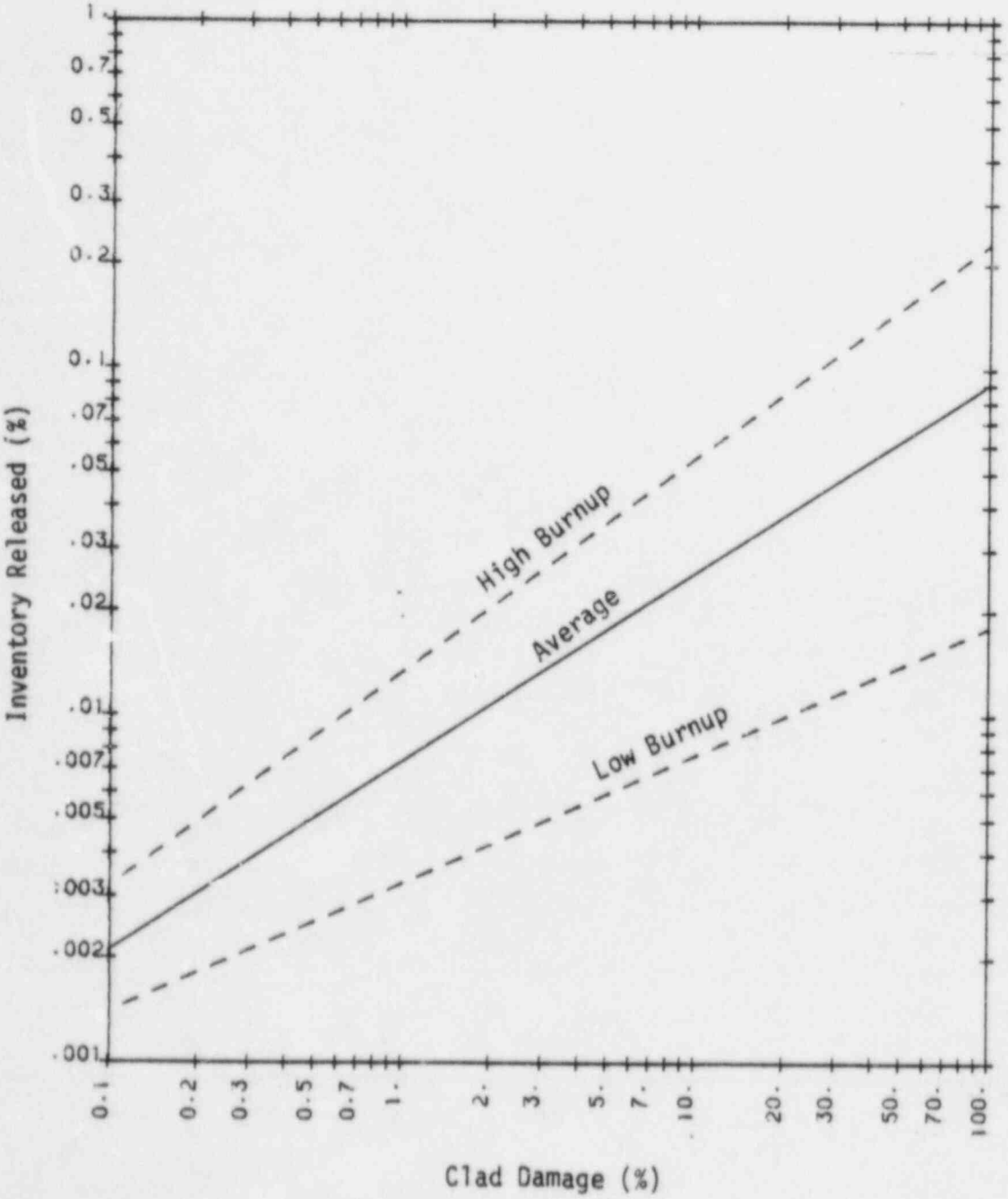

FIGURE 2-2 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF XE-133

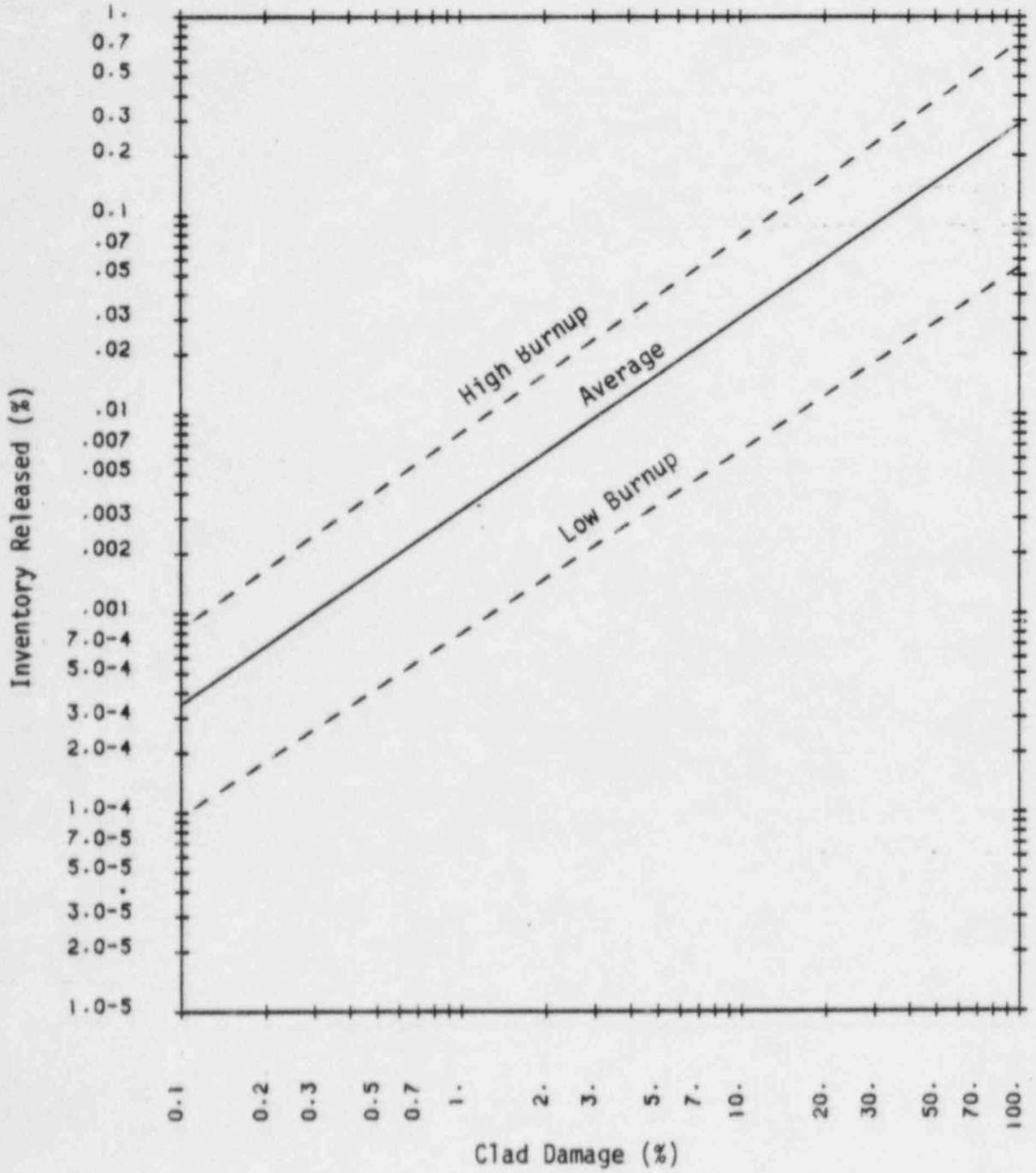

FIGURE 2-3 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-131

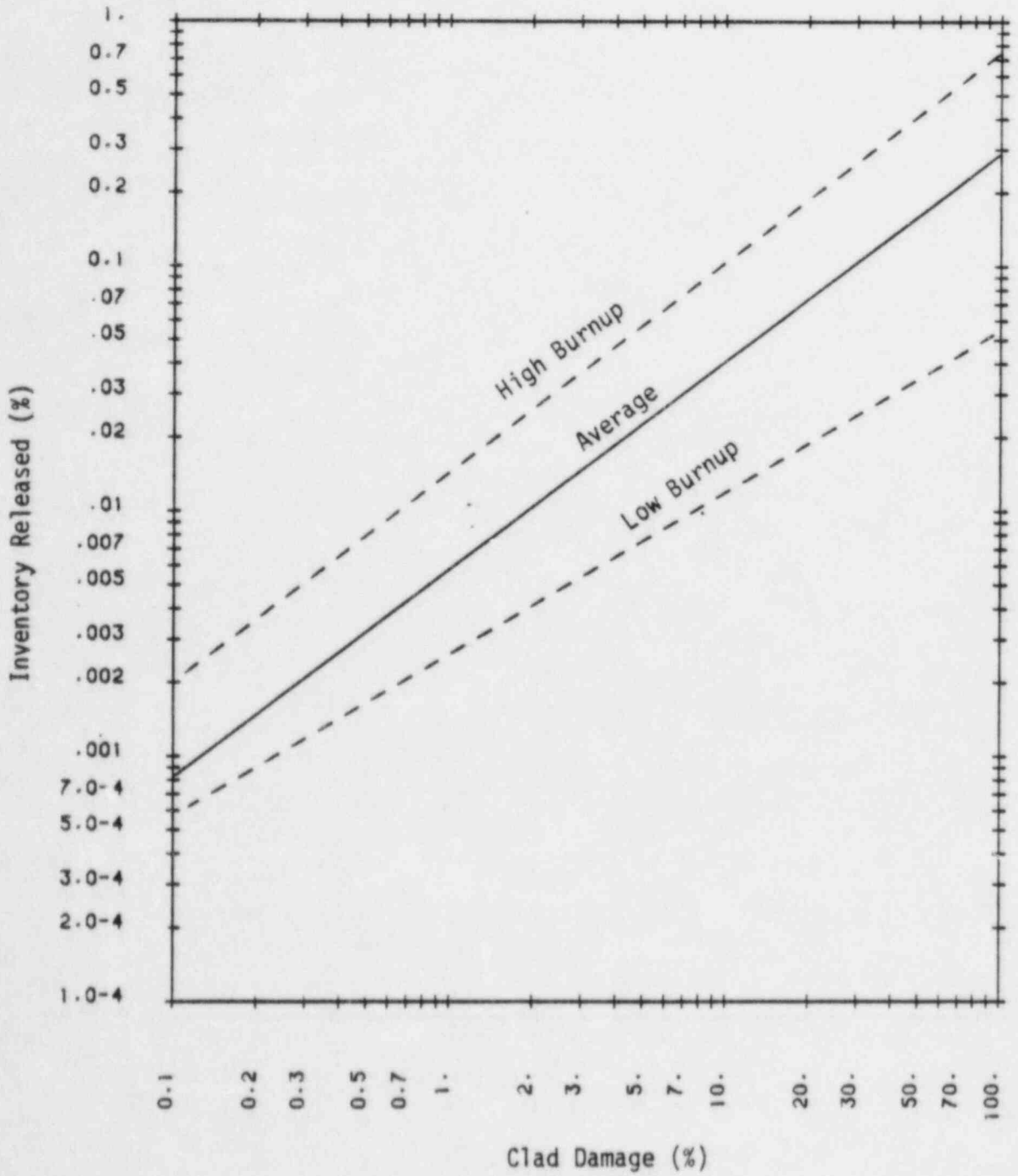

FIGURE 2-4 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-131 WITH SPIKING

