Staff Reports To
The President's Commission On

THE ACCIDENT AT THREE MILE ISLAND

Reports Of The Technical Assessment Task Force, Vol. II

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THREE MILE ISLAND

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REPORTS OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CHEMISTRY

THERMAL HYDRAULICS

CORE DAMAGE

WASH 1400 -- REACTOR SAFETY STUDY

ALTERNATIVE EVENT SEQUENCES

VOLUME II

October 1979
Washington, D.C.

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TECHNICAL ASSESSMENT TASK FORCE

ON

CHEMISTRY

BY

Robert E. English

October 1979
Washington, D.C.

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SUMMARY AND FINDINGS

The following topics at TMI-2 were reviewed: zirconium's reactions with water and the fuel uranium dioxide (UO_2); the fission products released from the fuel elements; and the hydrogen produced and its likelihood of exploding in the reactor vessel. The following are the findings:

- From the measured fission products, approximately 50 percent of the core is inferred to have exceeded 4,000°F.

- Ninety percent or more of the fuel rods ruptured their zirconium cladding.

- There is a limited amount of experimental data (references 4-6 and 32) indicating that the UO_2 fuel can dissolve in partially oxidized liquid zirconium (Zr) at about 3,450°F (1,900°C). The significance of the data is that some liquid reactor fuel could result from temperatures well below the 5,200°F temperature for melting UO . Chung (reference 4) further postulates that there is tie possibility of forming a low-temperature-melting eutectic between UO and either unsaturated alpha-phase or beta-phase zirconium at temperatures of 2,400-3,360°F (1,300-1,850°C). Honekamp (references 32 and 34) shows that the total amount of liquid formed must have been small at TMI-2.

- Some of the UO fuel is finely divided, and its fission products are being slowly leached by the reactor coolant.

- The amount of hydrogen burned in the containment building was estimated in two ways: from the height of the pressure pulse; and from the composition of the atmosphere in containment after the combustion. The amounts of hydrogen burned, based on these two observations, are as follows;

  -- pressure pulse: 294 pound-moles; and
  -- gas composition: 436 + 33 pound-moles.

- Additional amounts of hydrogen present after the combustion were as follows:

  -- H in containment atmosphere: 79 $\pm$ 4 pound-moles; and
  -- H^2 bubble in reactor loop: 92 + 22 pound-moles; and
  -- H2 dissolved in reactor coolant: 36 pound-moles.

- The total amounts of hydrogen generated are then as follows: 500 + 22 and 642 $\pm$ 40 pound-moles -- estimates with extreme ranges from 478 to 682 pound-moles. These extreme values range from 44 to 63 percent of the hydrogen capable of being produced from reacting with water, all the zirconium in the reactor. In turn, 44 to 63 percent of the zirconium in the core has been oxidized to produce hydrogen.

- **Because 18 percent oxidation severely embrittles zirconium, the upper 60 to 70 percent of the fuel clad is so embrittled that it has lost its structural integrity.**

- **In pressurized water reactors, oxygen produced by radiolysis is promptly consumed by an excess of hydrogen that is deliberately dissolved in the reactor coolant. Because of this back-reaction, there was never enough oxygen in the reactor loop for a hydrogen-oxygen explosion.**

- **At the time of the accident, information on the use of hydrogen to suppress the accumulation of radiolytically formed oxygen was available from some staff members of the Nuclear Regulatory Commission (NRC), from the national laboratories and the reactor manufacturers, and from textbooks on the chemistry of water for nuclear power plants.**

- **WASH 1400 (Rasmussen Report) concludes that an explosion .or detonation within a containment building of the type approved by NRC of all the hydrogen capable of being produced from the reactor's zirconium would not violate its ability to contain. An independent assessment by Los Alamos Scientific Laboratory (LASL) found that the containment building at TMI-2 may be marginal in its ability to withstand such a detonation of all the hydrogen which could be produced.**

- **The hydrogen production rate at TMI-2 was of the order of 500 times the capacity of the existing recombiner.**

INTRODUCTION

A number of topics in the chemistry and radiochemistry of the reactor at TMI-2 are important to understanding the accident. These are covered in this report. They are:

- the reactions of the nuclear fuel's zirconium clad, with both the cooling water of the reactor and its fuel, uranium dioxide;
- the information that measurements of released fission products tells about damage to the fuel;
- the hydrogen bubble in the reactor vessel, and the likelihood that it might have exploded;
- the hydrogen explosions in the containment building; and
- the recombination of hydrogen.

ZIRCONIUM REACTIONS

At temperatures up to about 3,000°F, the combined characteristics of zirconium, water, and UO_2 are well known (references 24 and 31). Inasmuch as the Zircaloy-4 clad is almost pure zirconium (98 percent zirconium), the discussion below focuses on the zirconium, although the experimental programs investigated the alloys as well as the pure metal.

Zirconium is a rare metal that has especially valuable properties as a clad for reactor fuel elements. It readily conducts the heat from the UO into the water, and its 3,320°F melting point is about 525°F above that of iron. It has the particularly desirable quality of not capturing many neutrons, thereby saving them so that they may be used to produce fission of the uranium.

At high temperatures, the zirconium also can react with water to produce hydrogen in the following way:

$$Zr + 2\ H_2O\text{-}ZrO_2 + 2\ H_2 \qquad (1)$$

Heat is also produced, and the oxidation of the zirconium can make it brittle if too much occurs. No oxygen is produced. The way to avoid this reaction is to keep the zirconium from becoming too hot. This problem is recognized in the design of all water-cooled commercial reactors in the United States, in part because the NRC requires that zirconium be kept below critical temperatures (reference 1). Therein the operating conditions of the zirconium are specified to remain within the following limits even during the design-basis accident: (1) peak clad temperature may not exceed 2,200°F; (2) oxidation may not exceed 17 percent of cladding thickness; and (3) hydrogen generation may not exceed one percent of that which would be produced if all the zirconium in contact with the fuel were to react. The ranges of operating conditions required to avoid these problems were specified in 1962 (reference 2). At TMI-2, all three limits were exceeded.

