WOLF CREEK GENERATING STATION

PLANT RECOVERY

CORE DAMAGE ASSESSMENT METHODOLOGY

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Revision 2

TABLE OF CONTENTS

		Page
1.0	INTRODUCTION AND PURPOSE	1
	1.1 Methodology	1
	1.2 Technical Bases for Core Damage Assessment Methodology	2
	1.2.1 Characteristic Fission Products	2
	1.2.2 Core Inventories	4
	1.2.3 Power Correction for Core Inventories	4
	1.2.4 Adjustments to Determine Activity Released	8
2.0	EVALUATION OF STATE OF CORE DAMAGE	19
	2.0.1 "No Damage" Criterion	19
	2.1 Relationship of Activity To Cladding Damage	19
	2.1.1 Gap Inventory	19
	2.1.2 Spiking Phenomena	21
	2.1.3 Activity Associated With Cladding Damage	21
	2.2 Relationship of Activity Release To Fuel Overtemperature Conditions	34
	2.3 Relationship of Activity To Fuel Melt Conditions	41
	2.4 Core Crumbling Conditions	47
3.0	AUXILIARY INDICATORS	48
	3.1 Containment Hydrogen Concentration	48
	3.2 Core Exit Temperatures and Reactor Vessel Water Levels	51
	3.3 Containment Radiation Monitors and Core Damage	54
	3.4 Gap Activity Ratios	60
4.0	GENERALIZED CORE DAMAGE ASSESSMENT APPROACH	62
	4.1 Sampling Locations	62
5.0	LIMITATIONS	65
6.0	EXAMPLE OF CORE DAMAGE ASSESSMENT METHODOLOGY	67
7.0	REFERENCES	87

LIST OF TABLES

		Page
1-1	Selected Nuclides for Core Damage Assessment.	3
1-2	Fuel Pellet Inventory	5
1-3	Sump Water Volume Relation.	11
1-4	Parent-Daughter Relationships	15
1-5	Source Inventory of Parent Nuclides Not Listed in Table 1-2 . . .	17
2-1	Gap Inventory	20
2-2	Normal Operating Activity	22
2-3	Expected Iodine Spike	33
2-4	Expected Fuel Damage Correlation With Cladding Temperature. . . .	35
2-5	Percent Activity Release for TMI-2 Accident	37
2-6	Percent Activity Release for 100 Percent Overtemperature	37
2-7	Percent Activity Release for 75 Percent Core Melt Conditions. . .	42
3-1	Instantaneous Gamma Ray Source Strengths.	56
3-2	Instantaneous Gamma Ray Fluxes.	57
3-3	Isopotic Activity Ratios of Fuel Pellet and Gap	61
4-1	Characteristics of Categories of Fuel Damage.	63
4-2	Suggested Sampling Locations.	64
	Example Calculation	
6-1	Results of Sampling Analysis 6 Hours After Shutdown	68
6-2	Fission Product Inventory at Reactor Shutdown	68
6-3	Adjusted Specific Activity - Containment Atmosphere	73
6-4	Adjusted Specific Activity - Containment Sump	73
6-5	Adjusted Specific Activity - RCS.	74
6-6	Containment Atmosphere Activity	74
6-7	Containment Sump Activity	75
6-8	RCS Activity.	75
6-9	Total Activity Released Before Decay Correction	78
6-10	Decay Corrected Release Activities.	81
6-11	Release Percentage.	82
6-12	Activity Ratios of Released Fission Products.	86

LIST OF FIGURES

		Page
1-2	Water Density Ratio	13
2-1	% Clad Damage with Activity of Xe-133 Released.	24
2-2	% Clad Damage with Activity of I-131 Released	25
2-3	% Clad Damage with Activity of I-131 Released with Spiking. . . .	26
2-4	% Clad Damage with Activity of Kr-87 Released	27
2-5	% Clad Damage with Activity of Xe-131m Released.	28
2-6	% Clad Damage with Activity of I-132 Released	29
2-7	% Clad Damage with Activity of I-133 Released	30
2-8	% Clad Damage with Activity of I-135 Released	31
2-9	% Fuel Overtemperature with % Core Inventory Released of Xe, Kr, I, or Cs.	39
2-10	% Fuel Overtemperature with % Core Inventory Released of Ba or Sr.	40
2-11	% Fuel Melt with % Core Inventory Released of Xe, Kr, I, Cs, or Te.	44
2-12	% of Fuel Melt with % Core Inventory Released of Ba or Sr.	45
2-13	% of Fuel Melt with % Core Inventory Released of Pr	46
3-1	Containment Hydrogen Concentration Based on Zirconium - Steam Reaction.	50
3-2	Distribution of Thermocouples and Flux Thimbles	52
3-3	Percent Noble Gases in Containment.	59

1.0 INTRODUCTION AND PURPOSE

This Plant Recovery Methodology is intended for use after accident release dose estimates are completed and sent to the Duty Emergency Director. The Duty Emergency Director (DED) will decide when this methodology is to be implemented. The DED will consult the list of personnel qualified to use this methodology, and select one person from the list to provide the estimate. The list is maintained in EPP 01-12.1 (Transition to Recovery).

The purpose of this methodology is to provide a preliminary indication of several states of possible core damage. These indications will then be the starting point in the evaluation of the recovery to restart actions.

This methodology is based primarily on the Westinghouse Owner's Group (WOG) Generic Core Damage Assessment Methodology (1). The WOG Methodology is based partly upon an unpublished, draft study - NUREG-0956 (7).

The WOG methodology has been followed in general in the preparation of this document. The document has been extended to cover aspects of plant recovery not covered by the WOG (Section 2.4). Generic parameters have been reduced to the specific Wolf Creek values, and the structure of the methodology changed to reflect the Wolf Creek Emergency Plan. Changes have been made in the fuel overtemperature and fuel melt sections (2.2 and 2.3) to more accurately reflect the referenced sources.

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

1.2 TECHNICAL BASES FOR CORE DAMAGE ASSESSMENT METHODOLOGY

1.2.1 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 cladding damage, fuel overheat, and fuel melt. Core crumbling is also addressed in this methodology, though core crumbling does not release any characteristic nuclides. The selection of nuclides for this methodology was based on half-life, gamma energy, gamma yield, release characteristics, quantity present in the core, and practicality of measurement using standard gamma spectrometry techniques.

The nuclides selected for this methodology have sufficient core inventories and radioactive half-lives to ensure that there will be sufficient activity for detection and analysis of the nuclides for some time following an accident. Most of the nuclides selected have half-lives which enable them to reach equilibrium quickly within the fuel cycle. The list of selected nuclides includes 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. These 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 gamma rays from the nuclides along with the practicality of detecting and analyzing the nuclides.

Nuclides were chosen to be representative of the specific states of core damage by their release characteristics. As the core progresses through these damage states, it is assumed that certain nuclides associated with each damage state will 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 1-1.

TABLE 1-1

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

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

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

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 inventories of the selected fission products (and some additional fission products of interest) are shown in Table 1-2.

1.2.3 POWER CORRECTION FOR CORE INVENTORIES

The source inventory shown in Table 1-2 presents inventories for an equilibrium, end-of-life core that has been operated at 100 percent power. A source inventory at the time of an accident that accounts for the power history is needed for this methodology. 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 is provided to calculate the source inventory at the time of the accident.

TABLE 1-2

FUEL PELLET INVENTORY*

Inventory, Curies

Nuclide	Wolf Creek (3565 Mwt)
Kr 85m	2.2(7)**
Kr 87	4.0(7)
Kr 88	5.7(7)
Xe 131m	6.3(5)
Xe 133	2.0(8)
Xe 133m	2.8(7)
Xe 135	3.7(7)
I 131	9.8(7)
I 132	1.4(8)
I 133	2.0(8)
I 135	1.8(8)
Rb 88	5.8(7)
Cs 134	2.3(7)
Cs 137	1.1(7)
Te 129	3.3(7)
Te 132	1.4(8)
Sr 89	7.9(7)
Sr 90	7.2(6)
Ba 140	1.7(8)
La 140	1.8(8)
La 142	1.5(8)
Pr 144	1.2(8)

* Inventory based on ORIGEN run for equilibrium, end-of-life core.
** 2.2(7) = 2.2×10^7. This notation is used throughout this report.

These power correction factors are contained in the POCO computer routine available on the UNIVAC or IBM-PC. Use of the POCO routine is recommended for all non-steady state cases since all entries and results are automatically documented.

1.2.3.1 POWER CORRECTION FACTOR

A) Steady state power prior to shutdown.

1) Half-life of nuclide < 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for prior 4 days}}{\text{3565 (Mwt)}}$$

2) Half-life of nuclide > 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for prior 30 days}}{\text{3565 (Mwt)}}$$

3) Half-life of nuclide approximately 1 year

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for prior 1 year}}{\text{3565 (Mwt)}}$$

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

B) Transient power history in which the power had 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.

$$\text{Power Correction Factor} = \frac{\Sigma_j P_j (1-e^{-\lambda_i t_j}) e^{-\lambda_i t^\circ_j}}{RP[(1-e^{-\lambda_i 300}) + (1-e^{-\lambda_i 600}) + (1-e^{-\lambda_i 900})]/3}$$

where:

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

RP = rated power level of the core (3565 Mwt)

t_j = operating period in days at power P_j where power does not vary more than $\pm$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:

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

$$\text{Power Correction Factor} = \frac{\Sigma_j P_j (1-e^{-\lambda_i t_j}) e^{-\lambda_i t^\circ_j}}{3565}$$

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.

$$\text{Power Correction Factor} = \frac{EFPD_3 + 2/3\ EFPD_2 + 1/3\ EFPD_1}{600}$$

C) For Cs-134 the POCO routine is used to determine the power correction factor.

The WCGS Power Correction Factor Computer Routine (POCO) is available on the UNIVAC and IBM-PC for detailed calculations if Cs-134 or other nuclides values are not clear cut. Power history since the start of cycle is required for nuclides with a half-life greater than one month.

1.2.4 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 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.

1.2.4.1 DILUTION OF THE 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. A similar approximation is required for a sump sample. For the purposes of this methodology, the water is assumed to be distributed within the primary system and the sump. However, consideration should be taken if a significant primary system to secondary system leak rate is noted (as in the case of a steam generator tube rupture). The amount of water that is available for distribution is the initial amount of primary system water and the amount of water that has been discharged from the Refueling Water Storage Tank (RWST). Also, an adjustment must be made for water added via the containment spray systems, accumulators, and chemical addition tanks. 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 + Activities from other sources (accumulators, spray additive tanks, etc.).

Note: The specific activities will be decay corrected to reactor shutdown, and the RCS amount will be corrected to account for temperature and pressure differences between sample and RCS in Sections 1.2.4.2 and 1.2.4.3.

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

Total Activity Released = Total Water Activity + Containment Atmosphere Activity

1.2.4.2 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.

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

where:

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

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

where:

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

The total activity released to the containment atmosphere is:

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

The specific activity of the liquid samples requires no adjustment if the specific activity is reported on a gram basis (μCi/gm). If the specific activity is reported on a volume basis (μCi/cc), an adjustment is performed to convert the volume specific activity to a gram specific activity. The conversion is performed for consistency with later calculations. If the temperature of the sample is above 200°F, an adjustment is required to the conversion. In most cases, the sample temperature will be below 200°F and no adjustment is necessary. Figure 1-2 shows a relation of water density at some temperature relative to the water density at standard temperature and pressure.

The volume of water in the containment sumps is primarily determined by the sump level indicators (LI-7 and LI-8) in the two recirculation sumps. These indicators extend beyond the top of the sump pits, since the pits themselves are relatively small (6 feet on a side), and most of the water from a LOCA would remain on the containment floor. Table 1-3 relates the sump level indicators to sump volume.

The bottom of the Core Instrumentation Tunnel is roughly 30 feet lower than the containment floor. This tunnel is covered by a splash guard and surrounded by a six inch berm to prevent water entry during a LOCA. The splash guard is not watertight, and water level indications above this level may be inaccurate. The highest recorded level above 94" should be used in this calculation for this reason.

TABLE 1-3

SUMP WATER VOLUME RELATION

Recirculation Sump Level Indicators
(LI-7 and LI-8)

Level Indication - h (inches)	Sump Volume (cc)
0 - 72	(8.5E4) h
72 - 88	(6.2E6) + (1.6E7) (h-72)
88 - 94	(2.7E8) + (2.8E7) (h-88)
94 - 156	unreliable (4.3E8) + (2.9E7) (h-94)

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

A. RCS or Sump temperature > 200°F:

RCS or sump mass (gm) = 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 ration at medium (RCS or sump) temperature, Fig. 1-2.

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

B. RCS or sump temperature < 200°F:

RCS or Sump Mass (gm) = RCS of Sump Volume (ft^3) x ρ_{STP} x

$$\frac{28.3 \times 10^3 \text{ cc}}{ft^3}$$

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).

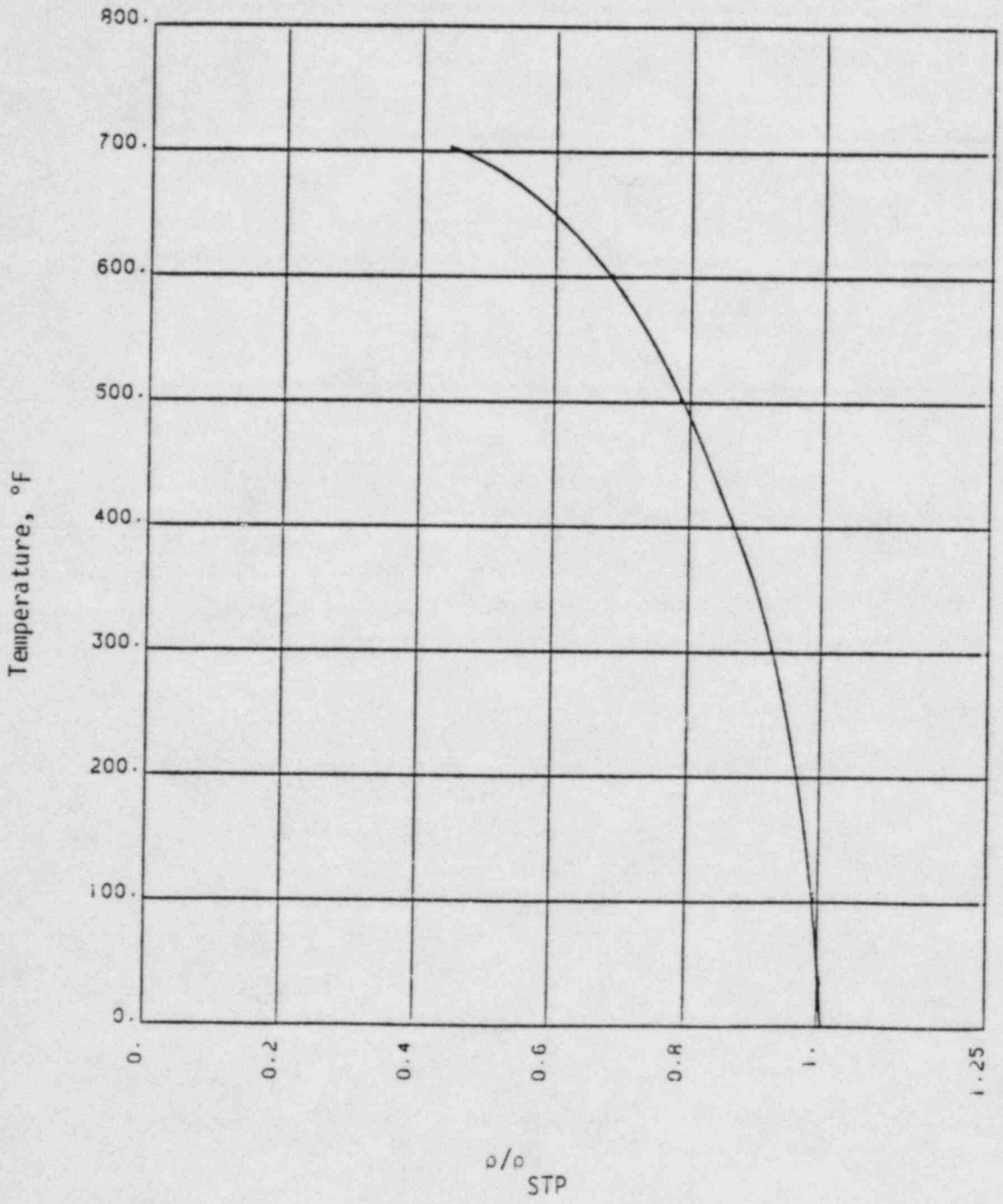

FIGURE 1-? WATER DENSITY RATIO (TEMPERATURE VS. STP)

1.2.4.3 DECAY CORRECTION

The released activity is decay adjusted to time of reactor shutdown using the following equation:

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

where:

λ_i = radioactive decay constant, 1/hr

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

Since this correction may also be performed by some analytical equipment, care must be taken to avoid duplicate correction. Also, consideration must be given to account for precursor effect during the decay of the nuclide. For this methodology, only the parent-daughter relationships are considered. Table 1-4 lists the significant parent-daughter relationships associated with the methodology. The decay scheme of the parent-daughter relationship is described by the following equation.