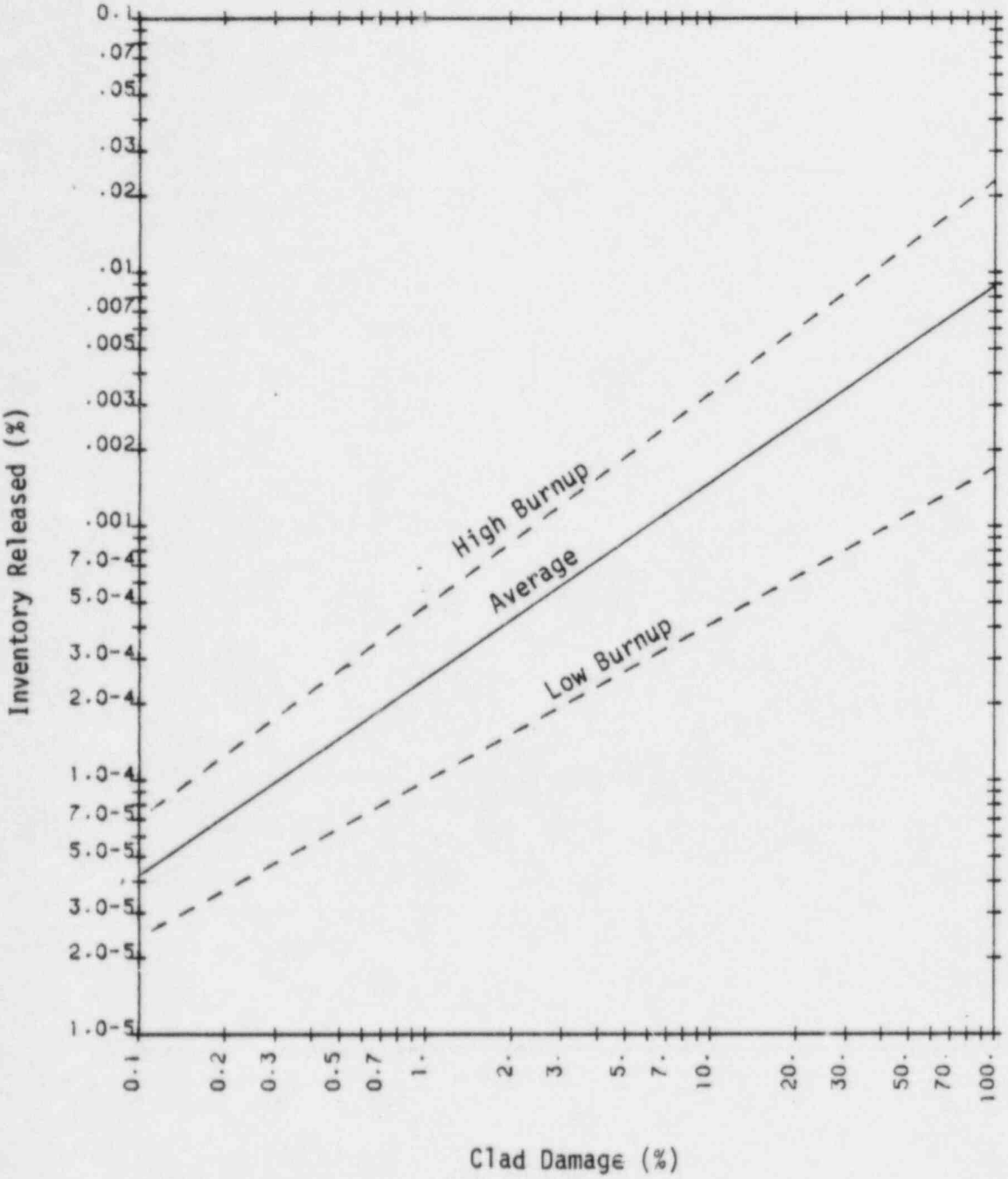

FIGURE 2-5 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF KR-87

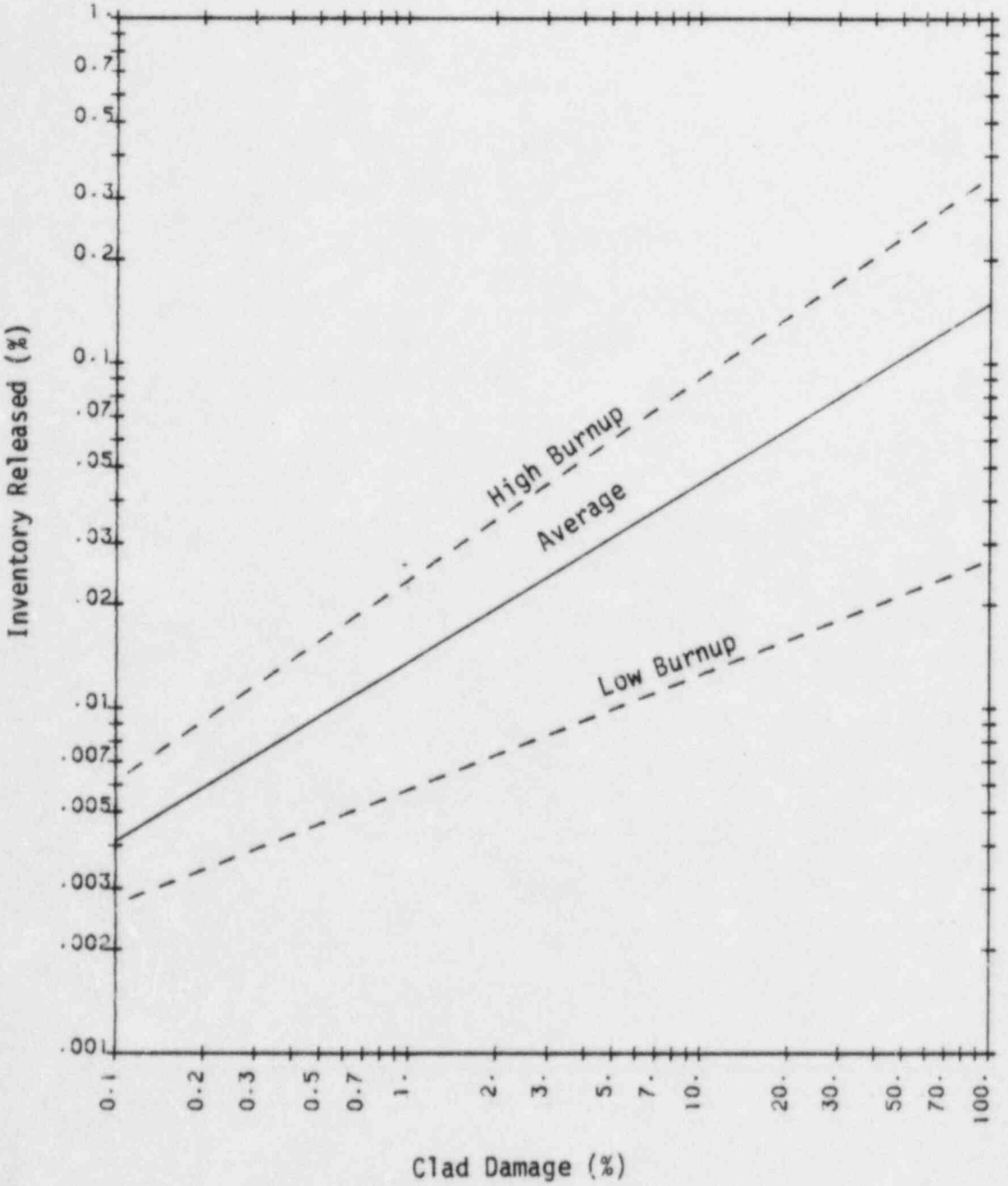

FIGURE 2-6 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF XE-131M

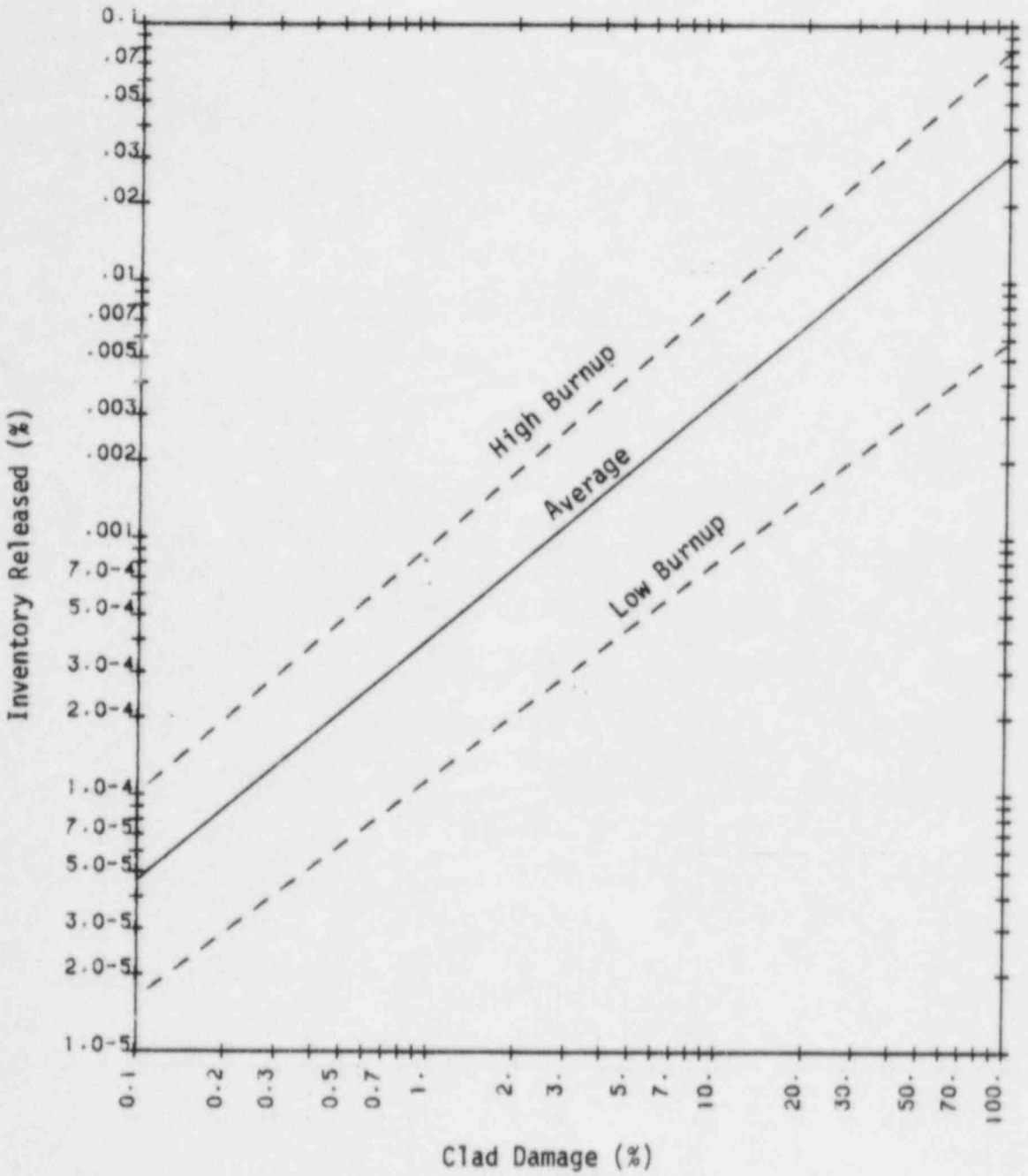

FIGURE 2-7 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-132

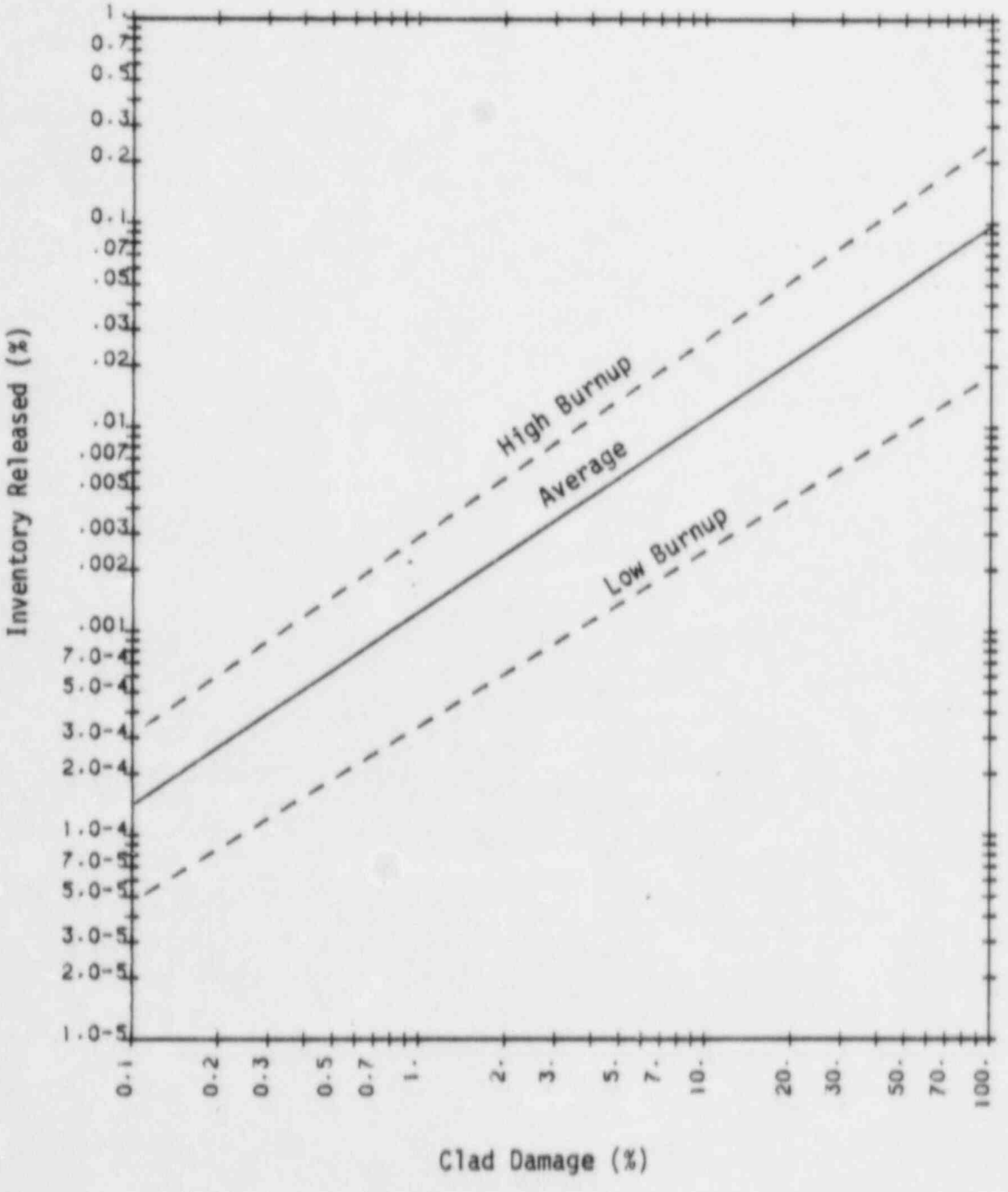

FIGURE 2-8 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-133

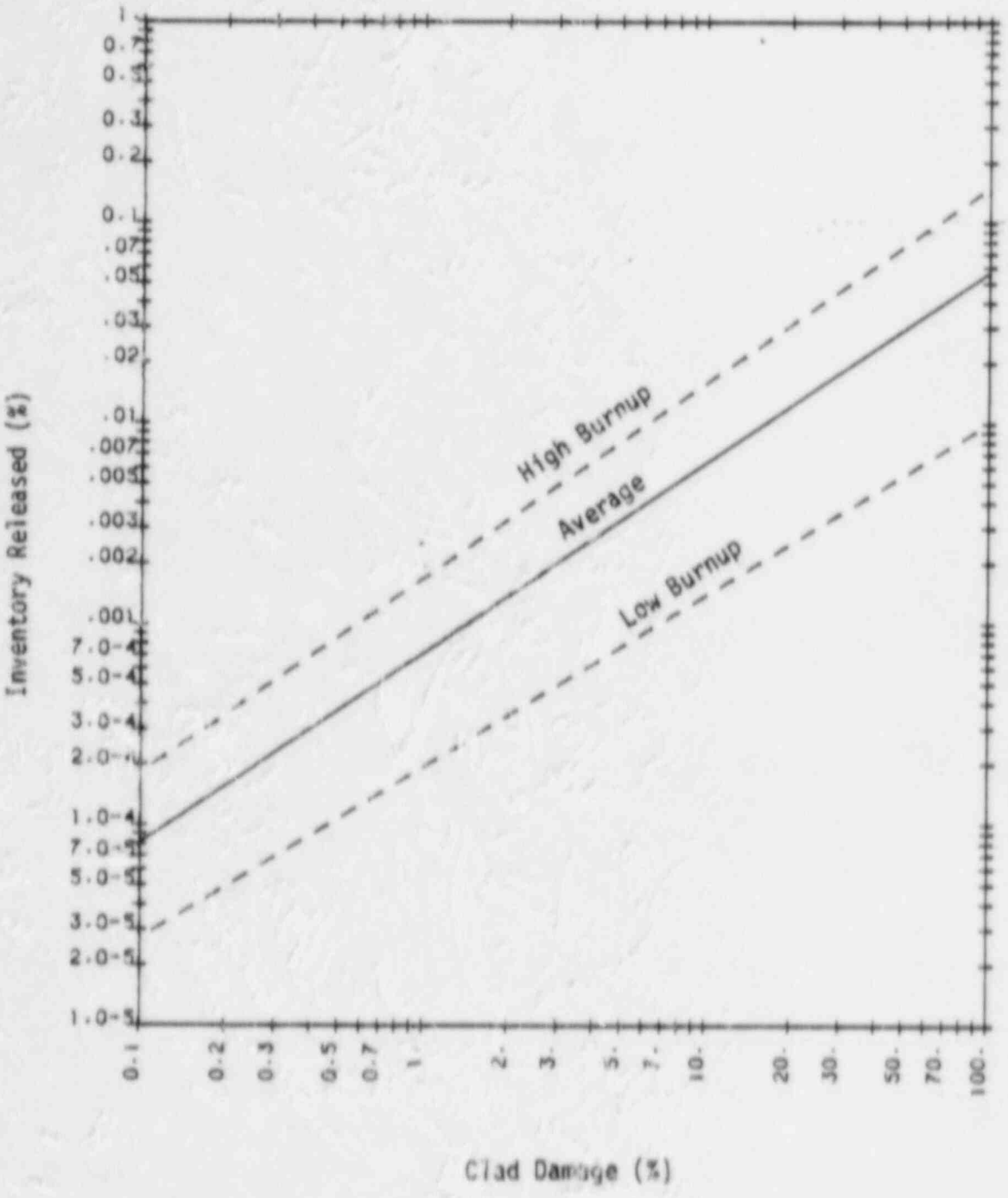

FIGURE 2-9 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-135

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 precentage 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 standard 3-Loop plant is used as the base plant for developing the relation. The total source inventory of Xe-133 for a Westinghouse standard 3-Loop plant is 1.6×10^{8} Curies (Table 2-2). For 100 percent clad damage, all of the gap inventory, which corresponds to 1.43×10^{5} Curies (Table 2-3) 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 $\mu Ci/gm$[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
** Additional nuclides

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. Table 2-3-1 shows the maximum and minimum values for the gap inventories. 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 2.6×10^8 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 standard 3-Loop plant from Table 2-3 for I-131 is 2.31×10^5 Curies. 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. The source inventory of I-131 for a 3-Loop plant is 8.0×10^7 Curies (Table 2-2). 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]. Figure 2-3 is adjusted to account for the release due to iodine spiking. The linear relationships of Figure 2.3 are adjusted 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. An assumption is made here that the only loss term is due to the decay of the nuclides which results in an equilibrium condition where the isotopic production equals the loss due to decay. 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 of 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 clad gap are known.

Assuming 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. 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. Table 2-6 presents the relative activity ratios for both the fuel pellet and the gap.

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

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

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

* Additional nuclides

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 RWST. Also, an account must be noted 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.

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

Note: The specific activities shown here have been decay corrected to reactor shutdown, and the RCS amount has been 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).

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.

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 sump 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.

A. Sample temperatures > 200°F

RCS or Sump Activity (μCi/gm) = Sample Activity (μCi/cc) x

$$\frac{\rho}{\rho_{STP}}(1)^{-1} \times 1/\rho STP$$

where:

$\frac{\rho}{\rho_{STP}}(1)$ = water density ratio at sample temperature, Figure 2-10

ρ_{STP} = water density at standard temperature and pressure = 1.00 gm/cc.

B. Sample temperature < 200°F

RCS or Sump Activity (μCi/gm) = Sample Activity (μCi/cc) x

$1/\rho_{STP}$

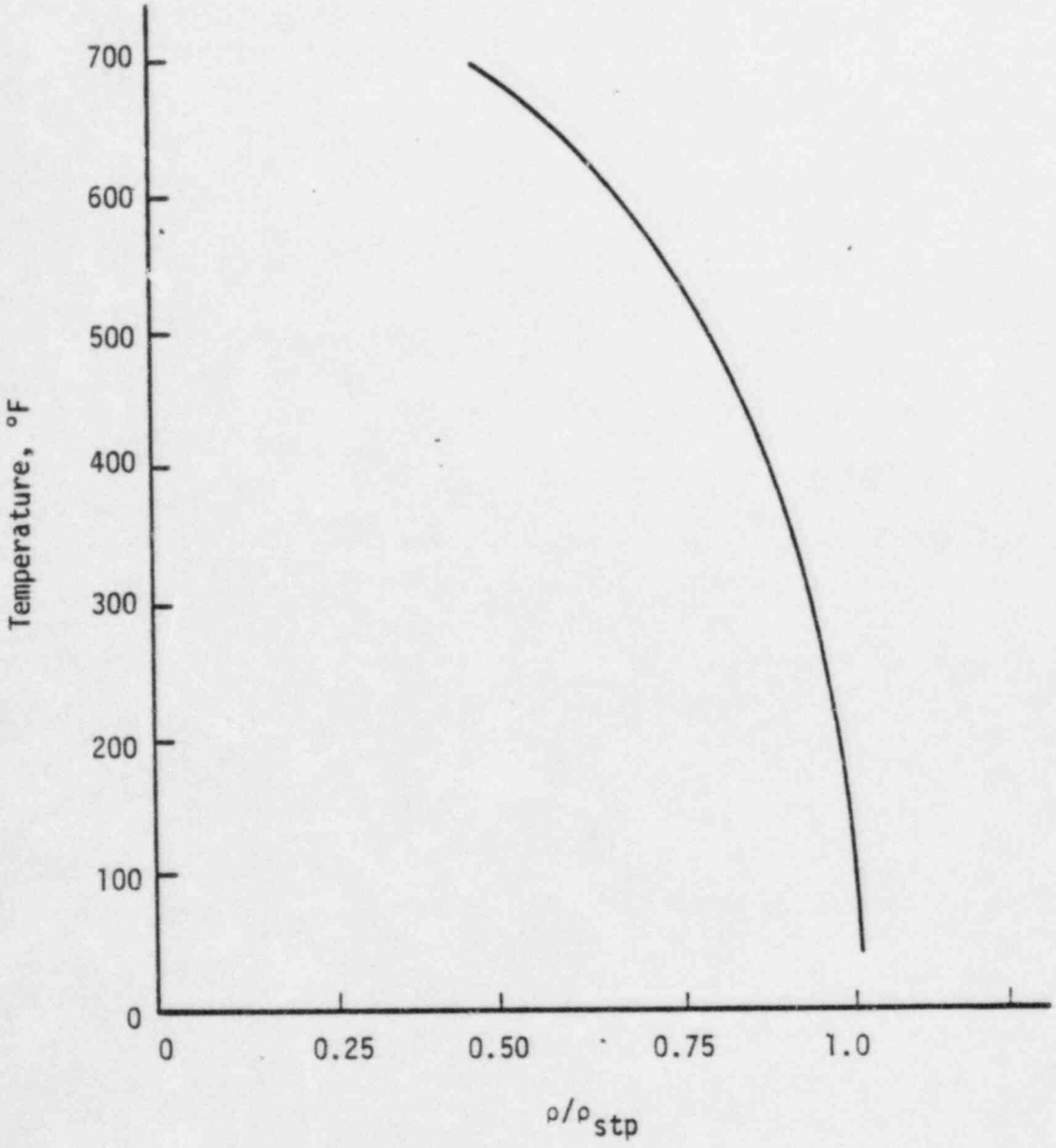

FIGURE 2-10 WATER DENSITY RATIO (TEMPERATURE VS. STP)

where:

ρ_{STP} = water density at standard temperature and pressure = 1.00 gm/cc.