The following examples will give some scope to the problem for the TMI-2 accident.

The Baker-Just equation (specified in reference 1 and discussed in reference 2) gives the oxidation rate, as follows:

$$\frac{dT}{dt} = \frac{0.3937}{T} \exp\left(-\frac{E}{RT}\right) \qquad (2)$$

where

T = zirconium thickness oxidized, cm
t = time, s
E = activation energy, 45500 cal/g-mole
R = 1.987 cal/g-mole -K
T = temperature, K

A similar equation by Cathcart and Pawel uses newer values for the coefficients and is, therefore, also commonly used (reference 4 discusses limitations of these equations).

If operated for 3 years at its normal temperature of about 650 °F, zirconium would oxidize to only 1/1000 to 1/100 of the second and third limits. If during an accident the zirconium were to reach 2,000°F, the 17 percent oxidation limit would be reached in just under an hour. Because the zirconium-oxidation rate varies with time, even at a given temperature (equation 2), and because the afterheat from radioactive decay of the fission products also varies with time, a direct comparison of the energies produced by the zirconium-water reaction and by the fission products requires a somewhat arbitrary selection of conditions. For the decay heat from the fission products, the time 140 minutes after trip was chosen, this being the time that the pilot-operated relief valve (PORV) was closed and the reactor began to heat up. In one minute at 2,000 °F, the heat produced by the zirconium-water reaction would be roughly double the decay heat produced in the 140th minute.

The zirconium-water reactions as they concern the TMI-2 accident have been summarized by Cohen in reference 3. Chung and Honekamp of Argonne National Laboratory (ANL) have summarized the literature for oxidation at 2,900°F, and above, and for clad-fuel interactions in the vicinity of 3,450°F (references 4 and 32).

FINDINGS

- The need to keep the zirconium below 1,000°F is well known and is a basic principle in reactor design.

- At temperatures above 1,500°F, the zirconium oxidizes in a steam atmosphere and produces hydrogen, but no oxygen. This oxidation leads to embrittlement of the clad and clad failure.

- At 3,450°F or above, partially oxidized zirconium can melt.

- There is a limited amount of experimental data (references 4-6 and 32) indicating that the UO_2 fuel can dissolve in this liquid, partially oxidized Zr. The significance of the data is that some liquid reactor fuel could result from a temperature well below the 5,200°F temperature for melting UO_2.

- Chung (reference 4) further postulates the possibility of forming a low-temperature-melting eutectic between UO and either unsaturated A-phase or B-phase zirconium at temperatures of 2,400-3,360°F (1,300-1,850°C).

- Honekamp (reference 32) shows that even at temperatures as high as 3,800°F (2,100°C) the total amount of liquid fuel must be small; English, however (reference 34) shows that the amount of fuel that could be dissolved is much higher at 4,352°F (2,400°C).

FISSION PRODUCTS

Measurements of the fission products released provide information on the extent of fuel damage during the accident. These fission products could be either gases that escaped to the atmosphere of the containment building, or substances dissolved in, or transported by, the reactor's cooling water. In either case, the damage is assessed by comparing the measured fission products with the total amount of that species produced by the reactor.

England and Wilson at LASL analyzed the operating history of the TMI-2 reactor, and from that history computed the quantities of the various fission products and actinides that were generated (reference 7). They also determined the amounts of these various radionuclides that remained at any given time after the accident, as well as the total quantity of decay heat that resulted from their radioactive decay. The computer codes CINDER and EPRI-CINDER were used for the analysis.

Samples of the reactor coolant at TMI-2 were taken from the let-down line first on March 29, and later on April 10, 1979. The first sample was sent to Bettis Laboratory for analysis of its fission products and the second was sent to Savannah River, Oak Ridge National Laboratory (ORNL), Bettis Laboratory, and Babcock & Wilcox (B&W); the results (reference 8) are presented in Table 1. Because the radioactivity of the various species declines with time, the decay of each individual species, from the time of shutdown on March 28, 1979, to the dates shown, is taken into account in computing the fractions of core inventory in the coolant.

On March 31, 1979, a gas sample was withdrawn from the air in the containment building, and the radioactivity of the sample was measured by Bettis Laboratory. The results are given below (reference 8):

Isotope	Concentration, P Ci/ml
Xe-133	676.000
Xe-133 m	16.000
Xe-135	8.100
1-131	0.063
1-133	below 0.03

The following evaluations of fuel damage are attributed to Bettis Laboratory (reference 9) following the water sample on March 29, and then the air sample on March 31, 1979: (1) Most of the volatile fission products were released to the reactor coolant, and 2 to 12 percent of the fuel reached 3,000-4,000°F. Based on this, and the amounts of strontium, barium, and uranium present, it was determined that little, if any, of the fuel melted. (2) About 90 percent of the 36,816 fuel rods burst their clads, and about 30 percent of the reactor fuel exceeded 3,500°F. Again, little, if any, of the fuel melted.

Cohen (reference 10) concludes that 57 percent of the xenon was released and that some of the fuel is probably in a finely divided form from which fission products are slowly being leached by the reactor's cooling water. Rest of ANL (reference 11) points out that the release