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

where:

Q_A^o = activity (Ci) or specific activity (μCi/gm or μCi/cc) of the parent at shutdown

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

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

TABLE 1-4

PARENT-DAUGHTER RELATIONSHIPS

Parent	Parent Half Life*	Decay Constant hr^{-1}	Daughter	Daughter Half Life*	Decay Constant hr^{-1}	K**
Kr-88	2.8 h	2.48(-1)	Rb-88	17.8 m	2.34(0)	1.00
I-131	8.05 d	3.59(-3)	Xe-131m	11.8 d	2.45(-3)	0.008
I-133	20.3 h	3.41(-2)	Xe-133m	2.26 d	1.28(-2)	0.024
I-133	20.3 h	3.41(-2)	Xe-133	5.27 d	5.48(-3)	0.976
Xe-133m	2.26 d	1.28(-2)	Xe-133	5.27 d	5.48(-3)	1.00
I-135	6.68 h	1.04(-1)	Xe-135	9.14 h	7.58(-2)	0.70
Xe-135m	15.6 m	2.67(0)	Xe-135	9.14 h	7.58(-2)	1.00
I-135	6.68 h	1.04(-1)	Xe-135m	15.6 m	2.67(0)	0.30
Te-132	77.7 h	8.92(-3)	I-132	2.26 h	3.07(-1)	1.00
Sb-129	4.3 h	1.61(-1)	Te-129	68.7 m	6.05(-1)	0.827
Te-129m	34.1 d	8.47(-4)	Te-129	68.7 m	6.05(-1)	0.680
Sb-129	4.3 h	1.61(-1)	Te-129m	34.1 d	8.47(-4)	0.173
Ba-140	12.8 d	2.26(-3)	La-140	40.22 h	1.72(-2)	1.00
Ba-142	11 m	3.78(0)	La-142	92.5 m	4.50(-1)	1.00
Ce-144	284 d	1.02(-4)	Pr-144	17.27 m	2.41(0)	1.00

*Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition
**Branching decay factor

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

λ_B = decay constant of the daughter, hr^{-1}

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

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 fraction, as assumption is made that the fraction of source inventory released of the parent and the daughter at time of shutdown are equal (for the nuclides used here within a factor of 2). The following steps should be followed to calculate the fraction of the measured activity due to the decay of the daughter that was released and then to calculate the activity of the daughter released at shutdown.

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

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

where:

Q_A^o = 100% source inventory (Ci) of parent, Table 1-2 or 1-5

Q_B^o = 100% source inventory (Ci) of daughter, Table 1-2 or 1-5

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

K = if parent has 2 daughters, K is the branching factor, Table 1-4

TABLE 1-5

SOURCE INVENTORY OF PARENT NUCLIDES NOT LISTED IN TABLE 1-2

Nuclide	Wolf Creek (3565 MWt)
Xe-135m	4.2(7)
Sb-129	3.2(7)
Te-129m	8.0(6)
Ba-142	1.6(8)
Ce-144	1.1(8)

λ_A = parent decay constant, hr^{-1}

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

t = time period from shutdown to time of sample, hr

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

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

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

M_B = Fr x released activity (μCi/gm or μCi/cc)

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

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

2.0 EVALUATION OF POSSIBLE STATE OF CORE DAMAGE

The activity released from the core is the primary method to identify possible damage states described in Sections 2.1 to 2.4. The exception to this method is the state of core crumbling. Core crumbling produces no characteristic activity. Indications of core crumbling are described in Section 2.4.

2.0.1 "NO DAMAGE" CRITERION

If coolant activity is less than 600 Ci/cc, then there is no cladding or fuel damage. This value is taken from the WCGS Emergency Plan (EPP 01-2.1) and is a value approximately five times the WCGS Technical Specification Limits for coolant activity. No significant cladding or fuel damage will have occurred below this value. If the coolant activity is below this value, this methodology is completed and the result is "no damage".

A failure to meet the "no damage" criterion of this section does not mean that some damage is inevitable. The result of this methodology can still be "no damage" even if this criterion is failed.

2.1 RELATIONSHIP OF CLADDING DAMAGE TO ACTIVITY

Cladding damage is defined as the fraction of rods with failed cladding.

2.1.1 GAP INVENTORY

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

To determine the fission product inventory of the gap, the ANS 5.4[4] Standard formulae were used with the average temperature and burnup of the fuel rod. The average gap inventory for the entire core for this methodology was estimated by assuming the core is divided into three regions- a low burnup region, a middle burnup region, and a high burnup region. Using the ANS 5.4 Standard, the gap fraction and subsquent 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 throughtout the core. Table 2-1 shows the calculated gap inventories of the noble gases and iodines. 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.

TABLE 2-1

GAP INVENTORY
WOLF CREEK
(3565 Mwt)

Nuclide	Nominal Value* (Curies)	Minimum - Maximum*** (Curies)
Kr 85m**	3.78(3)	6.90(2)-9.57(3)
Kr 87	3.61(3)	6.81(2)-9.22(3)
Kr 88**	7.98(3)	1.42(3)-1.99(4)
Xe 131m	8.85(2)	1.58(2)-2.21(3)
Xe 133	1.76(5)	3.33(4)-4.51(5)
Xe 133m**	1.68(4)	1.28(3)-1.77(4)
Xe 135**	8.98(3)	4.11(3)-5.61(4)
I 131	2.84(5)	5.39(4)-7.35(5)
I 132	4.56(4)	8.55(3)-1.17(5)
I 133	1.92(5)	3.53(4)-4.90(5)
I 135	9.80(4)	1.78(4)-2.49(5)

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

** Additional nuclides; no graphs provided.

*** 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).

2.1.2 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. Therefore, other isotopes should always be used to verify indications of gap release based on cesium.) Spiking is noted by an increase in reactor coolant iodine concentrations during some time period after the transient. In most cases, the iodine concentration would return to normal operating activity at a rate based on the system purification half-life. Spiking is a characteristic of the condition where an increase in the normal primary coolant activity is noted, but no damage to the cladding has occurred.

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 2-2. The use of this data is demonstrated in Section 2.1.3.2.

2.1.3 ACTIVITY ASSOCIATED WITH CLADDING DAMAGE

Cladding 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. Cladding damage can begin to occur in regions of high fuel rod peak 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 possible cladding failures. When 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 a break in the primary system boundary to the containment atmosphere, and iodines will stay in solution and travel with the primary system water during the accident. Iodine activity in the primary system water should be corrected to account for any significant iodine releases into the containment atmosphere.

TABLE 2-2

NORMAL OPERATING ACTIVITY*

Nuclide	Specific Activity in Reactor Coolant (μCi/gm)
Kr 85m	1.1(-1)
Kr 87	6.3(-2)
Kr 88	2.0(-1)
Xe 131m	1.9(-2)
Xe 133	5.2(+0)
Xe 133m	1.0(-1)
Xe 135	3.1(-1)
I 131	2.7(-1)
I 132	1.0(-1)
I 133	3.8(-1)
I 135	1.9(-1)

* Values obtained from FSAR Table 11.1-1 (Reference 13)

To determine an approximation of the extent of cladding damage (the number of fuel rods with damaged cladding), the total activity of a fission product release 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 cladding damage for each nuclide. Figures 2-1 through 2-8 present the direct correlations for each nuclide in graphical form. The contribution of the normal operating activity has been factored into the correlations shown in Figures 2-1 through 2-8. Examples of how to construct the correlations shown in Figures 2-1 through 2-3 are presented in the next two sections. Figures 2-4 through 2-8 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 2-1 need to be analyzed, but as many as possible should be analyzed to determine a reasonable approximation of cladding damage.

2.1.3.1 Xe-133

A graphical representation can be developed which describes the relationship of the measured release percentage of Xe-133 to the extent of cladding damage. The total source inventory of Xe-133 for Wolf Creek is 2.0×10^8 Curies (Table 1-2). For 100 percent cladding damage, all of the gap inventory, which corresponds to 1.8×10^5 Curies (Table 2-1) would be released. For 10 percent and 1 percent cladding damage, 1.8×10^4 Curies and 1.8×10^3 Curies would be released, respectively. These three values can be used to represent three points of the relationship between percentage of total inventory released and the extent of cladding damage. However, the normal operating activity needs to be included in the relation. From Table 2-2, the normal operating activity of Xe-133 is 5.2 $\mu Ci/gm$[6]. The primary coolant mass is 2.4×10^8 grams. The total normal operating contribution to the total release of Xe-133 is 1250 Curies. Thus, the adjusted releases are 1.8×10^5 Ci, 1.9×10^4 Ci, and 3.0×10^3 Ci (for 100, 10, and 1 percent cladding damage, respectively). This relation is shown in Figure 2-1, divided by the core inventory of Xe-133 from Table 1-2.