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 of Sump temperature > 200°F

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

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

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

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 dec corrected to time of shutdown.

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.

However, 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 (μ CI/gm or μ Ci/cc) of the parent at shutdown

Q_B^o = activity (Ci) or specific activity (μ Ci/gm or μ Ci/cc) of the daughter at shutdown

Q_B = activity (Ci) or specific activity (μ Ci/gm or μ Ci/cc) of the daughter at time of sample

λ_A = decay constant of the parent, sec^{-1}

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

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 the fracton, an assumption is made that the fraction of source inventory released of the parent and the daughter at time of shutdown are equal. 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.

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

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.

TABLE 2-8

SOURCE INVENTORY OF PARENT NUCLIDES NOT LISTED IN TABLE 2-2

Nuclide	3-Loop (2652 MWt)
Xe-135m	3.1(7)
Sb-129	2.4(7)
Te-129m	5.9(6)
Ba-142	1.2(8)
Ce-144	7.9(7)

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

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

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

A = Fr x measure specific activity (μ Ci/gm or μ Ci/cc)

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

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

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.

TABLE 2-9

EXPECTED FUEL DAMAGE STATE 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

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, 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. 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.

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.

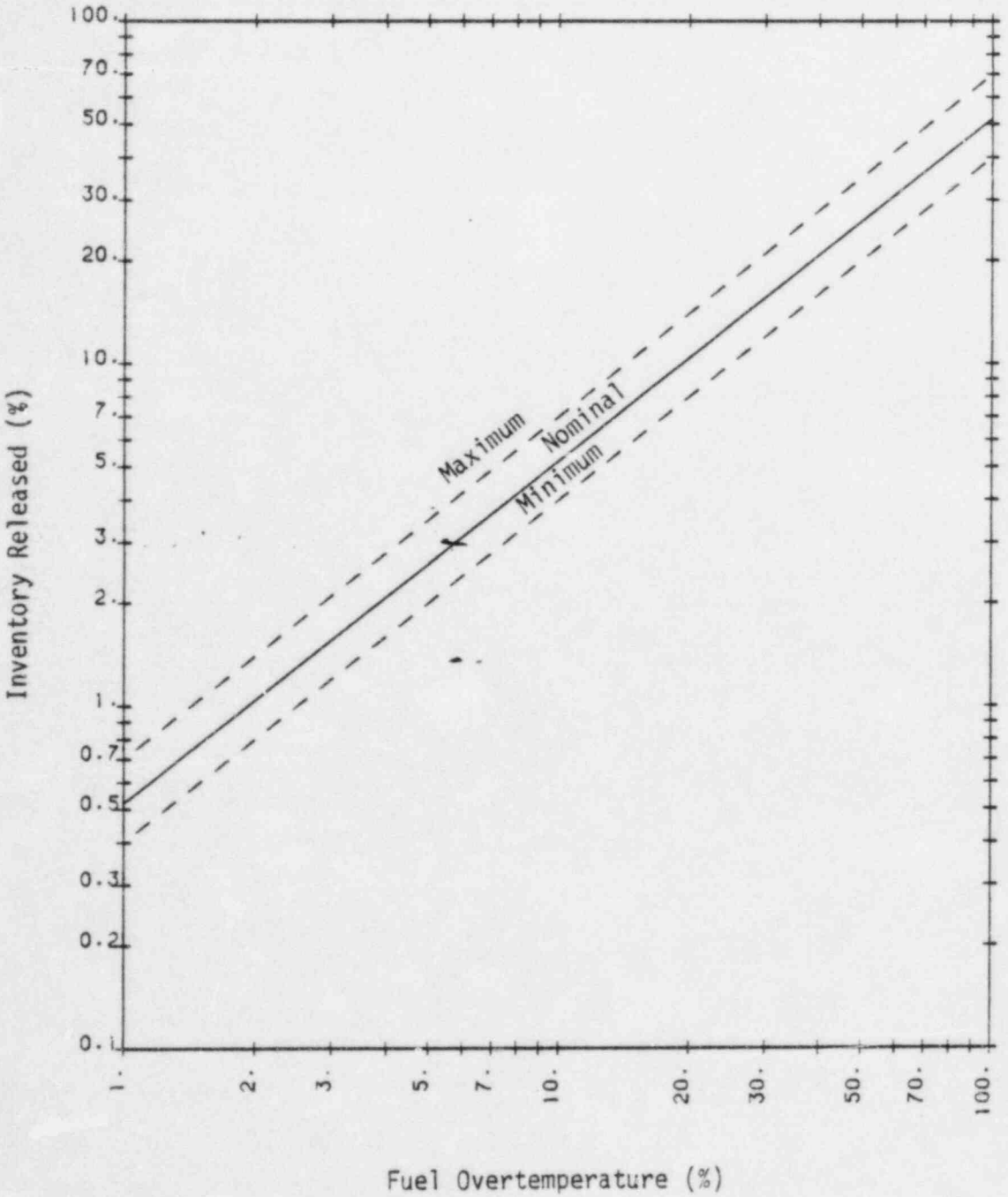

FIGURE 2-11 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % INVENTORY RELEASED OF XE, KR, I, CS

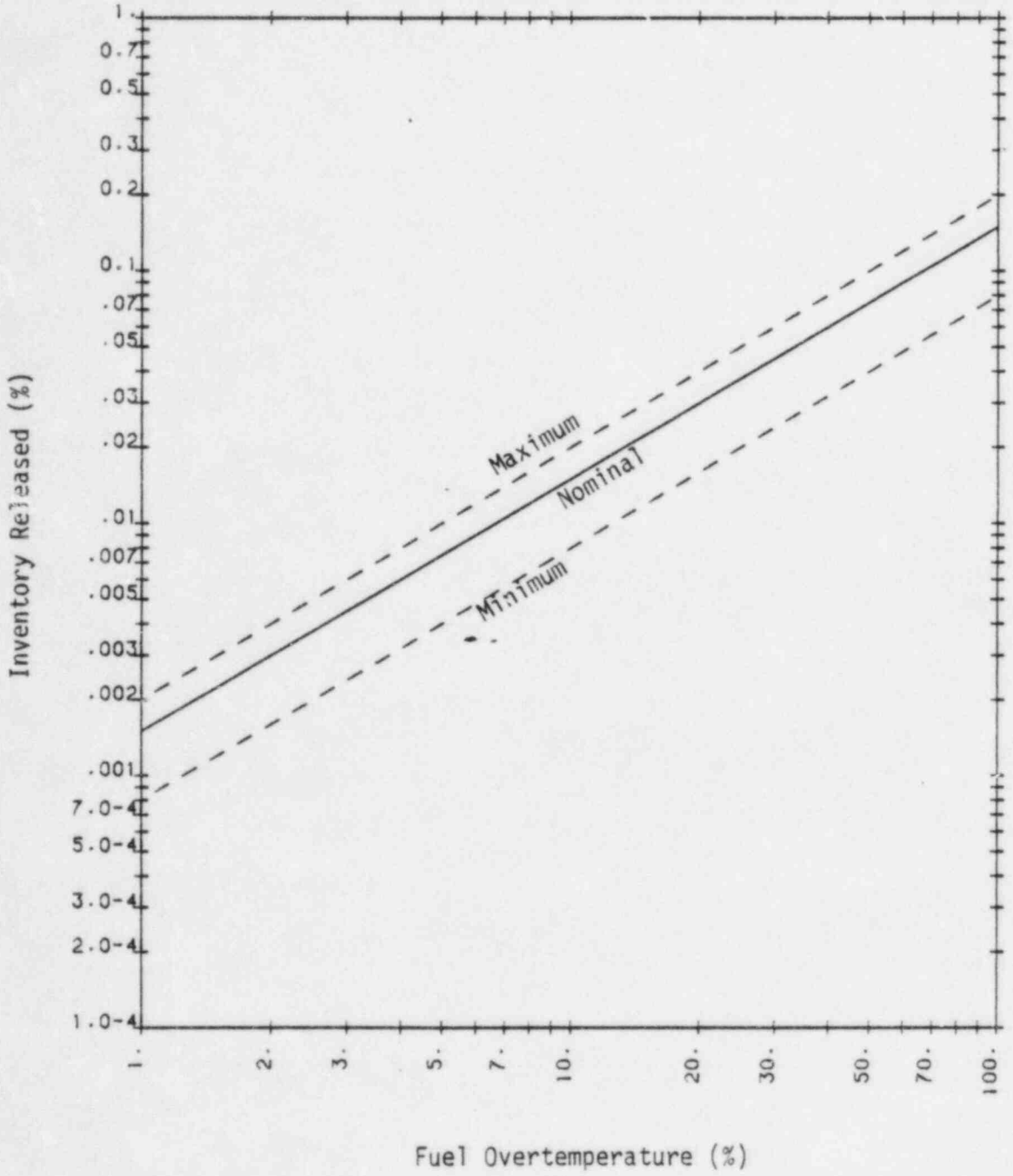

FIGURE 2-12 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % INVENTORY RELEASED OF BA, SR

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 constant

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 release. The assumption is also made that nuclides of any element i.e., 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 at least the principal locations, i.e., 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 specific sample locations to be used during 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	89	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.

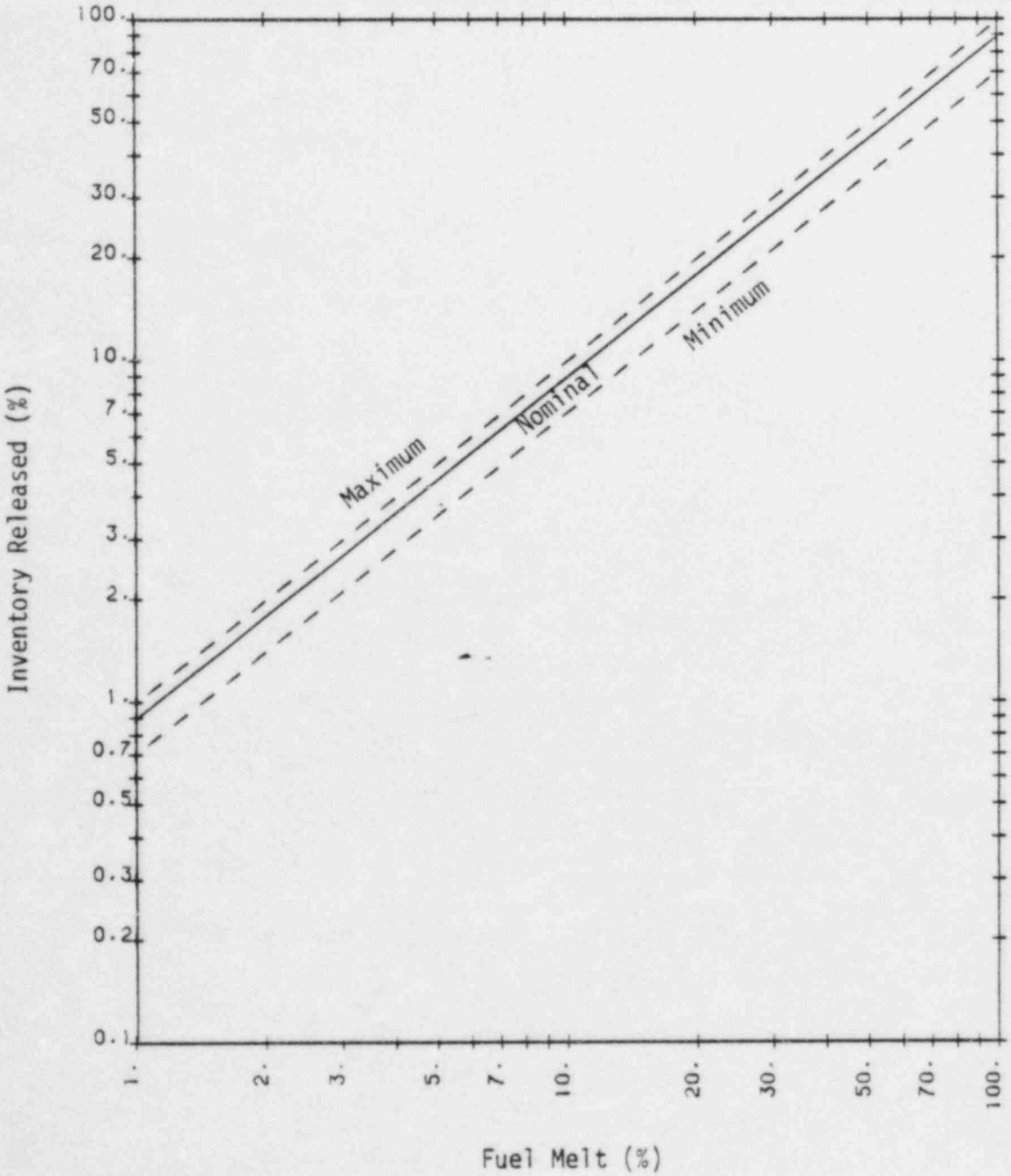

FIGURE 2-13 RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF XE, KR, I, CS, TE