		1st Primary Coolant Sample Analyses Corrected to 3/30/79		2nd Primary Coolant Analyses corrected to 4/11/79							
			Bettis	SRL		ORNL		Bettis		B&W	
Nuclide	T 1/2	Coolant concentration (μ Ci/cc)	Fraction* of Core in Primary Coolant	Coolant concentration (μ Ci/cc)	Fraction* of Core in Primary Coolant	Coolant concentration (μ Ci/cc)	Fraction* of Core in Primary Coolant	Coolant concentration (μ Ci/cc)	Fraction* of Core in Primary Coolant	Coolant concentration (μ Ci/cc)	Fraction* of Core in Primary Coolant
I-131	8d	1.4×10^{4}	0.095	4.5×10^{3}	0.086	8.2×10^{3}	0.155	8.5×10^{3}	0.16	6.7×10^{3}	0.13
I-133	20.8h	6.8×10^{3}	0.083								
CS-134	2Y	6.3×10^{1}	0.068	7.7×10^{1}	0.086	8.2×10^{1}	0.091	7.4×10^{1}	0.083	7.3×10^{1}	0.082
CS-136	13d	1.8×10^{2}	0.10	1.2×10^{2}	0.12	1.1×10^{2}	0.12	1.1×10^{2}	0.12	9.5×10^{1}	0.10
Cs-137	30Y	2.7×10^{2}	0.11	3.2×10^{2}	0.13	3.3×10^{2}	0.13	3.4×10^{2}	0.13	2.8×10^{2}	0.11
Sr-89	50d	5.4	0.000031	1.5×10^{3}	0.009	6.0×10^{2}	0.0042	7.3×10^{2}	0.004		
Sr-90	29Y.					5.0×10^{1}	0.022				
Ru-106	368d	3.6×10^{-1}	0.000039								
Ba-140	12.8d	2.1×10^{1}	0.000066	1.7×10^{2}	0.0010	2.9×10^{2}	0.0018	2.2×10^{2}	0.0014	1.5×10^{2}	0.00094
La-140	40h			1.4×10^{2}	0.00075	1.6×10^{2}	0.00086	1.4×10^{2}	0.00075		
Mo-99	66h			1.3×10^{2}	0.012	1.8×10^{2}	0.017			1.3×10^{2}	0.012
Te-132	78h	2.0×10^{2}	0.0012								
Ce-141	32d			1.05×10^{2}	0.00051						
Ba-136m	0.35s daughter of Cs-136			9.0×10^{1}	0.61						
Gross α		3.6×10^{-4}	1×10^{-10}	$<1 \times 10^{-3}$	$<1 \times 10^{-8}$	$<4.5 \times 10^{-4}$	$<4.6 \times 10^{-9}$	1.3×10^{-3}	1.3×10^{-8}		
U				<1 ppb		<20 ppb					
H^3	12Y					1.2	0.1				
pH				7.0		8.0		7.7		8.4	

*Based on a primary coolant volume of 3.8×10^{8} ml. If the make-up water (9×10^{8} ml) from the BWST is included and considered to be at the same concentration, the fraction of the core in the coolant would be about a factor of 3 higher.

TAHIF. 1: Fission Products in the Aaactor Ooolant frau Sales Taken on March 29 and April 10.

of fission products from the fuel depends on the degree of grain-boundary separation which, in turn, depends on the fuel's operating history. Lorenz of ORNL (reference 12) concludes that the sizeable release of gaseous fission products could be produced from having 40 percent of the fuel at 4,350°F (2,400°C) and the remainder at lower temperatures. Another interpretation is that the 57 percent release of xenon, all from the upper, hot two-thirds of the core, requires temperatures over 4,000°F throughout this region.

None of these temperature estimates is so high that the UO_2 itself would melt (5,200°F), but the estimates are generally high enough for formation of a ternary liquid of Zr-ZrO_2 -UO_2.

FINDINGS

- Approximately 50 percent of the core exceeded 4,000°F.
- Ninety percent or more of the fuel rods ruptured their zirconium clad.
- Some of the UO fuel is in a finely divided form from which fission products are being slowly leached by the reactor coolant.

HYDROGEN BUBBLE

Clearly, large amounts of hydrogen were produced by the reaction in equation one. This hydrogen created a bubble, or bubbles, of noncondensible gas in the reactor loop, dissolved in the reactor coolant, and escaped to the containment building, where it, in part, exploded. An inventory of all this hydrogen will be useful in assessing damage to the zirconium cladding of the UO_2 fuel from its reaction with water.

HYDROGEN BURNED IN CONTAINMENT

In constructing the inventory, a set of simultaneous measurements would be valuable, but these are lacking.. The hydrogen explosion in the containment building occurred on March 28, 1979. About 9 hours and 50 minutes after the accident began, pressure in the containment building rose above the setpoint of 28 pounds per square inch gauge (psig). That pressure initiated the water spray into the building, but it did not indicate the actual height of the pressure pulse. On the other hand, the reactimeter recorded data on the pressures in the steam generators, these pressures being the difference between steam pressure and building pressure; they were recorded every 3 seconds. The Electric Power Research Institute (EPRI) analysis of these data (reference 13) shows the following: (1) The magnitude of the pressure pulse was not greater than 28 psi; and (2) the time for the pressure to rise to its peak was between 6 and 9 seconds after the rise began. Overall, the pressure pulse was a relatively slow, gradual process and at least in the vicinity of the pressure sensors on the steam generators. No detonation was apparent.

The amount of hydrogen burned was estimated by the staff in two ways: first, from the size of the 28 psi pressure pulse; and second, from the depletion of oxygen in the air within the containment building.

The containment building is described in several places in the Final Safety Analysis Report (FSAR) for TMI-2. In paragraphs 3.8.1.1 and 6.1.1.1, the building is described as a vertical cylinder resting on a flat floor and topped by a dome that is a portion of a sphere. The inside diameter of the cylinder is 130 feet; the height of the cylindrical wall is 157 feet; and the radius of the spherical dome is 110 feet. In Table 6.2-1 and in paragraph 6.2.5.1, the free volume is given as 2.1 million cubic feet; in Table 6.2-9A, the free volume is stated to be 2,116,000 cubic feet.

The initial atmospheric conditions were taken as dry air mixed with hydrogen at 120°F and one atmosphere of pressure; final pressure was taken as 28 psig. The combustion was assumed to be adiabatic, to burn all the hydrogen, but to burn nothing else. The gas properties used (references 14 and 15) incorporate the changes in molecular species as well as the variations in their thermodynamic properties with temperature.

The results showed that the containment atmosphere contained 294 pound-moles of hydrogen, or 5.9 percent hydrogen by volume. Air temperature at the end of this combustion was 1,277°F. If the mixture

before combustion was not uniform (as assumed), the quantity of hydrogen burned would be unchanged, but the temperatures after combustion would vary with the local mixture and would be both higher and lower than 1,277°F. Although the air in containment was assumed to be dry initially, humidity has only a minor effect on the amount of hydrogen needed to reach 28 psig.

