



# DRESDEN NUCLEAR POWER STATION

## UNIT 3

Regulatory

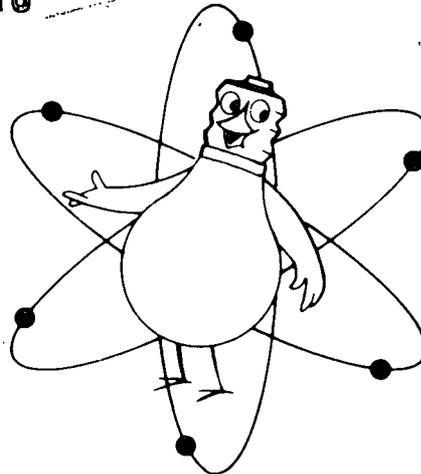
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### Hydrogen Generation in a Boiling Water Reactor

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**HYDROGEN GENERATION IN A BOILING WATER REACTOR**

## TABLE OF CONTENTS

	ABSTRACT	1
I.	SUMMARY OF GENERAL ELECTRIC APPROACH TO HYDROGEN PROBELM . . . . .	1
	A. Recommended Plan . . . . .	1
	B. Doses to Environs . . . . .	2
	C. Conclusions . . . . .	3
II.	INTRODUCTION . . . . .	3
III.	HISTORY . . . . .	4
IV.	HYDROGEN FROM ZIRCALOY-WATER REACTION . . . . .	4
	A. General . . . . .	4
	B. Effect of Metal-Water Reaction on Radiolysis . . . . .	7
V.	HYDROGEN FROM RADIOLYSIS . . . . .	13
	A. General . . . . .	13
	B. Coolant Energy Absorption Over the Post-LOCA Period . . . . .	18
	C. Summary of Experimental Programs to Investigate Radiolysis . . . . .	18
	(1) Operating Reactor Shutdown Tests . . . . .	18
	(2) Simulated BWR Radiolysis Tests at ORNL . . . . .	18
VI.	EFFECTS OF HYDROGEN ON CONTAINMENT . . . . .	20
	A. General . . . . .	20
	B. Assurance of Homogeneous Mixture of Combustible Gases . . . . .	21
	C. Lesser LOCA Within the Primary Containment . . . . .	24
VII.	HAZARDS OF INERTING . . . . .	27
VIII.	POTENTIAL SOLUTIONS . . . . .	30
	A. Venting the Containment . . . . .	30
	B. Recombiners . . . . .	30
IX.	CONCLUSIONS . . . . .	36
X.	RECOMMENDATIONS . . . . .	36
XI.	REFERENCES . . . . .	38
XII.	ACKNOWLEDGEMENT . . . . .	39
	APPENDIX A ENERGY DEPOSITION IN COOLANT FOLLOWING LOCA . . . . .	40
	APPENDIX B CONTAINMENT VENTING ASSUMPTIONS FOLLOWING LOCA . . . . .	43
	APPENDIX C DOSES TO ENVIRONS . . . . .	44

## LIST OF FIGURES

Figure	Title	Page
1	ECCS Improvements . . . . .	5
2	Typical Large BWR Loss-of-Coolant Accidents . . . . .	6
3	Typical Large BWR Recirculation Line Break Temperature vs Time . . . . .	9
4	Typical Large BWR Recirculation Line Break Metal Water Reaction vs Delay Time . . . . .	10
5	Recirculation Line Break - Flooding Only . . . . .	11
6	Typical Large BWR Recirculation Line Break - Metal-Water Reaction vs Time After Accident . . . . .	12
7	ORNL Radiolysis Tests - BWR Containment Simulation Air Atmosphere - Distilled H <sub>2</sub> O . . . . .	14

LIST OF FIGURES (Continued)

Figure	Title	Page
8	Hydrogen Generation Factor - ORNL - Simulated BWR Containment Radiolysis Tests . . . . .	15
9	Typical Large BWR-TID Release . . . . .	16
10	Typical Large BWR-GE Fission Product Release . . . . .	17
11	Flammability of O <sub>2</sub> -H <sub>2</sub> -N <sub>2</sub> Dry Mixtures . . . . .	19
12	Rapid Burning in Drywell . . . . .	22
13	Decay Energy Released and Radiolysis Following a Loss-of-Coolant Accident . . . . .	25
14	Effective Hydrogen Yield, G <sub>H<sub>2</sub></sub> , - TID Release . . . . .	26
15	Radiolysis Following a Loss-of-Coolant Accident . . . . .	42
16	Cloud Gamma Dose - TID Release Normal Leakage 0.635%/Day . . . . .	45
17	Cloud Gamma Dose - TID 14844 Release Containment Venting 10%/Day @78 Hours . . . . .	46
18	Thyroid Inhalation Dose - TID 14844 Release Normal Leakage 0.635%/Day . . . . .	47
19	Thyroid Inhalation Dose - TID 14844 Release Containment Venting 10%/Day @78 Hours . . . . .	48
20	Cloud Gamma Dose - GE Fission Product Release Normal Leakage 0.5%/Day . . . . .	49
21	Cloud Gamma Dose - GE Fission Product Release Containment Venting 8%/Day @78 Hours . . . . .	50
22	Thyroid Inhalation Dose GE Fission Product Release Normal Leakage 0.5%/Day . . . . .	51
23	Cloud Gamma Dose - GE Fission Product Release Containment Venting 8%/Day @78 Hours . . . . .	52

LIST OF TABLES

Table	Title	Page
1	Design Basis Accident Radiological Doses-Loss-of-Coolant Accident (TID Sources & Transport) . . . . .	2
1A	Design Basis Accident Radiological Doses-Loss-of-Coolant Accident (ECCS Sources & Transport) . . . . .	3
2	Heatup Test Results . . . . .	8
3	Effect of Initial Hydrogen on Time to Reach Specific Hydrogen Concentration Due to Radiolysis . . . . .	8
4	Initial Hydrogen . . . . .	13
5	Venting-Bleed Through Standby Gas Treatment System - Makeup Doses to Environs . . . . .	31
6	Fission Products (GE Source, AEC Source) . . . . .	32
7	Nonventing Containment Treatment Systems . . . . .	32
8	Thermal Recombination-Nonventing . . . . .	33
9	Catalytic Recombination-Nonventing . . . . .	34
10	Solid-Gas Recombination-Nonventing . . . . .	35
11	Hydrogen Problem . . . . .	37
B-1	Containment Venting Assumptions Following Design Basis LOCA . . . . .	43

## ABSTRACT

This report evaluates the potential hydrogen generation and its subsequent accumulation in the primary containment of a large BWR following a postulated loss-of-coolant accident. The report includes the application of experimental test data to a typical 3440MWt nuclear power plant and a generic evaluation of the effects of the pertinent parameters such as metal-water reaction, break area, and radiolytic decomposition rate. The conclusions are as follows:

1. Hydrogen from metal-water reactions is insignificant but radiolytic decomposition could eventually result in flammable mixtures if no corrective measures are taken.
2. Based on above inerting is not a solution because radiolysis also results in oxygen evolution and in addition, inerting imposes operational and safety problems and is therefore unwarranted.
3. Primary containment purging and venting through the Standby Gas Treatment System to the stack is recommended to remove any hydrogen that may form before reaching a combustible mixture as determined by appropriate monitoring.
4. The resulting doses to the environs due to controlled venting are well within the 10CFR100 guideline limits even using the ultraconservative TID-14844 fission product releases, and only an insignificant fraction thereof if the sources consistent with the functioning of any *one* of the ECCS are used.

### I. SUMMARY OF GENERAL ELECTRIC APPROACH TO HYDROGEN PROBLEM

#### A. Recommended Plan

The plan recommended by the General Electric Company to both control and preclude significant accumulations of combustible gases in the containment following a postulated loss-of-coolant accident (LOCA) is as follows:

1. The hydrogen concentration in both the drywell and suppression chamber free volumes would be monitored at various locations within the containment.
2. Should the hydrogen concentration show signs of approaching the lower flammability limits, the containment atmosphere would be purged with air and vented in a controlled manner through the standby gas treatment system to the stack. This is a high efficiency system for filtering fission products and is ideally suited for this purpose.
3. The containment fission-product activity would be determined at the location where the fission gases are vented from the containment following a LOCA. However, even if a TID-14844 source term is assumed, calculations show that offsite doses are only a fraction of 10CFR100 guideline limits, and an insignificant fraction thereof if fission-product sources consistent with the ECCS are used instead. (See Section V for comparative calculations)

Because many of the boundary conditions surrounding the accumulation of hydrogen in the containment are not based on purely technical consistent considerations but on licensing and other philosophical grounds, a final measure to control and preclude significant accumulation of hydrogen cannot be arrived at until these criteria are more clearly stated by regulatory bodies. The above solution is based on the assessment of the overall hydrogen problem strictly on technical grounds. In order to bring out all the current pertinent facts on hydrogen accumulation which may be of some assistance in arriving at such criteria, an overall summary starting with Section II is also included.

The calculated radiological doses to the environs using the TID-14844 source terms and the AEC fission product transport model are conservatively evaluated for a *typical* BWR in Section II - VIII. *More specific* calculations were performed for a 3440MWt plant and are presented in Section I, where a more detailed investigation was made of both the energy deposition in the water for radiolysis and the fission product transport model showing additional margins.

If, before venting, the halogens and solids are assumed to be trapped in the suppression pool, the corresponding total doses to the environs from both normal leakage and venting would be decreased due to the reduction of the airborne fission product activity. One method to ensure fission products are transported to the

suppression pool is to design for venting from the suppression chamber volume with the required air purge makeup being directed into the drywell volume. Thus, the inherent large water volume feature of the pressure suppression containment system can be used to advantage to reduce the fission product activity released from the containment.

**B. Doses to Environs**

The assumptions used to determine the doses to the environs for venting from the suppression chamber are included in Appendix B.

The resultant calculated doses to the environs at a typical exclusion area on the plume centerline and at a typical low population zone area are included in the following table for both the normal leakage (0.635%/day) and the 21 cfm venting rate (approximately 10%/day) starting at 78 hours after the accident to limit the hydrogen to less than 4% by volume in the primary containment. The 2-hour dose at the exclusion area due to leakage would only be 1/3 Rem and 15 Rem for the whole body and thyroid doses, respectively (see Table 1). The total 30-day dose to the environs due to leakage and venting at the low population zone would only be 0.8 Rem and 76 Rem for the whole body and thyroid, respectively (see Table 1). Note that the calculated thyroid inhalation dose contribution due to venting is about the same magnitude as that resulting from the normal leakage. The doses in terms of Rem/MWt for other distances are plotted on Figures 16-19 in Appendix C. It should be also noted that these doses would be significantly less than 100% of 10CFR100 guideline limits for all distances (see Figures 16-19).

**Table 1  
TID SOURCES – TID TRANSPORT MODEL  
DESIGN BASIS ACCIDENT RADIOLOGICAL DOSES  
LOSS-OF-COOLANT ACCIDENT**

	Radiological Exposure (REM)			
	Cloud Gamma		Thyroid Inhalation Dose	
	2-hour (1200m)	30-day (4827m)	2-hour (1200m)	30-day (4827m)
1. Normal Leakage (0.635%/day)	0.31	0.265	15	36
2. Containment Venting	*	0.52**	*	40**
3. TOTAL DOSE	0.31	0.785	15	76
4. Percent of 10CFR100	1.2%	3.1%	5%	25%

\* Venting starts in 78 hours to limit H<sub>2</sub> to < 4% by volume.

\*\* Exposures reflect venting from suppression chambers.

If instead of a TID fission-product source term, the sources associated with any one of several of the ECCS functioning are used along with a reasonably conservative fission product transport model<sup>(1)</sup> the doses are much lower as shown in Table 1-A. The doses in terms of Rem/MWt for other distances are given on Figures 20-23 in Appendix C. It should be noted that the halogen doses can be reduced to where they are not limiting by additional charcoal filtering.

**Table 1-A**  
**GE SOURCE MODEL<sup>(1)</sup> – TRANSPORT MODEL<sup>(1)</sup>**  
**DESIGN BASIS ACCIDENT RADIOLOGICAL DOSES**  
**LOSS-OF-COOLANT ACCIDENT**

	Radiological Exposure (REM)			
	Cloud Gamma*		Thyroid Inhalation Dose	
	2-hour (1200m)	30-day (4827m)	2-hour (1200m)	30-day (4827m)
1. Normal Leakage 0.635% day	$4 \times 10^{-5}$	$9 \times 10^{-4}$	$5.3 \times 10^{-6}$	$1.38 \times 10^{-4}$
2. Containment Venting	-----	$2.7 \times 10^{-3}$	-----	$1.72 \times 10^{-3}$
3. TOTAL DOSE	$4 \times 10^{-5}$	$3.6 \times 10^{-3}$	$5.3 \times 10^{-6}$	$1.86 \times 10^{-3}$
4. Percent of 10CFR100	$1.6 \times 10^{-4}\%$	$1.4 \times 10^{-2}\%$	$1.8 \times 10^{-6}\%$	$6.2 \times 10^{-4}\%$

\* Whole Body Dose

**C. Conclusions**

Therefore, it is concluded that even using both the TID fission-product source terms and the AEC fission-product transport model in conjunction with the assumptions listed in Appendix B, the primary containment can be vented to prevent the accumulation of a flammable mixture of hydrogen (4%) and still lie well within the 10CFR100 guidelines.

It should be noted that use of a GE source term which is consistent with having any one of the ECCS functioning and a transport model reflecting measured values for the key parameters<sup>(1)</sup> would result in doses to the thyroid and whole body of an insignificant fraction of the AEC guideline limits.

**II. INTRODUCTION**

The purpose of this section of the report is to discuss the problems associated with hydrogen generation following a loss-of-coolant accident (LOCA) in a boiling water reactor. The analysis and discussion will be generic

and will therefore apply to the current product line boiling water reactor. The current state of the art as well as the past history leading up to the present situation which is bounded by arbitrary assumptions is also summarized. Finally, the potential solutions as a function of imposed boundary conditions will also be described.

### III. HISTORY

The question of hydrogen generation and the resultant potential for hydrogen explosions has been of concern since the earliest days of power reactor design. For the last decade generation of power reactors, however, it first became a serious issue in 1964-65 during the licensing of the zirconium replacement core for the Humboldt Bay reactor. The chief concern was the hydrogen produced from zirconium-water reaction following a postulated core melt. Using analytical models which assumed that molten Zircaloy would run down the rods and quench in water at the bottom of the vessel, about 1/4 of the zirconium in the core could be shown to have oxidized. On this basis, the containment was reinforced and inerted with nitrogen. However, this was an inconsistent solution.

This inconsistency was partially resolved during the licensing of Dresden Nuclear Power Station, Units 2 and 3. After examining potential solutions, it was decided to greatly augment the ECCS to make the potential for a meltdown sufficiently improbable. Figure 1 graphically demonstrates the augmentation up to the present designs. Thus, the current boiling water reactor ECC systems were designed to prevent cladding melt (using grossly conservative models). It was later shown based on analysis and experimental data that the ECC systems limited the peak cladding temperatures to only 2000°F while the temperature limit to prevent fragmentation was shown to be 2700°F. Further, two independent systems were incorporated, each employing a different cooling phenomenon, i.e., spraying and flooding. In addition, complete break spectrum protection up to the rupture of the largest line was ensured even with no external power sources. Thus core melting was no longer the phenomenological basis for ECC system design. However, an arbitrary but unstated amount of hydrogen was still an AEC criterion for containment design.

Finally, in 1968 the hydrogen resulting from radiolysis of the water was demonstrated to be a second potential source of hydrogen. Recent tests have indicated that this source cannot be ruled out and they also show that a finite equilibrium in about 1000 hours after an LOCA does exist, although it is above the lower flammability limit. However, the average rate of evolution is about 1/4 of the previously accepted value ( $G_{H_2}=0.45$ ). The next two sections will discuss these two sources, i.e., metal-water and radiolysis, and their effects on the containment.

### IV. HYDROGEN FROM Zr-H<sub>2</sub>O REACTIONS

#### A. General

Following a LOCA for breaks in excess of about a 6-inch pipe, coupled with loss of normal a-c power, the calculated peak fuel rod cladding temperatures would increase to a maximum of about 2000°F for the largest break. Under such conditions over 90% of the cladding is considerably cooler than 2000°F and the transient only lasts for a maximum of 5 minutes (only 2 minutes with the flooders) before the entire core is quenched to ambient conditions. Since the amount of metal-water reaction which occurs is an exponential function of temperature and time, only a very small amount (~0.06%) of the Zircaloy is oxidized. The amount of hydrogen resulting from the oxidation of 0.06% of the Zircaloy would result in a hydrogen concentration of under 0.5% by volume in the containment; well below the commonly accepted lower flammability limit of 4% in air. The amount of zirconium oxidized is shown in Figure 2 as a function of break size. Also shown is the effect of postulated additional degeneration of the ECCS in terms of the number of pumps which are assumed to operate. Another concern is the effect of fuel rod failure on the metal water reaction. "A recent experiment(2) conducted with a full-size simulated BWR fuel bundle demonstrated that fuel rod cladding perforations have no significant effect on the ability to spray cool the bundle and therefore very little effect on cladding oxidation."(3)