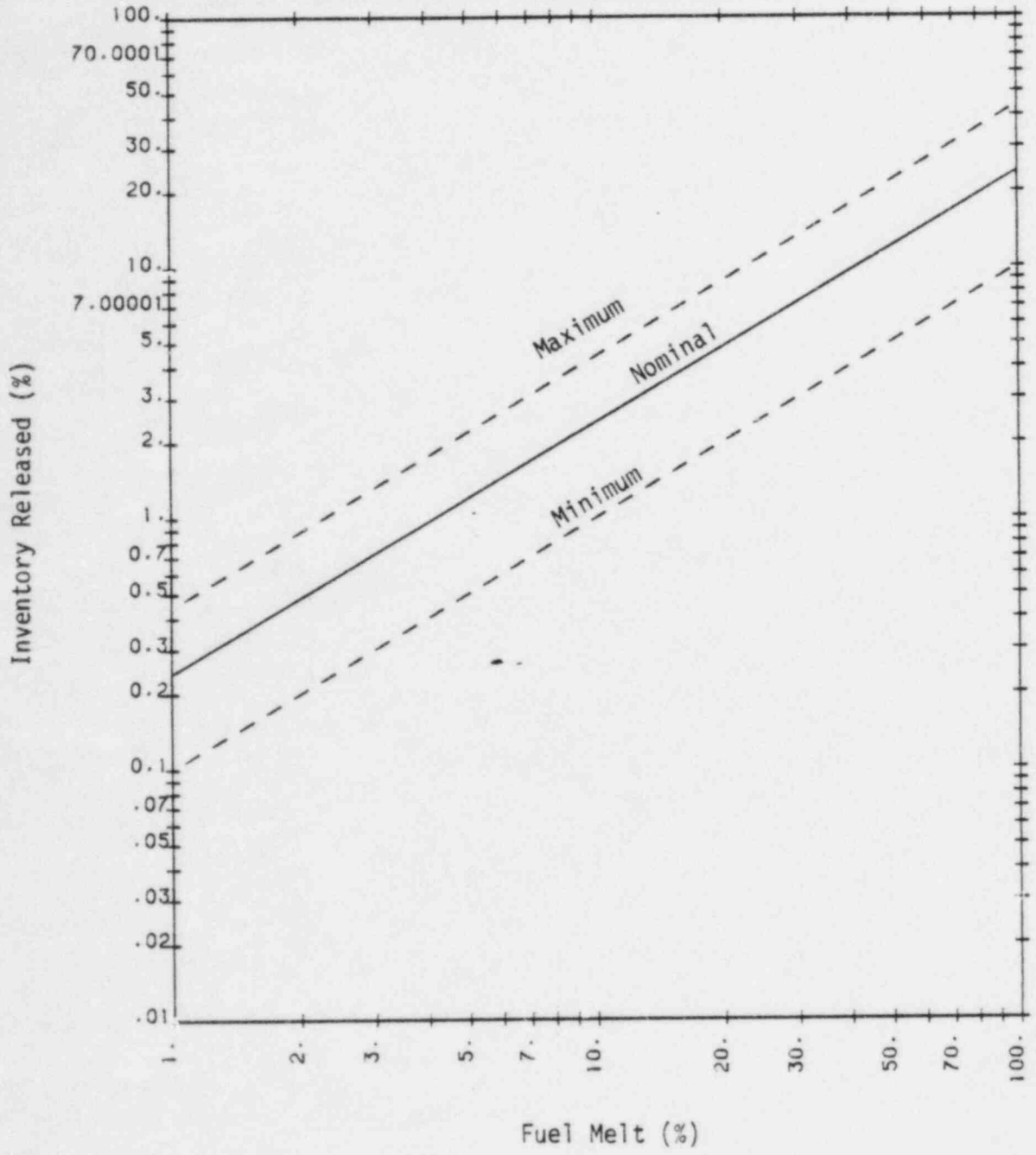

FIGURE 2-14 RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF BA, SR

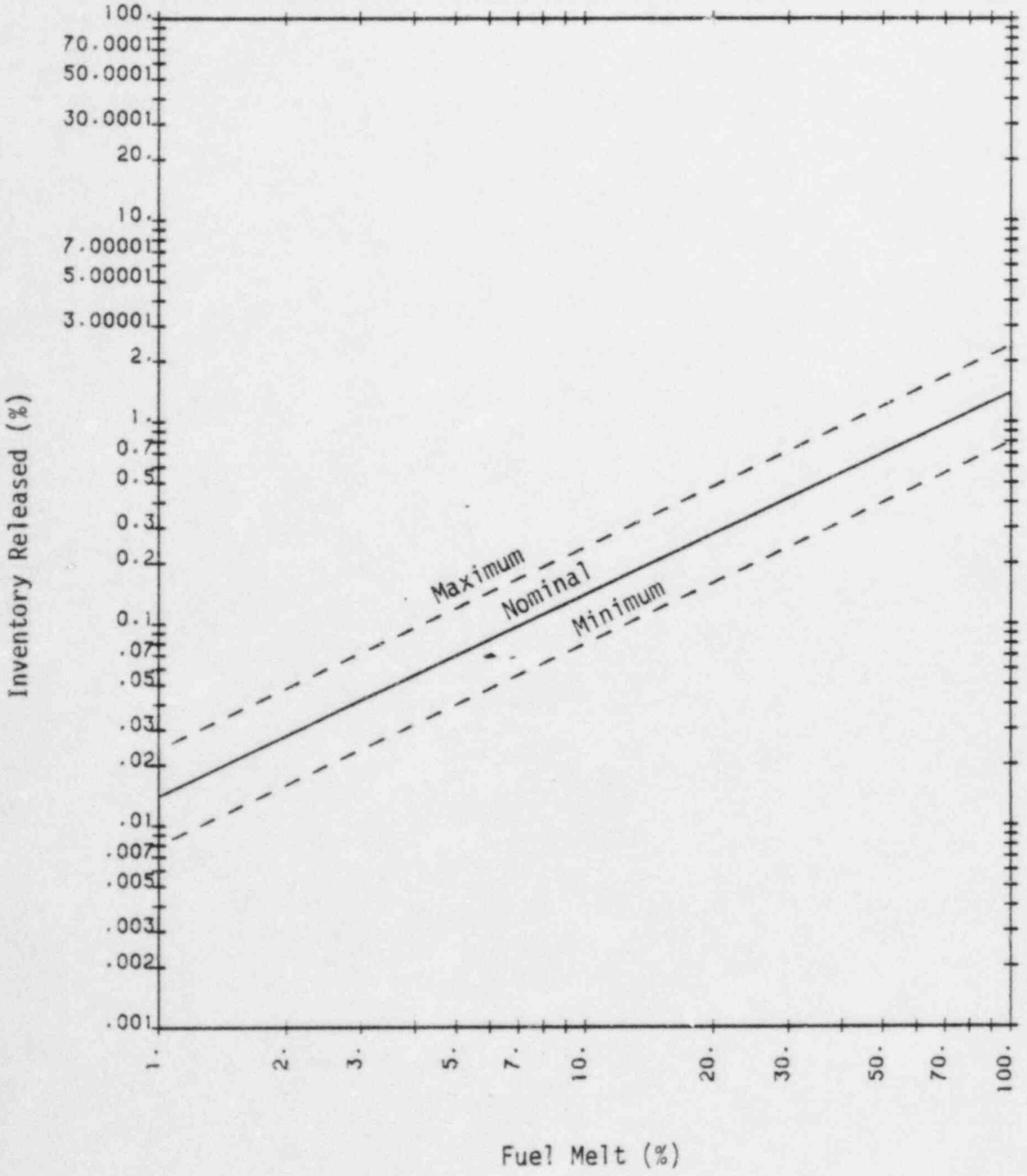

FIGURE 2-15 RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF PR

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.

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, Containment Atmosphere	RCS Pressurizer
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

TABLE 2-12 (Continued)

Suggested Sampling Locations

Scenario	Principal Sampling Locations	Other Sampling Locations
Indication of High Radiation Level in RCS	RCS Hot Leg	RCS Pressurizer

---

* Assume operating at that level for some appreciable time.

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

3.0 AUXILIARY INDICATORS

There are plant indicators which are monitored during an accident which 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. In absence of hydrogen control measures, monitoring this containment hydrogen concentration during the accident can provide an indication of the extent of zirconium water reaction can be approximated. 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. 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

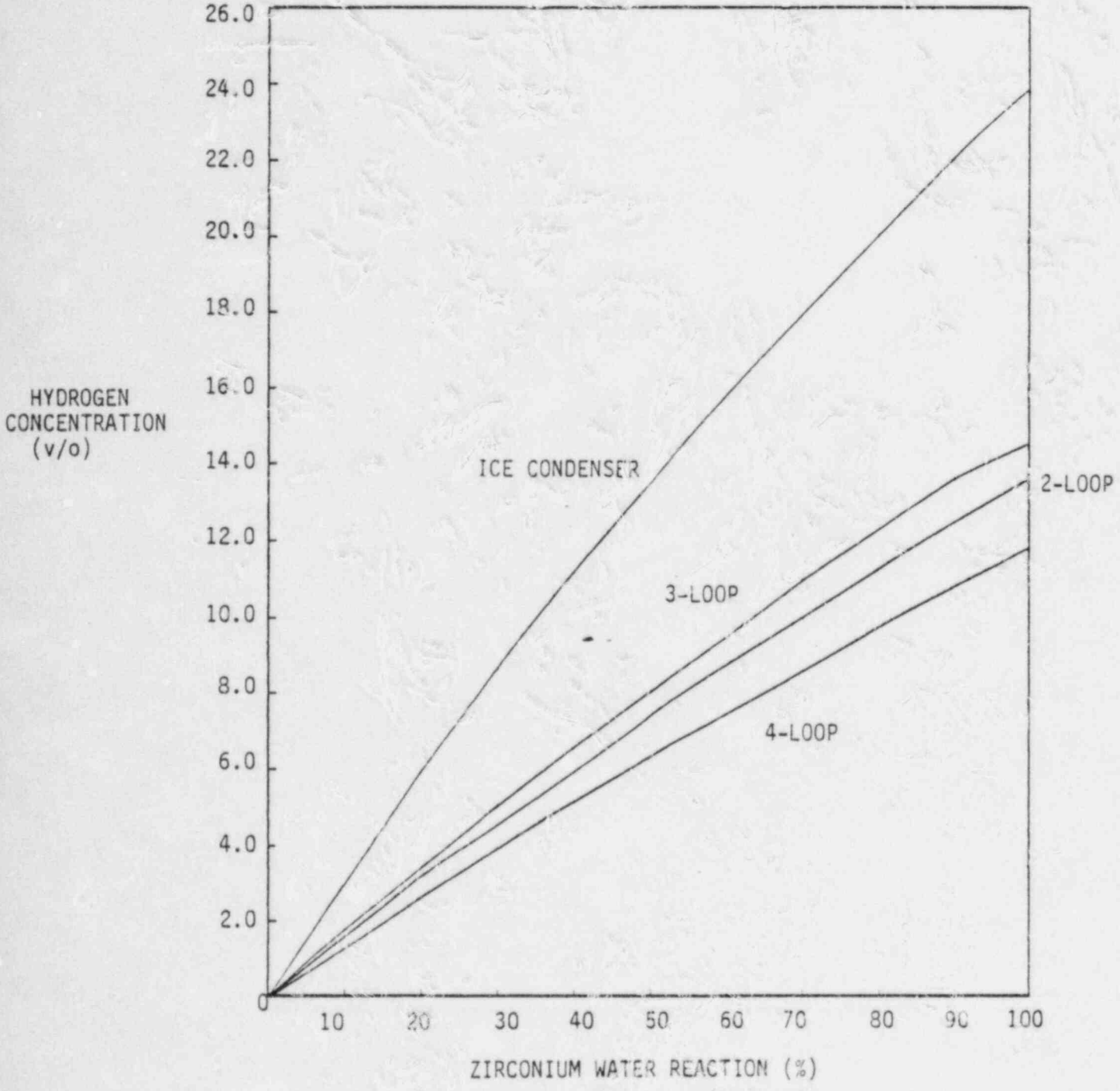

FIGURE 3-1 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM WATER REACTION

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. 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 the Westinghouse Standard two loop, three loop, and four loop plants and the ice condenser containment plant. 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 type of plant, 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.

TABLE 3-1

AVERAGE CONTAINMENT VOLUME AND ZIRCONIUM MASS

Plant Type	Average Zirconium Mass (lbm)	Average Containment Volume (SCF)
2-Loop	23,900	1.2×10^6
3-Loop	37,500	1.7×10^6
4-Loop	47,300	2.8×10^6
Ice Condenser	47,300	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 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].

- 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.

- 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 is likely to have occurred. 1300°F is the lower limit for cladding failures.

- If any RCPs are running, the RVLIS dynamic range is used. Due to the forced flow of the steam water mixture through the core, the CETCs will be good indicators of clad temperatures and no core damage should occur since the steam-water mixture will adequately cool the core.

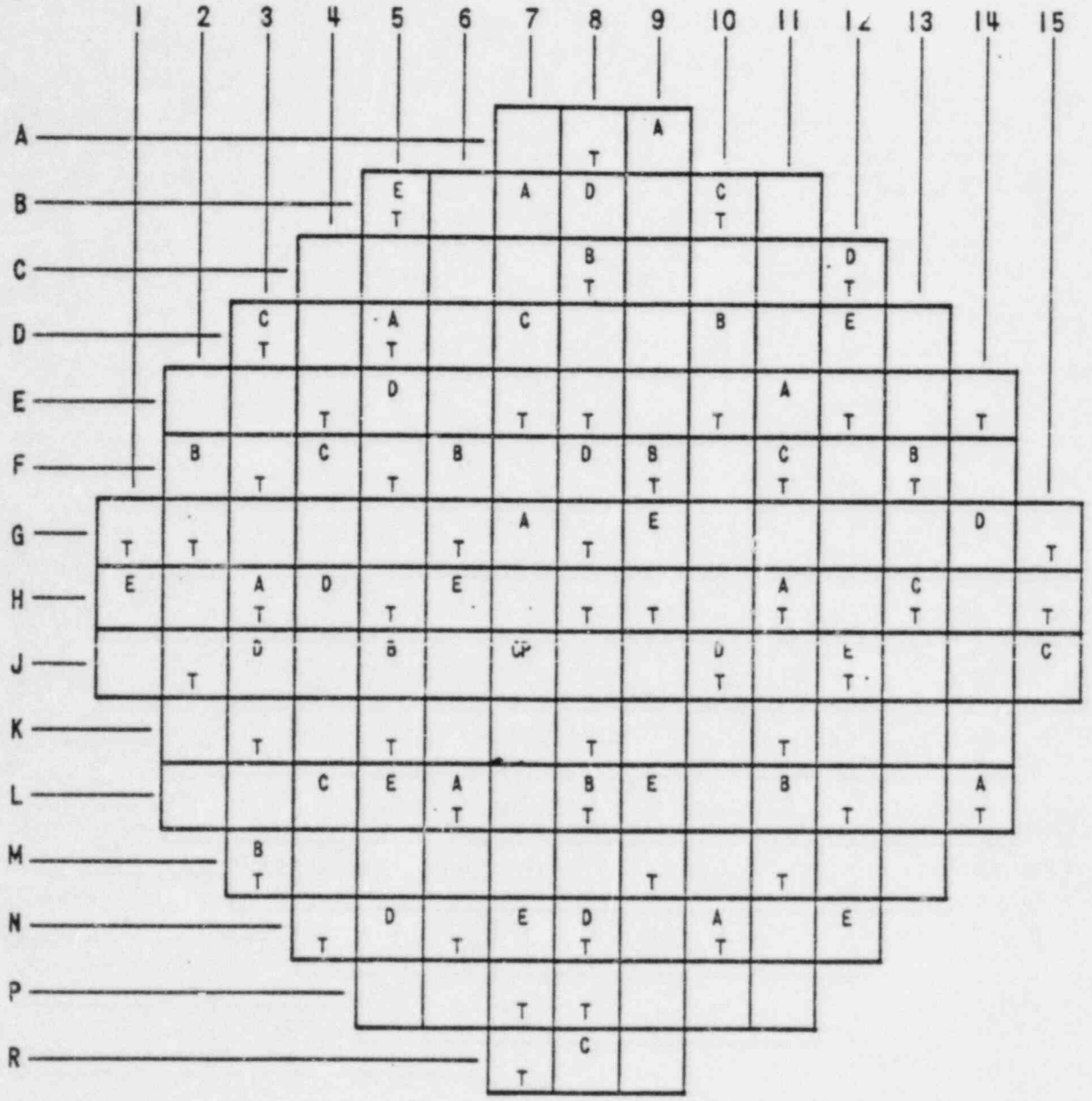

Figure No. 3-2
Core Distribution of Flux Thimbles and Thermocouples

FOR BVPS #1

- If RVLIS full range 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 full range 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[13] 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 in the containment in the range from 1 to 10^7 R/hr.

An analysis has been made to correlate these monitor readings in R/hr with radiogas release. 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. 1.0×10^{-3} percent of the gamma ray source strength is used to represent the release of the normal operating activity (ANS 18.1[6]) of the noble gases in the primary system.

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. 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

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

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.

An example of the relationship of the exposure rate of a detector as a function of time following reactor shutdown is presented in Figure 3-3. The exposure rates, which are expressed in units of R/hr-MWt, are based on a free containment volume of 2×10^6 ft^3 with a detector located 57.5 ft below the apex of the containment dome. No objects or components shield the detector from the noble gas sources which are assumed to be uniformly distributed throughout the containment building.

The methodology of using the relationship of containment radiogas monitors readings shown in Figure 3-3 is:

1. Determine time lapse between core shutdown and radiation reading.

2. Record containment monitor reading in R/hr at this time.

3. Correct the monitor reading for specific plant power via the relationship:

$$\text{R/hr-MWt} = \frac{\text{Radiation Monitor Reading}}{\text{Plant Power (MWt)}}$$

4. Determine core damage regime from Figure 3-3 at the time interval ascertained in step 1.

For plants which have the same monitor characteristics as the monitor described above, except for the containment volume, Figure 3-3 can be used provided a correction is made to the exposure rate (R/hr) as follows.

$$\text{R/hr-MWt} = \frac{\text{Radiation Monitor Reading (R/hr)} \times \text{Containment Vol. } (ft^3)}{\text{Plant Power (MWt)} \quad \times \quad 2 \times 10^6 \text{ ft}^3}$$