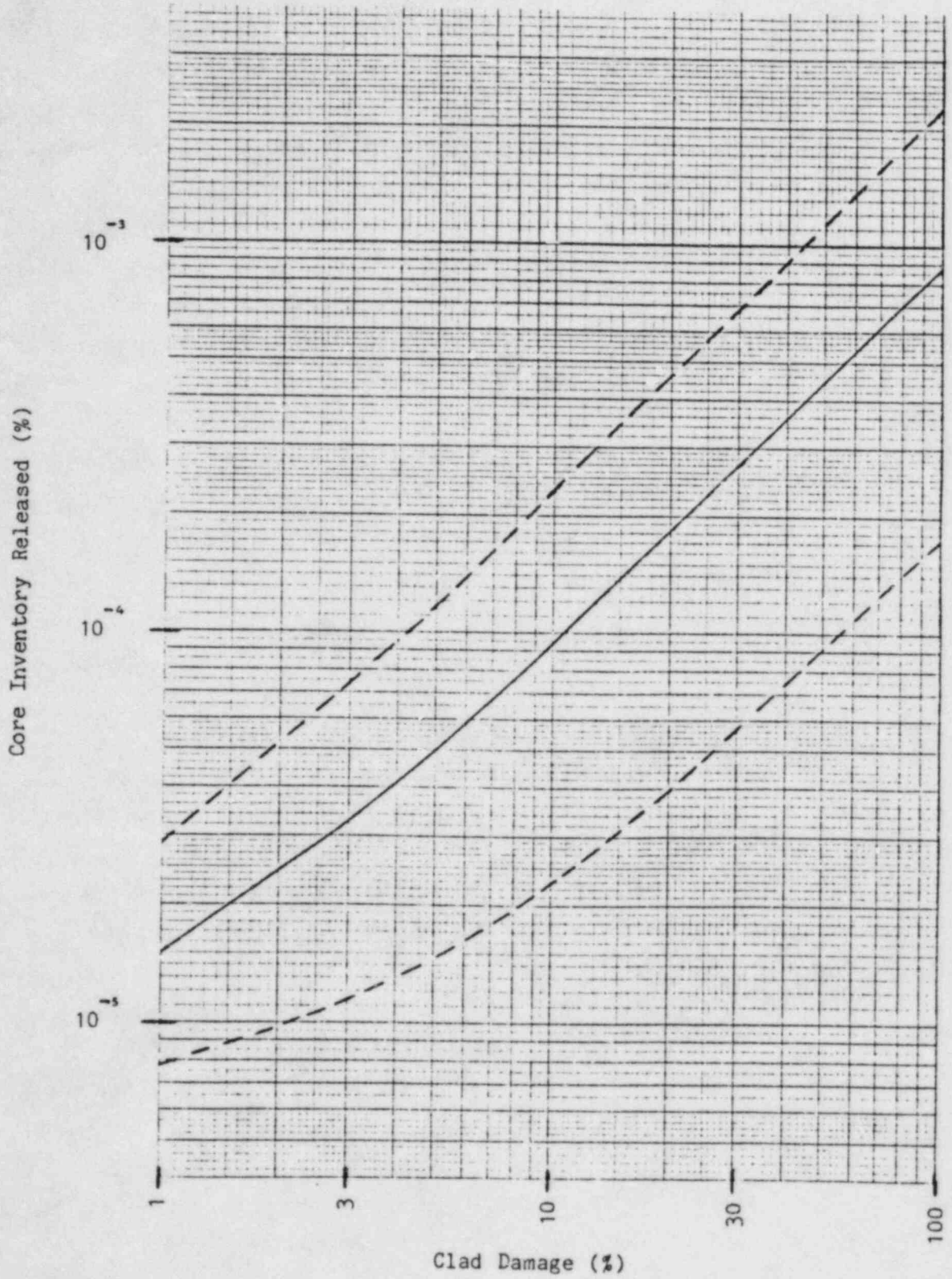

FIGURE 2-1 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF XE-133

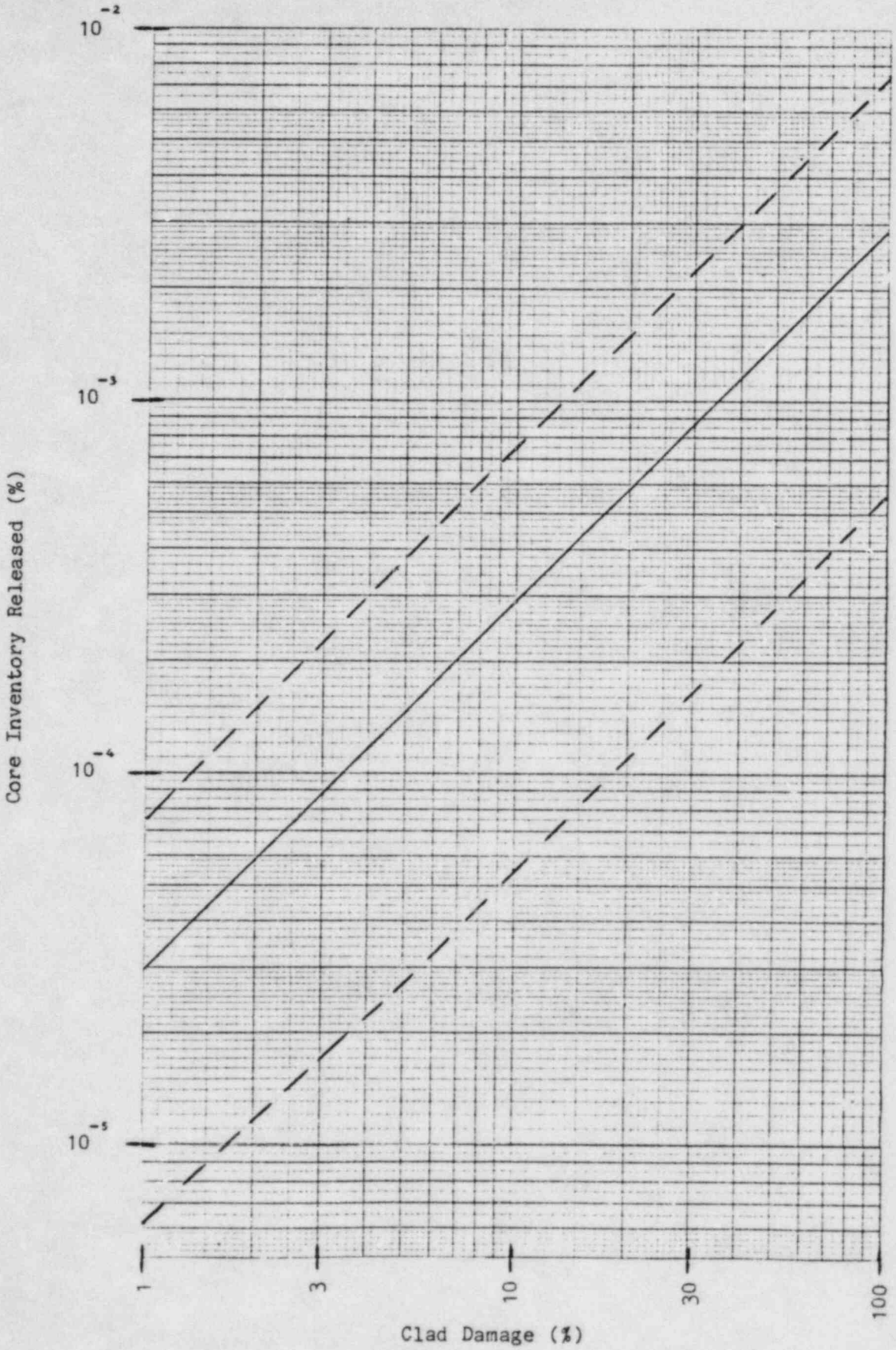

FIGURE 2-2 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF I-131

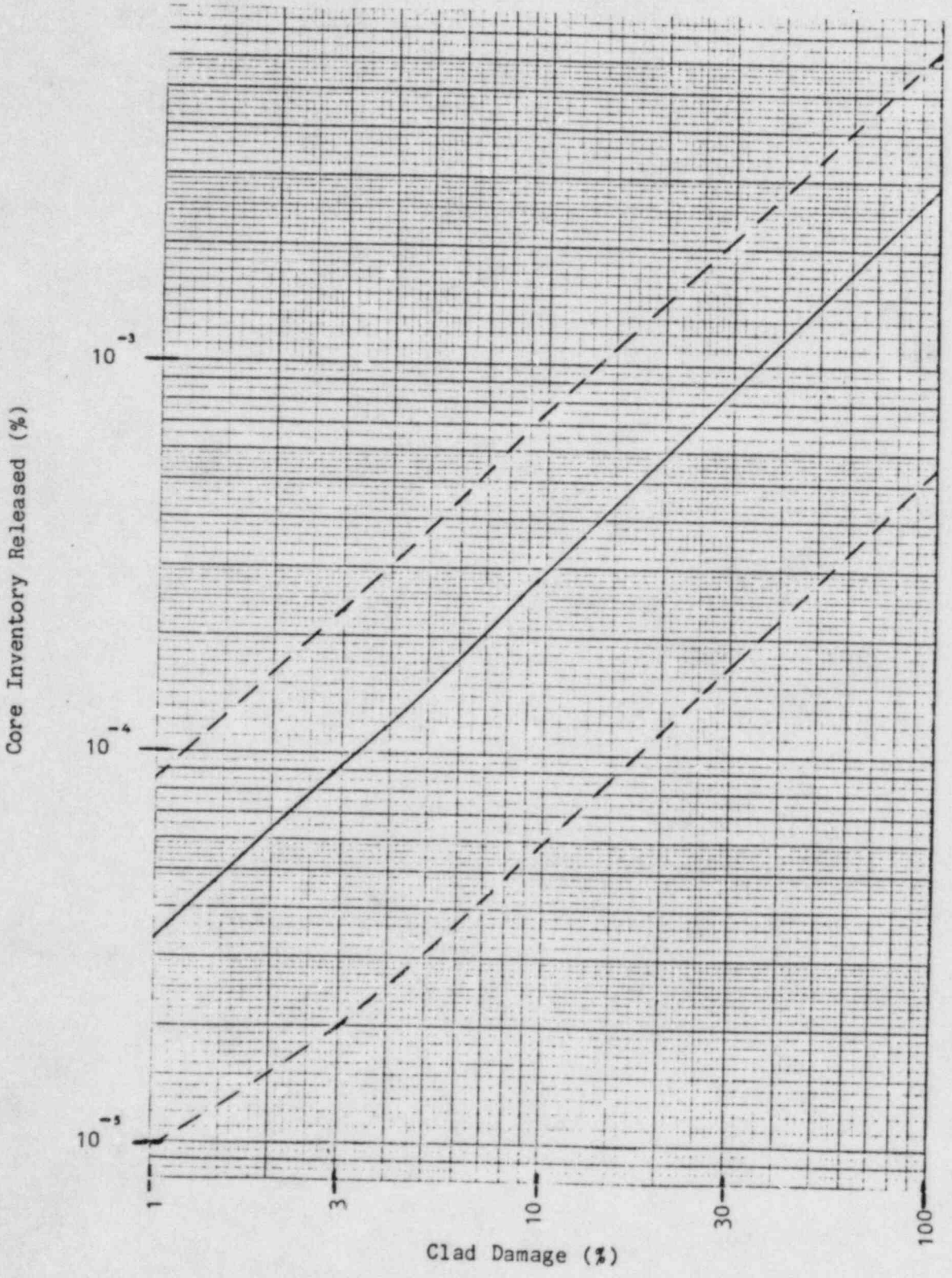

FIGURE 2-3 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF I-131 (with spiking)