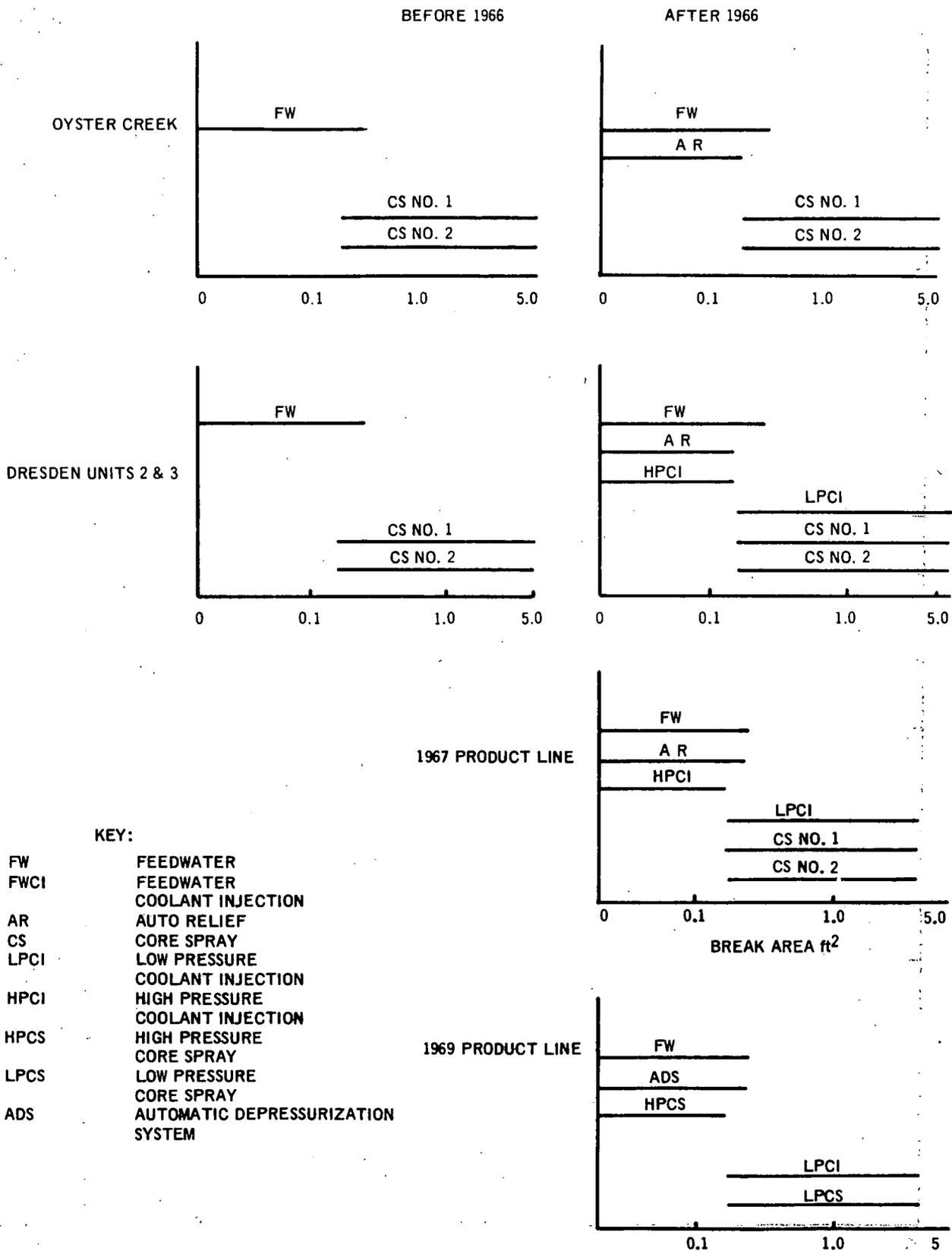


Figure 1. ECCS IMPROVEMENTS

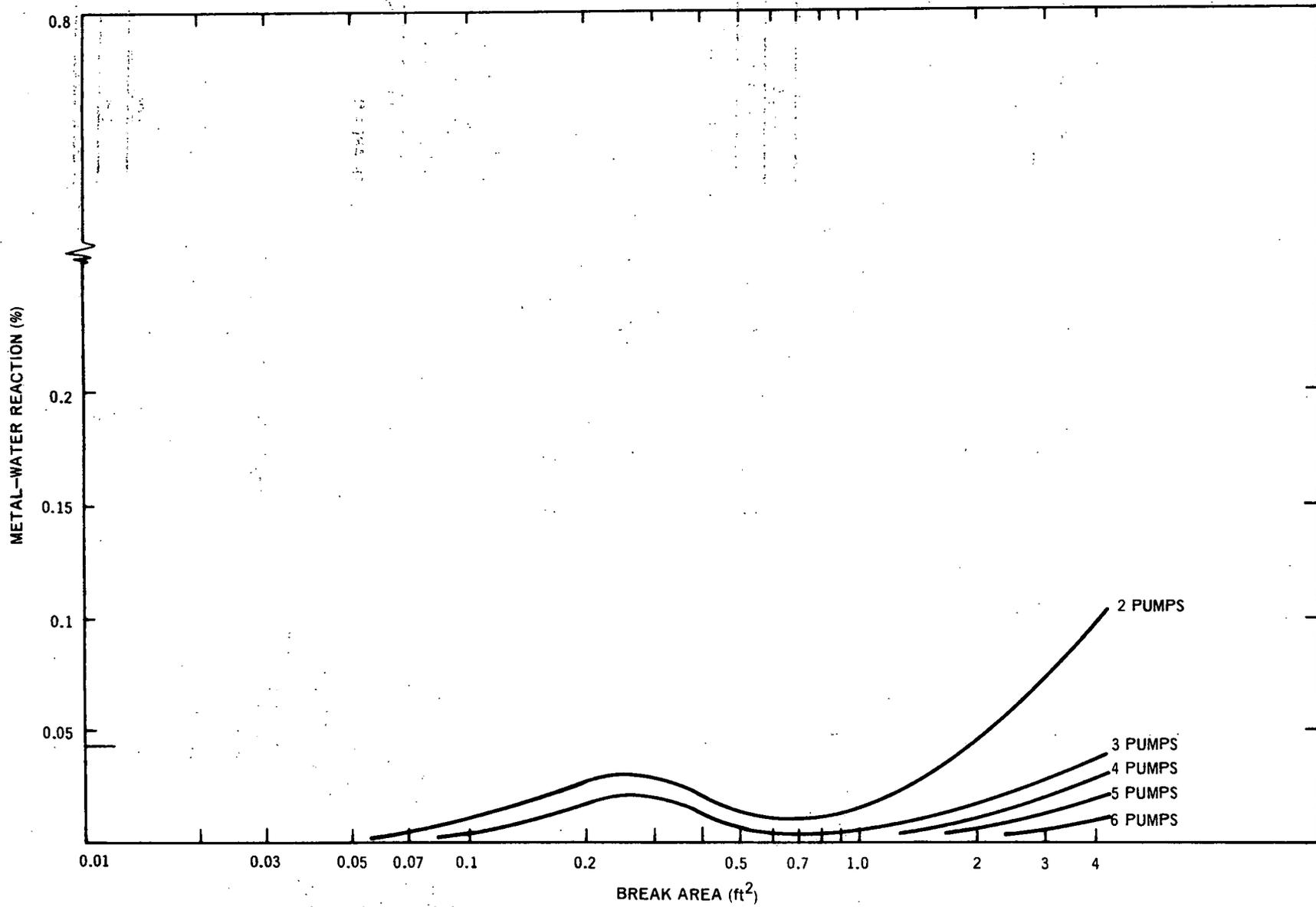


Figure 2. TYPICAL LARGE BWR LOSS-OF-COOLANT ACCIDENTS

For the large design break, Figure 3 shows typical peak cladding temperatures expected depending on the number of pumps assumed to operate. When it is recognized that the rates of reaction with Zircaloy are negligible below about 2100°F the reason for the small amounts of reaction expected becomes clear. Also indicated is the effect of one of the conservative assumptions, steam availability, which limits the reaction through both the cooling effect and the limited availability of steam for reaction with the Zircaloy.

Figure 4 shows the effect of *time degeneration* of the ECCS, in this case floodler activation time, on the amount of Zircaloy reacted. (The independent core sprays are assumed not to function and only 3 of 4 flooding pumps are assumed to operate.) For reference purposes a 0.1% M-W reaction is equivalent to about 3 # moles and would result in a hydrogen volumetric concentration of approximately 0.4% in the primary containment.

Figure 5 shows when in time the core would be *flooded* assuming various numbers of pumps to be operable along with the integrated metal-water reaction which would occur. Note the wide margins to the flammability limit. The "initial" sources equivalent to about 0.2% M-W is the hydrogen from UO<sub>2</sub> - H<sub>2</sub>O reactions and the original hydrogen within the primary system and containment.

Figure 6 shows the effect on metal-water of initial assumptions with respect to *blowdown heat transfer and subsequent number of ECCS pumps* postulated to operate. Again it can be seen that a large margin exists with respect to the degeneracy required to achieve large amounts of metal-water reaction.

Another conservatism worthy of note is in the relationship assumed for the rate of the Zircaloy-steam reaction. Tests<sup>(4)</sup> at General Electric conducted on single rods indicate that the "Baker Relation" used in the analytical model is conservative. The test rods were 6 feet long and heated in a steam atmosphere in a natural convection loop. The rods were segmented and analyzed for total integrated oxidation after the tests and were found to have oxidized only a fraction of that predicted using this relationship (see Table 2). Therefore, it is believed that the estimates discussed above the H<sub>2</sub> produced for a given set of assumptions are definitely conservative.

It is concluded<sup>(5)</sup> that *inerting should not be a requirement* for any BWR and especially for the Dresden 2 and subsequent class of reactors. The depth of ECCS protection and the small quantities of hydrogen produced even in event of degeneration beyond the design basis make inerting unwarranted even without consideration of the hazards of inerting, discussed in Section VII. Further, *the pressure suppression system can tolerate the very rapid burning of large quantities of hydrogen well above the flammability limit\** without violating the integrity of the containment, because the ability to accommodate large energy release rates is inherent in its design.

## B. Effect of Metal-Water Reaction on Radiolysis

The effect of any initial concentration of hydrogen in the containment due to metal-water reaction is given in Table 3. Metal-water reactions up to 0.75% are included which is more than 10 times the maximum value calculated. These calculations of times to reach flammability were made using the most representative of the ORNL BWR radiolysis test data (see Section IV) and both the GE fission product release model and the AEC "TID-14844 release" model. For conservatism, all of the TID release fission products (100% noble gases, 50% halogens, 1% solids) were used to cause radiolytic decomposition of the suppression pool water. (See Appendix A for discussion) Metal-water reactions in excess of approximately 1% would result in a flammable mixture of hydrogen of 4% by volume. The ORNL BWR simulated loop data indicates equilibrium hydrogen concentrations could exceed 4% by volume for radiolysis. Experimental verification of the effect of the initial hydrogen concentration on the equilibrium hydrogen concentration from radiolysis has been obtained by Zittel at ORNL. The results show that any initial hydrogen present does not affect the final equilibrium value achieved by radiolysis.

The effect of pre-accident containment pressure and temperature on the time to reach flammability is small. For example, an increase in containment pressure would provide additional moles of noncondensable gases, thereby causing an increase in the time to reach a flammable concentration. However, a 1 psi increase in the pre-accident containment pressure would only increase the noncondensibles by approximately 7% which would in turn result in an increase in the time to reach flammability of only an hour or two out of 50 hours. An increase in the drywell temperature from the normal value of approximately 135°F 35% relative humidity to, for instance, 180°F even at 100% relative humidity would only decrease the initial moles of noncondensable gases in the containment system by approximately 25%. This reduction in moles of noncondensable gases would in turn decrease the time to reach the 4% flammability limit by possibly 4-5 hours or a reduction in time of less than 10%.

\* See Section VI

Table 2  
HEATUP TEST RESULTS

Run No.	Max(a) Temp °F	Predicted Reaction (Baker's Equation) %	Measured÷Predicted(a) (Baker). %
1	3380	11.2	64.1
2	3300	13.4	40.3
3	3000	10.5	42
4	2460	6.53	61.3/76.6
5	2450	11.43	62
6	2870	18.27	27.9
7	3030	20.95	40.6
8	2440	11.95	54.5/63.5
9	2960	39.65	50.5/75.7
10	2870	35.5	30.7
11	3550	42.42	33

(a) Time at temperature different for each test. This effect included in prediction.

Table 3  
EFFECT OF INITIAL HYDROGEN ON TIME TO  
REACH SPECIFIC HYDROGEN CONCENTRATION  
DUE TO RADIOLYSIS

Core Rated Power = 2550 mwt }  
 Core Decay Energy to Coolant = 11% } (See Appendix A)  
 Fission Product Decay Energy to Pool = 100%

Infinite Irradiation

Initial H <sub>2</sub>	Fission Product Source	Time in Hours to Reach 6% H <sub>2</sub>		Time in Hours to Reach 4% H <sub>2</sub>	
		95°C†	125°C	95°C	125°C
0	GE	1200	***	300	900
0	AEC	400	***	150	350
7 # moles	GE	700	(4000)(a)	180	360
7 # moles	AEC	300	(1000)	120	240
25 # moles*	AEC	105	(190)	17	26
37 # moles**	AEC	40	(50)	-	-
25 # moles	GE	200	(330)	36	50

\* This means the % metal-water reaction is 7 times that calculated or 0.5% of entire core.

\*\* M-W reaction 11 times that calculated or about 0.75% of entire core.

\*\*\* H<sub>2</sub> equilibrium is less than 6%.

† Water Temperature.

(a) Although data indicate initial H<sub>2</sub> does not increase %H<sub>2</sub> equilibrium these calculations were conservatively made adding both contributions.

(These times were based on the hydrogen generation rate and equilibrium data measured at ORNL in the simulated BWR configuration radiolysis tests.)