Measurements of gas composition within the containment building also indicate the quantity of hydrogen burned because the hydrogen's combustion depletes the oxygen. On March 31, 1979, the first samples of the containment atmosphere were obtained; these two samples contained 1.7 and 1.9 percent hydrogen (reference 9). The oxygen levels were 15.7 and 16.5 percent (reference 13). Later measurements indicated both higher and lower oxygen concentrations. Why the oxygen level rose and fell with time is not clear, apart from experimental error. Some indication of the experimental errors is given by the April 1, 1979, data in Cohen (reference 9). The text indicates that the hydrogen concentration was measured eight times between 7:00 a.m. and 10:00 p.m. and that the concentration was 2.3 $\pm$ 0.3 percent hydrogen but it does not state whether the $\pm$ 0.3 is the standard deviation or the extreme range of the data.

The average values of the data obtained on March 31, 1979, were used in this report. The averages were used because the variations between the measurements on March 31 are probably not significant, the experimental errors shown above being considered. March 31 was also the first day that the bubble volume in the reactor loop and the composition of the containment atmosphere were both measured; the nearly simultaneous taking of these measurements is important in constructing a hydrogen inventory. The following composition is thus assumed for the containment atmosphere on March 31 on a dry-gas basis: 1.8 + 0.1 percent hydrogen, 16.1 $\pm$ 0.4 percent oxygen, 82.1 + 0.5 percent remainder (81.06 percent nitrogen plus 1.04 percent argon).

At the time of the explosion, much water vapor had been added to the containment atmosphere, so 100 percent relative humidity is assumed. Also, air temperature is taken to be 120°F, close to the 117°F on March 29, 1979, (reference 9). The result is that 436 $\pm$ 33 pound-moles of hydrogen are required in order to produce the measured depletion of oxygen, 148 percent of the 294 pound-moles computed to produce the 28-psi pressure spike. Thus, the hydrogen burned is taken to range from 294 to 469 pound-moles.

HYDROGEN INVENTORY

To these figures must be added the 1.8 $\pm$ 0.1 percent hydrogen present in the containment building's atmosphere on March 31, 1979, or 79 + 4 pound-moles.

On March 31, 1979, the hydrogen bubble was described as follows (reference 9): "The calculated volume of noncondensible gas was approximately 823 cubic feet at a reference pressure of 875 psia"; the uncertainty in this value is given as $\pm$ 200 cubic feet. For the purposes of this report, the bubble is taken as 823 + 200 cubic feet of

hydrogen at 875 psia at the loop temperature of 280°F; the resulting quantity of hydrogen is 91 + 22 pound-moles. In addition, 36 pound-moles of hydrogen were dissolved in the reactor coolant. The amount dissolved in the water within the containment building is negligible.

The sum of all these quantities of hydrogen ranges from 500 + 22 to 642 + 40, or 478 - 682 pound-moles of hydrogen.

The values herein differ significantly from the values presented in the NRC's "The Evaluation of Long-Term Post-Accident Core Cooling of TMI (reference 16 at A-12); NRC values are attributed to Butler (reference 17). The principal difference is in the hydrogen burned in the containment building.

Marino (reference 25) estimates the total amount of zirconium in the reactor at 49,711 pounds. Because reacting all this zirconium with water could produce 1,090 pound-moles of hydrogen, the inventory totals above amount to 44 to 63 percent of the amount theoretically possible. The proportion of zirconium embrittled severely by oxidation exceeds these proportions because even 18-percent oxidation produces severe embrittlement (reference 31).

GETTING RID OF THE HYDROGEN BUBBLE

Cohen (reference 19) and Jenks (reference 20) appraised the means by which the hydrogen bubble was removed from the reactor loop. Each of these references estimates that the differential solubility of hydrogen in water is not sufficient by itself to explain the rate at which the bubble disappeared, although this mechanism was a big contributor. Cohen postulates that a significant amount of gas may have leaked past the 0-ring seal between the reactor vessel and its head. During the period in which considerable hydrogen was trapped in the reactor vessel, the gas had access to this seal and could have escaped through any leak there. When the water level rose and covered this seal, this gas leakage would no longer have been possible, but it could have been a significant contributor for 2 days or so.

Another possibility is that the high temperatures reached at the reactor outlet may have overheated and damaged the synthetic rubber 0-rings that seal the top of the control rod drive mechanisms.

HYDROGEN EXPLOSION IN THE REACTOR?

The complete chronology concerning the views and actions concerning the possibility of a hydrogen explosion in the reactor vessel was reviewed and assessed. Included in this review were chronologies from the NRC (references 18, 27, and 33). This material was critiqued by a group at ANL led by Closs (reference 21). A second group at ANL led by Honekamp prepared a supporting technical analysis of the hydrogen bubble (reference 22). Cohen (reference 23) addressed various mechanisms by which oxygen could be produced within the reactor loop or transported into that loop from outside, by being dissolved in the water of the

emergency core cooling system (ECCS), for example. Jenks (reference 20) also investigated radiolytic production of oxygen within the reactor loop and described his contribution to the NRC's evolving view of the potential for explosion.

Basically, radiolytic decomposition of water always occurs in water-cooled nuclear reactors, both while they are operating and after they have been shut down. Knowledge of this phenomenon and of how to deal with it was evolved long ago and is discussed in considerable detail by Cohen (reference 24). The usual method of dealing with this oxygen (and the one used at TMI-2) is to add hydrogen gas to the make-up supply of water. This is accomplished by merely keeping hydrogen gas above the water in the make-up tank; at a pressure of 4-5 psi, enough hydrogen (20-25 Scc of hydrogen per kilogram of water) will dissolve in this water to suppress oxygen formation within the reactor loop; 0.1 Scc/kg will produce sufficient recombination (reference 18). In fact, the same radiolysis that produces oxygen also stimulates it to recombine with the excess of hydrogen that is present in the water, once more forming water.