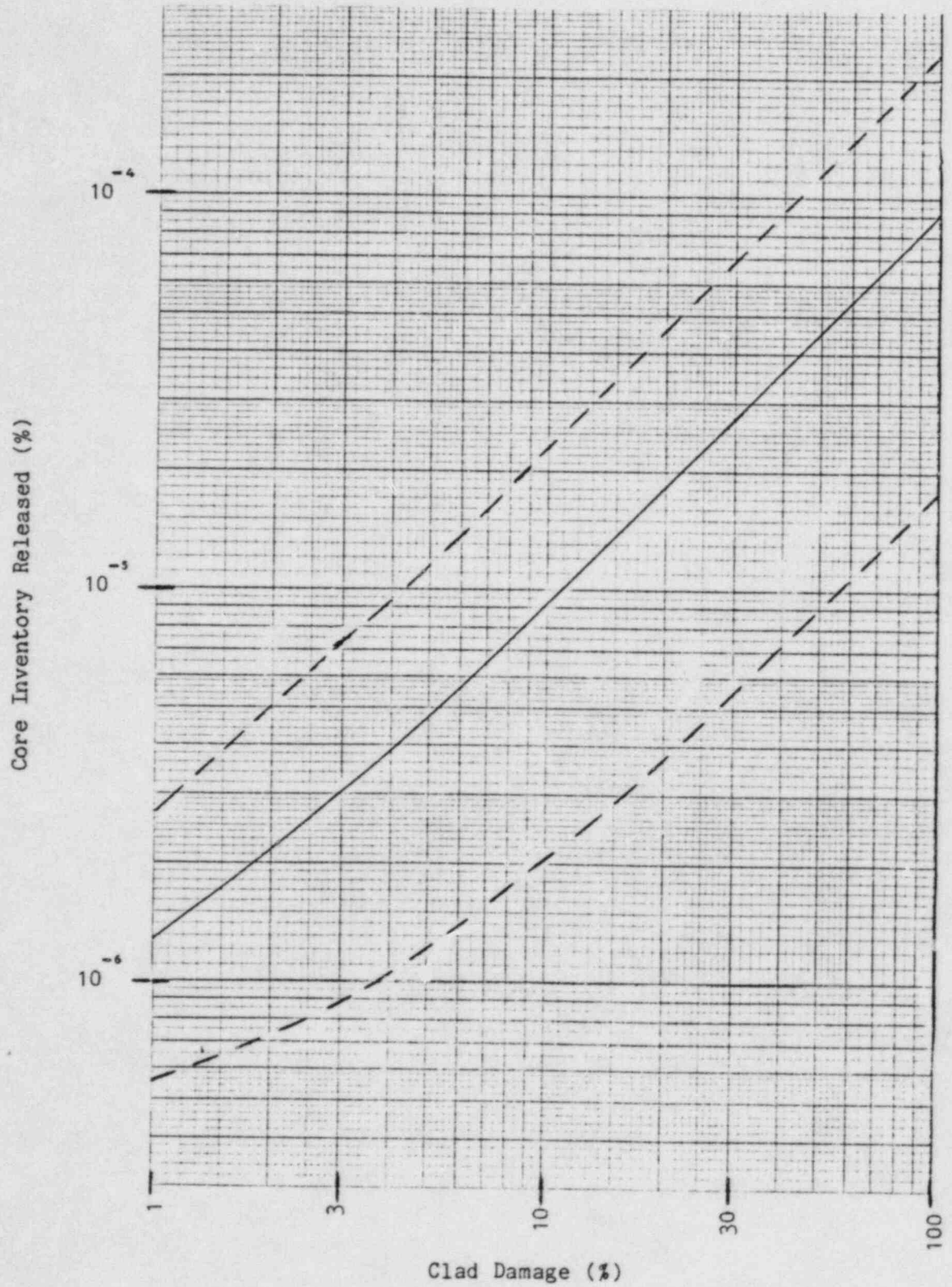

FIGURE 2-4 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF Kr-87

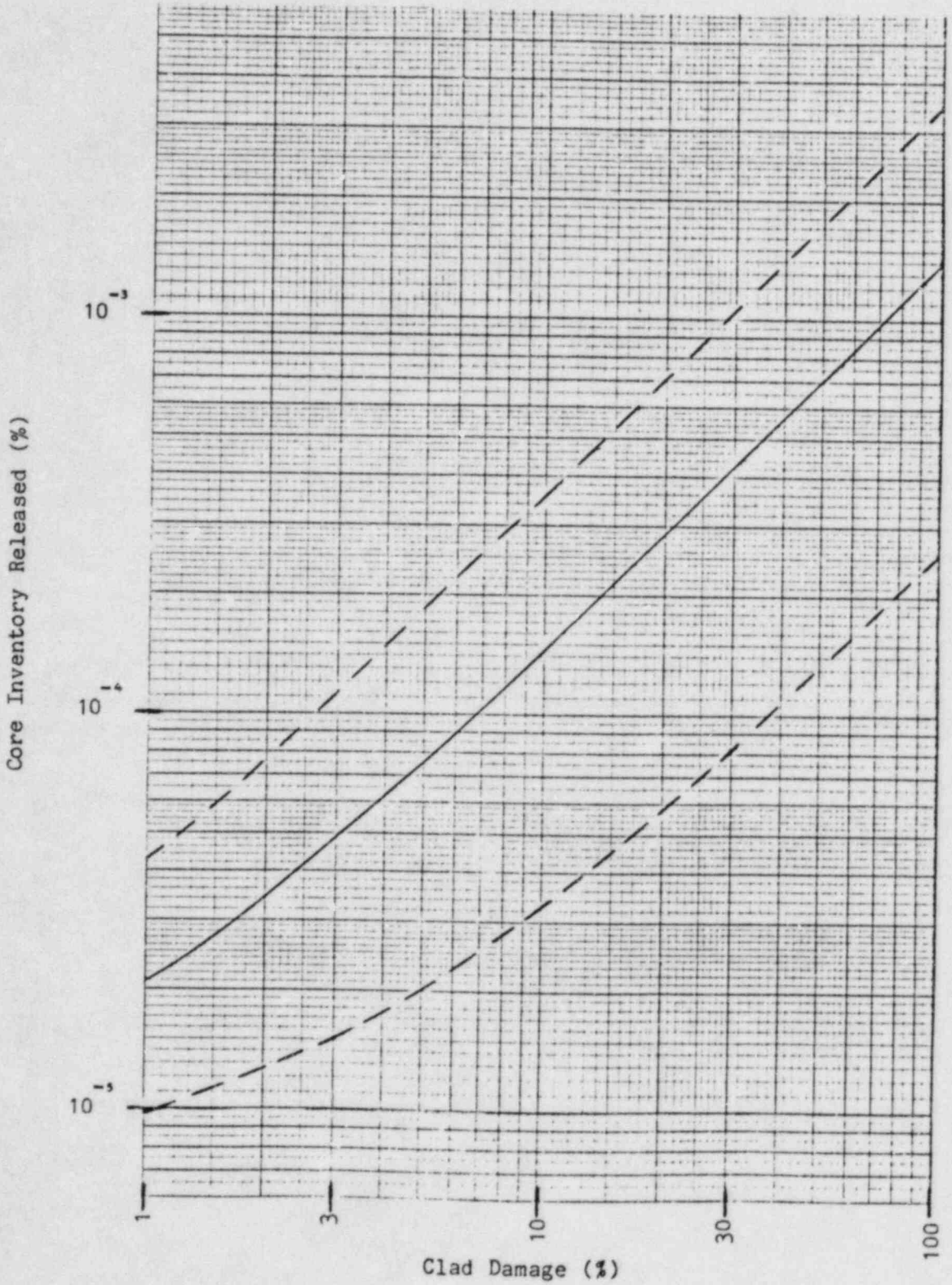

FIGURE 2-5 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF Xe-131

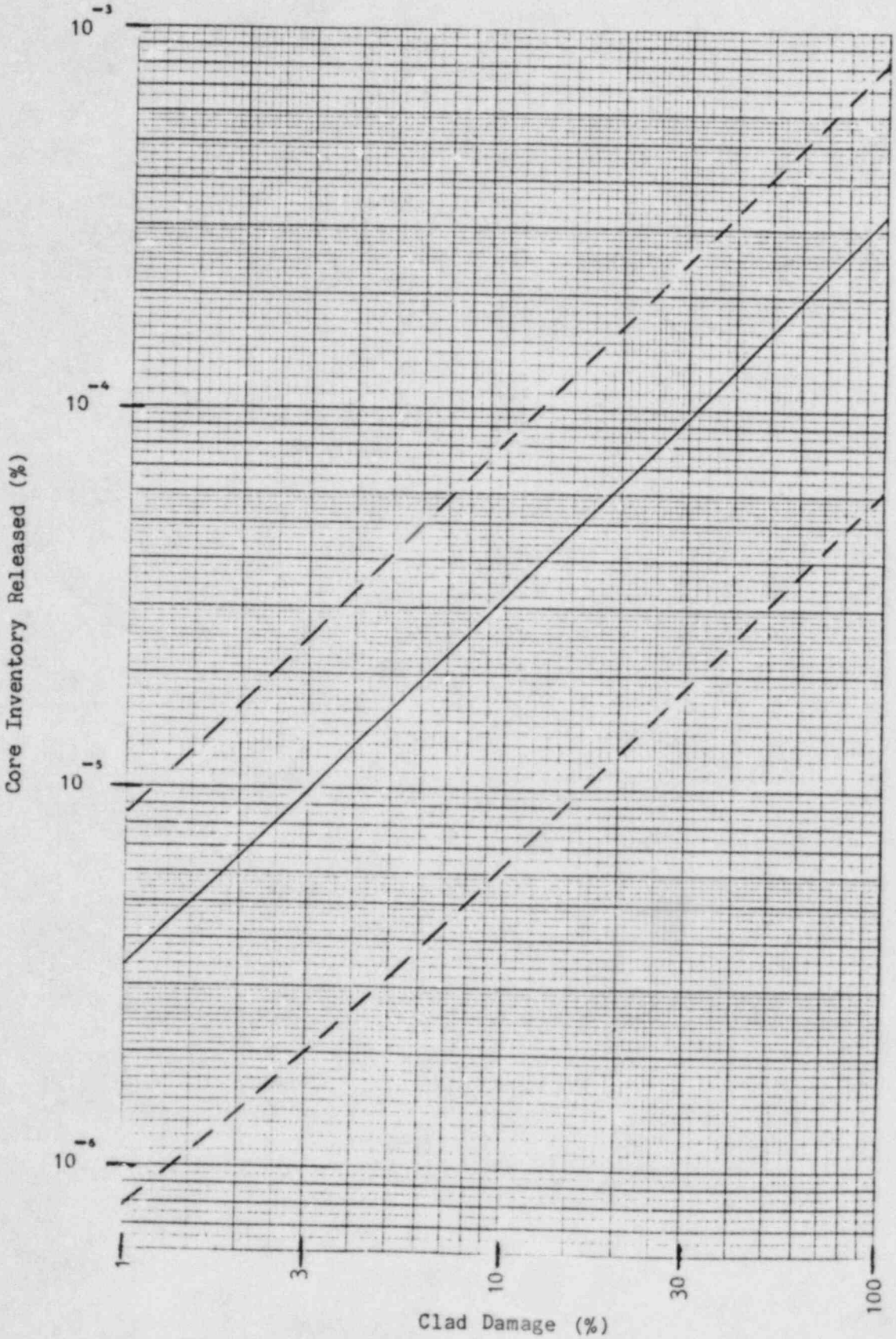

FIGURE 2-6 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF I-132

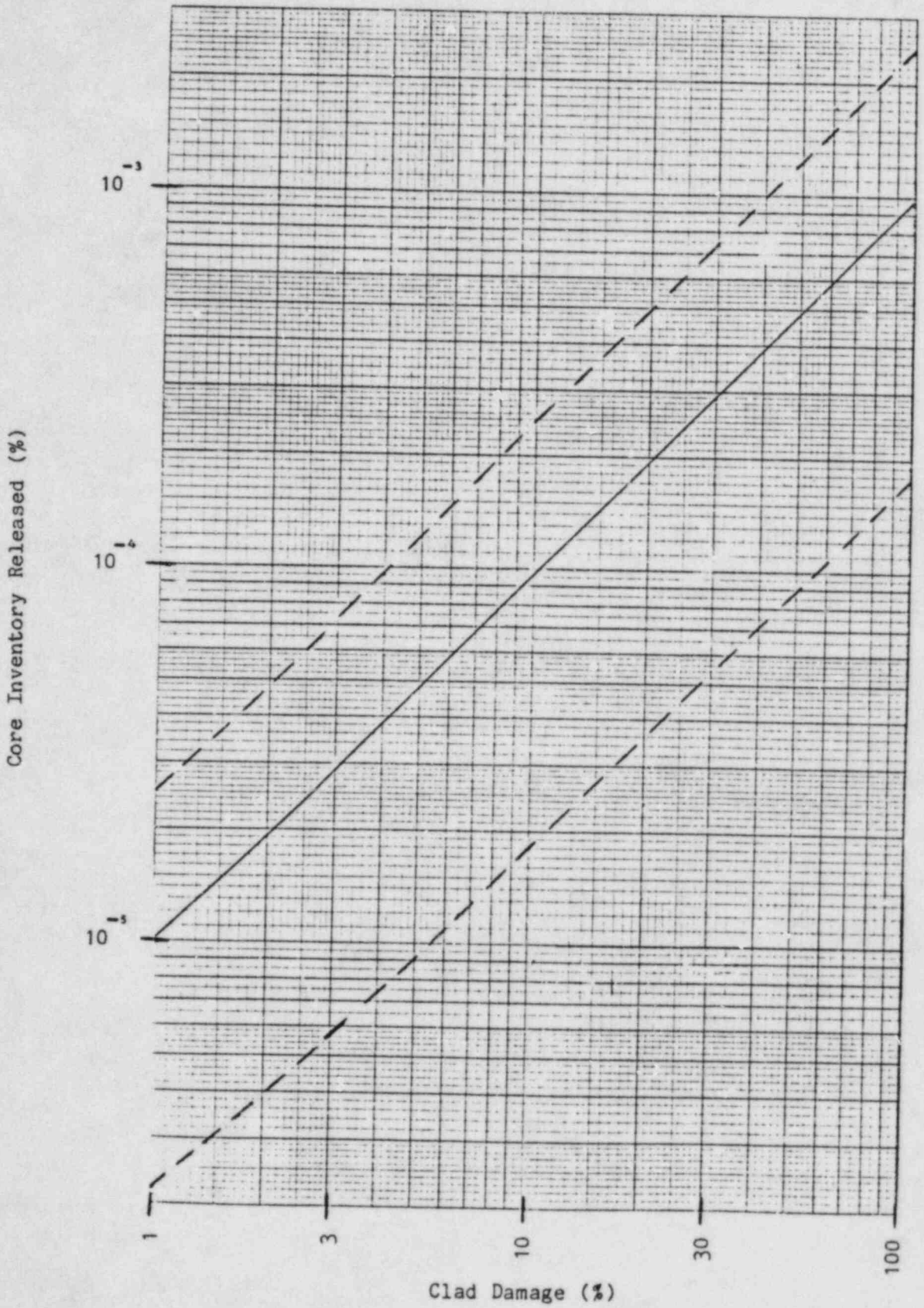

FIGURE 2-7 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF I-133

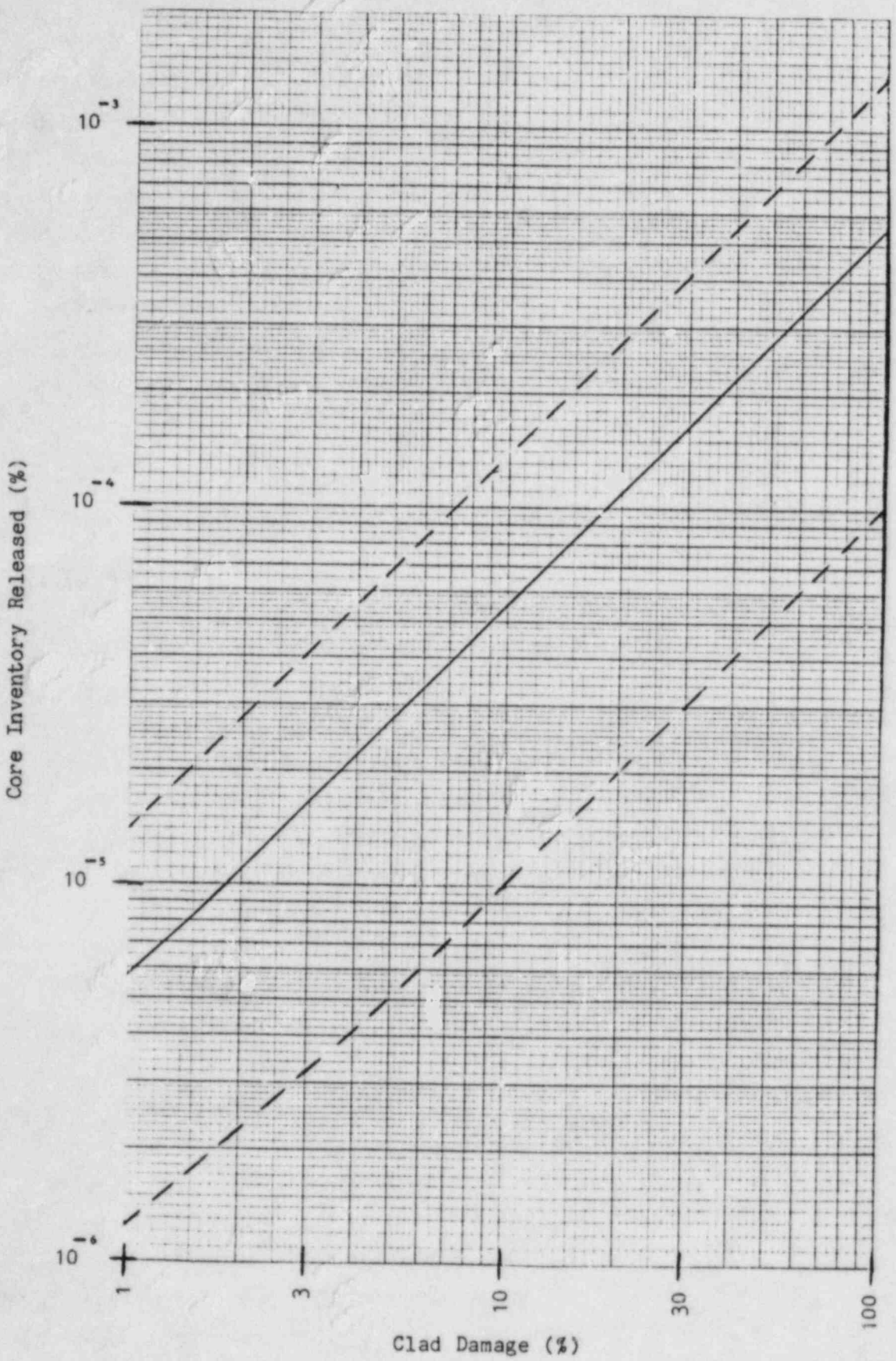

FIGURE 2-8 RELATIONSHIP OF % CLAD DAMAGE WITH ACTIVITY RELEASED OF I-135

Figure 2-1 also shows a minimum and a maximum relation which bound the best estimate line. The minimum a ; 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 2.1.1. The maximum gap inventory was determined by assuming the entire core was operating at the high burnup condition of Section 2.1.1. Table 2-3 shows the maximum and minimum values for the gap inventories. The points of the minimum and maximum relations are calculated in the same manner as discussed above.

2.1.3.2 I-131

The gap inventory from Table 2-3 for I-131 is 2.84×10^5 Curies. The minimum and maximum gap inventory for I-131 is 5.39×10^4 Ci and 7.35×10^5 Ci, respectively. The source inventory of I-131 is 9.8×10^7 Curies (Table 1-2). The normal operating specific activity for I-131 from Table 2-2 is 0.27 μCi/gm. With a primary coolant mass of 2.4×10^8 gm, the normal operating activity of I-131 is 65 Curies. The points of the nominal, minimum, and maximum relations are calculated in the same manner as described in Section 2.1.3.1. Figure 2-2 shows the percentage of I-131 activity as a function of cladding damage. The percentage release of I-131 calculated from the radionuclide analysis would be compared to Figure 2-2 to estimate the extent of cladding damage.

For I-131, the possibility of iodine spiking should be considered when distinguishing between no cladding damage and minor cladding damage. The contribution of iodine spiking is discussed in Section 2.1.2 and is estimated to be as much as 950 Curies of I-131 released to primary system with an average release of 380 Curies based on a normal operating I-131 activity of 0.27 μCi per gram[6]. The relationships of Figure 2-2 are adjusted to account for the release due to iodine spiking by adding 950 Curies of I-131 to the maximum release and by adding 380 Curies of I-131 to the minimum and nominal release. Figure 2-3 shows the percentage of I-131 released with iodine spiking versus cladding damage. Iodine spiking was not considered during the calculations of the correlations for the remaining iodines, I-132, I-133, and I-135, (Figures 2-6 through 2-8, respectively).

TABLE 2-3

EXPECTED IODINE SPIKE

Average, μCi/gm	I-131 Total Release, Curies
0.5 < SA* < 1.0	3400
0.1 < SA < 0.5	380
0.05 < SA < 0.1	200
0.01 < SA < 0.05	200
0.005 < SA < 0.01	100
0.001 < SA < 0.005	100
SA < 0.001	2

90/90 Upper Confidence Level, μCi/gm	
0.5 < SA < 1.0	6500
0.1 < SA < 0.5	950
0.05 < SA < 0.1	650
0.01 < SA < 0.05	650
0.005 < SA < 0.01	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.

2.2 RELATIONSHIP OF FISSION PRODUCT RELEASE TO OVERTEMPERATURE CONDITIONS

The temperature difference between the rod centerline and rod surface is less than 1000°F at power (less than 1400°F for the peak rod). Once the accident begins, the temperature difference should drop to less than 100°F. Decay heat (less than 7% power) is the only remaining energy source. This effect allows classification of the state of fuel overtemperature by the temperature of the cladding or the fuel rod surface temperature (Table 2-4). Fuel centerline temperatures of up to 2200°F can be a normal operational state for a new core.

Release of fission products from overheated fuel will be limited by the rod surface temperature, since this is the release interface for the fission products. One should also note that since release will not occur instantaneously upon fuel going "overtemperature", that calculated activity release may continue increasing even if the core overtemperature condition has stopped increasing or spreading. A stable release calculation with time indicates that core overtemperature has terminated.

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 2.1. Table 2-4 presents the expected fuel damage state associated with fuel rod temperatures.

Fission product release associated with fuel overtemperature 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. Reference 1 estimates the total release, including UO_2 diffusional release, to vary from 20 to 40 percent of the noble gas, iodine, and cesium inventories.