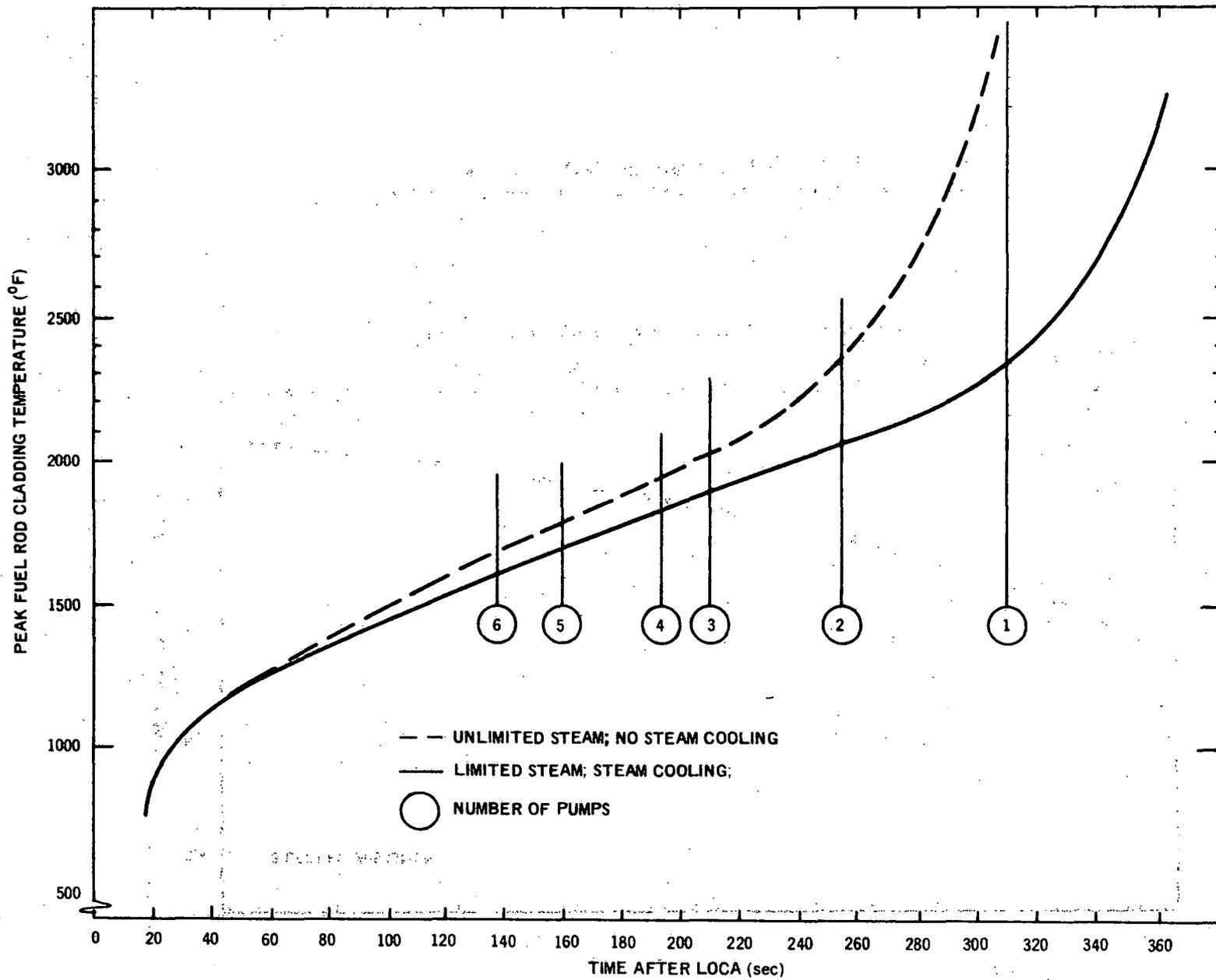


Figure 3. TYPICAL LARGE BWR RECIRCULATION LINE BREAK—TEMPERATURE VERSUS TIME

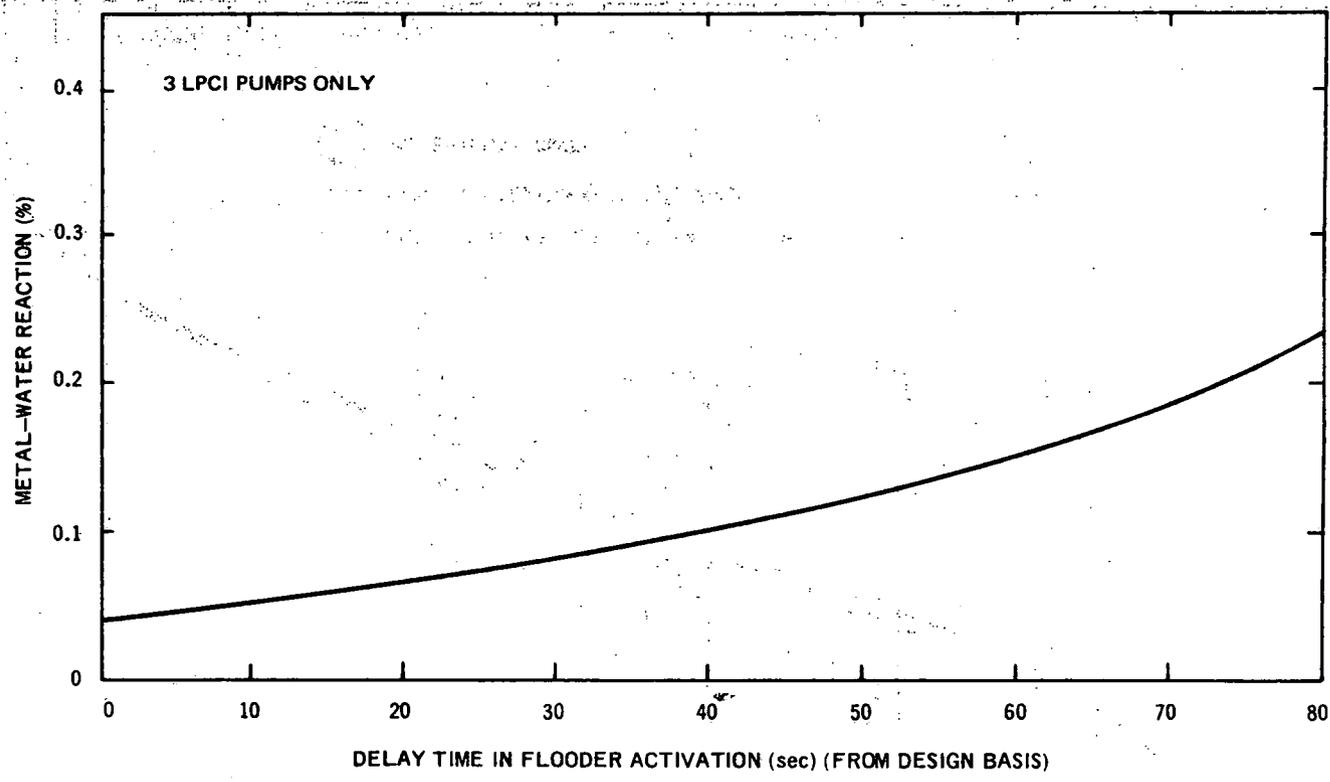


Figure 4. TYPICAL LARGE BWR RECIRCULATION LINE BREAK—METAL-WATER REACTION VERSUS DELAY TIME

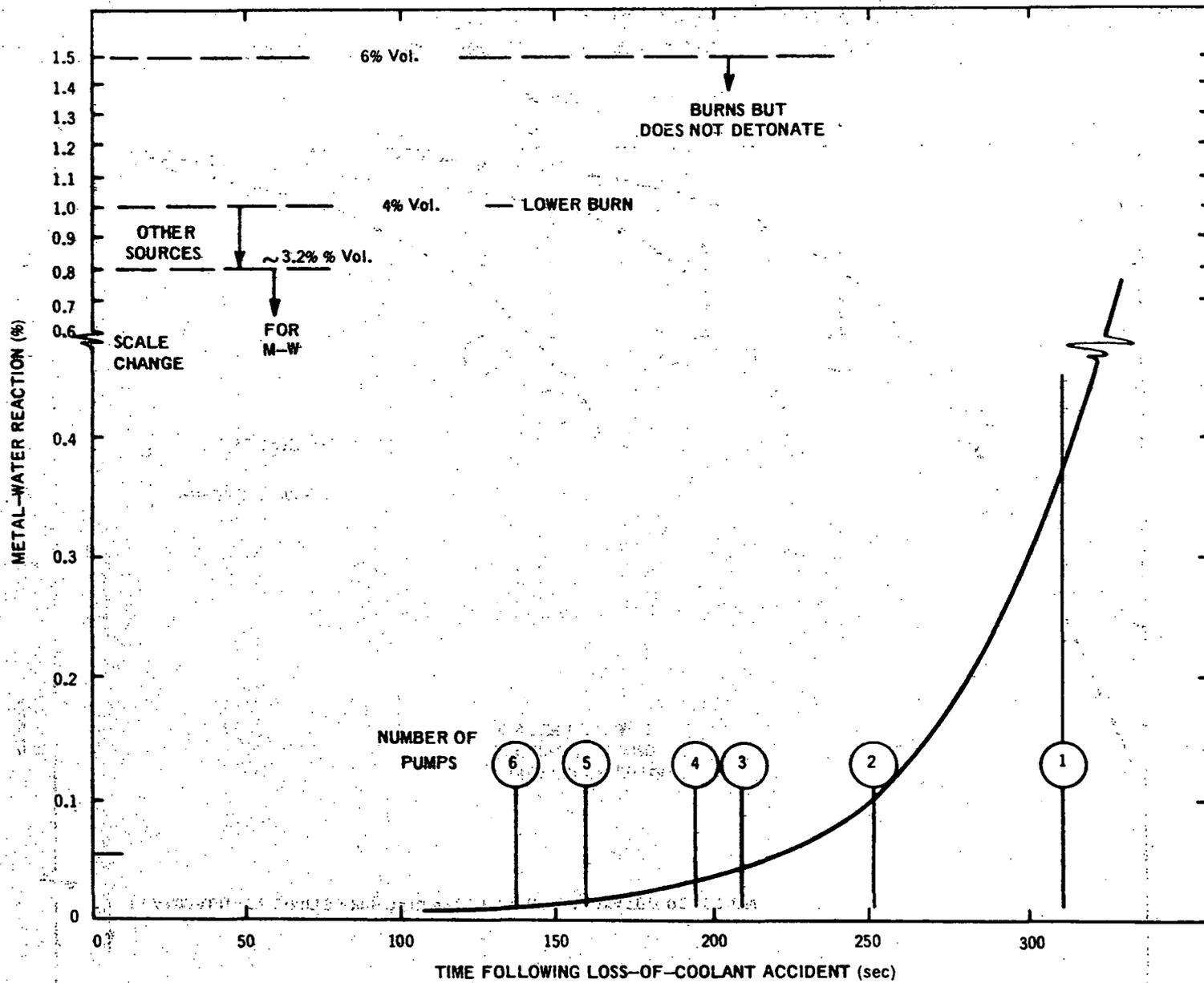


Figure 5. RECIRCULATION LINE BREAK—FLOODING ONLY

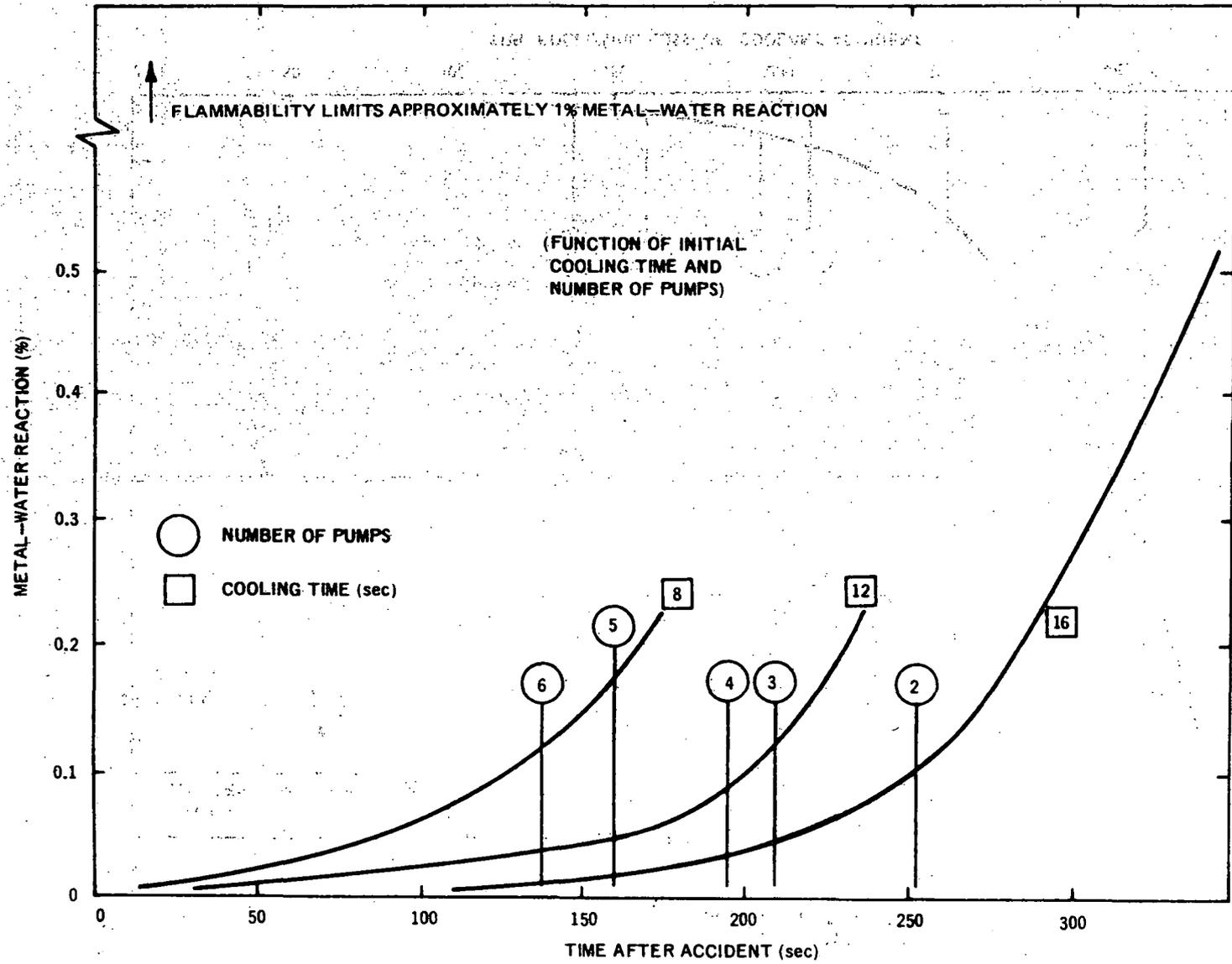


Figure 6. TYPICAL LARGE BWR RECIRCULATION LINE BREAK—METAL-WATER REACTION VERSUS TIME AFTER ACCIDENT

## V. HYDROGEN FROM RADIOLYSIS

### A. General

Following a LOCA the decay activity breaks down the water molecules and through a complex series of reactions results in liberation of hydrogen and oxygen. At some given hydrogen overpressure, the rate of recombination in the radiation field just equals the rate of production and the concentration comes to equilibrium. The equilibrium value must be determined experimentally from capsule or loop tests and ranges from H<sub>2</sub> overpressure of a few atmospheres (for solutions with chemical additives) to a few psi for the BWR solutions recently tested in the ORNL loop. Most of the ORNL data<sup>(6, 7)</sup> when converted to BWR accident conditions would indicate an equilibrium value as low as 3.6% by volume of hydrogen, for the case where the core is flooded and only boil-off is maintained, to as high as 14% with maximum cooling depending on temperature and residence time as shown in Figure 7.

The net H<sub>2</sub> generation rate determines how soon the flammability limit is reached. The commonly accepted value of 0.45 molecules of hydrogen released per 100 ev of absorbed energy is higher by at least a factor of 5 than that actually measured at the Humboldt Bay reactor. The ORNL loop tests from which G<sub>H<sub>2</sub></sub> values were extracted indicate that temperature and residence time in the radiation field are the key variables. As shown in Figure 8 the preliminary ORNL results also are lower - G<sub>H<sub>2</sub></sub> is as measured with the Humboldt Bay tests. Translated to reactor conditions following a LOCA the conservative assumption was made that 11%\* of the total decay activity still in the core is absorbed by the water. A detailed evaluation of the decay gamma energy spectrum and the material cross section resulted in only 5% of the remaining core decay energy being absorbed by the reactor core coolant. Assuming TID activity escape from the fuel\*\* an additional maximum value of 15% of all the decay energy could be available for radiolysis provided all of the released fission products are absorbed in the suppression pool. The rate of hydrogen buildup in a typical large BWR primary containment for these conservative assumptions (i.e., 11% core decay power to reactor coolant and ~ 15% core decay power to suppression pool) is illustrated in Figure 9. (See Appendix A for discussion.) Using assumptions with respect to fission product release, consistent with ECCS operation,\*\*\* i.e., only 11% of the core decay energy is absorbed by the water. Figure 10 shows the integrated amount of hydrogen generation due to radiolysis. Thus, with these assumptions it would take over 1200 hours to reach hydrogen concentrations of 6% by volume (see Table 2) and from 300 to 900 hours to reach 4% by volume depending on the overall average temperature over that period of time. With TID assumptions\* it would take over 400 hours to reach 6%, and 150 to 350 hours to reach 4%. In all cases negligible initial hydrogen was assumed. A summary of the times to reach flammability is shown in Table 3.

The "initial" quantities of H<sub>2</sub> present for the largest break are summarized in Table 4.

Table 4  
INITIAL HYDROGEN

<0.1% M-W Reaction . . . . .	< 3 # moles
UO <sub>2</sub> - H <sub>2</sub> O Reaction . . . . .	1 # mole
H <sub>2</sub> Primary System Leakage . . . . .	2 # moles
H <sub>2</sub> in Primary System . . . . .	1 # mole
Total . . . . .	7 # moles = 1% Vol.

\* 11% conservative, 5% more realistic (see Appendix A).

\*\* 100% of all noble gases, 50% of halogens, and 1% solids released.

\*\*\* 25% rod perforations, 1.8% noble gases, 0.32% halogens, and 1% solids released from perforated fuel.

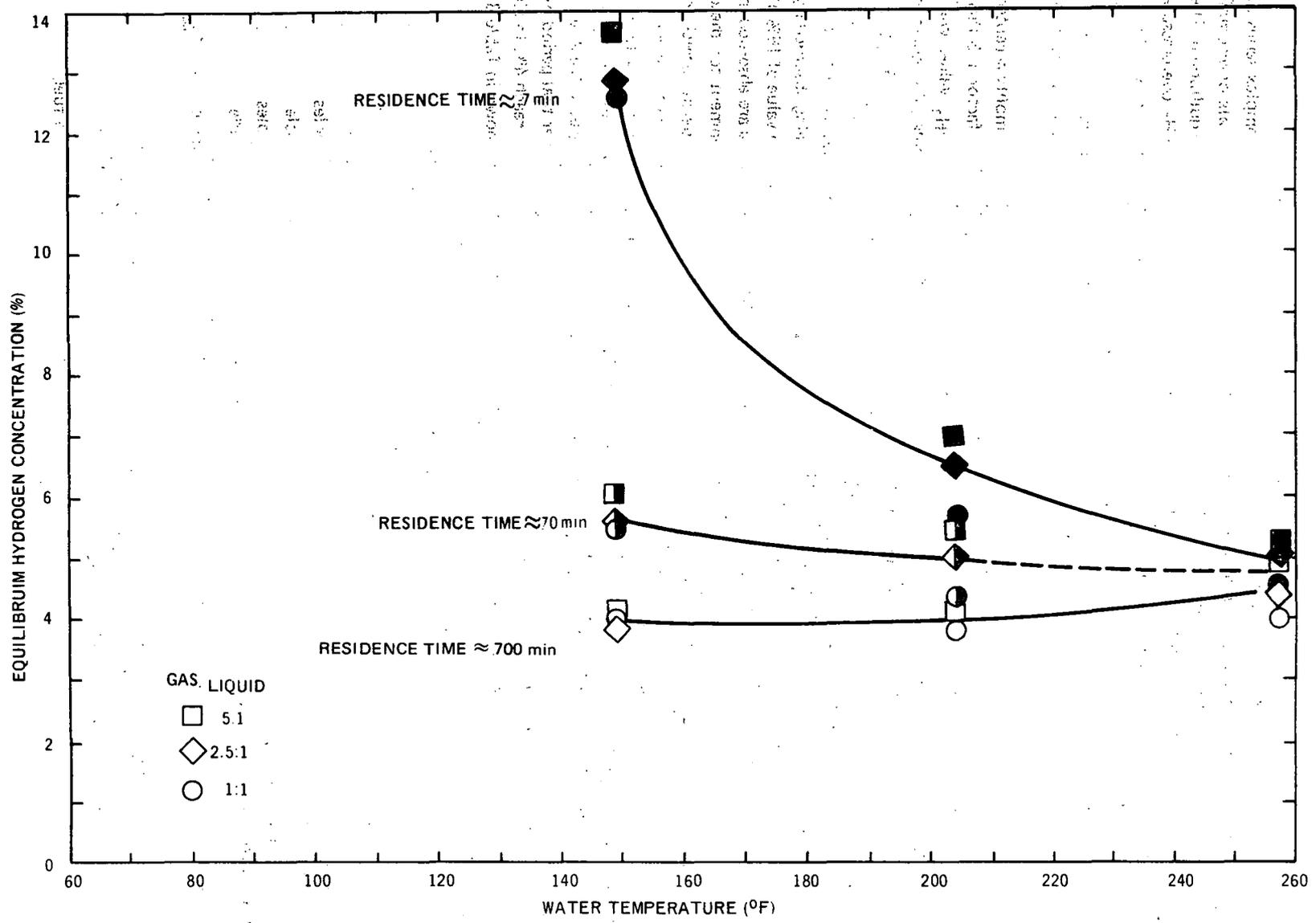


Figure 7. ORNL RADIOLYSIS TESTS<sup>6,7</sup> -BWR CONTAINMENT SIMULATION. AIR ATMOSPHERE-DISTILLED WATER

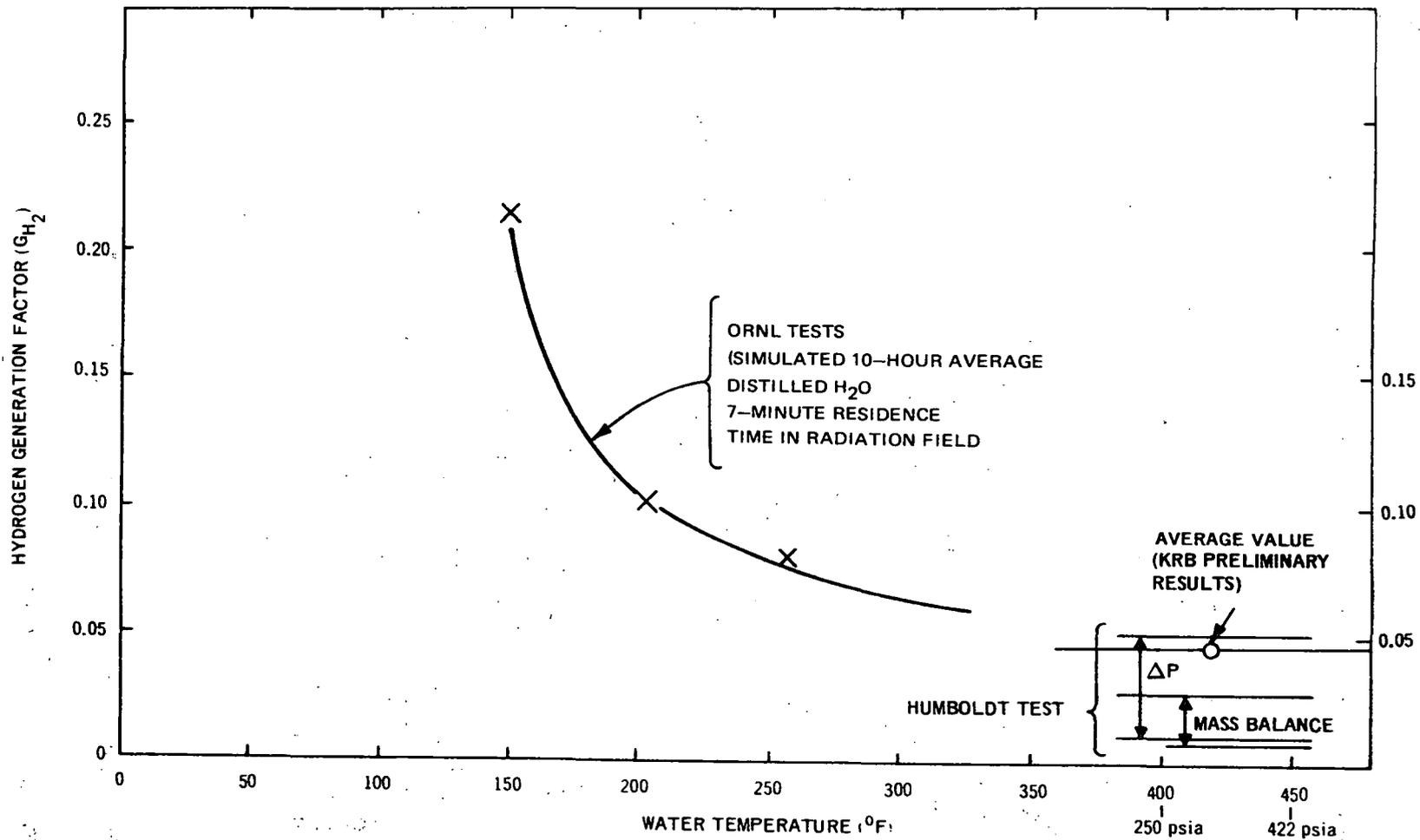


Figure 8. HYDROGEN GENERATION FACTOR—ORNL-SIMULATED BWR CONTAINMENT RADIOLYSIS TESTS

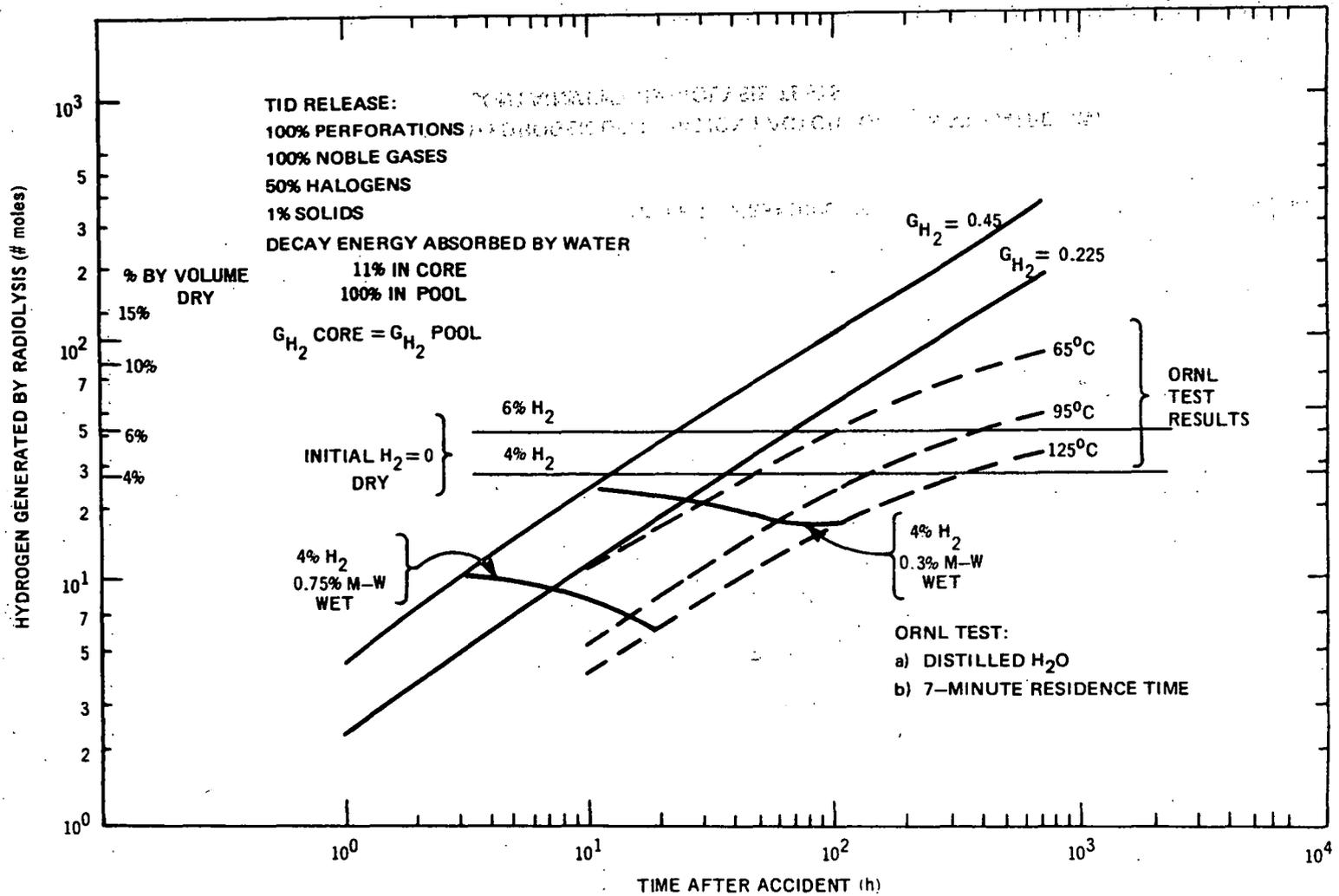


Figure 9. TYPICAL LARGE BWR-TID RELEASE

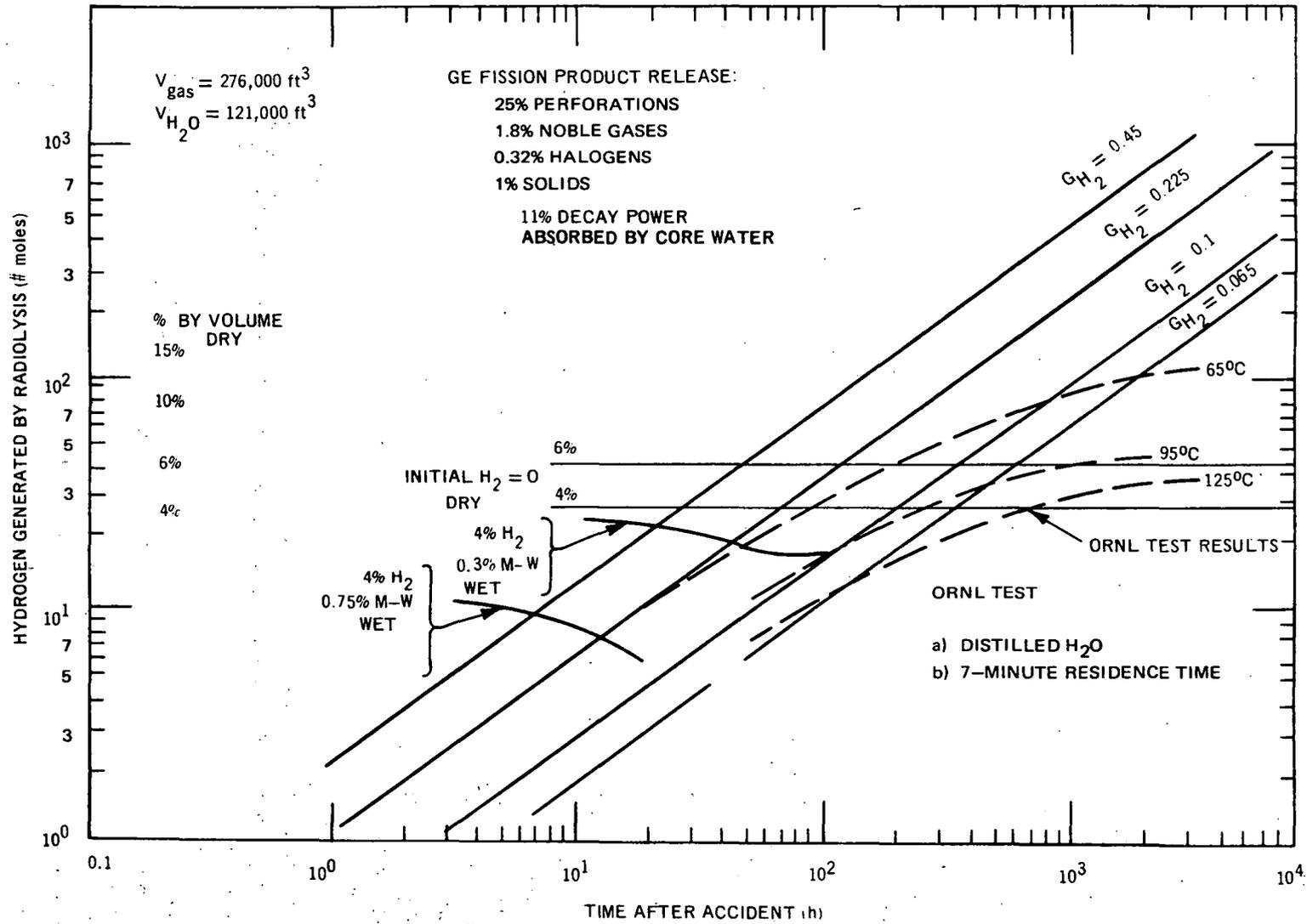


Figure 10. TYPICAL LARGE BWR-GE FISSION PRODUCT RELEASE

The metal-water reaction shown in Table 4 is what is actually calculated with any one of the two ECCS systems operating. The  $\text{UO}_2 - \text{H}_2\text{O}$  reaction was conservatively based on 25% of the fuel rods being perforated and oxidizing one-third of the pellets in a 1-foot section of each rod. The "hydrogen primary system leakage" is the hydrogen which leaks to the drywell with primary system fluid leakage. A 5 gpm continuous leak for 3 months was assumed. The total hydrogen within the primary system is less than 1 pound-mole.