When boiling occurs in the reactor, as it did at TMI-2, some of the radiolytically formed oxygen can (before recombination) escape from the liquid into the steam bubbles and be carried out of the liquid into the bubble above. Cohen (reference 23) points out that only about 5 percent of the decay heat is useful in radiolytically decomposing the water and that only 0.225 molecule of oxygen is produced for each 100 eV of radiolytic energy deposited in the water. Because of these factors, 20,000 times as many water molecules are released (or boiled) into the bubble as are oxygen molecules released by radiolysis, even if every oxygen molecule formed by radiolysis (that is, zero recombination) could escape into the bubble. With such a dilute concentration of oxygen, no combustion is possible (reference 30). In any real case, the amount of oxygen released into the bubble would be significantly smaller than the pessimistic case considered here.

From these references, the following overall judgments with respect to oxygen formation or explosion in the reactor vessel can be drawn.

- No explosion within the reactor vessel was possible at any time.

- The largest proportion of oxygen was released to the bubble when boiling first began. Even then, the concentration of oxygen in the bubble was far below any combustible limit.

- As boiling continued on March 28, 1979, and the reactor heated up, hydrogen was formed according to equation one, and this further diluted the oxygen and prevented any combustion within the reactor loop.

- Sixteen hours after the accident began, no additional boiling occurred within the reactor loop, preventing the release of any more oxygen (reference 21). Any oxygen present in the

water would have been completely recombined with hydrogen in less than 5 minutes. Any oxygen in the bubble at that time gradually dissolved in the water and there disappeared by recombination.

In spite of the impossibility of a hydrogen explosion within the reactor vessel, the NRC was greatly concerned about such an explosion from March 30 until April 2, 1979. ANL reviewed the chronology and the judgments concerning the handling of the hydrogen bubble in the reactor vessel at TMI-2 (reference 21) and reached the following conclusion:

> It is clear that the erroneous conclusions about dangerous concentrations of 0 in the H_2 bubble originated from a number of calculations neglecting the important back reaction. . . . Since the radiolysis of water has been studied for decades by radiation chemists, it is hard to understand why none of this country's outstanding radiation chemists were contacted, or as in the case of KAPL and Bettis, were asked so late in the incident. . . . Expertise in radiation chemistry is available at each of the National Laboratories. . . .
>
> Certainly, there was nothing in the TMI bubble incident for which the fundamental science was not well known . . . For example, the all-important H_2-0_2 back reaction, which was left out of the NRC estimates on oxygen formation, is the basis for adding H_2 to the primary cooling system under normal operating conditions.

FINDINGS

From assessments of the amount of hydrogen burned in the containment building, the inventory of all the hydrogen produced at TMI-2, and the handling of the hydrogen bubble, came the following findings:

- From the magnitude of the 28 psi pressure pulse, the amount of hydrogen burned in the containment building was computed to be 294 pound-moles. However, combustion of 436 t 33 pound-moles is required to account for the measured deficit in oxygen in the atmosphere within the containment building.

- Additional amounts of hydrogen present after the combustion were as follows:

  -- H in containment atmosphere: 79 + 4 pound-moles;
  -- H^2 bubble in reactor loop: 91 ± 22 pound-moles; and
  -- H2 dissolved in reactor coolant 36 pound-moles.

- The total amounts of hydrogen generated are then as follows: 500 ± 22 and 642 ± 40 pound-moles, estimates with extreme ranges from 478 to 682 pound-moles. These extreme values range from 44 to 63 percent of hydrogen capable of being produced from reacting with water and all the zirconium in the reactor. In turn, 44 to 63 percent of the zirconium in the core has been oxidized to produce hydrogen.

- Because 18 percent oxidation severely embrittles zirconium, the upper 60 to 70 percent of the fuel clad is so embrittled that it has lost its structural integrity.

- In pressurized-water reactors, oxygen produced by radiolysis is promptly consumed by an excess of hydrogen that is deliberately dissolved in the reactor coolant. Because of this back-reaction, there was never enough oxygen in the reactor loop for a hydrogen-oxygen explosion.

- Information on the use of hydrogen to suppress the accumulation of radiolytically formed oxygen is available from members of NRC, from the national laboratories and the reactor manufacturers, and from textbooks on the chemistry of water for nuclear power plants.

MAXIMUM HYDROGEN EXPLOSION IN THE CONTAINMENT BUILDING

Because only about half the zirconium in the reactor was reacted with water to produce hydrogen, one might ask if the containment building is strong enough to withstand a more severe hydrogen explosion. Marino estimates that the reactor contained 49,711 pounds of zirconium (reference 25). Complete reaction of this much zirconium with water would produce 1,090 pound-moles of hydrogen.

As an extreme case, consider that all this hydrogen was released to the containment building, uniformly mixed with the atmosphere there, and then ignited. Gordon (Reference 26) computed the pressure from a combustion for two cases: (1) thermodynamic equilibrium after a constant-volume adiabatic combustion; and (2) a one-dimensional Chapman-Jouguet detonation. In each case the initial conditions postulated were 120°F, 100 percent relative humidity, and pressure of one atmosphere. The overall results are as follows:

- equilibrium case:
  -- final pressure, psig = 79; and
  -- final temperature, °F = 3,668.

- detonation case:
  -- final pressure, psig = 166; and
  -- final temperature,°F = 4,042.

The containment building at TMI-2 was designed for an internal pressure of 60 psig and has been proof-tested at 69 psig. With its safety factor of 1.5, the building should withstand 90 psig without loss of its containment capability. The concrete shell might develop visible cracks, but the reinforcing steel should maintain the building's structural integrity. The steel plate that lines the inner surface of the walls and dome of the building should fulfill its role as a membrane that would prevent leakage of fission products even if the concrete were to crack. All this indicates that the building would successfully withstand the 79-psig gas pressure that would load the building shell for perhaps 5 seconds and then gradually decline when the water sprays inside the building cool the air.