FOR BVPS #1:

$$R/HR - MWT = \frac{RM\text{-}219\ READING\ (R/HR)}{2652\ MWt} \times .9$$

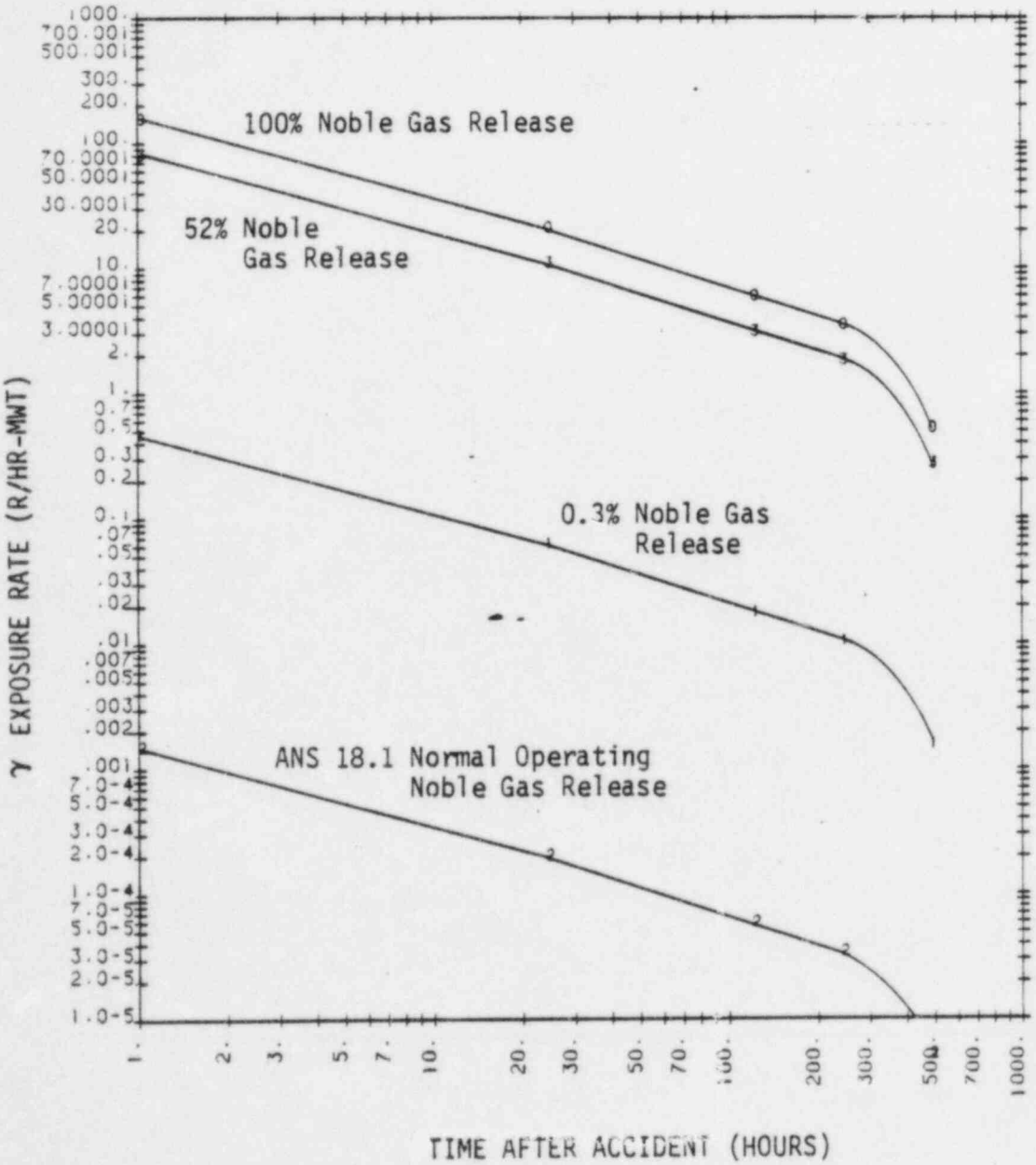

FIGURE 3-3 PERCENT NOBLE GASES IN CONTAINMENT

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 - Mwt) 10 hrs after shutdown**	Core Exit Thermocouples Readings (Deg F)	Extent and Duration of Core Uncovery (Minutes)	Hydrogen Monitor (Vol % H_2)***
No clad damage	Kr-87 < $1x10^{-3}$ Xe-133 < $1x10^{-3}$ I-131 < $1x10^{-3}$ I-133 < $1x10^{-3}$	Kr-87 = 0.022 I-133 = 0.71	-	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 10^{-3} - 0.1 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 - .08	750 - 1300	Core uncovery	0 - 7
50-100% clad damage	Kr-87 0.1 - 0.2 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	0.08 to 0.16	1300 - 1650	Core uncovery	7 - 14
0-50% fuel pellet overtemperature	Xe-Kr,Cs,I 1 - 20 Sr-Ba 0 - 0.4	Kr-87 = 0.22 I-133 = 2.1	0.16 to 21	> 1650	Core uncovery	7 - 14
50-100% fuel pellet overtemperature	Xe-Kr,Cs,I 20 - 40 Sr-Ba 0.4 - 0.8	Kr-87 = 0.22 I-133 = 2.1	21 to 42	> 1650	Core uncovery	7 - 14
0-50% fuel melt	Xe,Kr,Cs,I 40 - 70 Sr-Ba 0.2 - 0.8 Pr-Rb 0.1 - 0.8	Kr-87 = 0.22 I-133 = 2.1	42 to 70	> 1650	Core uncovery	7 - 14
50-100% fuel melt	Xe,Kr,Cs,I,Te > 70 Sr,Ba > 24 Pr,Rb > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 70	> 1650	Core uncovery	7 - 14

* 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 plant specific parameters and times other than 10 hours. These values are from Figure 3-3 and represent a specific detector geometry.

*** Ignitors may obviate these values.

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

5.0 LIMITATIONS

The emphasis of this procedure 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 representative sample, radiochemical analysis via gamma spectrum stripping techniques are used to caluclate the specific activity of various fission products released from the fuel. Radiochemical analysis of post-accident samples, which 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 be exercised in accounting for volumes of any water added via eccs and spray systems, accumulators, chemical addition tanks, amd 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 + or - 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 in 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 Section 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 22 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. Thus, estimates are just that, estimates, and 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 procedure 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.

6.0 EXAMPLE OF CORE DAMAGE ASSESSMENT METHODOLOGY

The following example is presented to illustrate the use of this methodology in assessing the extent of core damage.

6.1 SAMPLING RESULTS

For this example, a Westinghouse 3-Loop plant has experienced an accident where the plant's monitoring systems indicate that safety injection has initiated and a significant amount of water has accumulated in the sump. Samples are available from the primary coolant (RCS hot leg), the containment sump, and the containment atmosphere 6 hours after reactor shutdown. The results of the sampling are presented in Table 6-1.

6.2 DECAY CORRECTION

The specific activities determined by the sampling analysis are decay corrected to the time of reactor shutdown. A sample calculation is presented here.

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

where:

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

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

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

A_o = decay corrected specific activity μCi/gm or μCi/cc.

TABLE 6-1

RESULTS OF SAMPLING ANALYSIS TAKEN
6 HOURS AFTER REACTOR SHUTDOWN

Isotope	Atmosphere, μCi/cc	Specific Activity Sump, μCi/gm	RCS, μCi/gm
Kr 87	1.8(1)		
Xe 133	1.9(3,		
I 131		2.6(4)	6.9(4)
I 132		1.07(4)	2.9(4)
Cs 137		2.6(3)	6.5(3)
Ba 140		4.4(4)	1.3(5)

For I-131 primary coolant specific activity, Table 6-1,

$A = 6.9\ (4)$

$\lambda_i = 1.0 \times 10^{-6}\ sec^{-1}$

$t = 21600\ sec.$

$$A_o = \frac{6.9\ (4)}{e^{-(1.0(-6))\ (21600)}}$$

$A_o = 7.0\ (4)$

For I-132, parent daughter relationship must be considered in calculation of decay adjustment. Following the methodology outlined in Section 2.4.5.3, the decay correction calculation is as follows.

Parent-Daughter: Te-132 - I-132

1. Hypothetical activity of I-132 (daughter) 6 hours after shutdown, assuming 100 percent release of Te-132 and I-132.

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

where:

Q_A^o = 100% source inventory of Te-132, Table 6.2A = 1.2 (7) Ci

Q_B^o = 100% source inventory of I-132, Table 6.2A = 1.2 (8) Ci

K = decay branching factor, Table 6.2B = 1.00

λ_A = Te-132 decay constant = 2.48 (-6) sec^{-1}

TABLE 6-2A

SOURCE INVENTORY*

Nuclide	Inventory, Ci
Kr-85m	1.8(7)
Kr-87	3.3(7)
Kr-88	4.6(7)
Xe-131m	5.1(5)
Xe-133	1.6(8)
Xe-133m	2.3(7)
Xe-135	3.0(7)
I-131	8.0(7)
I-132	1.2(8)
I-133	1.6(8)
I-135	1.4(8)
Rb-88	4.7(7)
Cs-134	1.9(7)
Cs-137	8.7(6)
Te-129	2.7(7)
Te-132	1.2(7)
Sr-89	6.4(7)
Sr-90	5.9(6)
Be-140	1.4(8)
La-140	1.4(8)
La-142	1.2(8)
Pr-144	1.0(8)

* The source inventory of a 3-Loop (2900 Mwt) plant is used in this example.

TABLE 6-2B

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

λ_B = I-132 decay constant = 8.52 (-5) sec^{-1}

t = time from shutdown to sample time = 21600 sec

Q_B = 9.75 (6) Ci + 1.91 (7) Ci

Q_B = 2.89 (7) Ci of I-132 6 hours after shutdown

2. Contribution of decay of only I-132 to hypothetical activity of I-132.

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

$$= \frac{1.91 \, (7)}{2.89 \, (7)}$$

Fr = 0.66

3. Amount of measured sample specific activity due to decay of just I-132.

A = Fr x measured specific activity (i.e., RCS), Table 6-1

= 0.66 x 2.9 (4)

A = 1.9 (4)

4. Decay correct specific activity (A) of I-132 in RCS to shutdown.

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

$$= \frac{1.9 \, (4)}{.16}$$

A_o = 1.2 (5)

Table 6-2C lists the decay corrected specific activities of the sampling analysis.

6.3 PRESSURE AND TEMPERATURE CORRECTION

As discussed in Section 2.4.5.2, a correction is needed to the sample specific activity only if the temperature and pressure of the actual sample are different than the temperature and pressure of the medium from which the sample was taken. Since the measured specific activity of the RCS and sump samples are based on gram of water, no adjustment to the specific activities is required. The conditions of the medium and the sample are listed below.

Containment Atmosphere	Atmosphere Sample	Correction Factor
Pressure = 20 psia Temperature = 200°F	Pressure = 15 psia Temperature = 100°F	1.1

Containment Sump	Sump Sample	Correction Factor
Pressure = 20 psia Temperature = 125°F	Pressure = 20 psia Temperature = 125°F	1.0

Primary Coolant	RCS Sample	Correction Factor
Pressure = 1500 psia Temperature = 350°F	Pressure = 500 psia Temperature = 150°F	1.0

Correction factor calculations are shown below.

For containment atmosphere sample,

$$\text{Correction Factor} = \frac{P_2\ (T_1 + 460)}{P_1\ (T_2 + 460)}$$

TABLE 6-2C

DECAY CORRECTED SPECIFIC ACTVITIES OF SAMPLING ANALYSIS

Nuclide	Location	Measured Specific Activity*	X	Parent-Daughter Factor**	X	Decay Factor***	=	Decay Corrected Specific Activity*
Kr-87	Atmosphere	1.8(1)		N/A		26.6		4.8(2)
Xe-133	Atmosphere	1.9(3)		0.97		1.03		1.9(3)
I-131	Sump	2.6(4)		N/A		1.02		2.7(4)
I-131	RCS	6.9(4)		N/A		1.02		7.0(4)
I-132	Sump	1.07(4)		0.66		6.25		4.4(4)
I-132	RCS	2.9(4)		0.66		6.25		1.2(5)
Cs-137	Sump	2.6(3)		N/A		1.00		2.6(3)
Cs-137	RCS	6.5(3)		N/A		1.00		6.5(3)
Ba-140	Sump	4.4(4)		N/A		1.01		4.4(4)
Ba-140	RCS	1.3(5)		N/A		1.01		1.3(5)

* μ Ci/cc for atmosphere sample or μ Ci/gm for sump and RCS sample.

** Fraction of measured specific activity due to decay of only the daughter.

*** Decay factor = $1/e^{-\lambda it}$ where t = 21600 sec.

where:

P_1 = sample pressure = 15 psia

T_1 = sample temperature = 100°F

P_2 = containment pressure = 20 psia

T_2 = containment temperature = 200°F.

$$\text{Correction Factor} = \frac{20}{15} \quad \left(\frac{100 + 460}{200 + 460}\right) = 1.1$$

Tables 6-3, 6-4, and 6-5 lists the corrected specific activities due to pressure and temperature differences.

6.4 ACTIVITY OF EACH MEDIUM

The volume of the containment atmosphere and the mass of the sump and the primary coolant need to be known to determine the amount of Curies in each medium. Tables 6-6, 6-7, and 6-8 lists the activity of each medium.

1. Containment Volume

$$V = 1.7 \times 10^6 \text{ SCF} \times \frac{28.3 \times 10^3 \text{ cc}}{\text{SCF}} \times \frac{P_3}{P_2} \times \left(\frac{T_2 + 460}{T_3 + 460}\right) = 4.7 \times 10^{10} \text{ cc}$$

where:

P_2 = containment pressure = 20 psia
T_2 = containment temperature = 200°F
P_3 = standard pressure = 14.7 psia
T_3 = standard temperature = 32°F.

2. Sump Mass

The sump water level monitor indicates the sump is 50% full. For the purposes of this example, this corresponds to a water volume of 50,000 ft^3. The sump temperature is below 200°F and no adjustment is necessary in converting the sump volume to sump mass.

TABLE 6-3

ADJUSTED SPECIFIC ACTIVITY

DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

	Containment Atmosphere, μCi/cc		
Isotope	Specific Activity From Table 6-2c	Correction Factor	Specific Activity Adjusted
Kr 87	4.8(2)	1.1	5.3(2)
Xe 133	1.9(3)	1.1	2.1(3)
I 131			
I 132			
Cs 137			
Ba 140			
La 140			

TABLE 6-4

ADJUSTED SPECIFIC ACTIVITY
DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

	Containment Sump, μCi/gm		
Isotope	Specific Activity From Table 6-2c	Correction Factor*	Specific Activity Adjusted
Kr 87			
Xe 133			
I 131	2.7(4)	1.0	2.7(4)
I 132	4.4(4)	1.0	4.4(4)
Cs 137	2.6(3)	1.0	2.6(3)
Ba 140	4.4(4)	1.0	4.4(4)

* No correction is necessary since the nuclide analysis was performed on a per gram basis.

TABLE 6-5

ADJUSTED SPECIFIC ACTIVITY
DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

RCS, μCi/gm

Isotope	Specific Activity From Table 6-2c	Correction Factor*	Specific Activity Adjusted
Kr 87			
Xe 133			
I 131	7.0(4)	1.0	7.0(4)
I 132	1.2(5)	1.0	1.2(5)
Cs 137	6.5(3)	1.0	6.5(3)
Ba 140	1.3(5)	1.0	1.3(5)

* No correction is necessary since the nuclide analysis was performed on a per gram basis.

TABLE 6-6

CONTAINMENT ATMOSPHERE ACTIVITY

Isotope	Adjusted Specific Activity, μCi/cc	Atmosphere Volume, cc	Activity, Ci
Kr 87	5.3(2)	4.7(10)	2.5(7)
Xe 133	2.1(3)	4.7(10)	1.0(8)
I 131			
I 132			
Cs 137			
Ba 140			

TABLE 6-7

CONTAINMENT SUMP ACTIVITY

Isotope	Adjusted Specific Activity, μCi/gm	Sump Water Mass, gm	Activity, Ci
Kr 87			
Xe 133			
I 131	2.7(4)	1.4(9)	3.8(7)
I 132	4.4(4)	1.4(9)	6.2(7)
Cs 137	2.6(3)	1.4(9)	3.7(6)
Ba 140	4.4(4)	1.4(9)	6.2(7)

TABLE 6-8

RCS ACTIVITY

Isotope	Adjusted Specific Activity, μCi/gm	RCS Water Mass, gm	Activity, Ci
Kr 87			
Xe 133			
I 131	7.0(4)	2.3(8)	1.6(7)
I 131	1.2(5)	2.3(8)	2.8(7)
Cs 137	6.5(3)	2.3(8)	1.5(6)
Ba 140	1.3(5)	2.3(8)	2.9(7)

$$\text{Sump mass} = 50{,}000\ ft^3 \times \rho_{STP} \times \frac{28.3 \times 10^3\ cc}{ft^3}$$

$$= 1.4 \times 10^9\ gm$$

where:

$$\rho_{STP} = 1.00\ \frac{gm}{cc}$$

3) Primary Coolant Mass

The primary system monitors indicate the system is full. The volume of the primary system of a 312 plant is 8910 ft^3.

At the temperature of the RCS at time of sample (350°F)

$$\text{RCS mass} = 8910\ ft^3 \times \frac{\rho}{\rho_{STP}}\ (2) \times \rho_{STP} \times \frac{28.3 \times 10^3\ cc}{ft^3}$$

$$= 2.3 \times 10^8\ gm$$

where:

$\frac{\rho}{\rho_{STP}}$ (2) = water density ratio at RCS temperature (350°F), Figure 6-1
= 0.9

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

6.5 TOTAL ACTIVITY RELEASED

The total activity released is determined by adding the activity of the atmosphere, sump, and the reactor coolant system. Table 6-9 presents the total activity released.

6.6 ACTIVITY RATIOS OF THE RELEASED FISSION PRODUCTS

The activity ratios of the released fission products are shown in Table 6-10.