Thus, it is seen from Table 3 that with a conservative estimate of all the hydrogen sources, a significant period of time exists after an LOCA before the flammability limits are reached, ranging from 5 days to as long as 6 weeks. This is sufficient time to eliminate the hydrogen by various means. Furthermore, it is seen that even significant degenerations of the ECCS can be tolerated with respect to  $\text{H}_2$  buildup even before such accumulations occur (see Figures 4 and 5).

Finally, it can be seen from Figure 11<sup>(8)</sup> that *inerting is not a solution since with radiolysis the additional hydrogen generated is accompanied by oxygen and shifts the concentration along the dotted line such that by the time a 5% by volume of hydrogen exists, the oxygen content has increased to make the mixture flammable*. This would take between 200 and 700 hours as interpolated from Table 2 and hence does not differ significantly from the noninerted case.

### B. Coolant Energy Absorption Over the Post-LOCA Period

The energy transferred to the reactor coolant is a function of both the core decay power and the mass density of coolant present within the core. For conservatism the reactor was considered as having operated at rated power for an infinite time thereby resulting in the maximum core fission product decay energy. A realistic irradiation time would be 1000 days. For a unit cell within the reactor core approximately 11% of the total mass density (i.e., fuel, cladding, and water) consists of the coolant. Therefore, the conservative assumption was made that 11% of the remaining core total decay energy (gamma and beta) was deposited in the coolant with the remaining energy being deposited in the fuel and fuel rod cladding. Actual deposition in the coolant would be approximately 5% of the total decay energy. The beta decay energy, approximately 40% of the remaining reactor core total decay energy, would remain mostly within the fuel and cladding due to the relative short mean free path of the beta fission products (i.e., energy spectrum).

Consideration of the long term post-LOCA time period could result in a reduction in the total fraction of core decay power being available for radiolysis in the core region. This reduction in radiolysis would be a function of the total amount of water within the core region. The water swell within the reactor vessel does provide continuous and adequate core cooling even with the collapsed water level (i.e., two separate and distinct phases—steam and liquid) below the top of the active fuel. For this condition of void distribution within the core region the fractional amount of decay energy deposited in the reactor coolant is reduced due to the decrease in mass density of water.

Therefore it is conservative to assume that 11% of the total decay energy is deposited in the reactor coolant especially when the beta energy spectrum of the released fission products and any coolant swelling in the reactor core is considered.

### C. Summary of Experimental Programs to Investigate Radiolysis

The following experimental programs have been undertaken by the General Electric Company to better resolve the concerns relative to the  $\text{H}_2$  production by coolant radiolysis.

#### (1) Operating Reactor Shutdown Tests

A test was run at the PG&E Humboldt Bay Nuclear Power Plant Unit No. 3 during the scheduled refueling shutdown on November 4, 1968, to determine the radiolytic hydrogen production rate. Test data were taken during the first 2-1/2 hours of the shutdown when the reactor pressure was being reduced. The resultant hydrogen generation rate was calculated from the steam flow rate and the corresponding hydrogen concentration in the steam. Consideration of the experimental errors in the steam flow measurement (flow nozzle and reactor water mass balance) and the core decay power resulted in a hydrogen yield between 0.008\* and 0.05\* molecules

\* Based on 11% core energy to water.

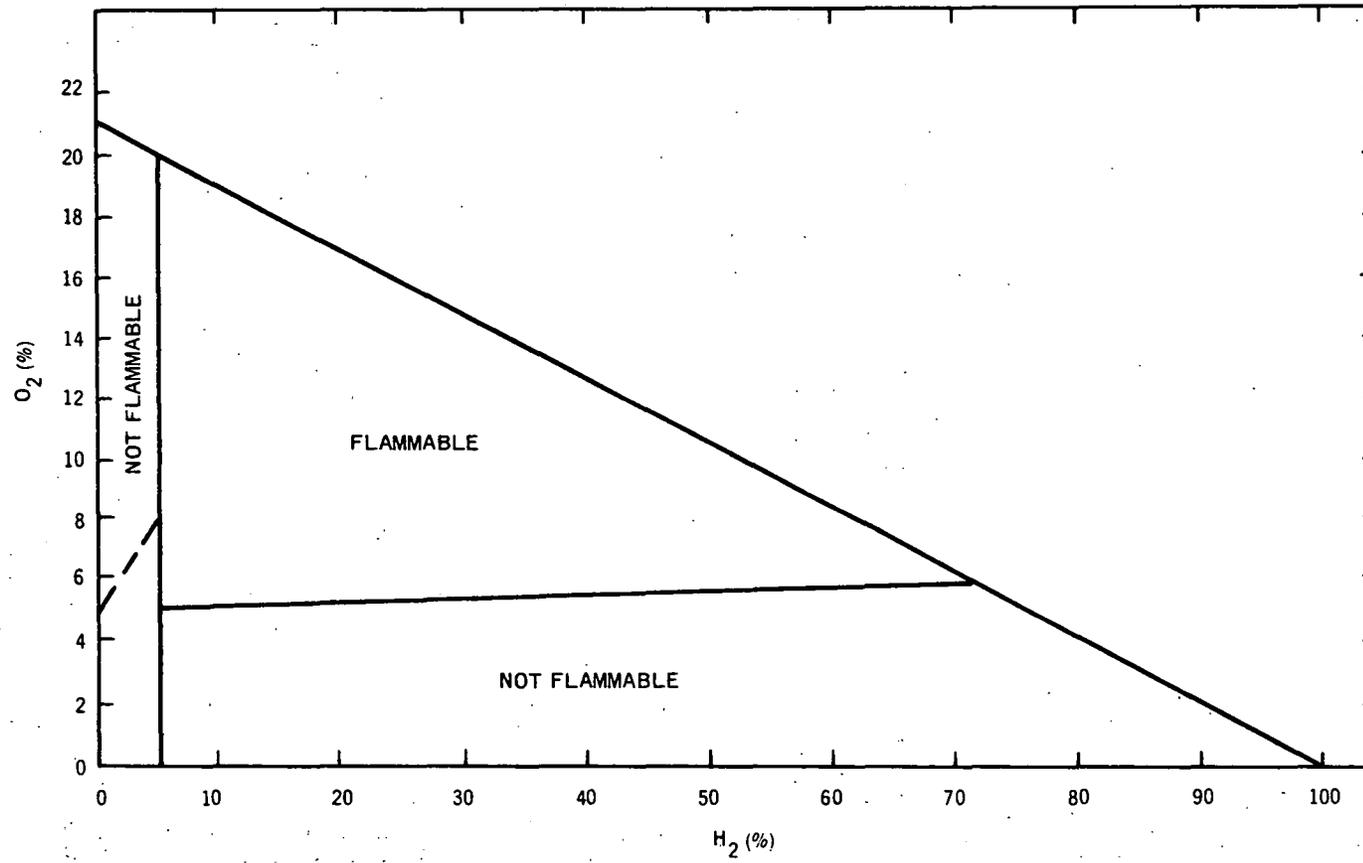


Figure 11.. FLAMMABILITY<sup>8</sup> OF O<sub>2</sub> - H<sub>2</sub> - N<sub>2</sub> DRY MIXTURES

H<sub>2</sub>/100ev compared to the previous values quoted in the literature from capsule tests of 0.45 molecules H<sub>2</sub>/100ev. A similar type shutdown radiolysis test has been conducted at the KRB nuclear power plant during the scheduled shutdown in June 1970. Improvements were made in both the experimental procedure and the measurement techniques used at Humboldt to thereby provide a more accurate determination of the radiolytic decomposition rate during the shutdown. The test at KRB also is more representative of the boundary conditions in a large BWR. The results of the test at KRB confirm the basic trend observed at Humboldt, i.e., a significantly lower hydrogen overall yield value than the previously accepted value of 0.45 molecules hydrogen released per 100 electron volts of energy absorbed. (See Figure 8).

#### (2) Simulated BWR Radiolysis Tests at ORNL

A series of tests were run at ORNL during the time period of July 1969 through December 1969 to determine the effect of various parameters on the coolant radiolysis. The experiments were conducted using a two-vessel closed loop with a gamma energy source (cobalt-60) surrounding one vessel and the other vessel simulating the containment free volume.\* Distilled water with iron impurities\*\* was circulated through the loop using a pump and with an air cover gas. The tests were conducted at three water temperatures (65°C, 95°C, 125°C), three flow rates (100 gpm, 1000 gpm, and 10,000 gpm) which correspond to residence times of 700 minutes, 70 minutes, and 7 minutes; and with three gas/liquid containment volumes (1:1, 2.5:1 and 5:1).

The preliminary results of the tests run at ORNL indicate that the radiolytic hydrogen generation rate is inversely proportional to the coolant water temperature at short residence times as shown on Figure 8. The maximum measured hydrogen yield\*\*\* with the ORNL tests is approximately a factor of 2 reduction over the 0.45 molecules H<sub>2</sub>/100ev previously used as a base index value. Furthermore, the reduction in H<sub>2</sub> generation rate was approximately a factor of 4 to 5 at coolant temperatures close to 212°F. The measured equilibrium hydrogen concentrations as a function of residence time and gas/liquid ratio are plotted in Figure 7(6,7). As can be seen from the figure, the maximum hydrogen concentration of approximately 14% results at a low water temperature and low residence time. Most of the test data resulted in hydrogen equilibrium concentrations between 4% and 7%. This is slightly above the lower flammability limit but below the minimum concentration for which detonation may occur.

Therefore, in summary, the Humboldt test, the KRB test, and the ORNL tests indicate a factor of 2 to 50 reduction in the hydrogen yield due to radiolysis over the 0.45 previously used. Also, the ORNL test data indicate that the equilibrium hydrogen concentration could eventually exceed the lower flammability limit of 4% in air, depending on conditions in the core. Analysis of the data is still underway and the foregoing represents the current best assessment of the situation.

## VI. EFFECTS OF HYDROGEN ON THE CONTAINMENT

### A. General

As indicated earlier, the pressure suppression system is inherently capable of absorbing high energy release rates, hence it has the ability to tolerate rapid (up to 200 ft/sec flame velocity) burning of hydrogen (i.e., "explosions") as long as detonation does not occur. Detonation in air has *never* been demonstrated below 15% to 18% by volume of hydrogen. There is some speculation that perhaps detonation *might* occur at values as low as 6% by volume but this has never been experimentally verified. Given the unlikely event that enough hydrogen is present to give even a 9% mixture, the factor that there is relatively little equipment inside a pressure suppression system that could be an ignition source makes the probability of detonation or combustion very low. Thus, it does not follow that just because the concentration of hydrogen is above 4% to 7% there will necessarily be a fire, let alone a detonation.

Much experimental effort has been expended in determining the flammable and detonable limits of hydrogen when mixed with oxygen and with various diluents. Lewis and von Elbe in their classic book,<sup>(9)</sup> review most of the pertinent work up to 1961. This book indicates that the predetonation compression and heating would not change the detonatable limit for hydrogen to a value below 14%. The 14% value has apparently been reached only with mixtures containing less than 50 ppm of water.

\* Vessel simulated free, volume also had water.

\*\* Corrosion products resulting from test not present at beginning of test.

\*\*\* Calculated from slope of increased pressure curve.

Comments that have been made about the possibility of combustion-induced impulses occurring with mixtures outside of the normally accepted detonatable limits are repeated in reference 10.

A careful search of the literature has failed to disclose any occurrence of hydrogen and air or hydrogen and oxygen detonations in mixtures below 14% hydrogen. In this general area where the theoretical calculations appear to outline the lower limits there exists a sound theoretical basis for nonoccurrence of detonation below 10% hydrogen.

Given that the hydrogen does burn, the *total* energy involved is not large. The LOCA blowdown energy is about  $400 \times 10^6$  Btu whereas the burning of  $H_2$  with all the  $O_2$  in the containment air would only liberate  $18 \times 10^6$  Btu. This is the burning of 170 # moles of  $H_2$ , which is 28% by volume of  $H_2$ . If all the energy were deposited in the containment walls the wall temperature would increase only  $180^\circ F$ . Actually upon ignition the air-stream mixture would be vented into the pool and quenched, and even though mixed mean-gas temperatures momentarily could reach  $2000^\circ F$ , the actual wall temperature would increase only  $10^\circ F$ .

Typical pressure-time curves as a function of burning rate are shown in Figure 12. Note that even for an "explosion" or rapid burn of the entire mixture in 500 milliseconds the containment would survive since a safety factor in excess of 2 has been applied in the design pressure rating. *Thus, even in event of the extremely low probability that a sufficiently large break were to occur, that a large amount of hydrogen would be allowed to accumulate, and that somehow it were to ignite, containment integrity would be retained.*

Even if it were assumed that, upon reaching a 6% concentration in 5 days, the integrity of both the primary and secondary containments were lost the thyroid dose at the site boundary would be 758 Rem and the whole body dose 0.77 Rem. For the same failure at 30 days the thyroid dose would be only 88 Rem and the whole body dose 0.55 Rem. Note that Table 3 shows that using the calculated value of initial metal-water reaction it would take from 30 to 170 days to reach the 6% flammability limit depending on the temperature. This assumes 25% of the rods have perforated and all of the activity in the plenum has escaped. Worst meteorology with ground level release, and a site boundary distance of 800 meters, was assumed.

#### **B. Assurance of Homogeneous Mixture of Combustible Gases**

Any combustible gas released to the primary containment during a postulated loss-of-coolant accident would first be mixed with the drywell atmosphere and would then be swept through the drywell vents due to the built-in pressure suppression action of the primary containment design. The gases would be cooled by the turbulent action in the suppression pool and would then mix with the other noncondensable gases in the suppression chamber. The heat removal and losses from the drywell would eventually reduce the gas temperature and thereby the drywell pressure. The pressure reduction would then result in the noncondensable gases being vented from the suppression chamber to the drywell via the vacuum breaker system to equalize the pressure between the two chambers. Thus, mixing of gases between the two chambers would occur.

Any combustible gases generated in the core region following the postulated LOCA due to either metal-water reaction or radiolytic decomposition of the coolant would be released to the drywell atmosphere. The core cooling flow, either LPCl or core spray, would exit from the vessel via the break and would provide a cooling action and the resulting thermal gradients within the drywell would cause natural convection currents, ensuring a reasonably homogeneous mixture of gases within the drywell. Furthermore, the diffusion of hydrogen gas is very rapid and any appreciable variation in  $H_2$  concentration should exist only for a short time period. Any significant release of noncondensable gases from the reactor vessel to the drywell would increase the drywell pressure which would then result in a venting action through the suppression pool water as soon as the differential pressure was sufficiently high to overcome the head of water in the downcomers. This venting action would cool the gases transferred to the suppression pool water. These noncondensibles would be reasonably well mixed in the suppression chamber due to both the thermal gradients resulting from heat losses, the diffusion of the gases, and the action of the suppression pool sprays.

Pocketing potential has been discussed also with Dr. Franklin S. Thomas of the Naval Research Laboratory. Based on his experiments in which stoichiometric hydrogen-oxygen mixtures were fed through a 6- to 8-inch\* porous plug, his measurements by schlieren techniques indicated that the mixture was detonatable only inside an

\* Diameter

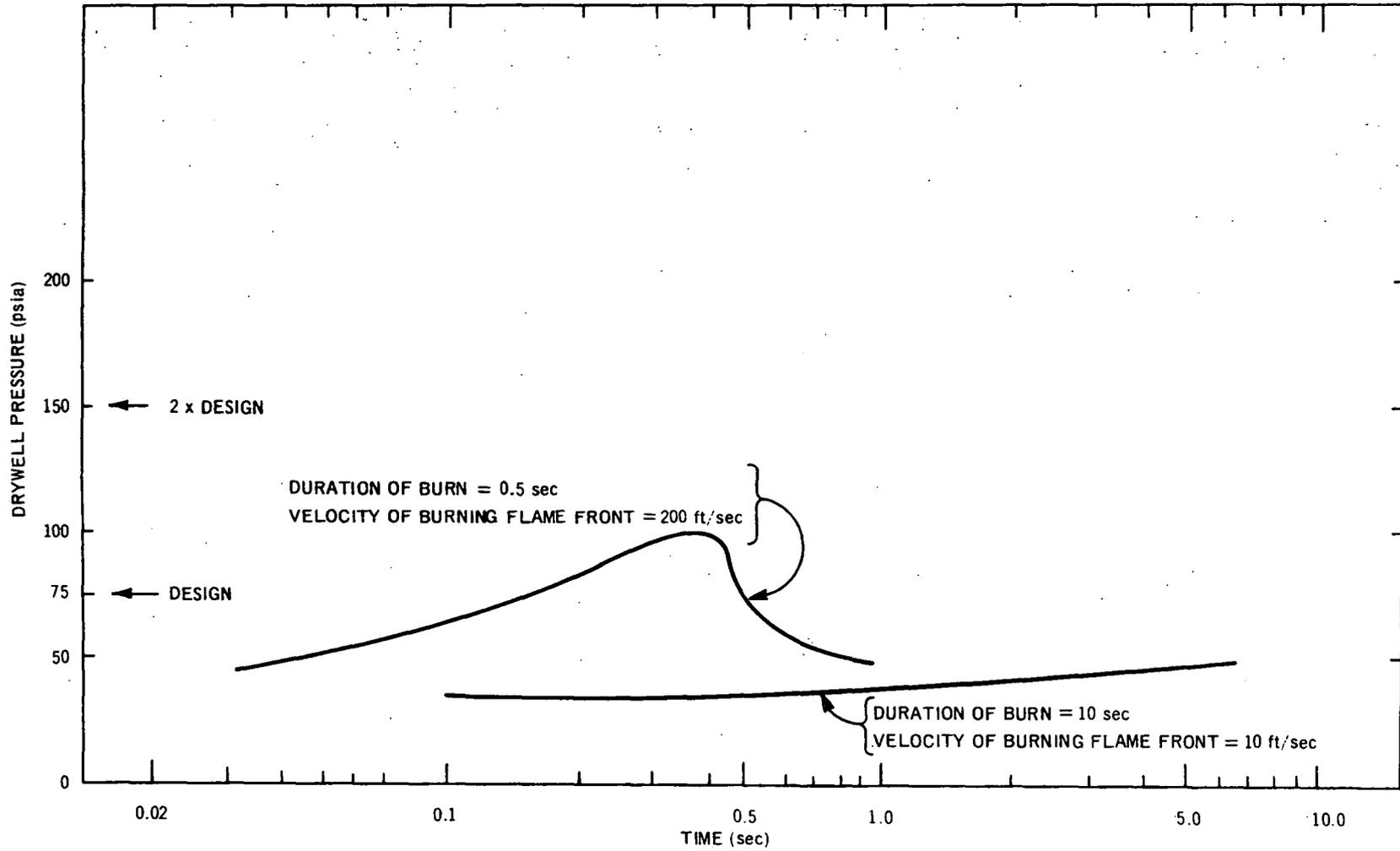


Figure 12. RAPID BURNING IN DRYWELL

18-inch high cone with the plug at the base. The concern for pocketing is generic to all containments, however, and not just for pressure suppression.

Furthermore, even if a pocket of hydrogen were to burn rapidly, the containment can structurally tolerate it (see Figure 12). Also there is no equipment, inside the containment, required following the LOCA that could be damaged if a localized burn were to occur.

Consideration of any significant fission product release from the fuel, such as that assumed in TID-14844, would have to be associated with a complete core meltdown which is inconsistent with the conservative design of the emergency core cooling network and its redundant systems. Actual release of fission products from the fuel therefore will be much less than that specified in TID-14844. G. W. Parker<sup>(6)</sup> states in an ORNL report\* that "the mechanistic method of evaluating potential volatile fission-product release may be useful in the justification of a significant reduction in presently accepted guidelines. For the average fuel rod linear power density, around 8kW/ft, the potential radioiodine release is lower than for UO<sub>2</sub> melting by perhaps a factor of 100; however, above 18kW/ft the potential is probably only a factor of 2 or 3 lower." Over 70% of a typical BWR core has a linear power of under 8 kW/ft with less than 0.3% having a linear power greater than 17 kW/ft. Therefore, a reasonable source term would be lower than TID-14844 by a factor of about 50 to 100.

The energy decay fraction contained in the TID-14844 release is plotted versus time on Figure 13. Note that the time average release fraction is initially approximately 14% and decreases as a function of time (i.e., 6.5% in 3 days). Equation (1) would give the effective hydrogen generation following a LOCA.

$$\text{Moles H}_2 \propto \left\{ G_{\text{H}_2 \text{ core}} \times (1-\text{Rel}) \times f_1 + G_{\text{H}_2 \text{ pool}} (f_2 \times \text{Rel}) f_3 \right\} \quad (1)$$

where:  $G_{\text{H}_2} = \frac{\text{molecules H}_2 \text{ generated}}{100 \text{ ev absorbed}}$

Rel = fraction of decay energy released from the core with the fission products, and

f<sub>1</sub> = fraction of remaining core decay energy into reactor coolant (conservatively assumed to be 0.11).