TABLE 2-4

EXPECTED FUEL DAMAGE CORRELATION WITH CLADDING 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
Control Rod Matrix Melts	> 1800 (ref. 9)
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Control Rod Cladding Melts	> 2550 (ref. 9)
Melting of Unoxidized Cladding	> 3350 (ref. 9)
Fuel Melt	> 3450
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
Melting of remaining UO_2	5100

* This temperature corresponds to the cladding or (in the case of clad failure) surface temperature of the fuel rods.

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. Core crumbling took place after fuel overtemperature and quenching. Core inventory fractions of various fission products released during the accident are given in Table 2-5. 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, in addition to diffusional release from grain boundaries and UO_2 grains, the release mechanism is believed to arise from UO_2 grain growth in steam.

The assumptions made in the WOG Methodology (1) were that: (1) TMI-2 experienced 100% fuel overtemperature (Reference 9 calculated a 50% to 60% overtemperature for TMI-2); (2) all nuclides that could be released during overtemperature, regardless of duration, were released (duration of fuel overtemperature is critical as shown in Section 2.3 and Reference 7); and (3) releases due to overtemperature occur instantaneously (as in cladding burst) and thus a linear relationship will hold between fuel overtemperature and inventory release (the time required for the processes described earlier in this section are ignored). There is no technical basis for these three assumptions made in the WOG Methodology.

Within the accuracy of this Methodology, portions of the core that experience fuel overtemperature for extended periods of time (over an hour for TMI-2) will release essentially all of the noble gases, cesiums, and iodines (9). This results in a one-to-one correlation between the percent of these isotopes released and the extent of fuel overtemperature. This relation can only be used, however, if the core RVLIS indicators or core exit thermocouples (see Section 3.2) indicate that portions of the core have uncovered for more than an hour. This correlation is consistent with the release model in Section 2.3 (from draft NUREG-0956).

TABLE 2-5

PERCENT ACTIVITY RELEASE FOR TMI-2 ACCIDENT
(Reference 9)

Nuclide	Min.	Max.	Nominal*	Min.	Max
Kr-85	47	70			
Xe-133	42	66	52	41	70
I-131	41	55			
Cs-137	45	60			
Sr-90	less than 0.08**				
Ba-140	0.1	0.2	0.13	0.08	0.2

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

** Only value available (upper bound).

TABLE 2-6

PERCENT ACTIVITY RELEASE FOR 100% OVERTEMPERATURE

Nuclide	Nominal	Minimum	Maximum
Kr, Xe, I, Cs	100	79	(135)*
Sr, Ba	0.25	0.15	0.38

* Calculational end-point

The relationship between extent of fuel damage and fission product release for several radioisotopes during overtemperature condition is depicted graphically in Figures 2-9 and 2-10. To construct the figures, the extent of fuel damage (expressed as a percentage of the core) is plotted as a linear function of the percentage of the source inventory released for various nuclides. The values used in constructing the graphs were obtained from Table 2-6. For example, if 100 percent of the core experienced overtemperatures, 100 percent of Xe-133 core inventory would be released. If 10 percent and 1 percent of the core experienced overtemperature, 10 percent and 1 percent, respectivley, of Xe-133 core inventory would be released. The assumption is also made that nuclides of any element (e.g., I-131 and I-133) have the same magnitude of release. 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 Tables 2-5 and 2-6 as the range of values; nominal values of release are simple averages of the measurements in Table 2-5.

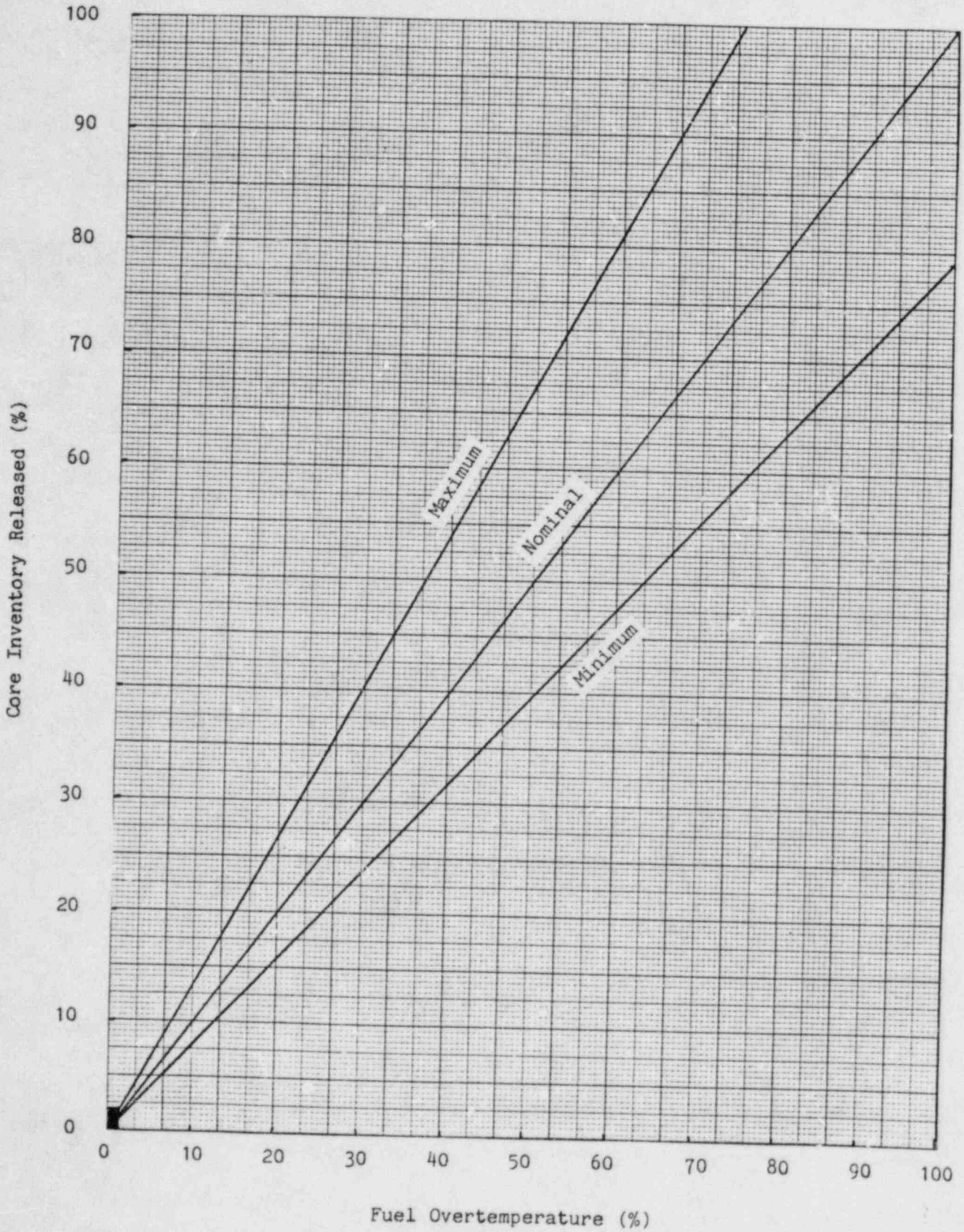

FIGURE 2-9 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF Xe, Kr, I, or Cs

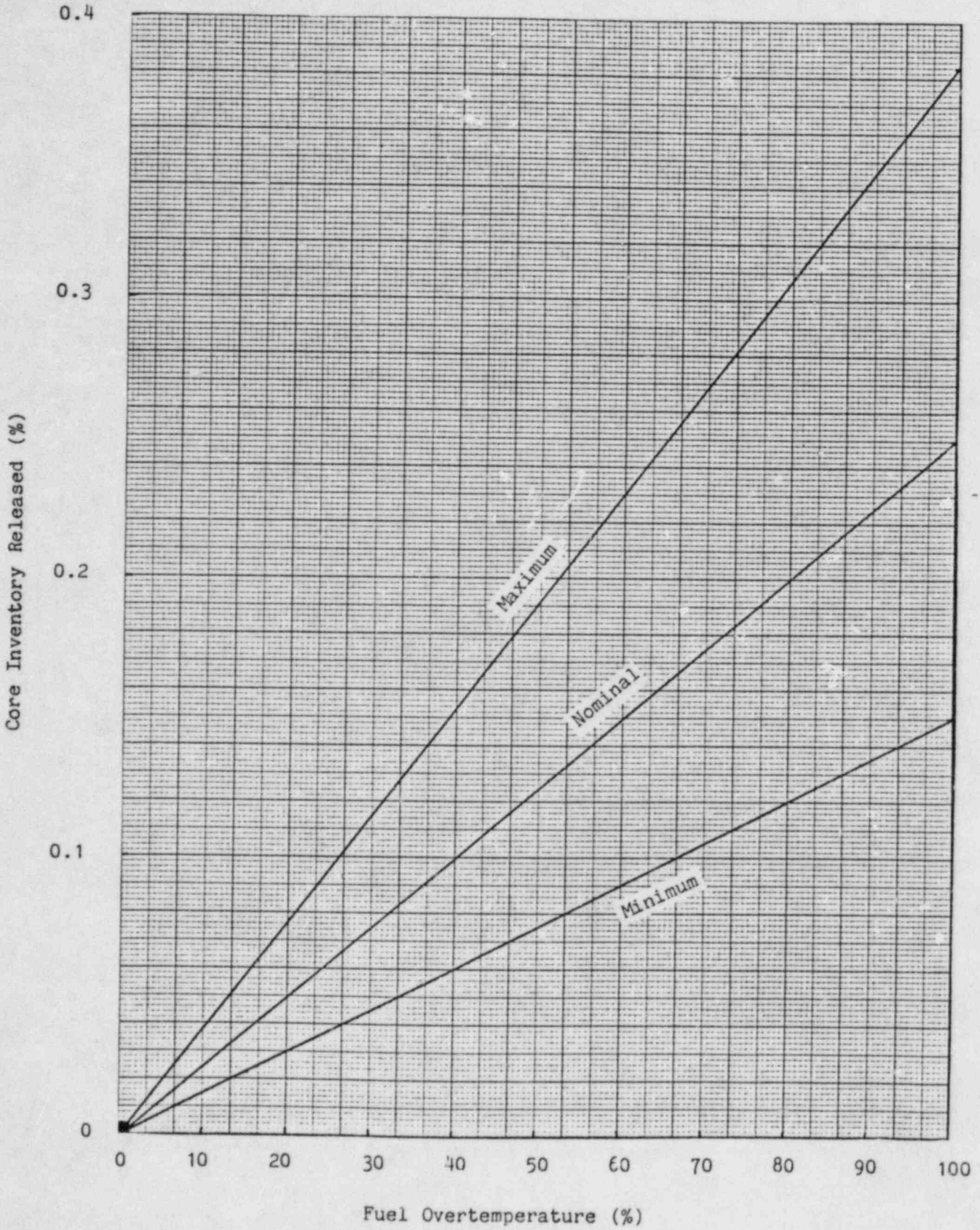

FIGURE 2-10 RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF Ba or Sr

2.3 RELATIONSHIP OF NUCLIDE RELEASE TO 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 fuel melt release conditions.

Values of the release of fission products during fuel melt conditions are obtained from the draft NUREG-0956 (ref. 7). These release calculations have been expressed as release rate coefficients for various temperature regimes. These release rate coefficients have been represented by a simple exponential equation in draft NUREG-0956. This equation has the form:

$K(T) = Ae^{BT}$ where

K(T) = release rate coefficient

A & B = constants

T = temperature

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

Fission product release percentages for three accident sequences which lead to 75 percent core melt are given in Table 2-7. Draft NUREG-0956 assumes that at most 75% of the core melts and terminates nuclide release at this point by cooling in the lower reactor vessel. The xenon, krypton, cesium, and iodine elements have been arranged into a single group because of similiarity in the expected magnitude of overtemperature release. The assumption is also made that nuclides of any element (e.g., Iodine 131 and Iodine 133) have the same magnitude of release. The differences in the calculated releases of the various elements for the different accident sequences were used to determine minimum and maximum values of expected release. Nominal values of release are simple averages of all release values within a group. Refractory releases were a source of considerable uncertainty in draft NUREG-0956. Core melt temperatures were artificially lowered to make these release terms "reasonable". Numerical estimates of fuel melt using these isotopes (Sr, Ba, Pr in Table 2-7) are highly uncertain.

TABLE 2-7

PERCENT ACTIVITY RELEASE FOR 75** PERCENT CORE MELT CONDITIONS

Species	Large* LOCA	Transient*	Small* LOCA	Large LOCA	Transient	Small LOCA
Xe	88.35	99.45	78.38			
Kr	88.35	99.45	78.38			
I	88.23	99.44	78.09	88	99	78
Cs	88.55	99.46	78.84			
Sr	10.44	28.17	14.80	15	36	19
Ba	19.66	43.87	24.08			
Pr	0.82	2.36	1.02	0.82	2.4	1.0

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

** Draft NUREG-0956 assumes a maximum of 75% core melt without RPV failure.

NOTE: The release of the refractory groups (Sr, Ba, and Pr) are higher for a small LOCA than for a large LOCA. This is due to the longer duration at fuel overtemperature conditions prior to core slumping for the small LOCA case. This arises because fuel temperatures were artifically "capped" at 5100°F (fuel melt temperature) in draft NUREG-0956.
The release of the volatile groups (Xe, Kr, I, and Cs) reflects the state of fuel overtemperature during these accidents, not the state of core melt.

The percentage release of various nuclides has been correlated to percentage of core melt with the linear extrapolations shown in Figures 2-11 through 2-13. There is no general technical basis for the linear assumption used here. Table 2-7 clearly shows that releases would be approximately linear for a small LOCA case (fuel melts slowly from top to bottom as the core is uncovered), but large LOCA's or transients release activity in a non-linear fashion. The volatile element release occurs (according to draft NUREG-0956) from the fuel overtemperature region. Fuel melt cannot, therefore, be determined from these nuclides. The strontium-barium and lanthanum groups must be used.

Figures 2-11 through 2-13 are therefore reliable for the case of a small LOCA. For any other accident type, these figures are at best uncertain as estimates of percent fuel melt. Since the error bands for the various inputs (see Section 5), uncertainties in draft NUREG-0956, and non-linearity exceed the 50%-75% damage spread available for the second fuel melt category available, no numerical estimate can technically be made for fuel melt during any accident but a small LOCA.