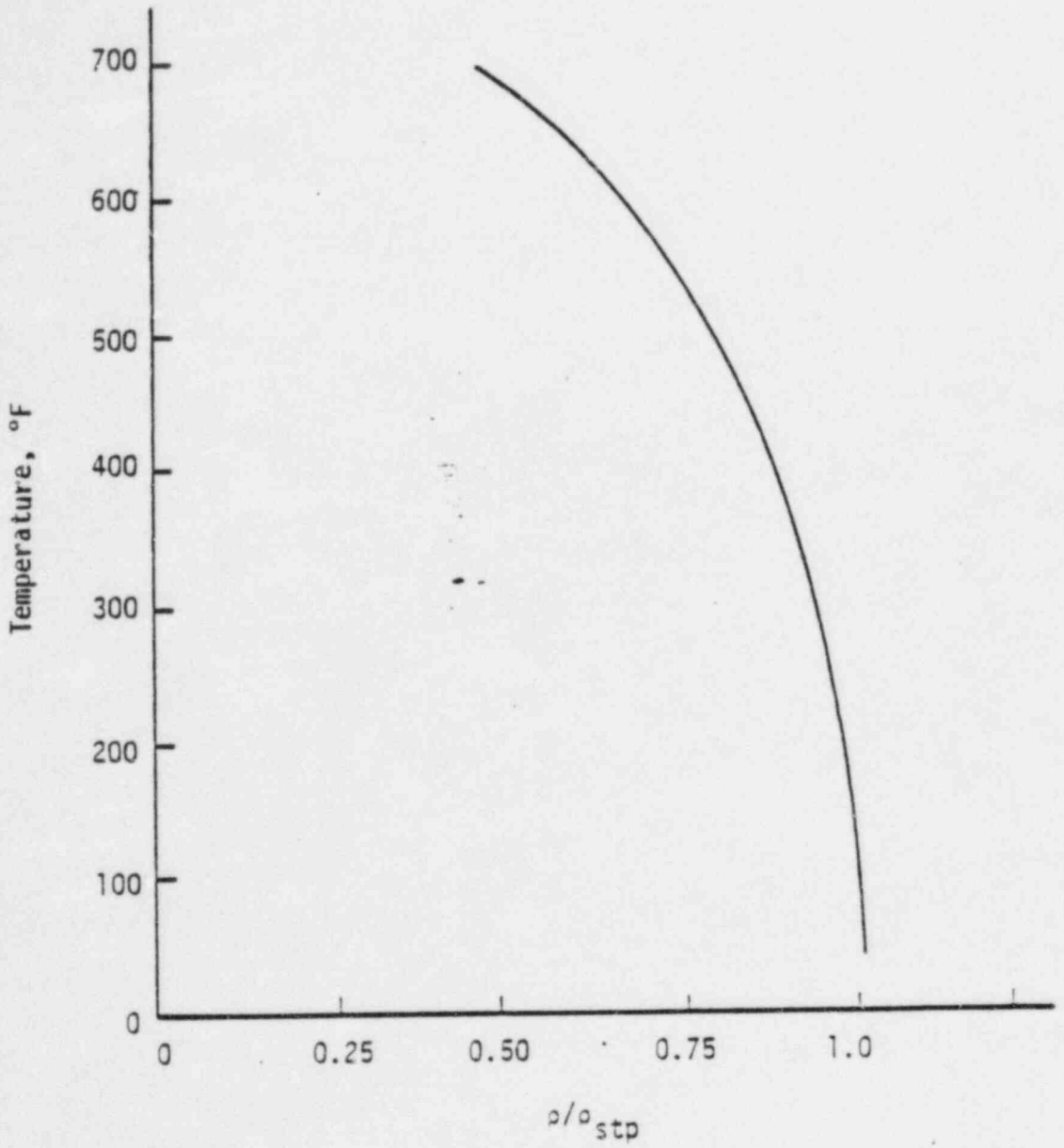

FIGURE 6-1 WATER DENSITY RATIO (TEMPERATURE VS. STP)

TABLE 6-9

TOTAL ACTIVITY RELEASED

Isotope	Atmosphere, Ci	Sump, Ci	RCS, Ci	Total, Ci
Kr 87	2.5(7)			2.5(7)
Xe 133	1.0(8)			1.0(8)
I 131		3.8(7)	1.6(7)	5.4(7)
I 132		6.2(7)	2.8(7)	9.0(7)
Cs 137		3.7(6)	1.5(6)	5.2(6)
Ba 140		6.2(7)	2.9(7)	9.1(7)

TABLE 6-10

ACTIVITY RATIOS OF RELEASED FISSION PRODUCTS

Isotope	Total Activity, Ci	Activity Ratio*
Kr 87	2.5(7)	.25
Xe 133	1.0(8)	1.0
I 131	5.4(7)	1.0
I 132	9.0(7)	1.7

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

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

6.7 INVENTORY AVAILABLE FOR RELEASE

To determine the total inventory of fission products available for release at reactor shutdown, the power history prior to shutdown needs to be known. For this example, the reactor has been operating continuously for 400 days with the following power history prior to shutdown.

20	days	at	75%	power = 2175 Mwt
10	days	at	100%	power = 2900 Mwt
10	days	at	50%	power = 1450 Mwt
5	days	at	75%	power = 2175 Mwt
45	days			

The new inventories are calculated by applying the power correction factors discussed to the equilibrium, end-of-life core inventories. The following sections present examples in determining the power correction factor for this scenario. The corrected core inventories are listed in Table 6-11.

1) Isotopes with half-lives <1 day

For isotopes with half-lives less than 1 day, it is assumed that they reach equilibrium in approximately 4 days. For this scenario the reactor is operating at 2175 Mwt for 5 days prior to shutdown. Thus, the power correction is as follows:

$$\text{Power Correction Factor} = \frac{2175 \text{ Mwt}}{2900 \text{ Mwt}} = .75$$

For I-133 ($t_{1/2}$ = 2 h),

$$\text{Corrected Inventory} = 1.2 \times 10^{8} \text{ Curies} \times .75$$
$$= 9.0 \times 10^{7} \text{ Curies}$$

TABLE 6-11

FISSION PRODUCT INVENTORY AT REACTOR SHUTDOWN

Isotope	Equilibrium Inventory at End-of-Life, Ci*	Power Correction Factor	Corrected Inventory, Ci
Kr 87	3.3(7)	0.75	2.5(7)
Xe 133	1.5(8)	0.68	1.0(8)
I 131	8.0(7)	0.68	5.4(7)
I 132	1.2(8)	0.75	9.0(7)
Cs 137	8.7(6)	0.60	5.2(6)
Ba 140	1.4(8)	0.65	9.1(7)

* Inventories for a 3-Loop plant

2) Isotopes with half-lives >1 day

Since the power is not constant during the 30-day period prior to shutdown, the transient power correction equation is applied.

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

For I-131 ($t_{1/2}$ = 8d, $\lambda_i = 8.7 \times 10^{-2}$ day^{-1})

since $\Sigma\, t_j$ = 45 days > $4 \times \frac{0.693}{\lambda_i}$ = 32 days,

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

$$= \frac{2175 \left(1-e^{-(8.7(-2))(20)}\right) e^{-(8.7(-2))(25)} +}{2900}$$

$$= \frac{2900 \left(1-e^{-(8.7(-2))(10)}\right) e^{-(8.7(.2))(15)}}{2900}$$

$$+ \frac{1450 \left(1-e^{-(8.7(-2))(10)}\right) e^{-(8.7(-2))(5)} + 2175\left(1-e^{-(8.7(-2))(5)}\right) e^{-(8.7(-2))(0)}}{2900}$$

$$= \frac{1972}{2900} = .68$$

3) Isotopes with half-lives around 1 year

For this scenario, the core has operated for 240 effective full power days during the 400 days of cycle operation.

For Cs - 137 ($t_{1/2}$ = 10 yr.)

$$\text{Power Correction Factor} = \frac{240 \text{ EFPD}}{400 \text{ D}} = .6$$

6.8 PERCENTAGE OF INVENTORY RELEASED

The corrected inventories are used to determine the percentage of inventory released for each isotope. The inventory released percentages are compared to Figures 6-2 through 6-9 to estimate the extent of core damage. Table 6-12 presents the release percentages for the isotopes of this example.

6.9 CORE DAMAGE ASSESSMENT BASED ON RADIONUCLIDE ANALYSIS

The results of the radionuclide analysis are used to determine an estimate of the extent of core damage. Table 6-12 shows the inventory released percentages of this accident scenario. These percentages are compared to Figures 6-2 thru 6-9 to estimate the extent of core damage.

The fission products analyzed after the accident are Kr-87, Xe-133, I-131, I-133, Cs-137, and Ba-140. The noble gases, iodines, and cesium are released during all stages of core damage with Ba-140 being a characteristic fission product of fuel overtemperature and fuel melt. The calculated release of Ba-140 is used to estimate the extent of fuel temperature and fuel melt. From Figures 6-2 and 6-3 the 0.025 percent release of Ba-140 corresponds to approximately 20 percent fuel overtemperature and less than 1 percent fuel melt. Based on the Ba-140 release percentage, the fission product release is primarily due to clad damage and fuel temperature.

The release percentage of the noble gases, iodines and cesium indicate from Figure 6-4 that approximately 15-25 percentage of the core has experienced overtemperature conditions. The activity ratios shown in Table 6-10 indicate that the release has progressed beyond gap release to fuel pellet release.

Comparing the release percentages of the noble gases and iodines to Figures 6-6 thru 6-9 greater than 100 percent clad damage is indicated. However, as stated previously, it is recognized that in actuality there is an overlap between the regimes of core damage states. It can be estimated that major clad damage (greater than 50 percent) has occurred.

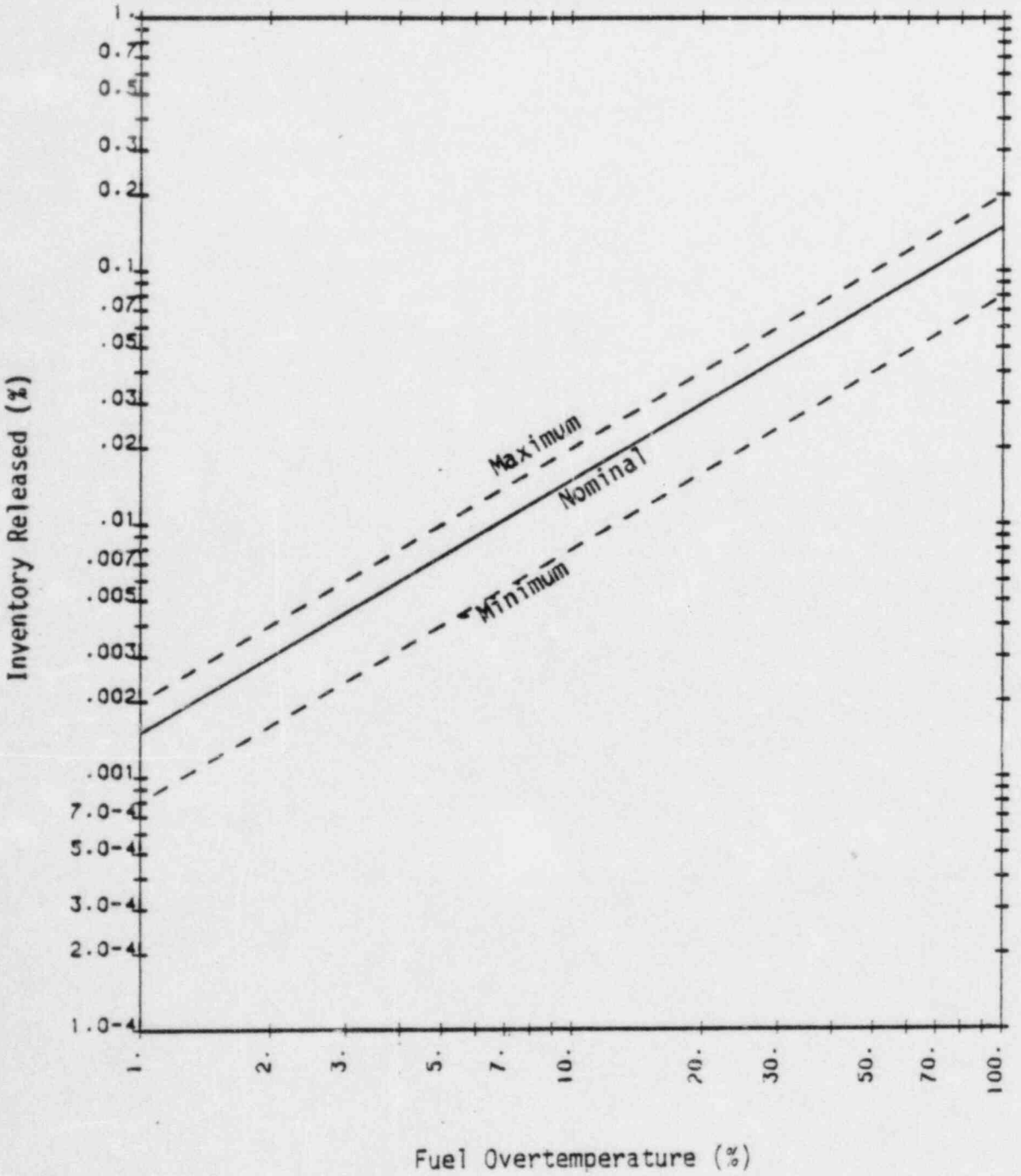

FIGURE 6-2 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % INVENTORY RELEASED OF BA, SR

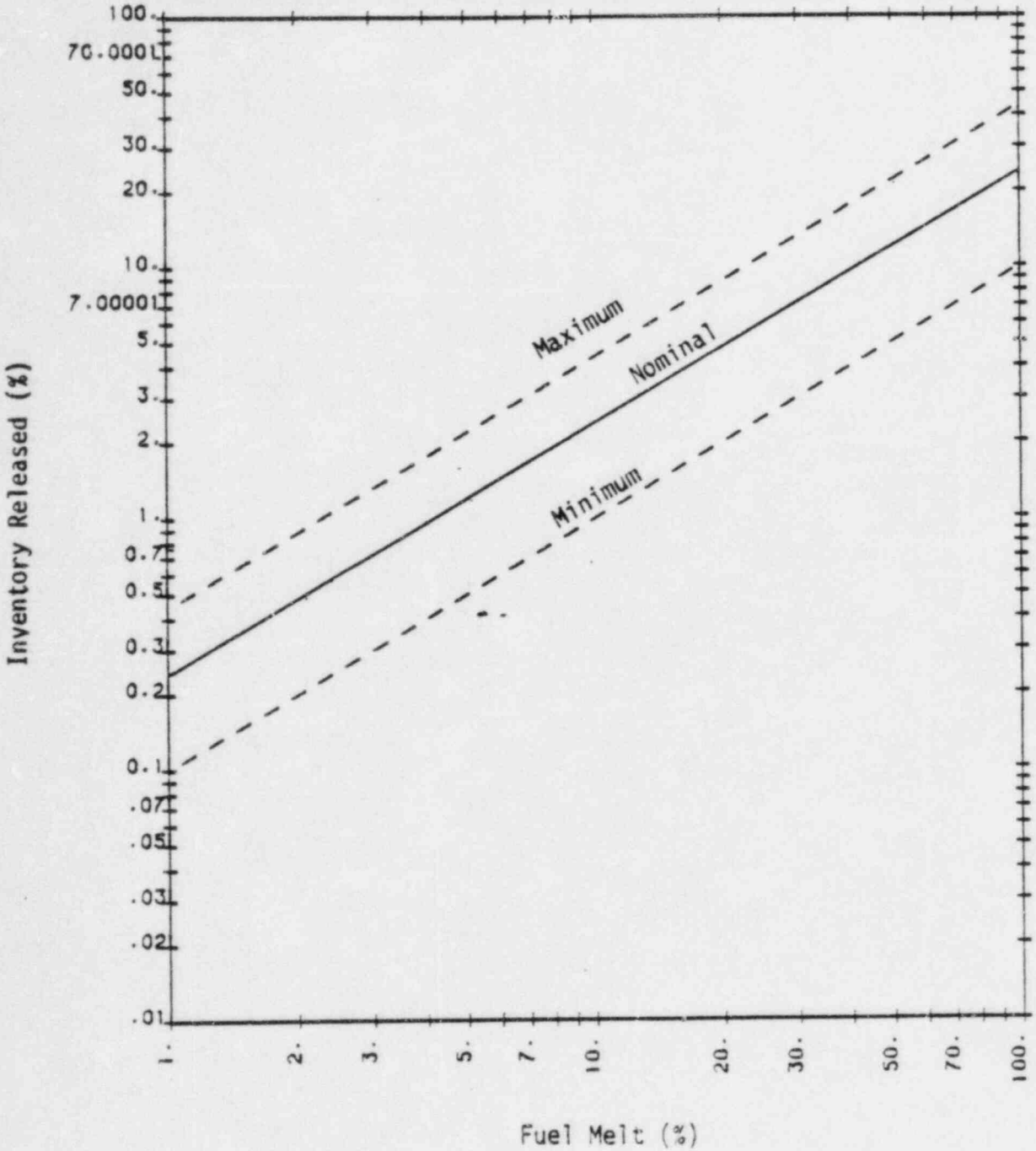

FIGURE 6-3 RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF BA, SR

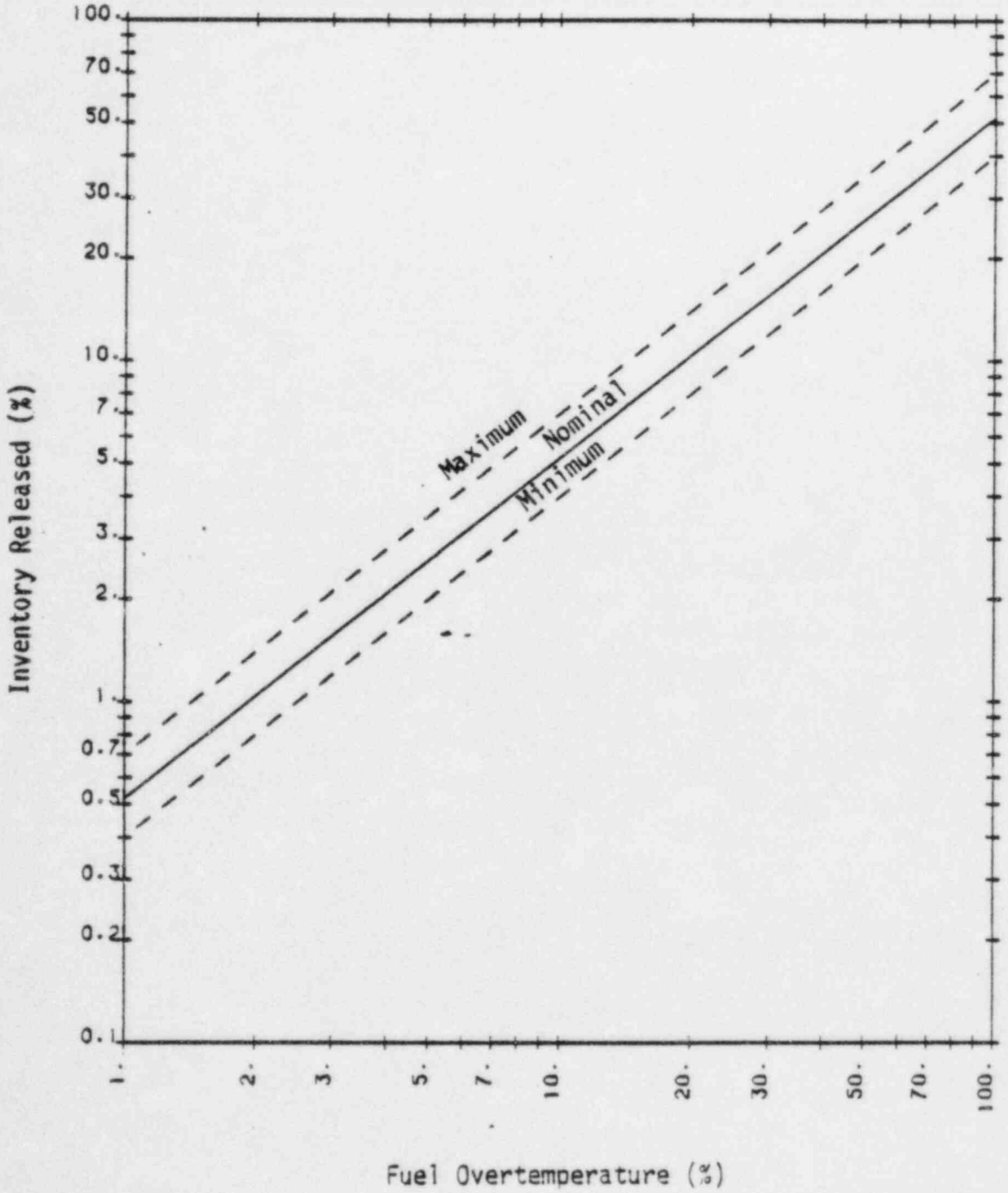

FIGURE 6-4 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % INVENTORY RELEASED OF XE, KR, I, CS