NOTE: The fraction of core decay energy deposited in the reactor coolant was conservatively evaluated to be 5% when consideration of the different gamma energy levels and material cross sections were used (see Appendix A for discussion of conservatism).

f<sub>2</sub>, fraction of released energy into pool (conservatively assumed to be 1.0)

NOTE: The fraction of released fission product energy which could be deposited in the pool would be approximately 50% since the noble gases would be airborne (see Appendix A).

f<sub>3</sub>, fraction of energy into pool absorbed by water (conservatively assumed to be 1.0)

For a stoichiometric release the amount of O<sub>2</sub> generated by radiolysis would be:

$$\text{Moles O}_2 = \frac{1}{2} \times \text{moles H}_2 \quad (2)$$

(radiolysis) (radiolysis)

Consideration of the extent of H<sub>2</sub> generated by radiolysis for the TID-14844 release compared to using the GE release\*\* is given by equation (3):

$$\frac{\text{Moles H}_2 \text{ (TID)}}{\text{Moles H}_2 \text{ (GE)}} = \frac{G_{\text{H}_2 \text{ core}} \times (1-\text{Rel}) \times f_1 + G_{\text{H}_2 \text{ pool}} (f_2 \times \text{Rel})}{\sim G_{\text{H}_2 \text{ core}} \times f_1} \quad (3)$$

\* ORNL Nuclear Safety Research and Development Program Bi monthly Report for Sept-Oct 1969 ORNL-TM-2777.

\*\* This release is described in APED-5756<sup>(1)</sup>, and consists of an average of 1.8% noble gases and 0.32% halogens.

\*\*\* The fission products released from the fuel (GE model) and if all transported to the pool would only cause a 1% or 2% increase in the radiolysis. The generation equals approximately G<sub>H<sub>2</sub> core</sub> x f<sub>1</sub> as shown in equation (3).

$$\text{or, } \frac{\text{Moles H}_2 \text{ (TID)}}{\text{Moles H}_2 \text{ (GE)}} = (1 - \text{Rel}) + \frac{G_{\text{H}_2 \text{ pool}} (f_2 \times \text{Rel})}{G_{\text{H}_2 \text{ core}} f_1} \quad (4)$$

$$\text{or, } \frac{\text{Moles H}_2 \text{ (TID)}}{\text{Moles H}_2 \text{ (GE)}} = 1 + 8.1 \text{ Rel} \quad (5)$$

for  $G_{\text{H}_2 \text{ pool}} = G_{\text{H}_2 \text{ core}}$  &  $f_2 = 1.0$

Figure 13 also shows the ratio of hydrogen generated due to radiolysis for both the TID-14844 release and the GE release as a function of time. It is evident from this figure that the use of TID-14844 release with all of the energy associated with all of the fission products released would result in a maximum increase in radiolysis of approximately 100% over the radiolysis associated with GE assumptions and would decrease with time (i.e., about 50% at 3 days).

Therefore, in summary, combustible gases generated in either the core region from metal-water reaction and/or by radiolysis or in the pool region by radiolysis would be reasonably mixed in the primary containment due to thermal gradients and spray action.

### C. Lesser LOCA within the Primary Containment

The predicted fuel rod cladding damage (i.e., perforations) as a result of a LOCA would be limited to break areas in excess of approximately 0.8 square feet. Furthermore, the predicted maximum fuel rod cladding perforations would be less than 13% of all the fuel rods in the core for the design break accident. The GE fission product release model would result in approximately 1.8% of the core average noble gases and approximately 0.32% of the core average halogen fission products being released from the fuel into the fuel rod plenum. The perforation of fuel rods would result in the transfer of the released fission products from the fuel rod plenum to the containment via the break area. The noble gases would remain airborne and the halogens could be absorbed in the pool water. Even if all the energy associated with the released halogens were deposited in the pool water the increase in total hydrogen production due to radiolysis would be negligible as compared to the case with no perforations. Therefore, the radiolytic decomposition of the water will occur essentially only in the core region and the hydrogen and oxygen gases thereby generated would be released to the drywell through the postulated break area. The increase in drywell pressure would cause the gases to be vented to the suppression chamber to equalize the pressure between the drywell and suppression chamber.

The contribution of total hydrogen production due to radiolysis was evaluated using the test data obtained from the simulated BWR containment configuration at ORNL. The resulting hydrogen yield or rate from the ORNL tests was observed to be a function of both water temperature and time following the accident as shown on Figures 7 and 8. The maximum measured hydrogen generation rate  $G_{\text{H}_2}$  was approximately 0.2 at a water temperature of 150°F and a residence time of 7 minutes and the rate continually decreased as a function of time as noted on Figure 14. Therefore, using the maximum measured ORNL radiolytic decomposition rate it would take approximately 95\* hours following the accident to reach a containment hydrogen concentration of approximately 4%, the lower flammability limit in air. The calculated metal-water reaction would be less than 0.06% across the entire break area spectrum which would result in an initial H<sub>2</sub> concentration in the containment of less than 1% (all sources included). The corresponding time to reach a 4% H<sub>2</sub> concentration in the containment considering these other sources would be approximately 58\* hours.

\* Based on GE fission-product release: 11% core power to water, infinite irradiation.

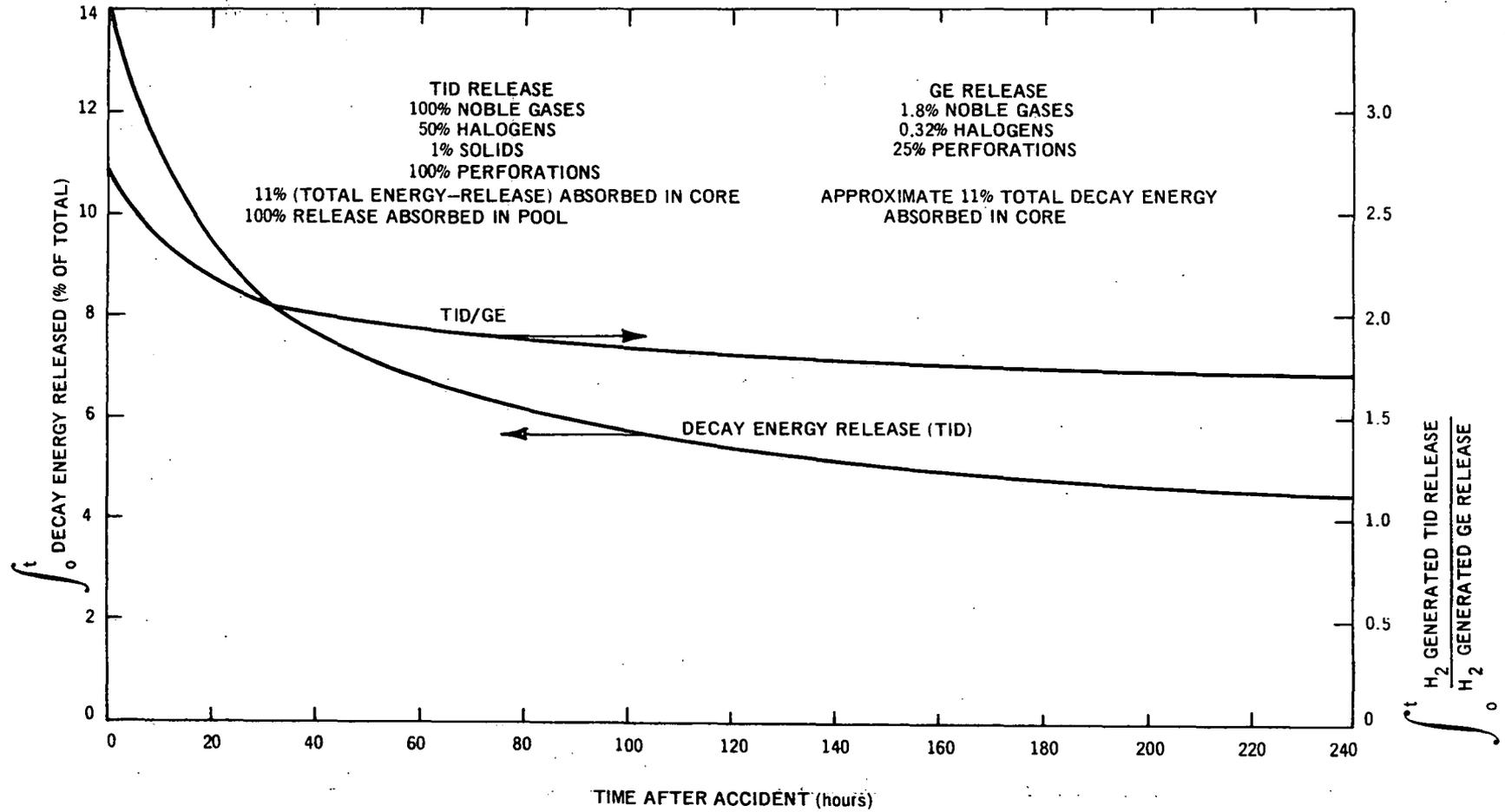


FIGURE 13. DECAY ENERGY RELEASED AND RADIOLYSIS FOLLOWING LOSS-OF-COOLANT ACCIDENT

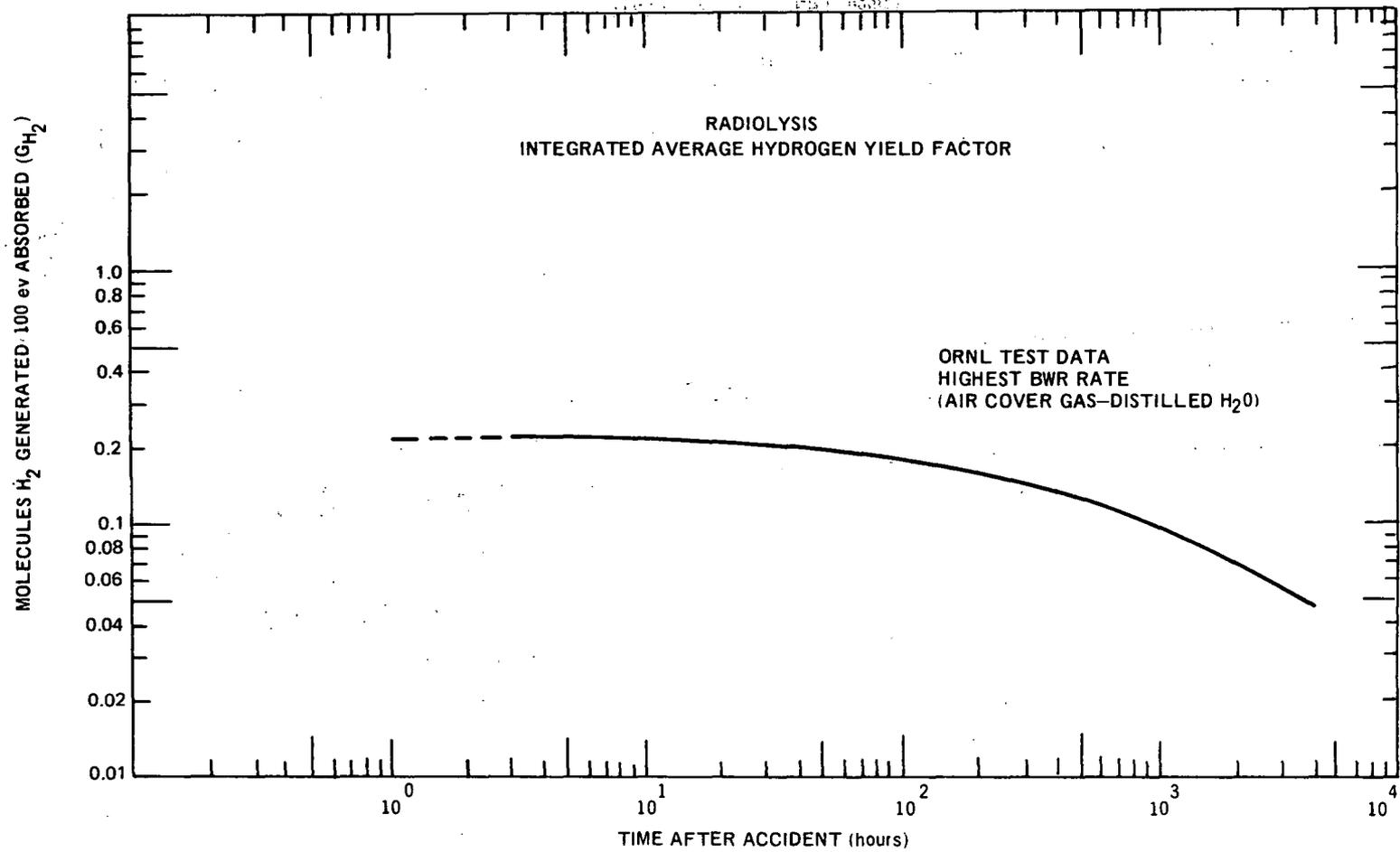


FIGURE 14. EFFECTIVE HYDROGEN YIELD, G<sub>H<sub>2</sub></sub>, TID RELEASE

Therefore, in summary, the break size and location would have a negligible effect on the total generation of hydrogen gas due to radiolysis and the time to reach a combustible gas concentration would only be a function of the initial hydrogen concentration in the containment immediately following the postulated break. For the small breaks the metal-water reaction would be small and would therefore reduce the initial hydrogen concentration in the containment immediately following the LOCA. See Figure 2 and Figure 14A.

## VII HAZARDS OF INERTING

The everyday adverse effect of inerting such as restricted inspections during startup, jeopardy to personnel, and placing an unfair burden and penalty on the operator (in making entry decisions for inspections should minor difficulties be encountered during operation) far outweigh any potential gains in safety for the highly unlikely event of a large rupture. For example, if the operator should note a slight indication of a leak or other adverse condition in the drywell, good operating practice would dictate that he assess the situation as soon as practical even though the condition is well below any Tech Spec limitations. Without inerting he could, within a few hours, reduce power, make his inspection, and be back on the line. With inerting, however, it is extremely unlikely that any entry would be allowed because of the danger of asphyxiation. (There are several instances on record where personnel were nearly asphyxiated from malfunctioning breathing apparatus or other incidents.) Thus he would be forced with having to de-inert the containment, flush it thoroughly and re-inert it. This could require as long as two shifts. Thus inerting, in essence, prevents the operator from exercising good operating practices. These have been an integral part of his training and experience, as well as the cornerstone of real safeguards against small events leading to larger and more serious accidents. Hence, it is believed that from an overall viewpoint a safer situation exists without inerting. Finally, inerting does not eliminate radiolysis which produces oxygen, thereby essentially de-inerting the containment.

In addition to the comments on the hazards of inerting already given the following item, 2B(1) to Amendment 22 Dresden Nuclear Power Station Unit Number 3, is cited in its entirety as an appropriate reference.

### 2. ACRS Letter Item Concerns

#### A. Introduction

*On September 10, 1969, a letter was written by the Advisory Committee on Reactor Safeguards on Dresden Nuclear Power Station Unit 2. Consideration was given to many items of plant safety. Most of these items of plant safety were addressed to the satisfaction of the committee, but some remained unresolved. The purpose of this section of this report is to review these unresolved items, to add information as it applies to these items, and to give a status report of those items which remain unresolved.*

#### B. Addressing of ACRS Letter Items

##### (1) CONTAINMENT INERTING

*During the ACRS review of Dresden Unit 2, Commonwealth Edison Company unequivocally stated its opposition to containment inerting. This position was founded on the fact that inerting discourages frequent access and inservice inspection of the reactor primary system and thereby fosters a safety posture less than desirable. Therefore, Commonwealth Edison is opposed to inerting of the Dresden Unit 3 containment.*

*An early requirement placed on containment vessels demanded that the containment be designed to withstand, without exceeding the design pressure, a substantial metal-water reaction resulting from an assumed reactor core meltdown. Since that time, major improvements have been made in the provisions for emergency cooling of the core. Some of these major improvements include:*

- (a) Redundant and completely independent core spray systems.
- (b) Addition of a high pressure coolant injection (HPCI) system.
- (c) Addition of a low pressure coolant injection (LPCI) system.
- (d) Completely redundant and independent emergency power systems.
- (e) Improved reliability of each electrical and instrumentation function by conformance to IEEE 279.

These added improvements to the emergency core cooling systems ensures that no reactor core meltdown will occur. This gives added support to the pressure suppression containment which, itself, will tolerate much larger quantities of hydrogen generation than what is calculated to occur. This is due to the containment's inherent design capability for transferring high energy and mass rates to the suppression pool. Actually, a large margin exists between the amount of hydrogen expected and that which can be tolerated by the containment.

The calculated metal-water reaction which will occur compared to the metal-water reaction which the containment can withstand is as follows: the maximum metal-water reaction expected to occur over the entire break spectrum would be less than 0.1%. Compare this with—

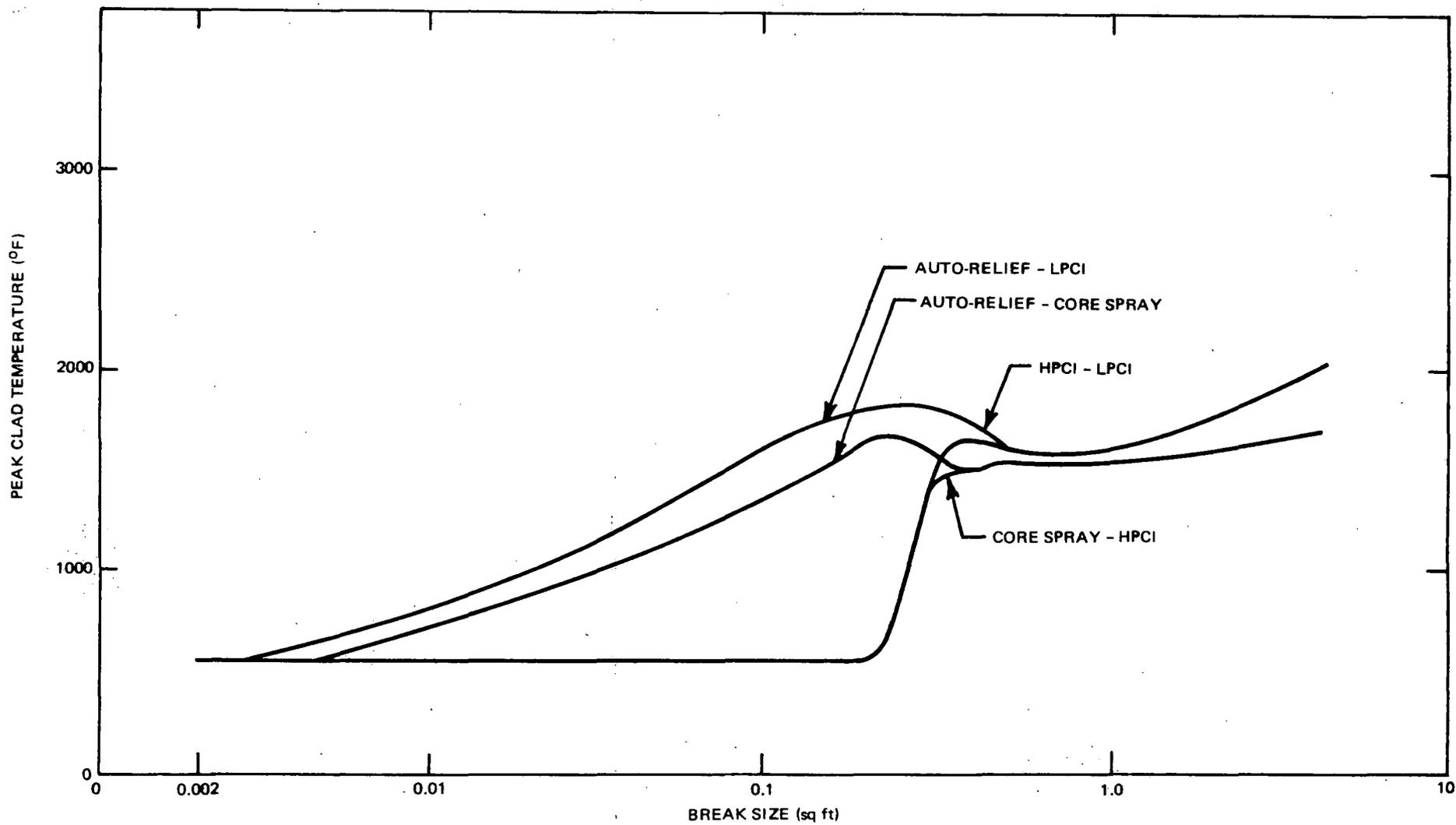
1. If the hydrogen burns as it is evolved, the containment is capable of tolerating at least a 30% metal-water reaction, a factor of 300 over that expected.
2. If ignition is postulated to occur in the drywell after hydrogen has been released, the containment can tolerate a 20% metal-water reaction, or a factor of 200 above that actually expected.
3. If the hydrogen is not burned at all, the containment can tolerate a 15% metal-water reaction, or a factor of 150 above that actually expected.

It should be noted that these large metal-water reactions are associated with failure of all emergency core cooling systems and a resulting gross core meltdown. Note also that the 20% metal-water reaction specified in case 2 is a theoretical value because the 15% metal-water reaction specified in case 3 is limiting.

It can be seen, then, that inerting of the containment to prevent a hydrogen explosion as a result of metal-water reaction is unwarranted. If but one of the many emergency core cooling systems functions, then the metal-water reaction is so small as to preclude a need for inerting. If all the emergency core cooling systems fail, gross core meltdown will occur—an event which cannot be limited by inerting the containment and the consequences of which cannot even be predicted.