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FIGURE 2-11 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF Xe, Kr, I, or Cs

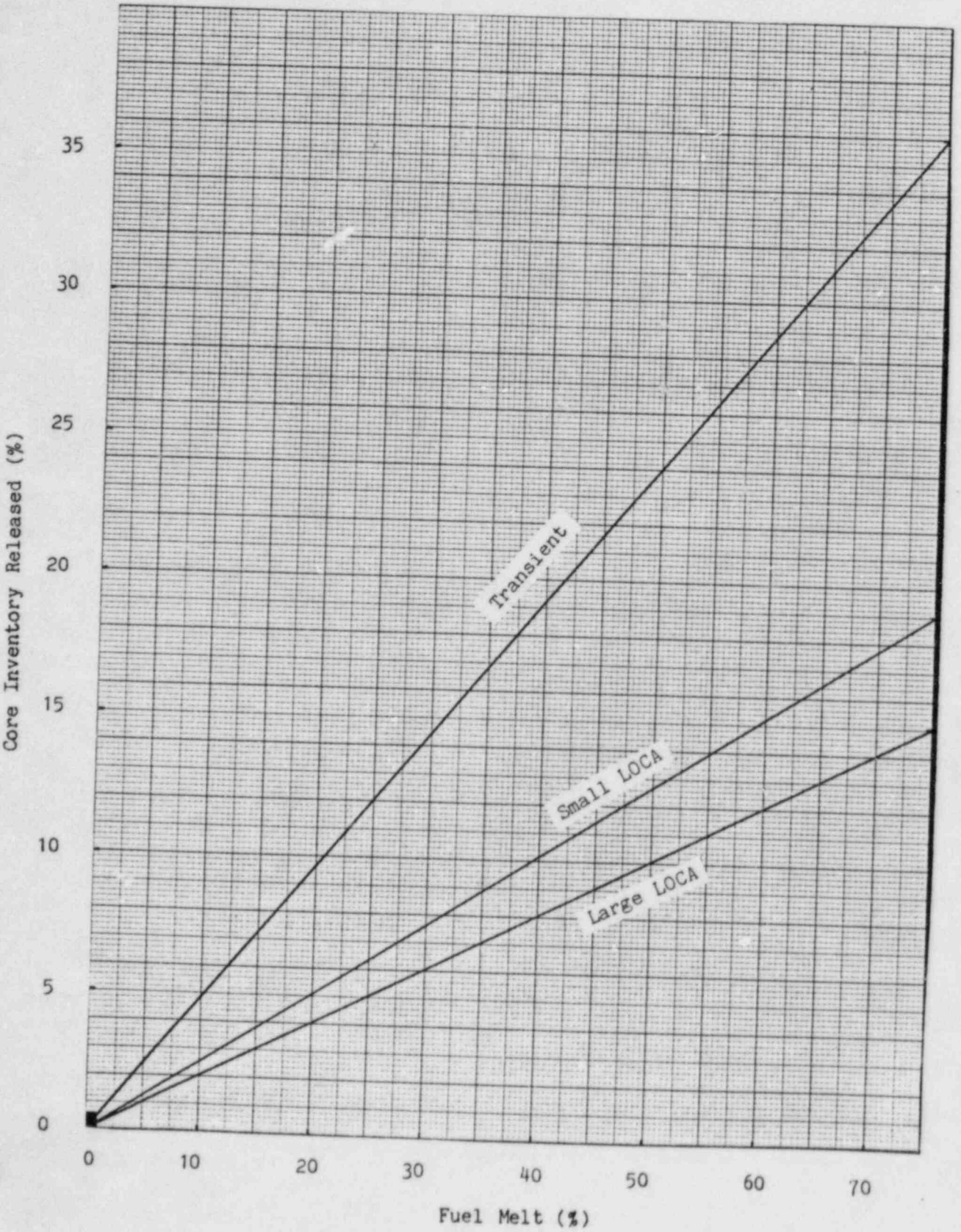


FIGURE 2-12 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF Ba or Sr

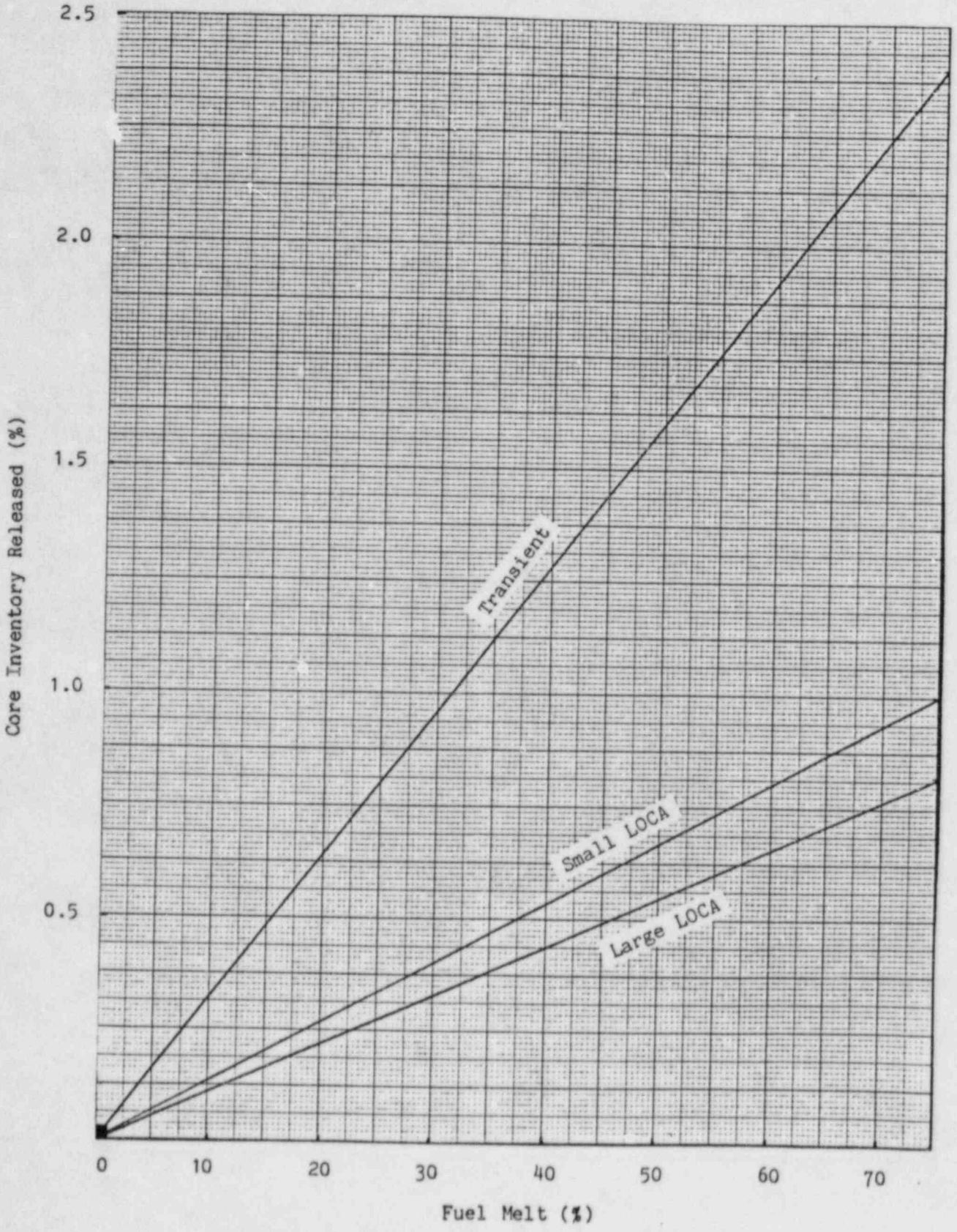


FIGURE 2-13 RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF Pr

2.4 CORE CRUMBLING

Core crumbling can occur after cladding failure has begun. Wholesale melting or chemical reaction of the cladding will remove the structural support from the fuel. Individual fuel pellets or pellet/cladding fragments may collapse under the combined physical loads of the fuel pin compression springs, steam and water dynamic loads, and vibration. The TMI-2 core crumbled at the time of core quenching (Reference 9).

Core crumbling does not release any characteristic nuclides or result in significant increases in overall activity. Core crumbling is a possibility anytime significant gap activity is released (50-100% gap release); and either (1) core exit thermocouples show a sustained temperature above 1600°F, or (2) containment hydrogen concentrations indicate a significant cladding/steam reaction.

According to reference 7, once the core loses its physical structure fuel melting and any further activity release from fuel overtemperature will cease unless the reactor pressure vessel is breached. Since some core crumbling could occur before large portions of the fuel overheat, significant gap release along with high containment hydrogen levels or core exit thermocouple temperatures may indicate loss of core structure- regardless of the absence of other indicators. Core crumbling may also, however, prevent further fuel overtemperature or fuel melting of those portions of the core that crumble.

3.0 AUXILIARY INDICATORS

There are plant indicators monitored during an accident which cannot provide useful estimates of the core damage state by themselves. They 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, containment radiation level, and activity ratios. When providing an estimate for core damage, these plant indicators (if available) should be used to confirm the results of the radionuclide analysis. For example, if the core exit thermocouple readings and reactor vessel water level indicate a possibility of cladding damage and the radionuclide concentrations indicate no cladding 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. This reaction produces hydrogen. The hydrogen production characteristic of the zirconium-steam reaction is that every mole of zirconium metal that reacts with steam produces two moles of hydrogen. It is assumed that all of the hydrogen produced is released to the containment atmosphere. The hydrogen dissolved in the primary system during normal operation is less than 1500 SCF. This is 0.06% of containment volume and contributes an insignificant amount of the total hydrogen released to the containment. In the absence of hydrogen control measures, monitoring this containment hydrogen concentration during the accident can provide an indication of the extent of zirconium-steam reaction. The percentage of zirconium-steam reaction does not equal the percentage of cladding damaged, but it does provide a qualitative verification of the extent of cladding damage estimated from the radionuclide analysis. Cladding damage estimates based on radionuclide analysis will generally be higher than estimates based on hydrogen generation since all gap activity is assumed to be released as soon as any part of the cladding is breached. Figure 3-1 shows the relationship between the hydrogen concentration and the percentage of zirconium-steam 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 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 that 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. 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 an extensive zirconium-steam reaction has occurred. Uncontrolled ignition of hydrogen and deliberate ignition outside the recombiners will hinder any quantitative use of hydrogen concentration as an auxiliary indicator. If the oxygen amount depleted during the burn is known, it can be used to estimate the amount of hydrogen burned. If the oxygen amount depleted is not known, it can be assumed that a minimum concentration of 4 percent hydrogen is needed for ignition of hydrogen to occur. 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 bewtween the hydrogen concentration (percent volume) and the percentage of zirconium-steam reaction. The hydrogen concentration shown is the result of the analysis of a dry containment sample. The curve was based on containment volume and the initial zirconium mass of the cladding in the core. To use the auxiliary indicator of hydrogen concentration, the assumptions are that all hydrogen from a zirconium-steam reaction is released to containment, a well-mixed atmosphere, and ideal gas behavior inside the containment.

The relationship between hydrogen concentration of a dry sample and the fraction of zirconium-steam reaction is based on the following formula:

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

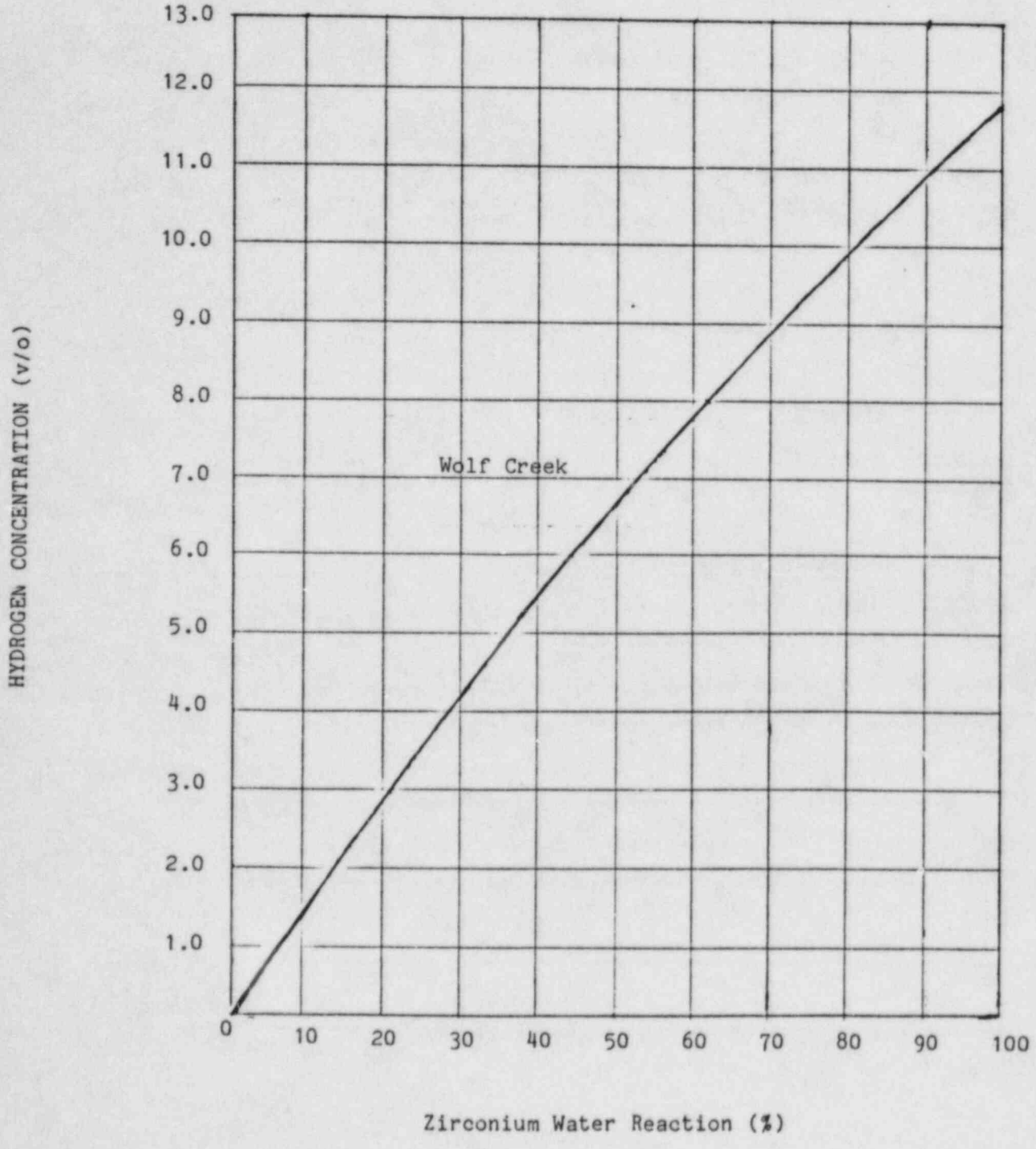


FIGURE 3-1 CONTAINMENT HYDROGEN CONCENTRATION BASED ON ZIRCONIUM-STEAM REACTION

where:

FZWR = fraction of zirconium-steam reaction

ZM = total zircaloy mass, 47,000 lbm (Reference 14)

H = conversion factor, 7.92 SCF of H_2 per pound of zircaloy (21wt% zirconium) reacted