The detonation case presents a more difficult problem because of the dynamic interaction between the detonation's impulsive load and the elasticity of the building. WASH-1400 (reference 28) concludes that the containment buildings of the type approved by NRC should withstand such a detonation. Los Alamos also evaluated this structural problem for TMI-2 by drawing on their background in explosions derived from the weapons program (reference 29). This evaluation showed that the force from the detonation would be imposed for a period much shorter than the building's periods of natural oscillations. As a result, the building's inertia as well as its strength would be called upon to resist the detonation; and the maximum load, as computed in reference 29, for the structure at TMI-2 would be below, but close to the building's structural limit. Inasmuch as the analyses of the detonation and of the structural dynamics were each on a somewhat simplified basis, additional study is

required before one could conclude with confidence that the containment building at TMI-2 could withstand such a detonation. In WASH 1400, the impulsive load from the shock was found to be less than 2.5 percent of the strength of the structure.

The key reason why the result in reference 29 is so different from that of WASH 1400, is that the duration of the imposed shock loads is so different -- 8 milliseconds in reference 29, and 10 microseconds in WASH 1400. The pressure load from the detonation wave would be imposed upon the wall in the following way: The shock front that is the detonation wave would strike the wall and be reflected, thereby producing a sudden rise in pressure on the wall. This increased pressure level would be sustained until rarefaction waves following the shock reached the wall and lowered the pressure there. Thus, the duration of this pressure loading of the wall would be a fraction of the following time: The radius of the building divided by the speed of sound in the gas within the building. The speed of sound in the product gases is 5,860 feet per second, and the radius of the containment building at TMI-2 is 65 feet. The duration of the pressure pulse on the wall would therefore be of the order of a few milliseconds.

The pulse duration of 10 microseconds in WASH 1400 (Appendix VIII at 123) appears to be based on the thickness of the shock front, rather than on the time interval between the shock wave's and the rarefaction wave's reaching the wall. For this reason, WASH 1400 is probably in error.

Even if this quantity of hydrogen were to be released to the containment building, explosions of either of these magnitudes appear extremely unlikely because of the likelihood that the hydrogen will be ignited before all of it enters the containment building. As TMI-2 demonstrates, the building does contain ignition sources, such as a switch that arcs. The peak pressure and the potential for damage would be reduced if the same amount of hydrogen were burned in several bursts, each individually smaller than the ultimate explosion. Perhaps deliberate introduction of an ignition source for this purpose would be prudent and installation of recombiners, as discussed below, might entirely avoid the problem, in most cases.

FINDINGS

- At TMI-2, combustion of all the hydrogen producible would not exceed the strength of the containment building.

- WASH 1400 concludes that detonation within containment buildings that meet NRC design criteria of all the hydrogen producible from the reactor's zirconium will not violate its ability to contain.

- In an independent assessment by the LASL the strength of the TMI-2 containment building was found to be somewhat above the loads conceivably imposed by detonation of hydrogen within the

building; however, the strength margin was less than the errors in the approximate analysis. Additional study is required to establish with confidence that the building will withstand the detonation.

- In WASH 1400, the method for analyzing containment building tolerance of hydrogen detonations is probably in error.

HYDROGEN RECOMBINATION

The FSAR specifies that a hydrogen recombiner be available for connection to containment and for peacefully reacting any hydrogen in the containment building. This recombiner is sized to remove hydrogen at the rate at which it would be formed by radiolysis from water in the sump of the containment building -- specifically, at the rate of 0.7 pound per hour (FSAR, paragraph 6.2.5). In addition, hydrogen could be produced by the zirconium-water reaction discussed above; the maximum amount of hydrogen allowed by NRC regulations (reference 1) is that resulting from reacting 1 percent of the reactor's zirconium.

A recombiner this low in capacity would require a long time to remove all the hydrogen released to containment at TMI-2. The hydrogen inventory herein treated this amount of hydrogen in two segments: the amount burned and the amount in the atmosphere on March 31, 1979. The oxygen deficit in gas analysis gave the larger value for the amount burned (436 pound-moles of hydrogen), and that value is used here. The hydrogen measured in the gas analysis was 79 pound-moles, for a total of 515 pound-moles. At 0.7 pound reacted per hour, the existing recombiner would require 6 weeks to reduce the hydrogen content of containment to 3 percent, just under the 4-percent limit discussed by Rose (reference 29). Expressed differently, the rate at which hydrogen was produced at TMI-2 exceeded the capacity of the recombiner by a factor of about 500.

Inasmuch as TMI-2 exceeded the capacity of the existing recombiner, some consideration was given during the accident investigation to reacting this hydrogen as rapidly as it was formed. The basic concept was to react the hydrogen with containment air (burn it) in a controlled manner and to vent the combustion products (water vapor, unburned air, and fission products) to the containment atmosphere. The combustion concepts were discussed with Larry Diehl, head of the Combustor Fundamentals Section, and Robert Jones, head of the Combustion Technology Section, both of NASA's Lewis Research Center where considerable research on hydrogen combustion has been conducted.

Two combustor concepts were selected: (1) a catalytic reactor; and (2) a conventional flame-type combustor. Catalytic reactors can readily react hydrogen with air over a wide range of flows and velocities up to a rated inlet-air velocity of 50 to 75 feet per second. This reaction would, for all practical purposes, consume all the hydrogen and release only a negligible unburned fraction to the containment atmosphere. The catalytic reaction of the hydrogen could be initiated by an electrically heated platinum grid on the entrance face of the catalyst bed. In concept, such a catalyst bed 13 inches in diameter could react 220 pounds of hydrogen per hour -- about 300 times the capacity of the present recombiner.

Conceivably, all the zirconium in the reactor could react with water to produce hydrogen -- about 2,200 pounds of hydrogen. Discharging this hydrogen through the catalytic reactor would permit recombination of up to 10 percent of this amount each hour or 100 percent in 10 hours.