The best way to reduce risk to the public health and safety due to a loss-of-coolant accident (LOCA) is to prevent the occurrence of an LOCA. The best way to prevent the occurrence of an LOCA is: (a) to have a well-designed and constructed primary system, (b) to provide a sensitive leak detection system, and (c) to provide the capability for convenient visual location and evaluation of leaks. Regarding item a, we have done all that we know how to do to ensure that the primary containment represents the best possible design and construction. Regarding item b, we have drawn on our extensive experience with the Dresden Unit 1 leak detection system and have spent an additional \$100,000 to install a leak detection system for Dresden Units 2 and 3 with superior sensitivity. Regarding item c, over 75 years of steam power plant experience has convinced us of the need for providing visual location and evaluation of leaks.

Inerting the containment makes the containment significantly less accessible for inspection of equipment and piping, and therefore militates against the capability for visual leak location and evaluation. The location and nature of leaks is very important to assessing the risk of an LOCA. Quantitative measure of leakage is insufficient for assessing this risk. Commonwealth Edison has installed a leak detection system which will allow detection of leaks on the order of ounces per minute. The design reflects the successful leak detection experience from Dresden 1 where the quantitative performance of the system has been demonstrated. While we have attempted to provide leak location capability in the Dresden 2/3 system, we will not know the extent to which this has been achieved in practice on a reliable basis until we have acquired sufficient operating experience with it. The only sure way to locate leaks is to enter the containment and visually inspect the primary system. With the leak detection system we



**FIGURE 14A. PEAK CLAD TEMPERATURES VS LIQUID BREAK SIZE**

have installed, and no containment inerting, an indication of a leak can be checked by bringing the unit to a hot standby (pressurized) condition during an off-peak load period (between midnight and 8 a.m., for example), and entering the containment for inspection. Once located, the leak can be repaired or isolated. If the leak cannot be repaired or isolated, the unit can then be brought to a cold shutdown condition whereby repairs can be made. Thus, a leak can be detected and repaired long before a more serious loss-of-coolant condition develops.

If the containment is inerted, the time to de-inert, look for and repair the leak, and re-inert the containment is such that inspection during the nightly off-peak load condition is not possible. We know of no protective clothing or equipment available which would enable personnel to enter an inerted containment and provide them with sufficient flexibility to locate and evaluate leaks. Thus, on receiving an indication of a possible leak, the incentive is to wait for an off-peak load condition longer than a few hours, or until the leak rate limits in the Technical Specifications are approached.

In addition, we are concerned with the safety of personnel who must enter the containment which has been inerted and then supposedly de-inerted. This will also result in an incentive to enter the containment as little as possible, and thus discourages the location and repair of small leaks.

From the standpoint of the public health and safety, the value of inerting is wrong, in our opinion, if the price is reduced ability to locate and repair leaks. In the first place, the original basis for inerting is gone. The inerting precedent in BWR's was established on Humboldt Bay when peak clad temperatures approached the melting point of Zircalloy cladding and a metal-water reaction of 33% could be postulated. As is well known, the peak clad temperatures were shown to be actually under 2300° F several years ago when the clad shattering phenomenon was discovered by ANL. Dresden Unit 3, like Dresden Unit 2, utilizes improved core cooling systems over those used at Oyster Creek and Nine Mile Point. With these systems, peak cladding temperatures for an LOCA are well below the melting point and well within the range where core geometry is preserved. This is true even for a degenerated case where equipment is assumed to fail. As a result of limiting cladding temperatures, the hydrogen in the containment produced from the zirconium-steam reaction will not reach concentration limits for flammability. It should also be pointed out that the hydrogen release calculation is conservative and has been verified by experimental work.\*

Secondly, in the event of radiolysis of water following an LOCA, containment inerting would be of little or no value since both hydrogen and oxygen would be produced by the radiolysis process.

Third, after 10 years of operation of Dresden Unit 1, we are convinced that it is much more desirable to locate leaks as soon as possible. Experience has shown that many small leaks through valve packing will probably develop, particularly during initial operation of the unit, and following unit shutdown and subsequent start-up. By correcting these packing leaks as soon as they occur, we are able to keep a relatively leak-free containment thus avoiding masking, and are able to identify new leak sources much more quickly than if we had a containment with several known and uncorrected leaks.

The ACRS recognized the validity of our position in its letter on operation of Dresden Unit 2 (September 10, 1969) by stating that "inerting increases problems of inspecting for and repairing leaks in the primary system." The committee also recommended that the requirement for inerting be reviewed periodically as operating experience is gained. We believe that recent experiences with leaks in the Elk River, LaCrosse and Nine Mile Point reactors provides additional evidence as to the need for providing access to the containment as soon as indications of a possible leak are obtained. This can be accomplished in a practical manner on the Dresden class reactors only if the containment is not inerted.

In summary, we oppose inerting because, in our judgment, inerting increases the risk of an LOCA, and this risk far outweighs the very limited value of inerting in mitigating the consequences of an LOCA. Moreover, we see no justification for exposing our employees to the added risk of working in an inerted environment. We will inert Dresden Unit 3 only if the AEC tells us we must do so as a condition for issuance of an operating license.

\* H.C. Brassfield, J.F. White, et al., "Recommended property and reaction kinetics data for use in evaluating a light water cooled reactor loss-of-coolant incident involving Zr-4 or 304 stainless-steel clad UO<sub>2</sub>" GEMP-482 (April 1968).

## VIII. POTENTIAL SOLUTIONS

From the foregoing discussions it is apparent that given the remote combination of conditions associated with the design basis accident (DBA) the hydrogen concentration could indeed build up over a period of time. Even if it should build up to the equilibrium values implied by the ORNL tests and then be ignited, the containment integrity would be retained. However, the H<sub>2</sub> generation rate is low, thus giving the order of days or weeks in which to maintain the concentration to low values. Thus, if for regulatory reasons more margin is desired there are various ways in which this can be done.

### A. Venting the Containment

One way is to simply vent the containment through the standby gas treatment system filters and replenish the noncondensibles. The key issue here is the acceptability of doses and venting time. Venting would be considered by the operator only if measurements after an LOCA indicated the hydrogen concentration were approaching, say, a 4% concentration.

It is very important to point out that if one approaches the problem on a strictly technical consistent basis it is clear that venting is a reasonable, possible, and practical solution. If any of the key inputs are made inconsistent or are arbitrarily set, a different solution becomes optimum and the effect will be serious. The key boundary conditions which govern the feasibility of venting are the fission-product source term, the dose transport model, the rate of hydrogen evolution and the amount of initial metal-water reaction postulated. Various combinations of assumptions have been calculated as shown in Table 5. Shown in this table are the effect on doses of perturbing the hydrogen sources, rate, and fission-product source terms. The "GE Source" is consistent with ECCS and the "AEC Source" consistent with TID-14844 as defined earlier and in Table 5. However, as previously mentioned these values were conservatively evaluated using 11% core decay power for radiolysis in the core region, all of the released energy to the pool and infinite irradiation. All of these times would increase significantly using the realistic assumptions discussed in Appendix A. Correspondingly, the doses to the environs would also decrease significantly.

The most reasonable case, No. 10, (which is still conservative) shows that the doses are within the ½ Rem yearly dose allowed under 10CFR20 by over an order of magnitude. Even if the initial metal-water reaction is increased by over an order of magnitude the offsite doses are within ½ Rem as shown by case No. 4.

Case No. 1 demonstrates that a simultaneous increase in both initial hydrogen and hydrogen evolution rate results in well under a ½ Rem dose provided a reasonable source term is used.

Cases 3, 6, and 9 show the result of adding the gross conservatism associated with the fission product source terms and transport model to the hydrogen source and generation terms. The key differences in the fission-product source and transport models are shown in Table 6. One cannot be too emphatic that use of the TID-14844 source term is incorrect and inconsistent with use of the ECCS. Even using the results from G. W. Parker's ANS paper<sup>(11)</sup> which states that for linear power densities above 18 kW/ft the release is within a factor of 2 to 3 TID-14844, the fact remains that less than 0.1% of the core is above this level. The integrated source term would still be at least a factor of 50 to 100 below the values presented in TID-14844.

Venting could be combined with a cleanup system having a greater decontamination than the standby gas treatment system. The degree of decontamination achieved would depend on the investment one would be willing to make. Hence, feasibility is highly dependent on allowable doses, venting rate, and heat load to be accommodated.

### B. Recombiners

If venting is not an acceptable solution there are a variety of nonventing recombination schemes. However, these are significantly more complex and expensive. Three classes of possible recombiners are summarized in Table 7. Tables 8 through 10 describe the various recombiners possible within each class along with its advantages and disadvantages. It is estimated that actual incorporation of one of these systems as a finished product would

Table 5  
VENTING –  
BLEED THROUGH STANDBY GAS TREATMENT SYSTEM – MAKEUP  
DOSES TO ENVIRONS

Reactor Power = 2550 MWt

Core Decay Energy to Coolant = 11%

Fission Product Decay Energy to Pool = 100%

(See Appendix A)

**INFINITE IRRADIATION**

Source	% M-W*	G <sub>H<sub>2</sub></sub> **	Transport Model	Venting Rate CFM	Initiation Time Hr	Thyroid Dose Rem*** (Halogens)	Whole Body Dose Rem*** (Noble Gases)
1. GE	0.3	0.45	GE	111	18	0.05	0.14
2. AEC	0.3	0.45	GE	111	18	15	23
3. AEC	0.3	0.45	AEC	111	18	3400	23
4. GE	0.75	0.1	GE	27	24	<0.05	<0.1
5. AEC	0.75	0.1	GE	27	24	2.7	5.8
6. AEC	0.75	0.1	AEC	27	24	930	5.8
7. GE	0.3	0.07	GE	15	120	0.003	0.012
8. AEC	0.3	0.07	GE	15	120	0.9	1.9
9. AEC	0.3	0.07	AEC	15	120	310	1.9
†10. <u>GE</u>	<u>0.06</u>	<u>0.1</u>	<u>GE</u>	<u>15</u>	<u>175</u>	<u>&lt;0.003</u>	<u>&lt;0.012</u>
10CFR100 Limits						300	25

\* Best estimate is 0.06%

\*\* Best estimate is 0.1

† Reasonably conservative best estimate

\*\*\* 30-day dose at 800 meters (stack ~ 100 meters)

**Table 6  
FISSION PRODUCTS**

<b>Halogens</b>	<b>GE Source Release</b>	<b>AEC Source Release</b>
Core Effect . . . . .	25%	100%
Halogen Plenum Activity . . . . .	0.32%	50%
Plateout Factor . . . . .	2	2
Airborne in Primary Containment . . . . .	0.04%	25%
	<b>GE Transport Model</b>	<b>AEC Transport Model</b>
Methyl Iodide Formation . . . . .	1%	8%
Partition Factor* . . . . .	$10^2$	1
SGTS Filter Efficiency . . . . .	99%	90%
Duration of Continuous Venting . . . . .	30 days	30 days
Meteorological Conditions . . . . .	same	same
	<b>GE Source Release</b>	<b>AEC Source Release</b>
<b>Noble Gases</b>		
Core Effect . . . . .	25%	100%
Noble Gas Plenum Activity . . . . .	1.8%	100%
Plateout Factor . . . . .	1	1
Airborne in Primary Containment . . . . .	0.45%	100%
	<b>GE Transport Model</b>	<b>AEC Transport Model</b>
Partition Factor . . . . .	1	1
SGTS Filter Efficiency . . . . .	0	0
Duration of Continuous Venting . . . . .	30 days	30 days
Meteorological Conditions . . . . .	same	same

\* Partition Factor defined mass of iodine per unit volume of water to mass per unit volume of air (see APED-5756, p. 2-38).

**Table 7  
NONVENTING CONTAINMENT TREATMENT SYSTEMS**

- Thermal**    When  $H_2$  and  $O_2$  are heated together, the reaction  $2H_2 + O_2 \rightarrow 2H_2O$  occurs. This reaction may be used to remove  $H_2$ . (At high temperatures\* the reaction is reversible.)
- Catalytic**    The  $2H_2 + O_2 \rightarrow 2H_2O$  reaction occurs at temperatures near room temperature in the presence of some materials (i.e., platinum and palladium).
- Solid Gas**    Copper and its oxides react with gaseous  $H_2$  and  $O_2$  under some conditions. By adjusting and maintaining the correct bed temperatures it may be possible to effectively remove  $H_2$  and/or  $O_2$  from the containment.

\* Probably need temperatures well in excess of 2000°F to decompose  $H_2O$ .

7

Table 8  
THERMAL RECOMBINATION—NONVENTING

System	Description	Advantages	Disadvantages	Ranges of Feasibility		
				Air Containment $H_2 < 4\%$	Inert Containment $H_2 < 4\%$	Inert Containment $H_2 > 4\%$
Flame	Flame systems must operate with starting point $H_2$ concentrations greater than 4% and $O_2$ greater than 5%. Flame may be supplied by another system but would add non-condensibles or need stored $H_2$ & $O_2$ in large quantities.	The use of sintered metal flame arrestor burner is possible.  Nonpoisoning	Provides ignition source and could create accident.  Will not work below 4% $H_2$ or 5% $O_2$ in burner.	No (1)	No (1)	Yes With much added $O_2$
Hot Surface (pebble bed or glo-bar)	Nonflame burning process. Any surface at temperatures above $750^\circ C$ acts as a reaction site for $2H_2 + O_2 \rightarrow 2H_2O$ . Needs external heat supply and temperature control.	Works at all concentrations of $H_2 > 1/2\%$  Simple Nonpoisoning	Provides ignition source (safety concern) (2)	Yes	Yes	Yes With sufficient $O_2$
Hot Gas	Nonflame burning process. At $T > 750^\circ C$ the auto-catalytic reaction of $H_2 + O_2$ to form $H_2O$ is usually rapid.	Works at all concentrations of $H_2 > 1/2\%$  Simple Nonpoisoning	Provides ignition source (safety concern) (2)	Yes	Yes	Yes With sufficient $O_2$

- (1) Hydrogen might be added to bring input composition up to flammability.  
 (2) Flame and detonation arrestors are possible.

11-53

Table 9

CATALYTIC RECOMBINATION—NONVENTING

System Description	Advantages	Disadvantages	Ranges of Feasibility		
			Air H <sub>2</sub> <4%	Inert H <sub>2</sub> <4%	H <sub>2</sub> >4%
Gas is treated to reduce relative humidity less than 30%, then passed through the catalyst bed where H <sub>2</sub> + O <sub>2</sub> combine to form water.	Works on any concentration of H <sub>2</sub> and O <sub>2</sub> at temperatures between 100 and 500°C.  Can be used to remove O <sub>2</sub> by adding H <sub>2</sub> . Is used for normal off-gas.	Potentially susceptible to poisoning.  If not cooled, can be a source of ignition.	Yes	Yes	Yes If inert and O <sub>2</sub> sufficient.

**Table 10**  
**SOLID-GAS RECOMBINATION—NONVENTING**

System Description	Advantages	Disadvantages	Ranges of Feasibility		
			Air H <sub>2</sub> <4%	Inert H <sub>2</sub> <4%	H <sub>2</sub> >4%
By properly adjusting the temperature of a bed of finely divided Cu the O <sub>2</sub> in a stream of containment atmospheric gas can be removed.	Nonpoisoning  Operating range — ~350°C	Acts as catalyst for H <sub>2</sub> + O <sub>2</sub> reaction.  Easily overheats and loses surface area (efficiency).	Yes	Yes	Yes
By properly adjusting the temperature of a bed of finely divided CuO the H <sub>2</sub> in a stream of containment atmospheric gas can be removed.		Feasibility not yet demonstrated over composition range of interest.			
May be shown regenerable in both modes.					

35

take about 2 to 3 years. The major contribution to the time period is the engineering required to design a system to operate over the large range of conditions that could follow a LOCA. Of course, the system must be highly reliable, it must be instrumented, and it must not in itself generate the very accident that the system is there to prevent.

In case the initial mixture was postulated to be near the detonable range, i.e., ~5% metal-water\* reaction, the recombiner systems described would have to be modified significantly. All systems would have to be isolated from the containment by flame and detonation arresters. The catalytic systems could also incorporate inerting with steam or nitrogen within the process stream as is done in the catalytic recombiner offgas systems. If the calculated initial hydrogen amount with appropriate safety factors is not to be used, it is of utmost importance to the system design to know what arbitrary amount of hydrogen must be considered.

It is concluded that if one adopts a technically consistent attitude venting results in reasonable and acceptable doses and is thus the leading candidate to solve the hydrogen problem.

Whether this or the other solutions are mutually satisfactory depends on the ground rules and regulatory decisions imposed. This is best illustrated by the matrix in Table 11. The potential solutions are listed vertically. Across the top the key issues which must be agreed upon are indicated. Thus, for example, for inerting alone to be a solution the flammability limit must be at least 6% and the radiolytic hydrogen equilibrium must also be under 6%. For solutions other than venting, the state of the art is such that it would be 2 to 3 years after agreement on boundary conditions before operating units could actually be installed at the sites.

## IX. CONCLUSIONS

1. The calculated metal-water reaction across the entire break area spectrum would be approximately 0.06% with the current ECCS design thereby resulting in a hydrogen concentration of less than 0.5% by volume.
2. The ECCS can withstand degeneration without large increases in hydrogen production due to metal-water reaction.
3. Fuel rod failure due to perforations from the internal gas pressure following the LOCA does not impede the performance of the ECCS. Therefore perforations cause an insignificant increase in the amount of hydrogen released.
4. The hydrogen equilibrium value in the large BWR primary containment due to radiolysis of the coolant could be in the 4% to 7% range.
5. Even if sufficient hydrogen accumulated to result in a flammable mixture, the containment can withstand, without failure, rapid burning of hydrogen concentrations up to approximately 15% by volume.
6. The rate of hydrogen buildup within the containment would permit controlled venting and the resultant doses to the environs would be in the range of a few percent of the 10CFR100 guidelines.
7. The solution to the hydrogen problem is heavily influenced by policy and nontechnical decisions. Ground rules must be established before the best solution can be chosen.
8. Inerting alone is not a solution to the overall hydrogen problem.
9. Controlled venting is the best solution from a technically consistent point of view and most practical from a backfit viewpoint.

## X. RECOMMENDATIONS

It is recommended that:

- a. The TID-14844 issue be realistically assessed and source terms decided upon in a manner which is technically consistent.
- b. A sensible approach be taken to the metal-water reaction source margin requirements taking due account of ECCS.
- c. Controlled venting be given very serious consideration as the best solution.
- d. Policy and philosophical ground rules be explicitly stated as soon as possible so that detailed engineering toward a solution can begin.

\* Implies an inert atmosphere required.

**Table 11  
HYDROGEN PROBLEM**

Resolution	G.E.-AEC Issues									
	A	B	C	D	E	F	G	H	I	J
Do Nothing . . . . .					X		X	X		
Vent . . . . .			X	X	X	X				
Inert . . . . .							X	X		
1. Vent and Inert . . . . .	X			X						
2. Vent and Inert . . . . .			X	X	X	X				
3. Vent and Inert . . . . .		X		X	X	X				
Recombine . . . . .					X				X	X
Inert and Recombine . . . . .									X	X
Vent and Recombine . . . . .			X	X	X	X			X	X
Recombine and Cleanup . . . . .					X				X	X
Vent and Cleanup . . . . .		X	X		X	X				

**GE/AEC ISSUES**

- A Evolution rate < 0.2 vs. 0.45
- B Source term < TID
- C G.E. Transport model
- D 10CFR100 limit vs. 10CFR20
- E Metal-water 5 to 10 times calculated vs. arbitrary 10%
- F Time to vent—days vs. months
- G Flammability of 4% to 6% does not detonate
- H Radiolysis H<sub>2</sub> equilibrium < 6%
- I Safety problem not introduced
- J Feasibility of actual design

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## XII. ACKNOWLEDGMENT

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## APPENDIX A – ENERGY DEPOSITION IN COOLANT FOLLOWING LOCA

### 1. Reactor Region

The energy transferred to the reactor coolant is a function of both the core decay power and the mass density of coolant present within the core. Consideration of the realistic 1000-day irradiation time\* (instead of infinite as conservatively used in the generic study) would result in a reduction in the total integrated core decay energy. Furthermore, in the generic calculations the fraction of core decay energy deposited in the coolant in the reactor core region was based on the *mass density* fraction of water, fuel, and cladding within the core region. However, consideration of the different decay gamma energy levels and a detailed evaluation of the cross sections showed that a conservative value of 8% of the gamma decay energy is deposited in the core water. Since the gamma energy is approximately 60% of the core total decay energy and since all of the beta energy would remain within the fuel rod, the resulting amount of the core *total* decay power deposited in the reactor core water would be less than 5%, therefore, a decay energy fraction of 5% was used in the specific section while a conservative value of 11% of the core decay energy fraction was used in the generic sections. This reduction in gamma energy absorption in the core water to 5% coupled with the decay power generation for a 1000-day irradiation results in an integrated hydrogen production at 100 hours of 7,700 SCF instead of 23,000 SCF calculated using the higher value. A lower hydrogen production rate has three effects: *first*, it allows a longer waiting period before venting is required; *second*, it results in decreased venting rates; and *third*, it reduces the total fission products released from the containment.