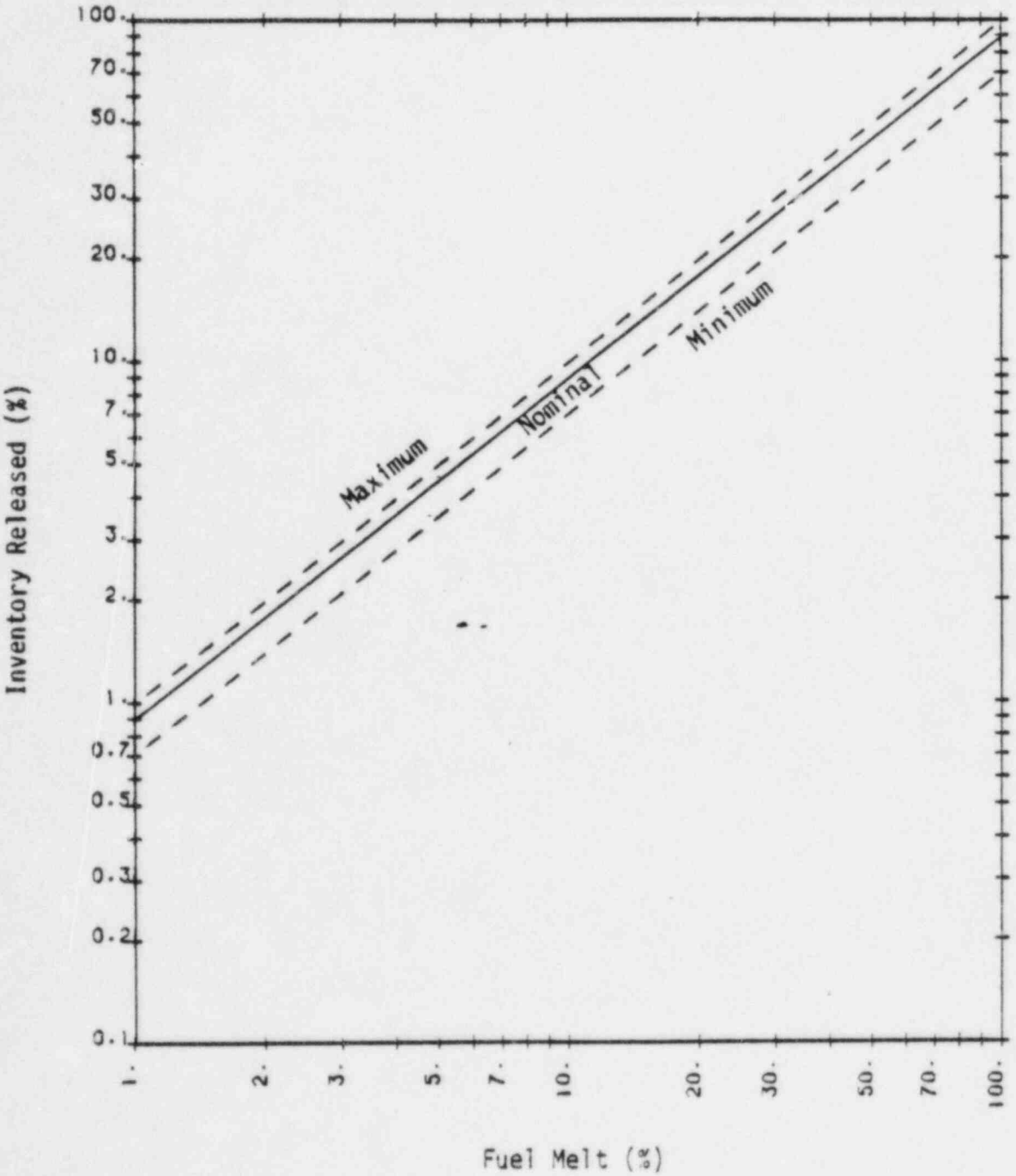

FIGURE 6-5 RELATIONSHIP OF % FUEL MELT WITH % INVENTORY RELEASED OF XE, KR, I, CS, TE

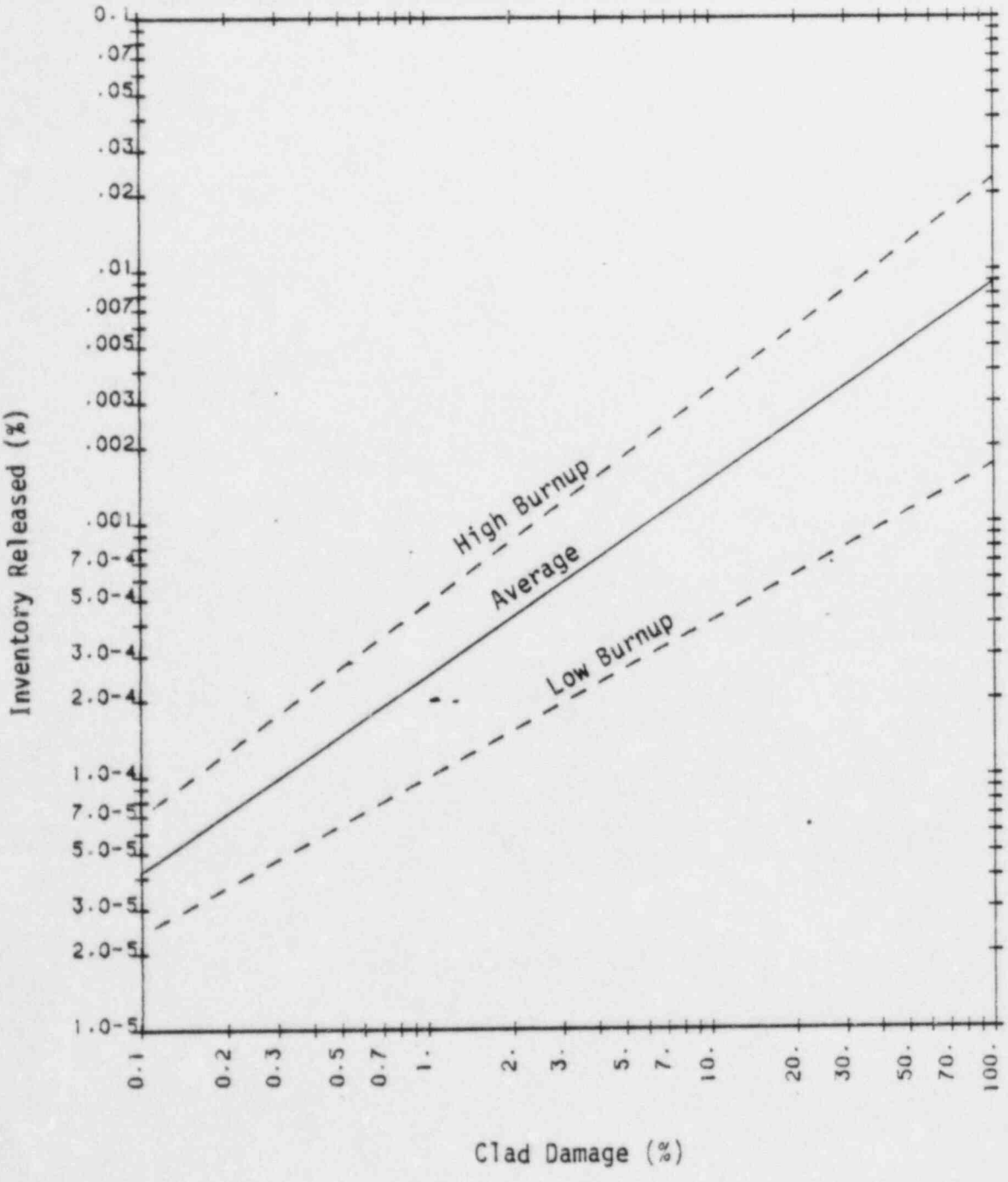

FIGURE 6-6 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF KR-87

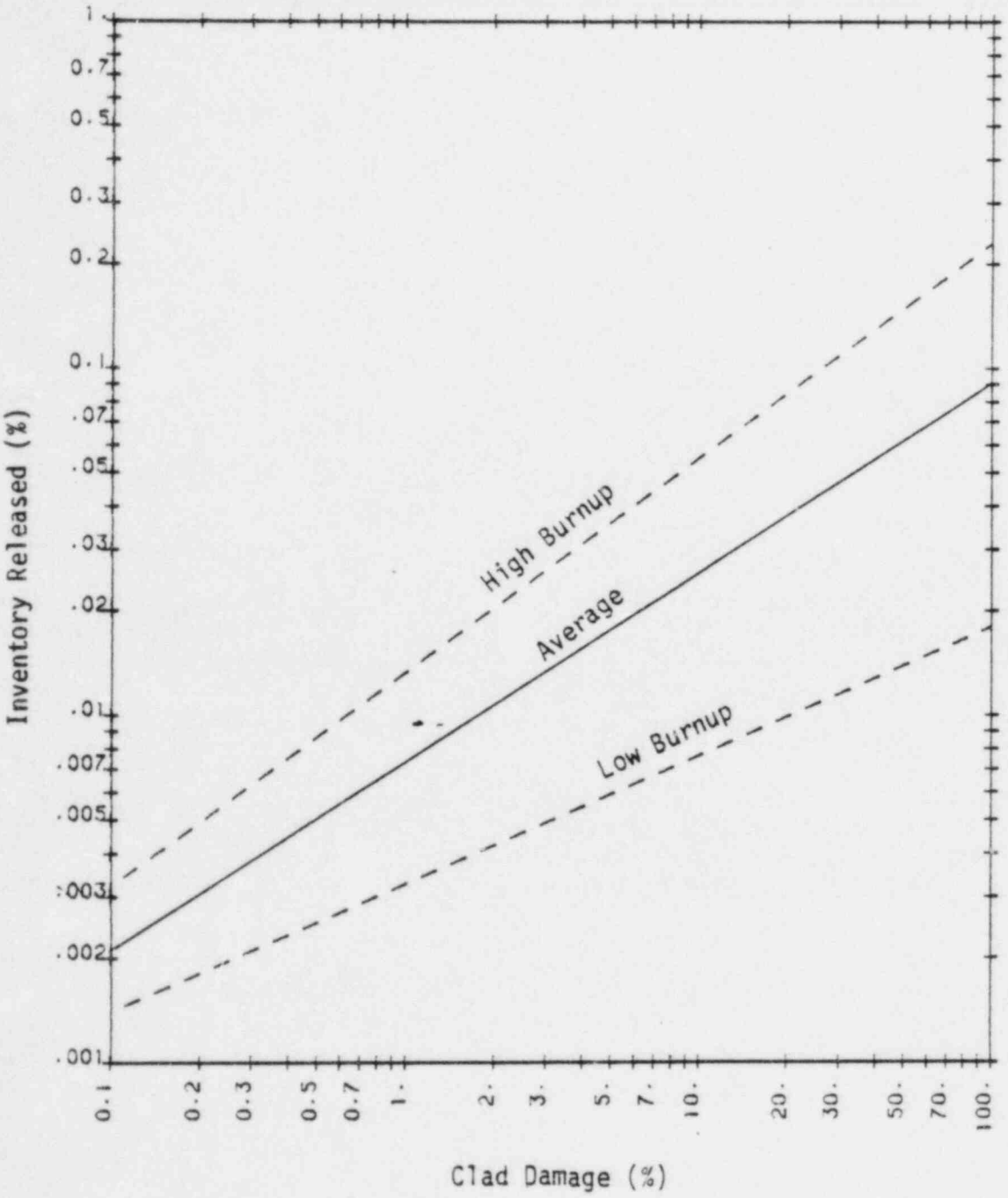

FIGURE 6-7 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF XE-133

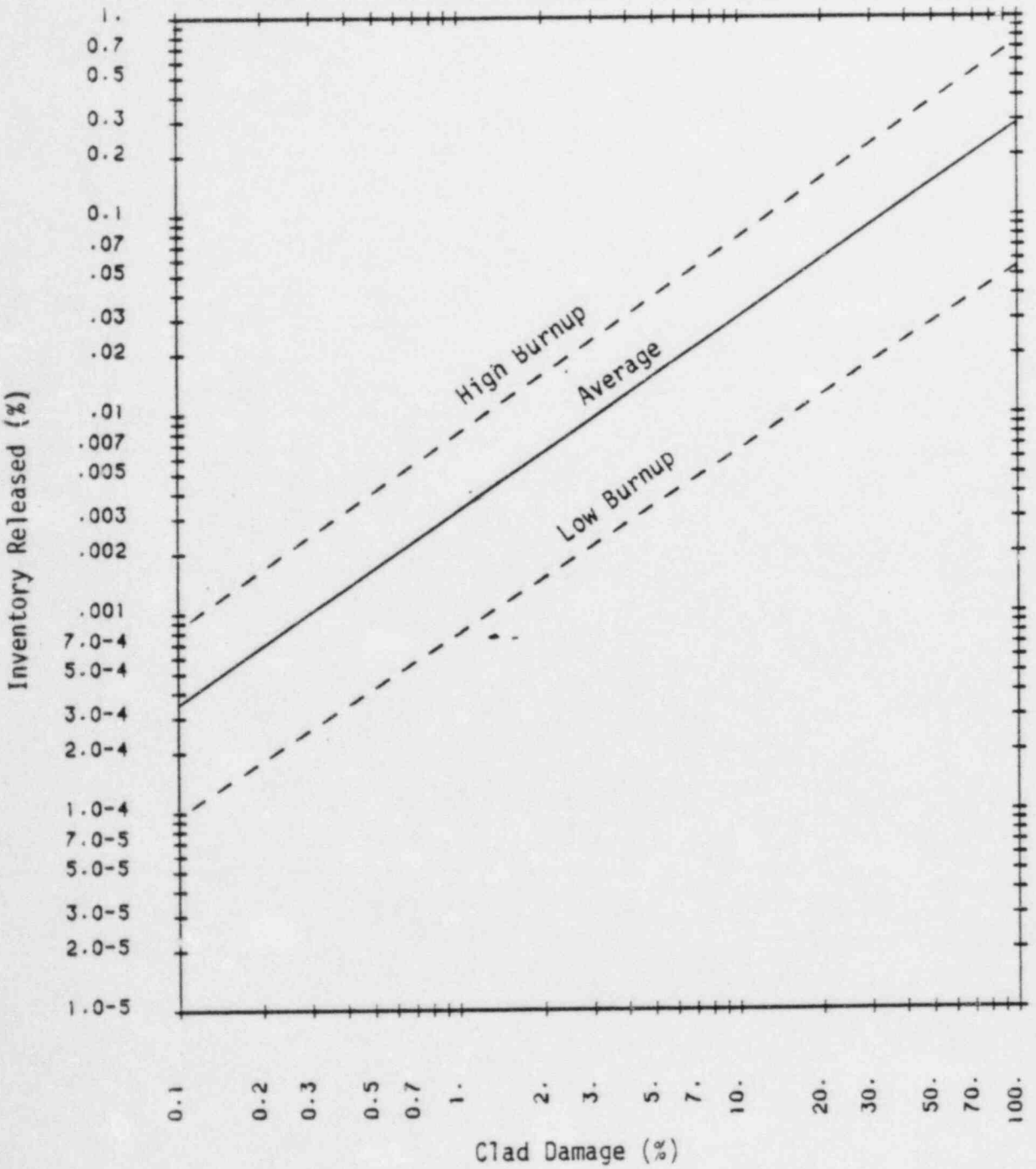

FIGURE 6-8 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-131

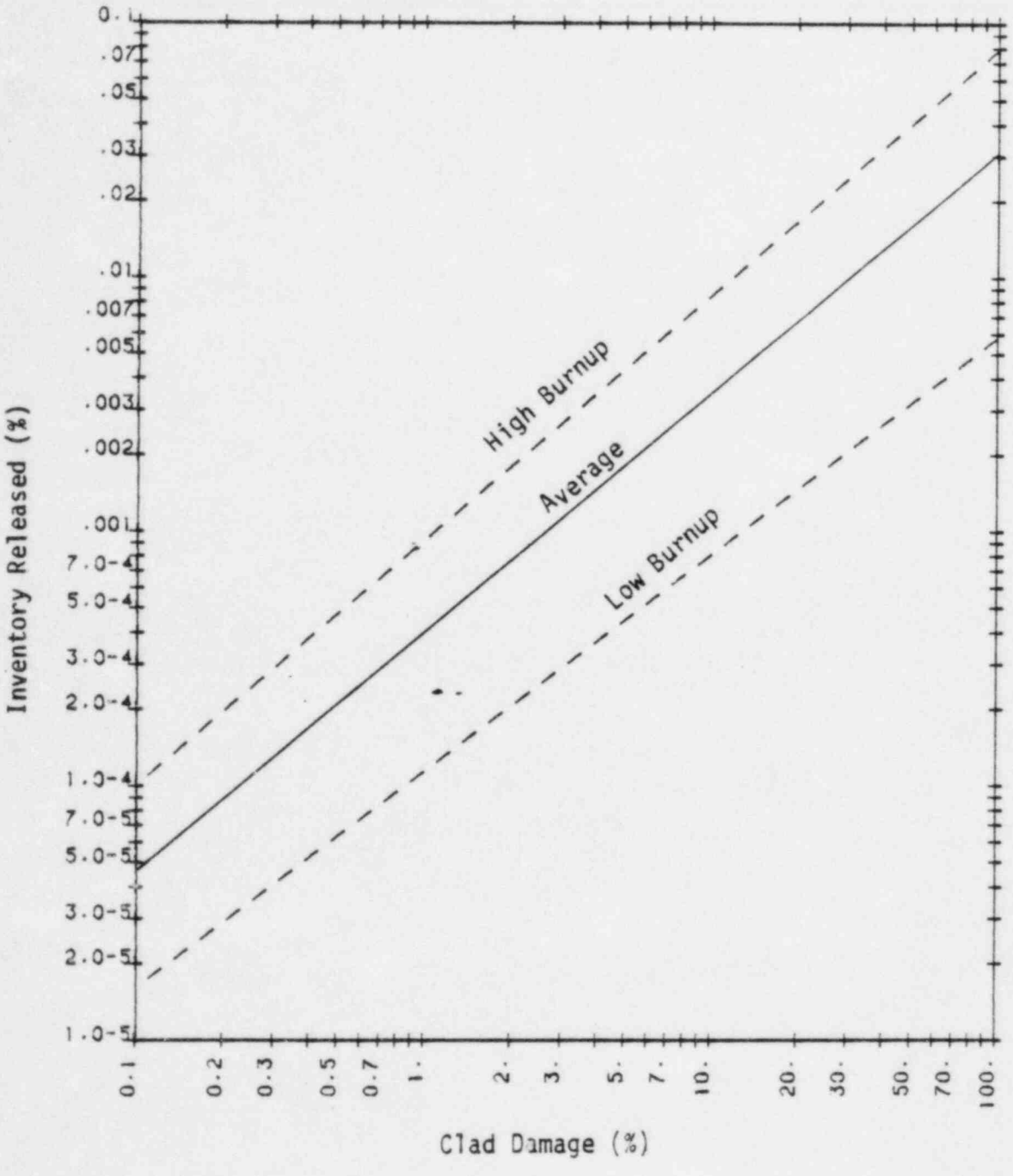

FIGURE 6-9 RELATIONSHIP OF % CLAD DAMAGE WITH % INVENTORY RELEASED OF I-132

TABLE 6-12

RELEASE PERCENTAGE

Isotope	Total Avtivity Released, Ci	Corrected Inventory, Ci	Release Percentage, %
Kr 87	2.0(6)	2.5(7)	8.0
Xe 133	8.3(6)	1.0(8)	8.3
I 131	4.6(6)	5.4(7)	8.5
I 132	7 4(6)	9.0(7)	8.2
Cs 137	4.1(5)	5.2(6)	7.8
Ba 140	2.3(4)	9.1(7)	.025

The conclusion drawn from the radionuclide analysis is that the damage states of the core are major clad damage (greater than 50 percent), less than 50 percent fuel overtemperature, and the possibility of very minor fuel melt (less than 1 percent).

6.10 AUXILIARY INDICATORS

To verify the conclusion of the radionuclide analysis, the auxiliary indicators (containment hydrogen concentration, core exit thermocouple temperature, reactor vessel water level and containment radiation monitor readings) are used.

6.10.1 CONTAINMENT HYDROGEN CONCENTRATIONS

The containment hydrogen monitor indicated a hydrogen concentration in the containment of 10 v/o. From Figure 6-10, 10 v/o hydrogen concentration corresponds to approximately 75 percent zirconium water reaction. Thus, the hydrogen concentration indicates that there is a high probability that greater than 50 percent of the clad is damaged, Table 6-13. It also indicates that the core had uncovered during the accident.

6.10.2 CORE EXIT THERMOCOUPLE READINGS AND REACTOR VESSEL WATER LEVEL

The core exit thermocouple readings during this accident rose greater than 1650°F for the center regions of the core and ranged between 900°F to 1100°F for the outer regions of the core. The reactor vessel water level monitor indicated that the core uncovered during the accident for an extended period of time. From Table 6-13, these readings indicate a possibility of the core experiencing fuel overtemperature.

6.10.3 CONTAINMENT RADIATION MONITOR

The containment radiation monitor indicated a gross gamma dose rate of 1.02×10^4 R/hr at 6 hours after reactor shutdown. To use Figure 6-11, the dose rate is normalized based on power and containment volume. It is assumed that the only difference between this plant and the plant of Section 3.3 is the containment volume. This plant has a volume of 1.7×10^6 ft^3. The normalized dose rate is calculated as follows:

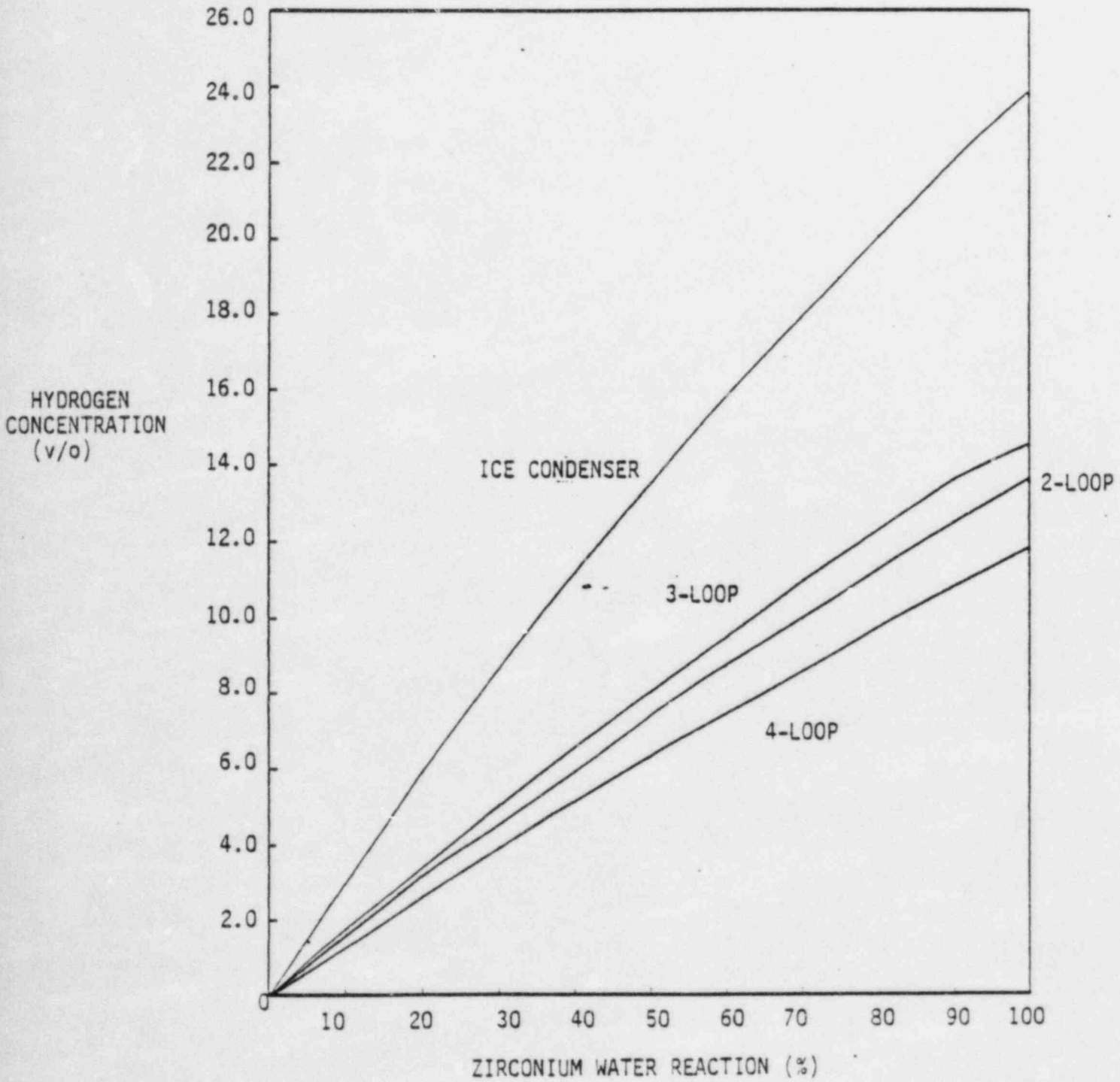

FIGURE 6-10 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM WATER REACTION

TABLE 6-13
CHARACTERISTICS OF CATEGORIES OF FUEL DAMAGE*

Core Damage Indicator Core Damage Category	Percent and Type of Fission Products Released	Fission Product Ratio	Containment Radiogas Monitor (R/hr - Mwt) 10 hrs after shutdown**	Core Exit Thermocouples Readings (Deg F)	Extent and Duration of Core Uncovery (Minutes)	Hydrogen Monitor (Vol % H_2)*** & Plant Type