V = containment volume, 2.4×10^6 SCF

3.2 CORE EXIT TEMPERATURES AND REACTOR VESSEL WATER LEVELS

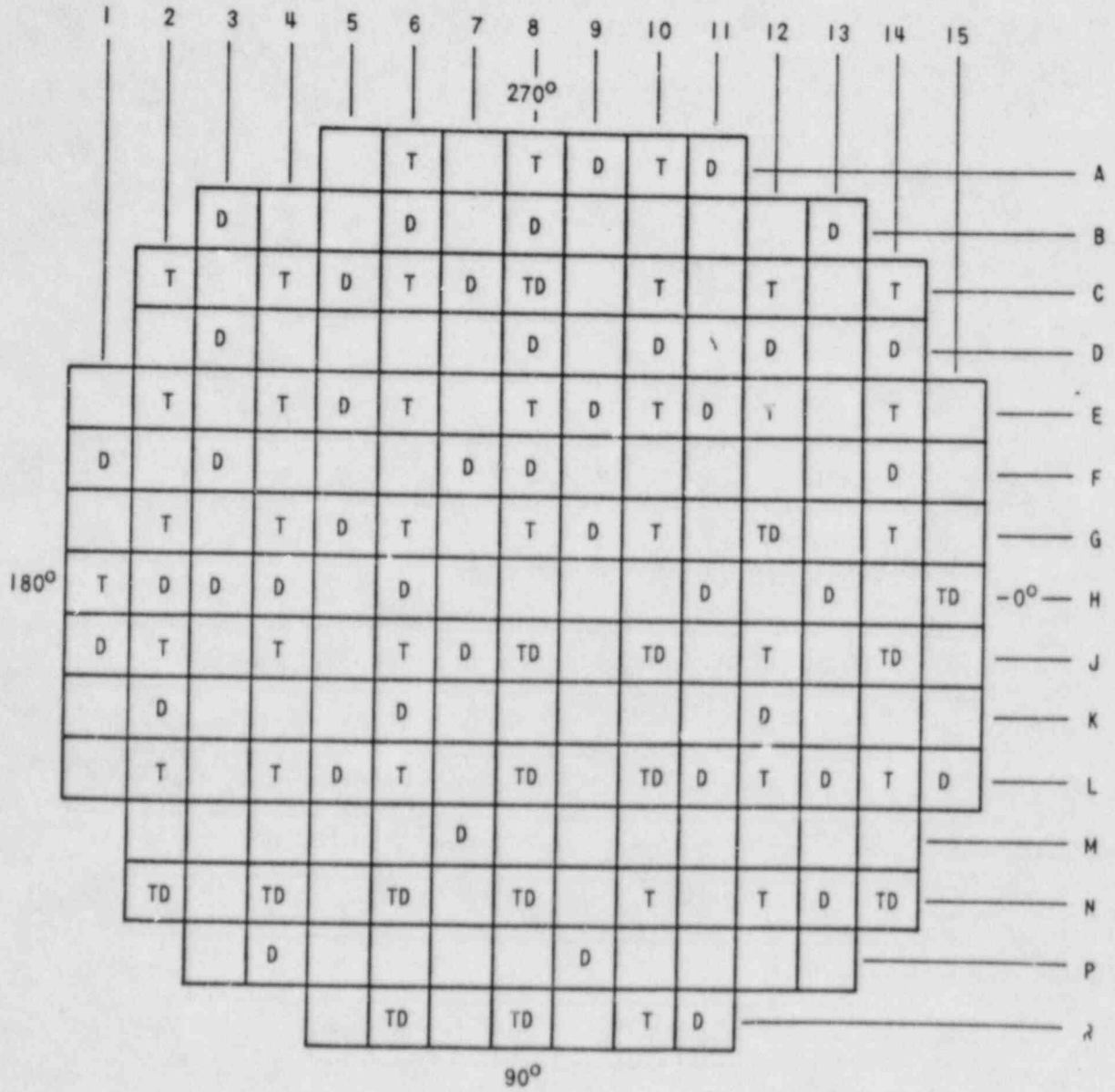
Core exit thermocouples measure the temperature of the fluid at the core exit at various radial core locations (Figure 3-2). The typical thermocouple system is qualified to read temperatures as high as 1650°F. This is the ability of the system to measure the fluid temperatures at the incore thermocouples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

The reactor vessel level indication system (RVLIS) uses 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.

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

- Due to the heat transfer mechanisms between the fuel rods, steam, and thermocouples, the highest cladding temperature will be higher than the CETC readings. Therefore, if thermocouples read greater than 1300°F, clad failure may have occurred. 1300°F is the lower limit for cladding failure. Thermocouple readings in excess of 1600°F may indicate core crumbling. 1600°F is the lower limit for oxidation of the cladding.

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



Distribution of Thermocouples and Flux Thimbles WCGS

Figure 3-2

If RCPs are not running, the following apply.

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

- If RVLIS indicates a collapsed liquid level of less than 3.5 feet 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 cladding temperatures start to reach 1200°F. Ten minutes later they can be as high as 2200°F. These times will shorten as the break size increases due to the core uncovering faster and to a greater depth.

- If the RVLIS indication shows a collapsed liquid level between 3.5 feet 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. It is recommended in Reference 11 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 core damage has occurred.

3.3 CONTAINMENT RADIATION MONITORS AND CORE DAMAGE

Post-accident radiation monitors at WCGS can be used to estimate the xenon and krypton concentrations in the containment.

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

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

2. Accidents were considered in which 100% of the noble gases, 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 developed in Reference 2 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 the assumed noble gas release, Table 3-1, and the plant characteristics of the detector. The gamma ray source strengths presented in Table 3-1 are based on 100 percent release of the noble gases.

A relation developed specifically for the Wolf Creek Plant (Reference 16) indicates that a 100% noble gas release will result in an initial reading of 1.1×10^7 R/hr on both high-range monitors, at t = 0 hr. Correcting this value for one hour of decay gives 3.7×10^6 R/hr. To determine the exposure rate of the detector based on 0.3 percent noble gas release, 0.3 percent of the gamma ray source strength is used. The same decay curve (adjusted for initial activity) is used for WCGS.

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

In general, values below 0.3% releases are indicative of cladding failures, values between 0.3% and 100% release are in the fuel pellet overtemperature and the core melt regimes. To represent the release of the normal operating noble gas activity in the primary coolant as obtained from ANS 18.1[6], 1.0×10^{-3}% of the gamma-ray source strength is used. In actual practice it must be recognized that there is overlap between the regimes because of the nature in which core heating occurs. The hottest portion of the core is in the center due to flux distribution and hence greater fission product inventory. Additionally, heat transfer is greater at the core periphery due to proximity of pressure vessel walls. 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 cladding 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.

TABLE 3-1

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 Shutdown (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	3.5×108	6.3×10^{5}	4.5×10^{6}	2.6×10^{6}	9.7×10^{4}
4.00 - 5.00	3.1×107	4.4×10^{4}	3.6×10^{2}	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	1.8×105	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

TABLE 3-2

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

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

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

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, are representative of the WCGS high-range detectors.

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

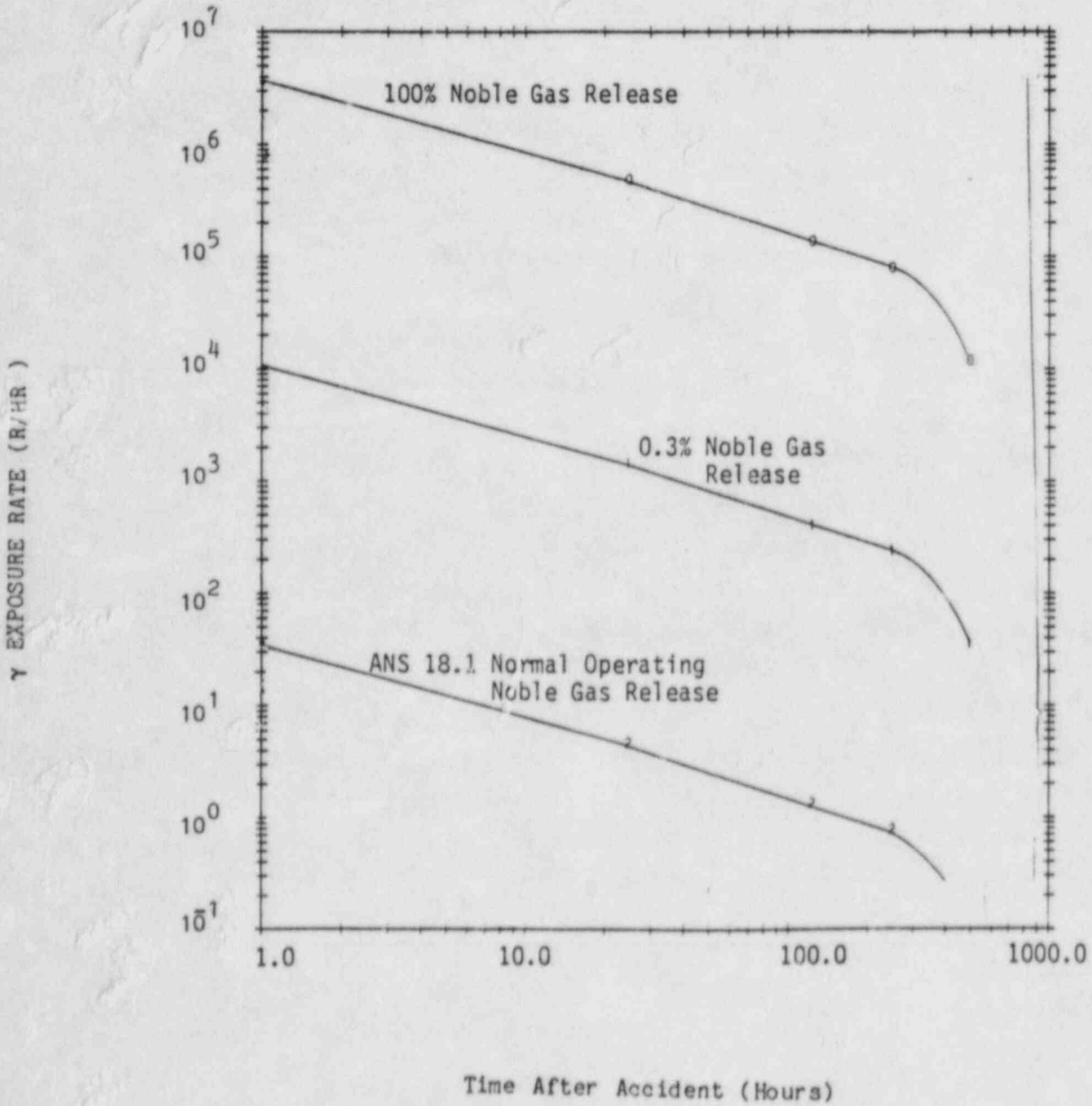


FIGURE 3-3 PERCENT NOBLE GASES IN CONTAINMENT - WOLF CREEK

3.4 GAP ACTIVITY RATIOS

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

Equilibrium conditions can also be considered to exist in the fuel rod gap. Under this condition the gap inventory of the nuclides is fixed. The distribution of the nuclides in the gap are not in the same proportions as the nuclides in the fuel pellet since the migration of each nuclide into the gap depends on its particular diffusion rate. The relative diffusion rates of these nuclides under various operating conditions are approximately constant. Therefore the relative ratios of the nuclides in the gap are known.

In the presence of other indicators of a major release, the relative ratios of the nuclides can be compared with the relative ratios of the nuclides analyzed (corrected to shutdown) during an accident to determine the source of the fission product release. Table 3-3 presents the relative activity ratios for both the fuel pellet and the gap. The relative ratios for gap activities are significantly lower than the fuel pellet activity ratios. Measured relative ratios greater than gap activity ratios are indicative of more severe failures (e.g., fuel overheat).

This procedure is to be used only after significant cladding failure has been identified. Activity ratios from pinhole leaks can mimic the activity ratios in the fuel rods. This procedure then can discriminate more severe releases from the fuel - after significant cladding failure is noted.

TABLE 3-3

ISOTOPIC ACTIVITY RATIOS OF FUEL PELLET AND GAP

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

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

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

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

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

1-50%	cladding failure
50-100%	cladding failure
1-100%	fuel pellet overtemperature
1-100%	core crumbling
1-50%	fuel melt
50-75%	fuel melt

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 cladding damage, fuel overtemperature, and core melt considerable judgment needs to be applied.

4.1 SAMPLING LOCATIONS

To obtain the most accurate assessment of core damage, it is recommended to sample and analyze radionuclides from the reactor coolant system, the containment atmosphere, and the containment sump (if available). Other samples can be taken depending on availability. The specific sample locations to be used during the initial phases of an accident should be selected based on the type of accident in progress. If the type of accident scenario is unknown, known plant parameters (pressure, temperature, level indicators, 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 4-2 presents a list of the suggested sample locations for different accident scenarios based on the usefulness of the information derivable from the sample.

TABLE 4-1

CHARACTERISTICS OF CATEGORIES OF FUEL DAMAGE*

Core Damage Category	Core Damage Indicator: Percent and Type of Fission Products Released	Fission Product Ratio	Containment Radiogas Monitor (R/hr) 10 hours after shutdown**	Core Exit Thermocouples Readings (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)***
No damage	Kr-87 < 0.001 Xe-133 < 0.001 I-131 < 0.001 I-133 < 0.001	Not Applicable	<10.	< 1300	No uncovery	Negligible
1-50% clad damage	Kr-87 0.001 - 0.01 Xe-133 0.001 - 0.01 I-131 0.001 - 0.3 I-133 0.001 - 0.1	Kr-87 = 0.022 I-133 = 0.71	1.0(1) to 2.8(2)	1300 - 1650	Core uncovery	0 - 6
50-100% clad damage	Kr-87 0.01 - 0.02 Xe-133 0.1 - 0.2 I-131 0.3 - 0.5 I-133 0.1 - 0.2	Kr-87 = 0.022 I-133 = 0.71	2.8(2) to 5.7(2)	1300 - 1650	Core uncovery	6 - 12
1-100% fuel pellet overtemperature	Xe-Kr, Cs, I 1 - 40 Sr-Ba 0 - 0.2	Kr-87 = 0.22 I-133 = 2.1	5.7(2) to 1.5(5)	> 1650	Core uncovery	6 - 12
1-100% core crumbling	Xe-Kr, Cs, I 1 - 40 Sr-Ba 0 - 0.2	Kr-87 = 0.22 I-133 = 2.1	5.7(2) to 1.5(5)	> 1650	Core uncovery	6 - 12
1-50% fuel melt	Xe, Kr, Cs, I 40-70 Sr-Ba 0.2 - 0.8 Pr 0.1 - 0.8	Kr-87 = 0.22 I-133 = 2.1	1.5(5) to 2.5(5)	> 1650	Core uncovery	6 - 12
50-75% fuel melt	Xe, Kr, Cs, I, Te > 70 Sr, Ba >24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 2.5(5)	> 1650	Core uncovery	6 - 12

* 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 for times other than 10 hours. These values are from Figure 3-3.

*** Hydrogen recombiners and ignitions may obviate these values.

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

TABLE 4-2

Suggested Sampling Locations

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

* Assume operating at that level for some appreciable time.

** 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.

5.0 LIMITATIONS

The emphasis of this methodology is on radiochemical analysis of appropriate liquid and gaseous samples. The assumption has been made 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 analyses via gamma spectrometry are used to calculate the specific activity of various fission products released. Radiochemical analyses of fission products under normal plant operating conditions are accurate to $\pm$10 percent. Radiochemical analyses of post accident samples which may be much more concentrated, and contain unfamiliar nuclides, and which must be performed expeditiously may have an error band of 20 to 50 percent.