Consideration of the long-term post-LOCA time period could result in a reduction in the total fraction of core decay power being available for radiolysis in the core region. This reduction in radiolysis would be a function of the total amount of water within the core region. The water swell within the reactor vessel does provide continuous and adequate core cooling even with the collapsed water level (i.e., two separate and distinct phases—steam and liquid) below the top of the active fuel. For this condition of void distribution within the core region, the fractional amount of decay energy deposited in water is reduced since the quantity of water would be less by a maximum factor of 1.5.

### 2. Suppression Pool Region

The energy decay fraction contained in the TID release is plotted versus time on Figure 13. Note that the time average release fraction is initially approximately 14% and decreases as a function of time (i.e., 6.5% in 3 days). Of this decay energy, the 100% noble gases and 50% halogens equally comprise essentially all of the decay energy. Furthermore, the beta energy fraction is about 30% of the total energy contained in the halogens. Essentially all of the energy, beta and gamma, associated with those halogens captured in the pool water would be available for radiolysis. With an infinite partition factor for the suppression pool, all of the halogen energy would be absorbed in the pool water. This boundary condition would result in the maximum amount of hydrogen and oxygen generated due to radiolysis. Equation (A.1) would give the effective hydrogen generation following a LOCA.

$$\text{Moles H}_2 \propto \left\{ G_{\text{H}_2} \text{ core} \times (1 - \text{Rel}) \times f_1 + G_{\text{H}_2} \text{ pool} (f_2 \times \text{Rel}) F_3 \right\} \quad (\text{A.1})$$

where:

$$G_{\text{H}_2} = \frac{\text{molecules H}_2 \text{ generated}}{100 \text{ eV absorbed}}$$

Rel , fraction of decay energy released from core

$f_1$  , fraction of remaining core energy into water, 0.05

$f_2$  , fraction of released energy into pool,  $\sim 0.5$  if all halogens and solids absorbed

$f_3$  , fraction of energy into pool absorbed by water, assumed to be 1.0

Both Rel and  $f_2$  vary as a function of time

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\* Equivalent to a core average exposure of approximately 19,000 MWd/ton.

For a stoichiometric release, the amount of O<sub>2</sub> generated by radiolysis would be:

$$\frac{\text{Moles O}_2}{\text{(radiolysis)}} = \frac{1}{2} \times \frac{\text{moles H}_2}{\text{(radiolysis)}} \quad (\text{A-2})$$

Consideration of the extent of H<sub>2</sub> generated by radiolysis for the TID release compared to using the GE release\* is given by equation (A.3):

$$\frac{\text{Moles H}_2 \text{ (TID)}}{\text{Moles H}_2 \text{ (GE)}} = \frac{G_{\text{H}_2 \text{ core}} \times (1-\text{Rel}) \times f_1 + G_{\text{H}_2 \text{ pool}} (f_2 \times \text{Rel})}{\sim G_{\text{H}_2 \text{ core}} \times f_1} \quad (\text{A-3})$$

or

$$\frac{\text{Moles H}_2 \text{ (TID)}}{\text{Moles H}_2 \text{ (GE)}} = (1-\text{Rel}) + \frac{G_{\text{H}_2 \text{ pool}} (f_2 \times \text{Rel})}{G_{\text{H}_2 \text{ core}} \times f_1} \quad (\text{A-4})$$

Figure 15 shows the ratio of hydrogen generated due to radiolysis for both the TID release and the GE release as a function of time. It is evident from this figure that the use of TID release with all of the energy associated with the halogens and the solids would result in a maximum increase in radiolysis of approximately 100% over the radiolysis associated with GE assumptions and would decrease with time. Also included in Figure 15 is the fraction of the hydrogen generated in the suppression pool for the TID release compared to the total H<sub>2</sub> generated for the TID release (i.e., pool and core) as a function of time. A maximum of approximately 60% of the radiolytic decomposition (H<sub>2</sub> production) would occur in the pool and the fraction would decrease with time to approximately 30% after 10 days.

It should be pointed out that if all of the energy associated with the halogens were deposited in the suppression pool the fission product activity due to halogens in the containment atmosphere would be zero. The other extreme would be for all of the halogen activity to remain in the gas phase, thereby reducing the overall radiolytic hydrogen to less than that calculated using the GE model. Actually, plateout and partition factor would reduce the airborne activity significantly such that the radiolytic hydrogen production in the pool could be mostly due to the energy decay associated with the halogens. The airborne halogen activity would then correspondingly be significantly reduced from the activity associated with 50% of the core inventory of halogens assumed to be released with the TID source term.

Therefore, in summary, combustible gases generated in either the core region from metal-water reaction and/or by radiolysis or in the pool region by radiolysis would be reasonably mixed in the primary containment due to thermal gradients and spray action. Furthermore, even for a significant fission product release fraction from the fuel, the additional H<sub>2</sub> contribution due to radiolysis in the pool would result in a small increase on the total radiolysis, approximately 25%.

\* This release is described in APED-5756(1) and consists of an average of 1.8% noble gases and 0.32% halogens.

\*\* The halogen fission products released to the pool in the GE model would only cause a 1% or 2% increase in the total radiolytic hydrogen production rate. The generation therefore equals approximately G<sub>H<sub>2</sub></sub> (core) x f<sub>1</sub> as shown in equation A-3.

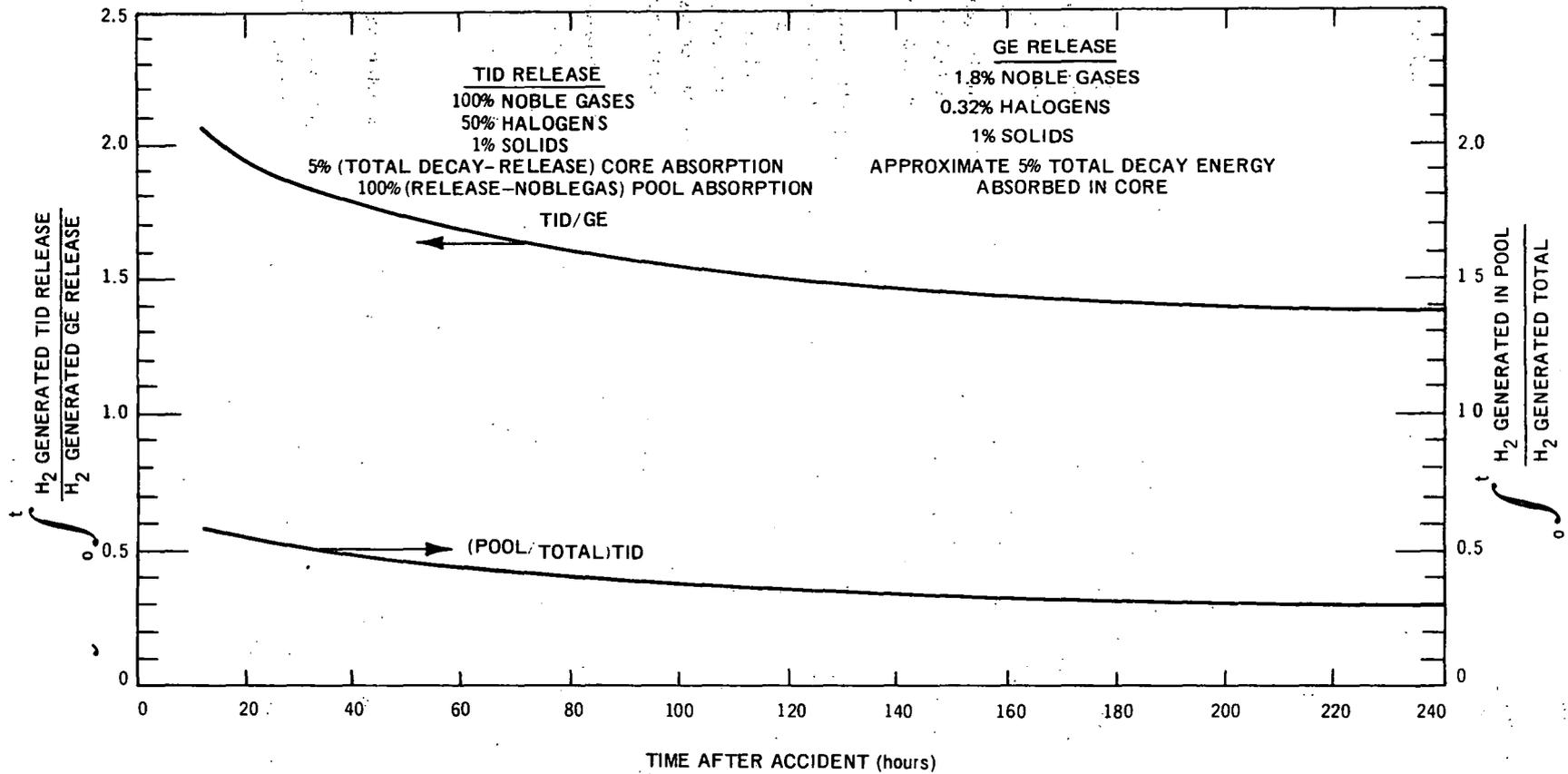


FIGURE 15. RADIOLYSIS FOLLOWING A LOSS-OF-COOLANT ACCIDENT

## APPENDIX B CONTAINMENT VENTING ASSUMPTIONS

The following assumptions were used to determine the doses to the environs for venting from the suppression chamber of a standard 3440 MWt BWR nuclear power plant.

**Table B-1**  
**CONTAINMENT VENTING ASSUMPTIONS**  
**FOLLOWING DESIGN BASIS LOCA**

	Assumed Value For AEC Doses	Calculated or Measured Value	Margin Present
1. Metal-Water Reaction	0.3%	≤ 0.06%	≥5
2. Fission Product Release			
Noble Gases	100%	1.8%	56
Halogens	50%	0.32%	~75
Solids	1%		
3. Fuel Rod Perforations	100%	≤ 13%	≥7.7
4. Plateout Reduction Factor			
Halogens (Elemental Iodine Only)	2	25	12.5
5. Core Decay Energy	1000-Day	NA	NA
	Irradiation		
6. Radiolytic Hydrogen Production*			
a. Energy Deposited			
(1) Core	5%	<5%	<2
b. Net Hydrogen Yield	0.2	0.05-0.2	1-4
Molecules H <sub>2</sub> /100 eV			
7. Venting Rate	21 cfm (10% per day)	NA	NA
8. Time to Initiate Venting	78 hours	NA	NA
9. Methyl Iodide Production	5% of Halogens	1%	5
10. Suppression Pool Fission Product Reduction Factor			
a. Elemental Iodine	10	>40	>4
b. Methyl Iodide	2	~2	1
11. Standby Gas Treatment Filter Fission Product Reduction Factor			
Halogens	10	100	10
12. Meteorological Parameters	1	4	4**
Same as used for normal leakage,			

\* See Appendix for discussion on radiolytic hydrogen production in both core and pool.

\*\* Includes Reflection Factor, Atmospheric Dispersion.

## APPENDIX C DOSES TO ENVIRONS

The overall doses to the environs as a result of the postulated loss-of-coolant accident are included for both normal leakage and venting. The assumption of TID-14844 fission-product release and the fission-product releases determined using the General Electric model<sup>(1)</sup> are also included. The doses are normalized per MWt of reactor power and are plotted as a function of distance. The fission-product transport model used with the TID-14844 fission-product release to determine the doses is consistent with the assumptions specified in Appendix B. The fission-product transport model used with the GE fission-product release is also given. Reference 1 (APED-5756).

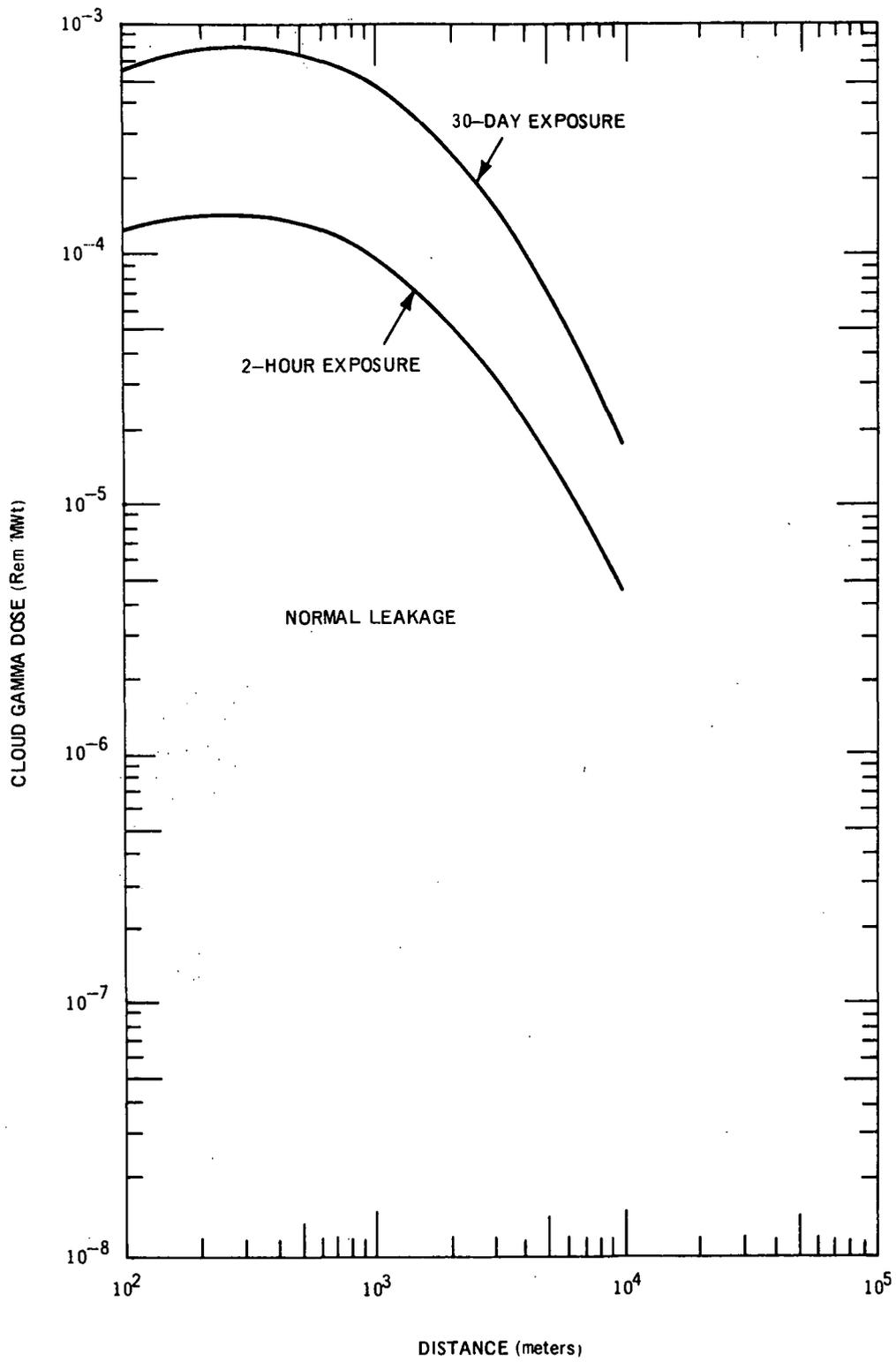


FIGURE 16. CLOUD GAMMA DOSE (Rem/MWt) VERSUS DISTANCE - TID-14844 SOURCES. LEAKAGE RATE - 0.635%/day.

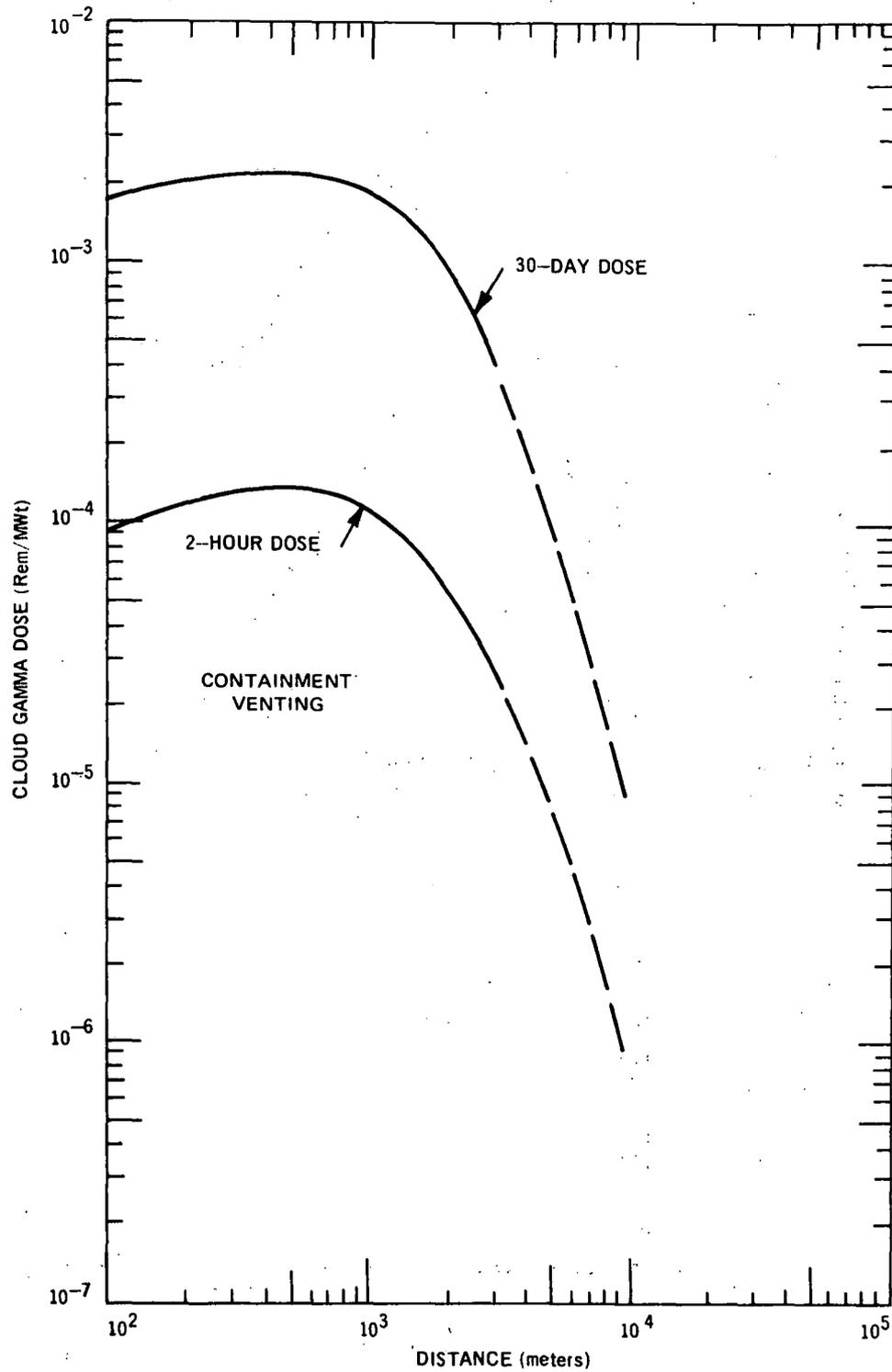


FIGURE 17. CLOUD GAMMA DOSE VERSIS DISTANCE. AEC SOURCES. CONTAINMENT VENTING - 78 HOURS DECAY PRIOR TO VENTING.

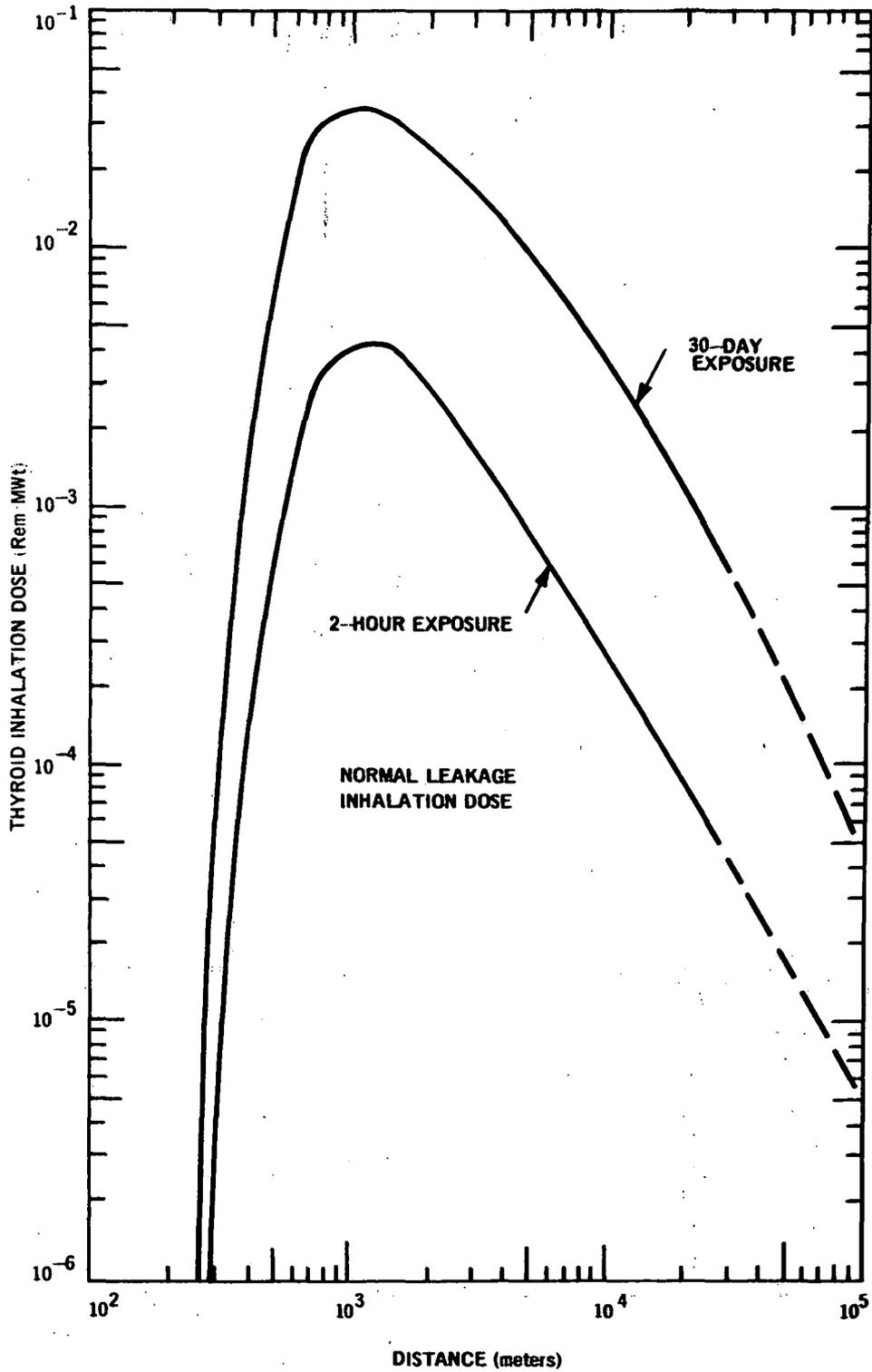


FIGURE 18. THYROID INHALATION DOSE VERSUS DISTANCE - AEC SOURCES. 0.635%/day LEAK RATE.