No clad damage	Kr-87 < $1x10^{-3}$ Xe-133 < $1x10^{-3}$ I-131 < $1x10^{-3}$ I-133 < $1x10^{-3}$	Kr-87 = 0.022 I-133 = 0.71	-	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 10^{-3} - 0.1 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 - .08	750 - 1300	Core uncovery	2 Loop 0 - 6 3 Loop 0 - 7 4 Loop 0 - 6 ice 0 - 13
50-100% clad damage	Kr-87 0.1 - 0.2 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	0.08 to 0.16	1300 - 1650	Core uncovery	2 Loop 6 - 13 3 Loop 7 - 14 4 Loop 6 - 11 ice 13 - 24
0-50% fuel pellet overtemperature	Xe-Kr,Cs,I 1 - 20 Sr-Ba 0 - 0.4	Kr-87 = 0.22 I-133 = 2.1	0.16 to 21	> 1650	Core uncovery	2 Loop 6 - 13 3 Loop 7 - 14 4 Loop 6 - 11 ice 13 - 24
50-100% fuel pellet overtemperature	Xe-Kr,Cs,I 20 - 40 Sr-Ba 0.4 - 0.8	Kr-87 = 0.22 I-133 = 2.1	21 to 42	> 1650	Core uncovery	2 Loop 6 - 13 3 Loop 7 - 14 4 Loop 6 - 11 ice 13 - 24
0-50% fuel melt	Xe,Kr,Cs,I 40 - 70 Sr-Ba 0.2 - 0.8 Pr-Rb 0.1 - 0.8	Kr-87 = 0.22 I-133 = 2.1	42 to 70	> 1650	Core uncovery	2 Loop 6 - 13 3 Loop 7 - 14 4 Loop 6 - 11 ice 13 - 24
50-100% fuel melt	Xe,Kr,Cs,I,Te > 70 Sr,Ba > 24 Pr,Rb > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 70	> 1650	Core uncovery	2 Loop 6 - 13 3 Loop 7 - 14 4 Loop 6 - 11 ice 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 plant specific parameters and times other than 10 hours. These values are from Figure 3-3 and represent a specific detector geometry.

*** Ignitors may obviate these values.

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

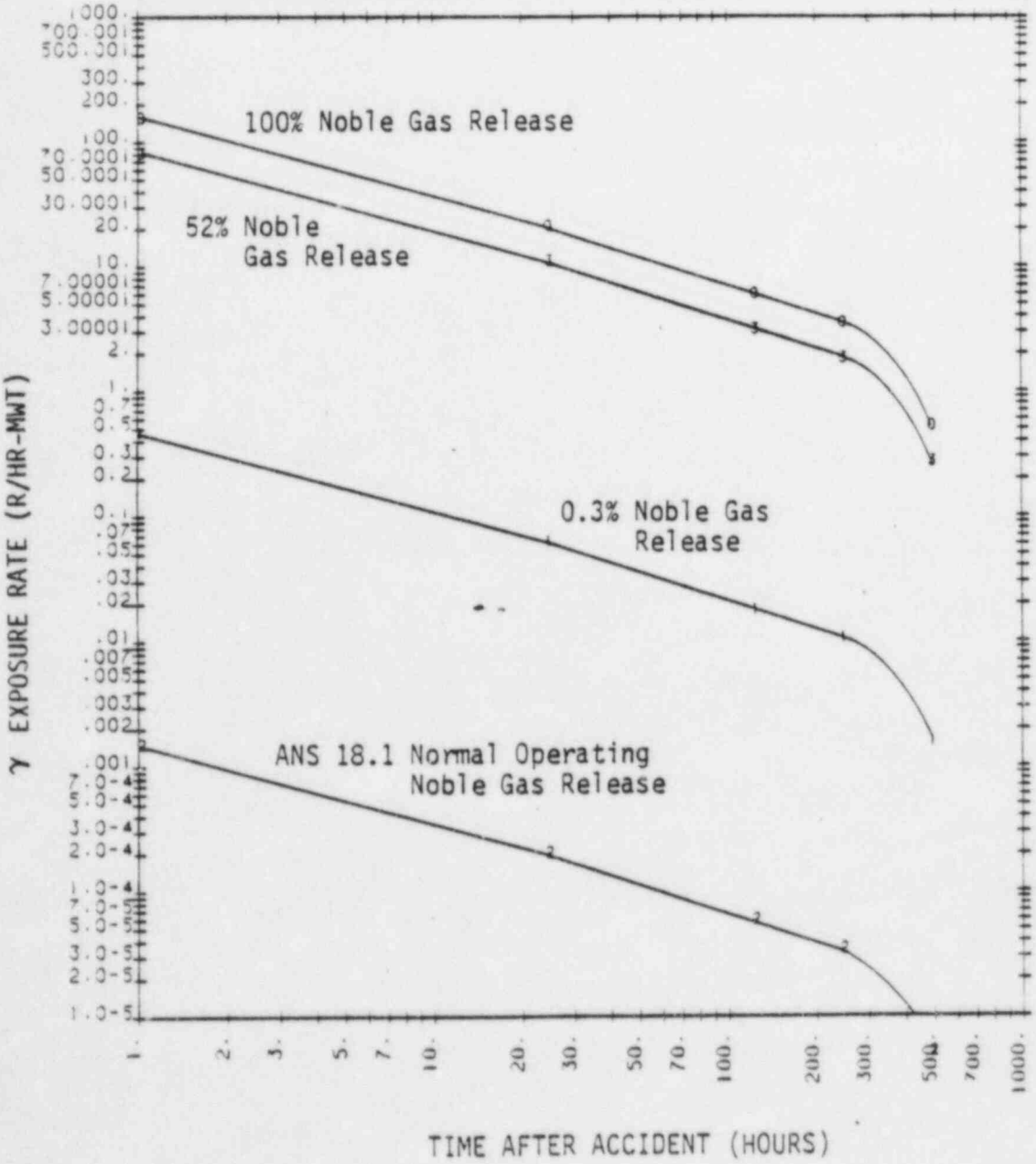

FIGURE 6-11 PERCENT NOBLE GASES IN CONTAINMENT

$$\text{Dose Rate (Normalized)} = \frac{1.02 \times 10^4 \text{ R/hr} \times 1.7 \times 10^6 \text{ ft}^3}{2900 \text{ MWt} \times 2 \times 10^6 \text{ ft}^3}$$

$$= 3 \text{ R/hr - MWt}$$

From Figure 6-11, 3 R/hr-MWt corresponds to an overtemperature release and a significant gap release which confirms the radionuclide analysis.

6.11 SUMMARY

The radionuclide analysis indicated greater than 50 percent clad damage, less than 50 percent fuel overtemperature, and a possibility of very minor fuel melt. This conclusion was substantiated from the analyses of the auxiliary plant indicators.

This example was provided to illustrate the use of this core damage assessment methodology in determining the extent of core damage. Although this example was for a Westinghouse 312 plant, the methodology can be employed for the other Westinghouse pressurized water reactors.

7.0 REFERENCES

1. "Clarification of TMI Action Plan Requirements," NUREG-0737, USNRC, November 1980.

2. "A Report to the Commission and to Public, NRC Special Inquiry Group," M. Rogovin, 1980.

3. "ORIGEN Isotope Generation and Depletion Code," Oak Ridge National Laboratory, CCC-217.

4. Method of calculating the fractional release of fission products from oxide fuel, ANSI/ANS 5.4 - 1982.

5. WCAP-9964, Westinghouse Electric Corporation.

6. "Source Term Specification," ANS 18.1 Standard 1976.

7. "Radionuclide Release Under Specific LWR Accident Conditions," NUREG-0956, USNRC, January 1983.

8. "Release of Fission Products From Fuel in Postulated Degraded Accidents," IDCOR DRAFT Report, July 1982.

9. "TMI-2 Accident: Core Heat-up Analysis," NSAC/24, January 1981.

10. "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, August 1983.

11. Westinghouse Emergency Response Guidelines.

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