Having obtained specific activity analysis, the calculation of total release requires knowledge of the total water volume from which the samples were taken. Care must thus be exercised in accounting for volumes of any water added via ECCS and spray systems, accumulators, and chemical addition tanks. 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 $\pm$10 percent.

The specific activity also requires a correction to adjust for the decay of the nuclide in which the measured specific activity is decay corrected to time of reactor shutdown. For some nuclides, precursor effects must be considered in the decay correction calculations. The precursor effect is limited to parent-daughter relationships for this methodology. A major assumption is made that the release percentages of the parent and daughter are equal. For overtemperature and melt releases, this assumption is consistent with the technical basis presented in Sections 2.2 and 2.3, but the gap releases could be different by as much as a factor of 2.

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

The models employed for estimates of release at higher temperature have not been verified by experiment. Additionally, calculations of expected core temperatures for severe accident conditions are still incomplete. 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 core melt conditions. Consideration of the melt release estimates shown in Table 2-6 for the refractory nuclides indicate a range of approximately ±70 percent.

From these considerations it is clear that the combined uncertainties are such that core damage estimates using this methodology are sufficient only to establish major categories of core damage.

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

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

6.2 POWER CORRECTION FOR CORE INVENTORY

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	=	2674 Mwt
10	days	at	100%	power	=	3565 Mwt
10	days	at	50%	power	=	1783 Mwt
5	days	at	75%	power	=	2674 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 6-2.

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	1.3 (0)		
Xe 133	1.8 (2)		
I 131		2.6 (4)	6.6 (3)
I 132		3.9 (4)	9.3 (3)
Cs 137		2.5 (4)	5.9 (2)
Ba 140		1.4 (2)	3.2 (1)

TABLE 6-2

FISSION PRODUCT INVENTORY AT REACTOR SHUTDOWN

Isotope	Equilibrium Iventory at End-of-Life, Ci	Power Correction Factor	Corrected Inventory, Ci
Kr 87	4.0 (7)	0.75	3.0 (7)
Xe 133	2.0 (8)	0.68	1.4 (8)
I 131	9.8 (7)	0.68	6.7 (7)
I 132	1.4 (8)	0.75	1.0 (8)
Cs 137	1.1 (7)	0.60	6.6 (6)
Ba 140	1.7 (8)	0.65	1.1 (8)

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 2674 MWt for 5 days prior to shutdown. Thus, the power correction is as follows:

$$\text{Power Correction Factor} = \frac{2674 \text{ MWt}}{3565 \text{ MWt}} = 0.75$$

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

$$\text{Corrected Inventory} = 1.4 \times 10^{8} \text{ Curies} \times 0.75$$
$$= 1.0 \times 10^{8} \text{ Curies}$$

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.

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

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

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

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

$$= \frac{2674 \left(1-e^{-(8.7E-2)(20)}\right) e^{-(8.7E-2)(25)} +}{3565}$$

$$+ \frac{3565 \left(-e^{-(8.7E-2)(10)}\right) e^{-(8.7E-2)(15)} +}{3565}$$

$$+\frac{1783\ (1-e^{-(8.7E-2)x(10)})\ e^{-(8.7E-2)x(5)} + 2674(1-e^{-(8.7E-2)x(5)})\ e^{-(8.7E-2)x(0)}}{3565}$$

$$= \frac{2424}{3565} = 0.68$$

3) Isotopes with half-lives greater than 1 year:

For this scenario, the core has operated for 60 effective full power days (EFPD) during the current cycle operation, 300 EFPD during the previous cycle, and 300 EFPD during the earlier cycle.

For Cs-137 ($t_{1/2}$ = 30 yr.)

$$\text{Power Correction Factor} = \frac{60 + (2/3)300 + (1/3)300}{600} = 0.6$$

6.3 PRESSURE AND TEMPERATURE CORRECTION

As discussed in Section 1.2.4.2, 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 grams 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,

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

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

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

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

6.3.1 ACTIVITY OF EACH MEDIUM

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

1. Containment Volume

$$V = 2.45 \times 10^6 \text{ SCF} \times \frac{28.3 \times 10^3 \text{ cc}}{\text{SCF}} \times \frac{P_3}{P_2} \times \frac{(T_2 + 460)}{(T_3 + 460)} = 6.4 \times 10^{10} \text{ cc}$$

where:

P_2 = containment pressure = 20 psia
T_2 = containment temperature = 200°F
P_3 = standard pressure = 14.7 psia
T_3 = standard temperature = 73°F

TABLE 6-3

ADJUSTED SPECIFIC ACTIVITY
DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

Containment Atmosphere, μCi/cc

Isotope	Specific Activity From Table 6-1	Correction Factor	Specific Activity Adjusted
Kr 87	1.3(0)	1.1	1.4(0)
Xe 133	1.8(2)	1.1	1.9(2)

TABLE 6-4

ADJUSTED SPECIFIC ACTIVITY
DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

Containment Sump, μCi/gm

Isotope	Specific Activity From Table 6-1	Correction Factor	Specific Activity Adjusted
I 131	2.6(4)	1.0	2.6(4)
I 132	3.9(4)	1.0	3.9(4)
Cs 137	2.5(4)	1.0	2.5(4)
Ba 140	1.4(2)	1.0	1.4(2)

TABLE 6-5

ADJUSTED SPECIFIC ACTIVITY
DUE TO PRESSURE AND TEMPERATURE DIFFERENCES

RCS, μCi/gm

Isotope	Specific Activity From Table 6-1	Correction Factor*	Specific Activity Adjusted
I 131	6.6(3)	1.0	6.6(3)
I 132	9.3(3)	1.0	9.3(3)
Cs 137	5.9(2)	1.0	5.9(2)
Ba 140	3.2(1)	1.0	3.2(1)

* No Correction is necessary since the nuclide analysis was performed on a gram basis

TABLE 6-6

CONTAINMENT ATMOSPHERE ACTIVITY

Isotope	Adjusted Specific Activity, μCi/cc From Table 6-3	Atmosphere Volume, cc	Activity, Ci
Kr 87	1.4(0)	6.4(10)	9.0(4)
Xe 133	1.9(2)	6.4(10)	1.2(7)

TABLE 6-7

CONTAINMENT SUMP ACTIVITY

Isotope	Adjusted Specific Activity, μCi/gm From Table 6-4	Sump Water Mass, gm	Activity, Ci
I 131	2.6(4)	1.4(8)	3.7(6)
I 132	3.9(4)	1.4(8)	5.4(6)
Cs 137	2.5(4)	1.4(8)	3.5(6)
Ba 140	1.4(2)	1.4(8)	1.9(4)

TABLE 6-8

RCS ACTIVITY

Isotope	Ajusted Specific Activity, μCi/gm From Table 6-5	RCS Water Mass, gm	Activity, Ci
I 131	6.6(3)	2.9(8)	1.9(6)
I 132	9.3(3)	2.9(8)	2.7(6)
Cs 137	5.9(2)	2.9(8)	1.7(5)
Ba 140	3.2(1)	2.9(8)	9.3(3)

2. Sump Mass

The sump water level monitor indicates 80 inches of water. From Table 1-3, this corresponds to a water volume of 1.4×10^8cc. The sump temperature is below 200°F and no adjustment is necessary in converting the sump volume to sump mass.

$$\text{Sump mass} = 1.4 \times 10^8 \text{cc} \times \rho_{STP}$$

$$= 1.4 \times 10^8 \text{gm}$$

where:

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

3. Primary Coolant Mass

The primary system monitors indicate the system is full. The volume of the primary system is 11,393 ft^3.

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

$$\text{RCS mass} = 11{,}393 \text{ ft}^3 \times \frac{\rho}{\rho_{STP}} \text{ (2)} \times \rho_{STP} \times \frac{28.3 \times 10^3 \text{ cc}}{\text{ft}^3}$$

$$= 2.9 \times 10^8 \text{ gm}$$

where:

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

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

6.3.2 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, before decay correction.

6.4 DECAY CORRECTION

The released 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 activity, Ci
λ_i = decay constant of isotope i, hr^{-1}
t = time elapsed from reactor shutdown to time of sampling, hr
A_o = decay corrected activity, Ci

For I-131 activity, Table 6-9.

A = 5.6 E+6 Ci

λ_i = 3.59 E-3 hr^{-1}

t = 6 hr

$$A_o = \frac{5.6\ E+6}{e^{-(3.6\ E-3)\ x\ (6)}}$$

A_o = 5.7 E+6 Ci

TABLE 6-9

TOTAL ACTIVITY RELEASED
BEFORE DECAY CORRECTION

Isotope	Table 6-6 Atmosphere, Ci	Table 6-7 Sump, Ci	Table 6-8 RCS, Ci	Total, Ci
Kr 87	9.0(4)			9.0(4)
Xe 133	1.2(7)			1.2(7)
I 131		3.7(6)	1.9(6)	5.6(6)
I 132		5.4(6)	2.7(6)	8.1(6)
Cs 137		3.5(6)	1.7(5)	5.2(5)
Ba 140		1.9(4)	9.3(3)	2.8(4)

For I-132, a parent-daughter relationship must be considered in calculation of decay adjustment. Following the methodology outlined in Section 1.2.4.3, 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 1-2 = 1.4E8 Ci

Q_B^o = 100% source inventory of I-132, Table 1-2 = 1.4E8 Ci

K = decay branching factor, Table 1-3 = 1.00

λ_A = Te-132 decay constant = 8.92E-3 hr^{-1}

λ_B = I-132 decay constant = 3.07E-1 hr^{-1}

t = time from shutdown to sample time = 6 hr

$Q_B(t)$ = 1.14E8 Ci + 2.22E7 Ci

$Q_B(t)$ = 1.36E8 Ci of I-132 6 hours after shutdown

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

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

$$= \frac{2.22E7}{1.36E8}$$

Fr = 0.16

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

M_B = Fr x measured activity, Table 6-9

= 0.16 x 8.1E+6

M_B = 1.3E+6

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

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

= 1.3E+6 x 6.3

M_B^o = 8.2E+6

Table 6-10 lists the decay corrected activities of the sampling analysis.

6.5 PERCENTAGE OF INVENTORY RELEASED

The corrected inventories are used to determine the percentage of inventory released for each isotope. The total activities released (Table 6-10) are compared to the available fuel inventory (Table 6-2). Table 6-11 presents the release percentages for the isotopes of this example.

TABLE 6-10

DECAY CORRECTED RELEASE ACTIVITIES

Nuclide	Table 6-9 Activity (Ci)	x	Parent-Daughter Factor*	x	Decay Factor**	=	Decay Corrected Activity (Ci)
Kr-87	9.0 (4)		N/A		26.6		2.4 (6)
Xe-133	1.2 (7)		0.97		1.03		1.2 (7)
I-131	5.6 (6)		N/A		1.02		5.7 (6)
I-132	8.1 (6)		0.16		6.3		8.2 (6)
Cs-137	5.2 (5)		N/A		1.00		5.2 (5)
Ba-140	2.8 (4)		N/A		1.01		2.8 (4)

* Fraction of activity due to decay of only the daughter

** Decay factor = $1/e^{-\lambda_i t}$ where t = 6 hr

TABLE 6-11

RELEASE PERCENTAGE

Isotope	Total Activity Released (Ci)	Corrected Inventory (Ci)	Release Percentage (%)
Kr 87	2.4(6)	3.0(7)	8.0
Xe 133	1.2(7)	1.4(8)	8.6
I 131	5.7(6)	6.7(7)	8.5
I 132	8.2(6)	1.0(8)	8.2
Cs 137	5.2(5)	6.6(6)	7.9
Ba 140	2.8(4)	1.1(8)	2.5(-2)

6.6 CORE DAMAGE ASSESSMENT BASED ON RADIONUCLIDE ANAYLSIS

The results of the radionuclide analysis are used to estimate the extent of core damage. Table 6-11 shows the percentages of inventory released in this accident scenario. These percentages are compared to Figures 2-1, 2-2, 2-4, 2-6, and 2-9 through 2-12 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 cesiums are released during all stages of core damage. Ba-140 is a characteristic fission product of fuel overtemperature or fuel melt. The calculated release of Ba-140 is used to estimate the extent of fuel overtemperature or fuel melt. From Figures 2-10 and 2-12 the 0.025 percent release of Ba-140 corresponds to approximately 10 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 cladding damage and fuel overtemperature.

The release percentages of the noble gases, iodines, and cesium indicate from Figure 2-9 that approximately 10 percent of the core has experienced overtemperature conditions. Thus some unknown fraction of the fuel has experienced overtemperature. Only if one knows that the accident was a small LOCA can the numerical estimate of 1-50% fuel overtemperature be made.

Comparing the release percentages of the noble gases and iodines to Figures 2-1, 2-2, 2-4 and 2-6; cladding damage greater than 100 percent is indicated. However, as stated previously, it is recognized that in actuality there is an overlap between the regimes of core damage states. Unfortunately, it cannot be estimated from the radionuclide analysis the extent of cladding damage. The release due to overtemperature dominates the release due to cladding damage.

The conclusion drawn from the radionuclide analysis is that the core has experienced some cladding damage (but the extent is not known from solely the radionculide analysis), some fuel overtemperature, possible core crumbling, and no fuel melt (less than 1 percent). In addition, if the accident is known to have been a small LOCA, the fuel overtemperature can be further classified as between 1% and 50% of the core.

6.7 AUXILIARY INDICATORS

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

6.7.1 CONTAINMENT HYDROGEN CONCENTRATIONS

The containment hydrogen monitor indicated a hydrogen concentration in the containment of 10 v/o. From Figure 3-1, 10 v/o hydrogen concentration corresponds to approximately 80 percent zirconium-water reaction. Thus, the hydrogen concentration indicates that there is a high probability that greater than 50 percent of the cladding is damaged (Table 4-1).

6.7.2 CORE EXIT THERMOCOUPLE READINGS AND REACTOR VESSEL WATER LEVEL

The core exit thermocouple readings during this accident reached 1650°F in the center of the core and ranged between 900°F to 1500°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 4-1, these readings indicate a possibility of the core experiencing fuel overtemperature in the center regions and cladding damage in the outer regions. Also, the high hydrogen concentration measured in the containment confirms that the core had uncovered during the accident.

6.7.3 CONTAINMENT RADIATION MONITOR

The containment monitor indicated a gross gamma dose rate of 1.0×10^4 R/hr at 6 hours after reactor shutdown. From Figure 3-3, 10^4 R/hr corresponds to an overtemperature release and a significant gap release which confirms the radionuclide analysis.

6.7.4 ACTIVITY RATIOS

The activity ratios of the released fission products are shown in Table 6-12. The activity ratios shown in Table 6-12 indicate that the release has progressed beyond gap release to release from fuel pellets.

6.8 SUMMARY

The combination of the radionculide analysis and the auxiliary measurements indicated greater than 50 percent cladding damage, 1-50% fuel overtemperature, possible core crumbling, and no fuel melt.

TABLE 6-12

ACTIVITY RATIOS OF RELEASED FISSION PRODUCTS

Isotope	Total Activity, Ci	Activity Ratio*
Kr 87	2.4(6)	2.0(-1)
Xe 133	1.2(7)	1.0
I 131	5.7(6)	1.0
I 132	8.2(6)	1.4

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

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

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