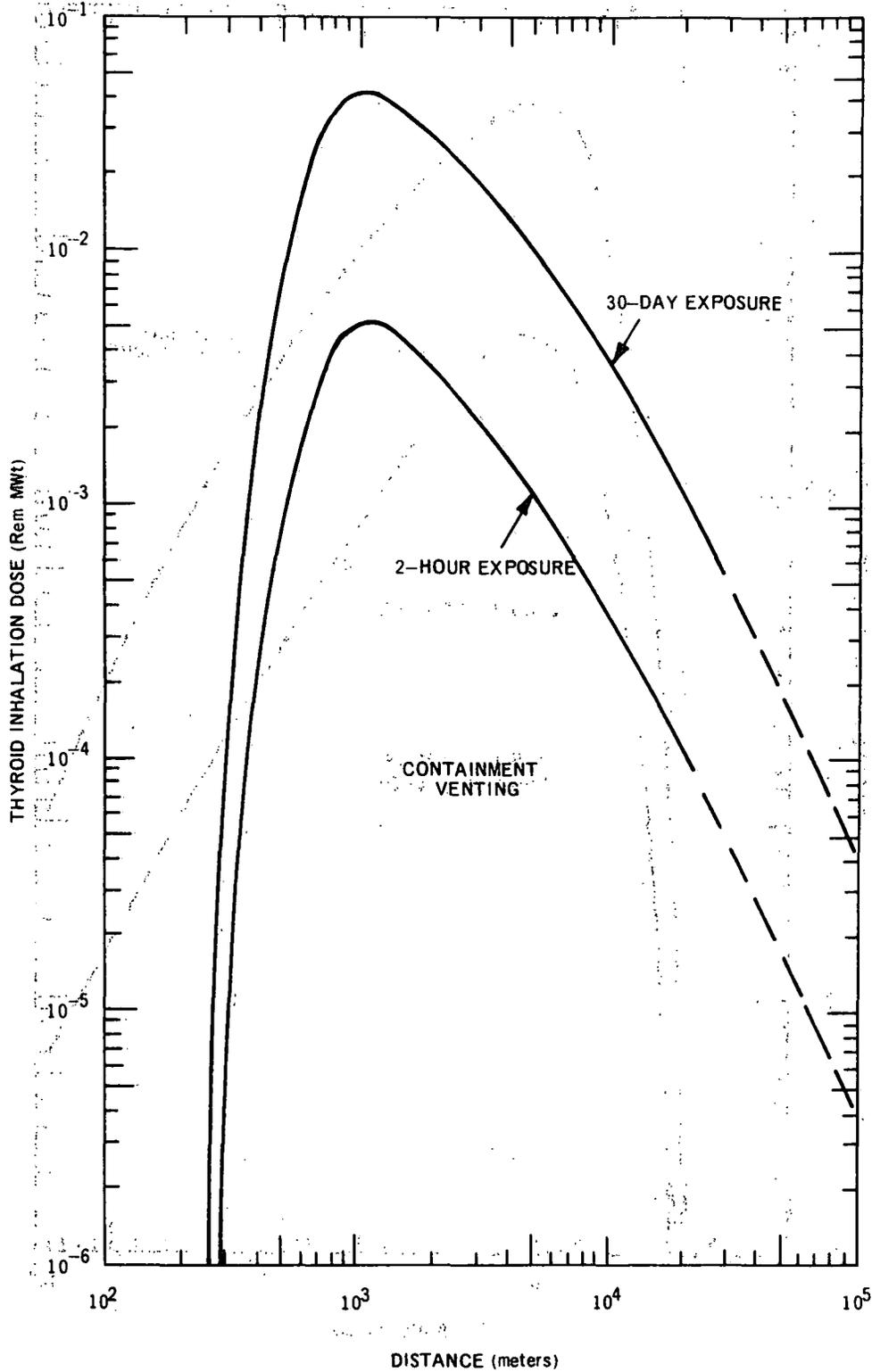


FIGURE 19. THYROID INHALATION DOSE VERSUS DISTANCE - DOSE IN Rem/MWt. TID-14844 SOURCE TERMS USED. 78 HOURS BEFORE VENTING BEGINS.

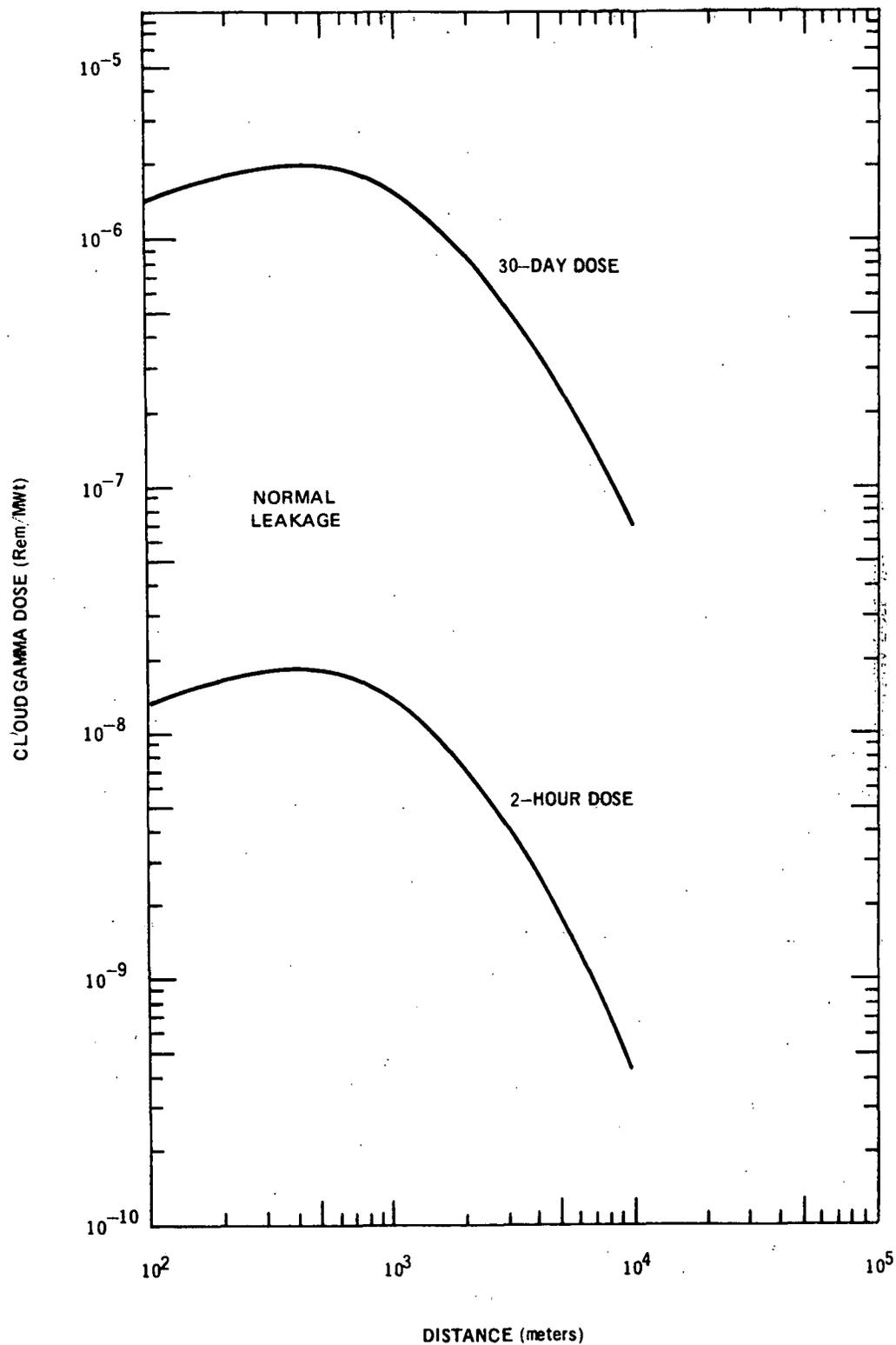


FIGURE 20. CLOUD GAMMA DOSE VERSUS DISTANCE—GE ASSUMPTIONS. NORMAL LEAKAGE 0.5%/day.

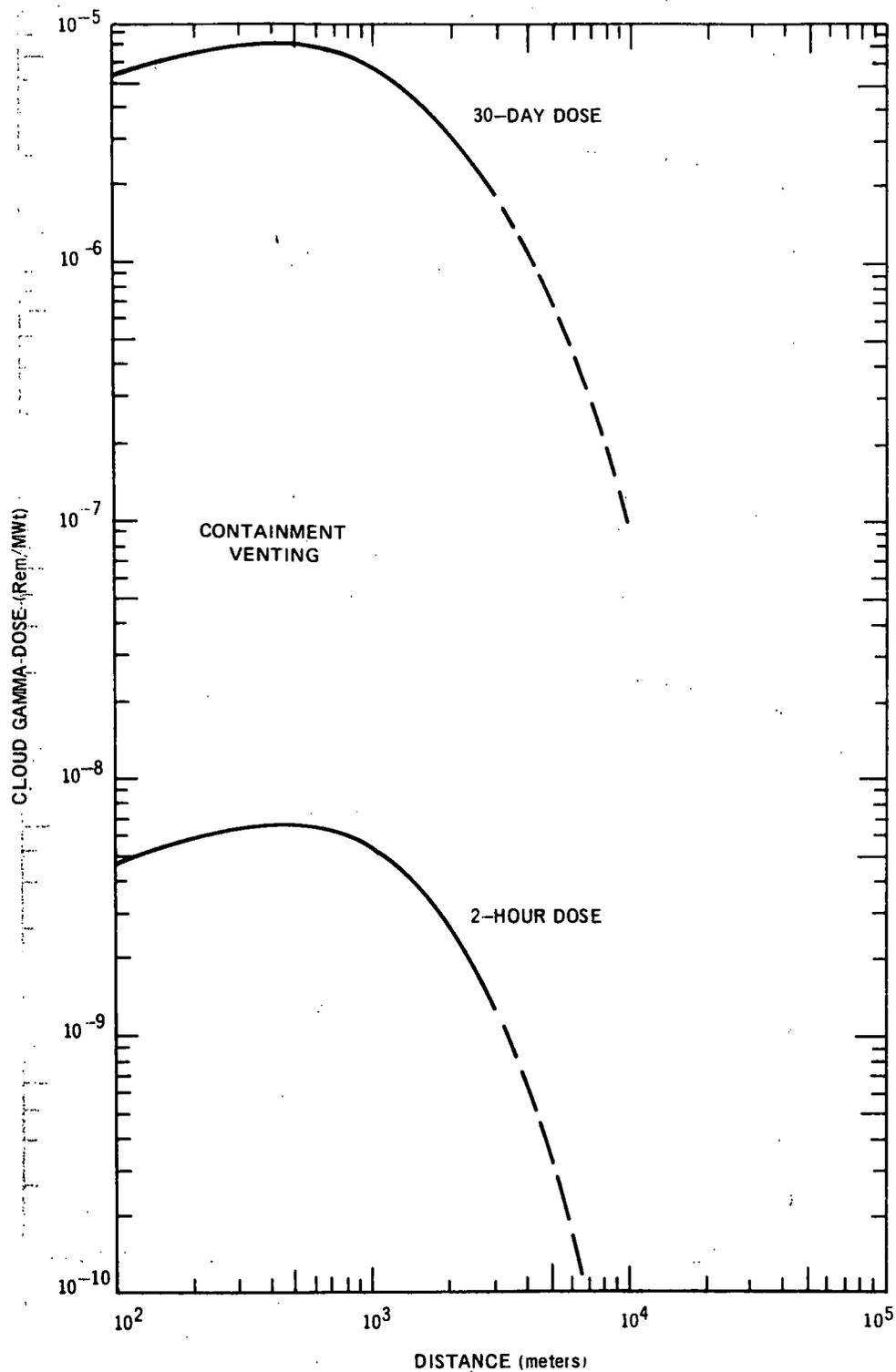


FIGURE 21. CLOUD GAMMA DOSE VERSUS DISTANCE - GE ASSUMPTIONS.  
CONTAINMENT VENTING 8%/day AT 78 HOURS.

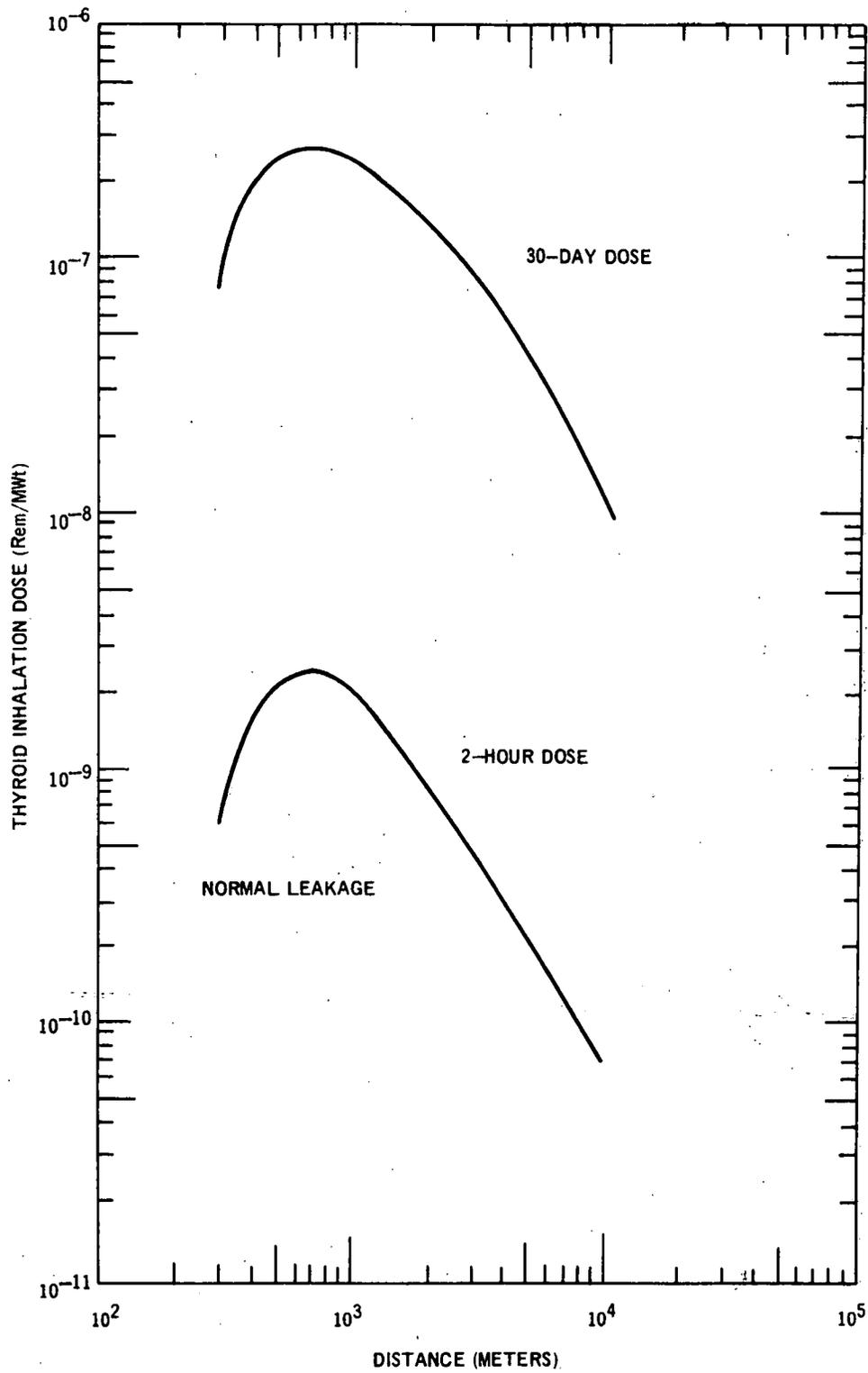


FIGURE 22. THYROID INHALATION DOSE VERSUS DISTANCE—GE SOURCES.  
0.5%/DAY LEAK RATE

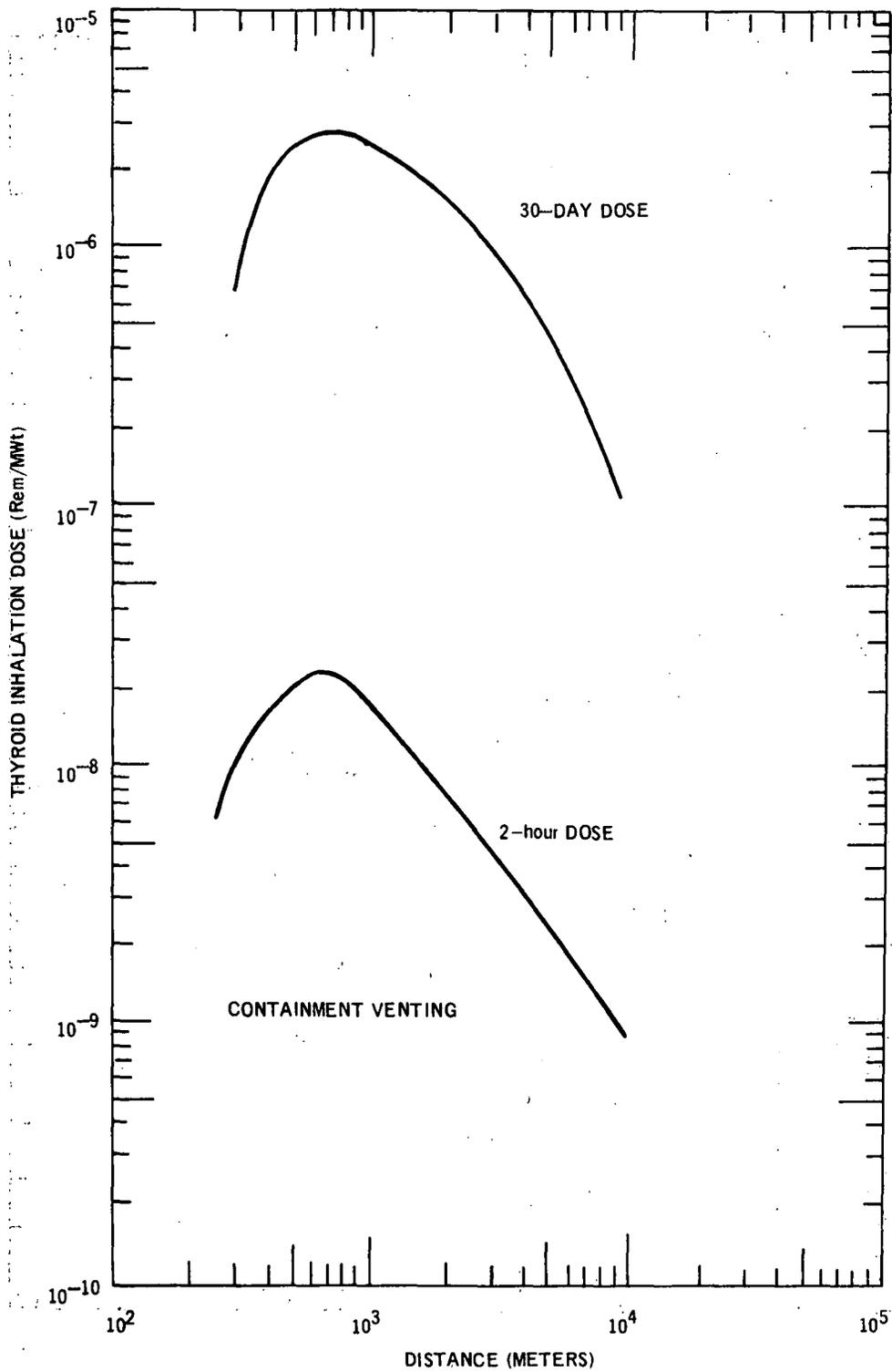


FIGURE 23. THYROID INHALATION DOSE VERSUS DISTANCE-GE SOURCES. 78 HOURS PRIOR DECAY

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