A conventional flame-type combustor could handle higher flow rates. Although such a combustor cannot successfully consume all the hydrogen at the very lowest flows (which the catalytic reactor can handle very well), it can react larger flows than the catalytic type inasmuch as an inlet velocity of 150 feet per second is acceptable, and it could tolerate the stoichiometric mixture as well. Such a combustor, 24 inches in diameter, could recombine 2,200 pounds of hydrogen, or all the hydrogen that could conceivably be produced, in one hour's time. Water sprays totaling 300,000 gallons would be sufficient to cool these combustion products and to condense the water vapor produced.

Although neither of these concepts was carried to the design stage, they show that devices for reacting all the hydrogen from the reactor loop can be small in relation to the other components of the power plant. Through their use, the actual explosion in the containment building at TMI-2 as well as the largest hydrogen explosion possible at TMI-2 could have been avoided.

For both the catalytic and flame-type combustors, the discharge of hydrogen from the reactor loop was assumed to pass directly through the combustor. This presumes that the combustors, or recombiners, would all be connected to the reactor coolant vent system and ready to receive the hydrogen. (This contrasts with TMI-2 where several days were required to put the recombiners into service -- according to PNO-79-67F and -67K.) It also presumes that the hydrogen would be vented as designed for. In case of an accident, this might not occur, and the hydrogen could conceivably be vented directly to the containment atmosphere through, say, a ruptured pipe. In that event, the catalytic reactor described above could remove the hydrogen from the containment atmosphere by processing 70 cubic feet of that atmosphere each second. In 8.33 hours, a volume equal to the entire containment atmosphere would pass through the catalyst bed, and the concentration would be diminished by 63 percent. The concentration would be diminished by 90 percent in 19.2 hours and by a further 90 percent in each successive 19.2-hour period of operation. The hydrogen released to containment at TMI-2 could be recombined to below the flammable limit in 10.3 hours rather than the 6 weeks for the existing recombiner at TMI-2.

FINDINGS

- The hydrogen production rate at TMI-2 exceeded the capacity of the existing recombiner by a factor of about 500.

- Alternate concepts assessed only superficially make increasing the recombiner capacity appear feasible.

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REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

THERMAL HYDRAULICS

BY

Robert E. English

October 1979
Washington, D.C.

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SUMMARY AND FINDINGS

Thermal-hydraulic events and processes were analyzed in considerable detail for the first 3-1/2 hours of the accident by means of the Transient Reactor Analysis Code (TRAC) computer code. Also considered were additional selected thermal-hydraulic problems affecting reactor cooling during the period 100-210 minutes from start of the accident. The following are the principal findings:

1. Thermal-hydraulic analysis of the Three Mile Island Unit 2 (TMI-2) reactor loops (or reactor coolant loops) by means of the TRAC computer code fairly accurately reproduces the observed operating conditions over the first 3 hours. At about this time, a peak fuel rod temperature of about 3,900 °F was reached.

2. At 101 minutes after start of the accident, the inability of the reactor coolant pumps to operate while pumping a water-steam mixture having a very high proportion of steam made it necessary to turn off the pumps. Stopping the pumps interrupted the reactor cooling provided by this two-phase mixture, and the reactor fuel elements rose in temperature to 3,500-4,000°F.

3. When the reactor coolant pumps were stopped, water was trapped in the lower portion of each steam generator. The geometry of the reactor loops prevented this water from draining into and cooling the reactor.

4. During the period that the reactor coolant system lacked an adequate supply of circulating water to cool the reactor, natural circulation had the potential to cool the reactor by boiling water in the reactor, condensing the resulting steam in the steam generators and allowing the condensate to flow back into the reactor. This was not achieved for several reasons:

(a) The geometry of the reactor loops did not allow the condensate to drain back to the reactor, as cited above.

(b) The steam pressure on the secondary side of the steam generators was not regulated so as to be lower than the steam pressure in the reactor loops.

(c) After 150 minutes from the start of the accident, hydrogen in the reactor loops prevented flow of steam into the steam generators. Remotely operated vents at the high points in the loops might have permitted venting this hydrogen to the containment building.

INTRODUCTION

This report is concerned with both the flow of water and steam throughout the reactor loops (or reactor coolant loops) and with the ability (or inability) of these fluids to remove heat from the nuclear reactor. Emphasis is placed on those situations causing trouble or leading to overheating of the reactor. A group of consultants led by Peter Griffith reviewed the thermal hydraulic phenomena at TMI-2 (reference 4), and what is reported here draws on their work.

Because the reactor was damaged severely during the period 100-210 minutes after start of the accident, the thermal history during the first 210 minutes was analyzed in considerable detail by means of the TRAC computer code (references 1 and 2). Some thermal hydraulic processes were examined for the period 100-210 minutes in order to evaluate some means by which the core damage might have been delayed, mitigated or avoided.

The principles for keeping the reactor cool after the reactor is shut down are simple: (1) keep the reactor full of water; (2) circulate that water throughout the reactor loops (the circulation can be produced by natural convection or by pumping); and (3) provide a heat sink, that is, a place to dump the heat. The heat sink can be supplied by either the steam generators or by injecting water, via high pressure injection (HPI), which is boiled and then discharged through the relief valves (either the pilot-operated relief valve (PORV) or the safety relief valves). The term "water" as used herein always refers to liquid water.

The generation of decay heat by the fission products is quantified in great detail in reference 3. The amounts of decay heat at various times are illustrated by the following table.

Time After Shutdown	Decay Power (megawatts)
1 second	168
1 minute	97
1 hour	36
12 hours	17
1 day	13
1 week	5.1
1 month	2.1
6 months	.40

In one second, decay heat was 6 percent of the rated power of 2,772 megawatts. The decay heat dropped by almost half in the first minute and to 10 percent of its initial value in 12 hours. The decay heat continued to decline but at a slower and slower rate as time went on. A key problem in protecting the reactor is to properly dispose of the still fairly substantial amounts of heat over the first several hours. As time goes on the heat removal is easier and easier, but heat must, of course, still be removed.

ANALYSIS OF REACTOR TRANSIENTS AT TMI-2

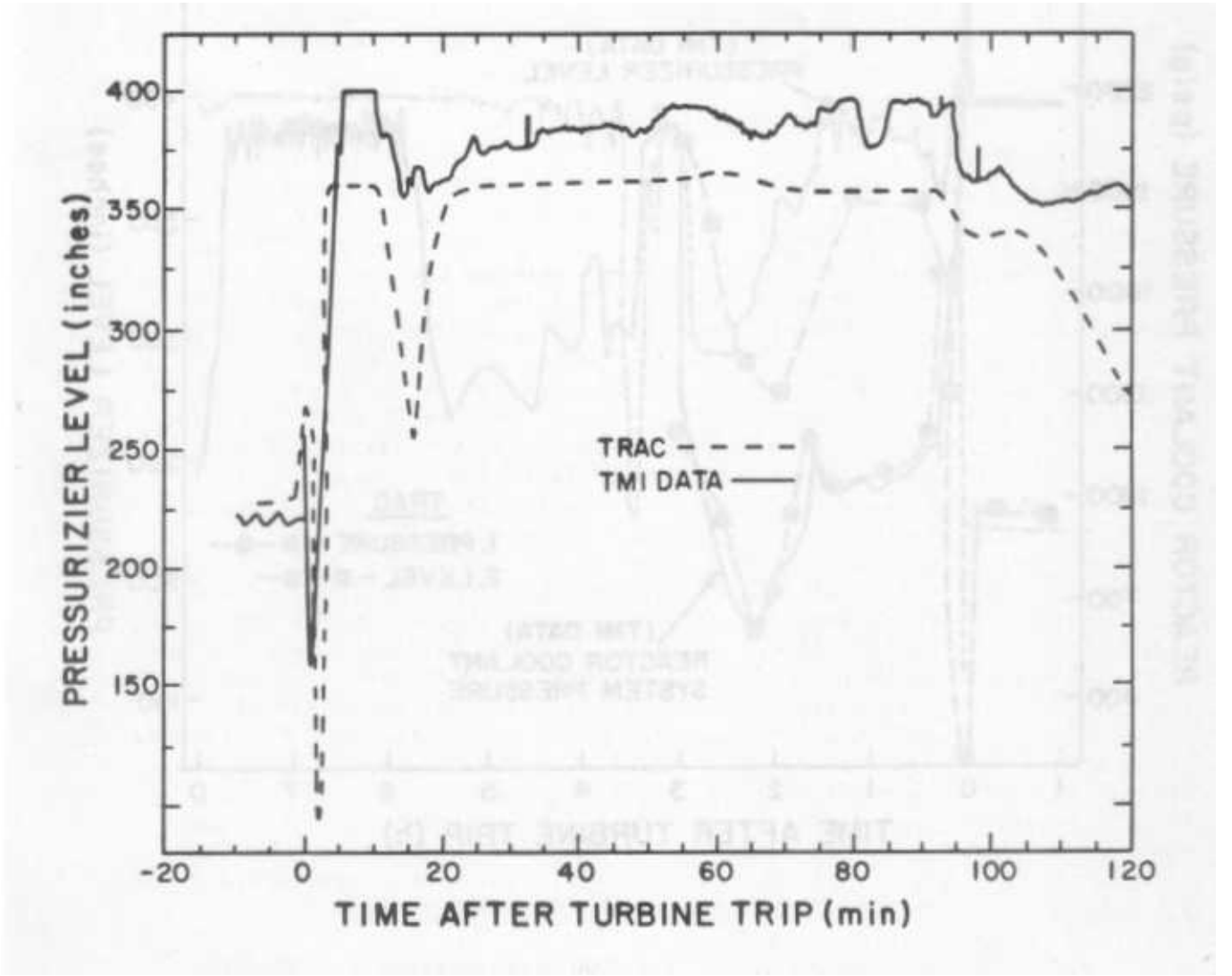
References 1 and 2 are applications of the Transient Reactor Analysis Code (TRAC) to analysis of the course of damage to the TMI-2 reactor. Picklesimer estimated reactor damage, but in making his estimates simply and quickly, he introduced some serious compromises in thermal hydraulics (reference 5). For the the period of 100-180 minutes, he assumed a constant rate of fall of water level in the core of 12 feet per hour; instead, the water level should fall rapidly when the core is fully covered, and the rate of fall should asymptotically approach zero as the water level approaches the bottom of the core. He also neglected convective heat transfer to the steam evolved by boiling water in the core; the result is that in his figures 3 to 7, he has a severe temperature inversion at the top of the core. Reference 4 shows that only small temperature differences occurred between the fuel rods and the steam. Thus, the hottest portions of the fuel rods were cooled somewhat by the steam; this heat was then transported upward by the steam, and some of it transferred to the cooler portions of the fuel rods. The result was an evening out of the spatial distribution of temperature in the core, the hottest parts being somewhat cooled and the cool upper portion being heated in comparison with the results in reference 5. Damage to the core was accordingly also evened out. Several attempts are under way to improve on reference 5 (references 1 and 6, for example).

The application of the TRAC code takes on a more ambitious effort, viz., analysis of the history of TMI-2 reactor operation during the whole first 3-1/2 hours. TRAC is a very complete, elaborate code designed for analyzing fairly brief reactor transients. For this reason, its speed of computing was accelerated for the rather long transient at TMI-2 by using a small number of spatial nodes. For the first 81 minutes, 24 hypothetical cells were used in the reactor vessel and 42 in the two system loops; this permitted three levels vertically within the core. At 81 minutes, the reactor vessel was more finely divided in order that the core might have five divisions vertically. The time increment for integration was chosen to be 0.1 second. The coolant was considered to be homogeneous within each cell.

In addition to two-phase flow, the TRAC calculations included the energy release from the zirconium-water reaction, which at high temperature produces more heat than radioactive decay of the fission products. Input data for the calculations were the estimated values of high pressure injection (HPI), let down, and pilot-operated relief valve (PORV) flows into and out of the reactor loops.

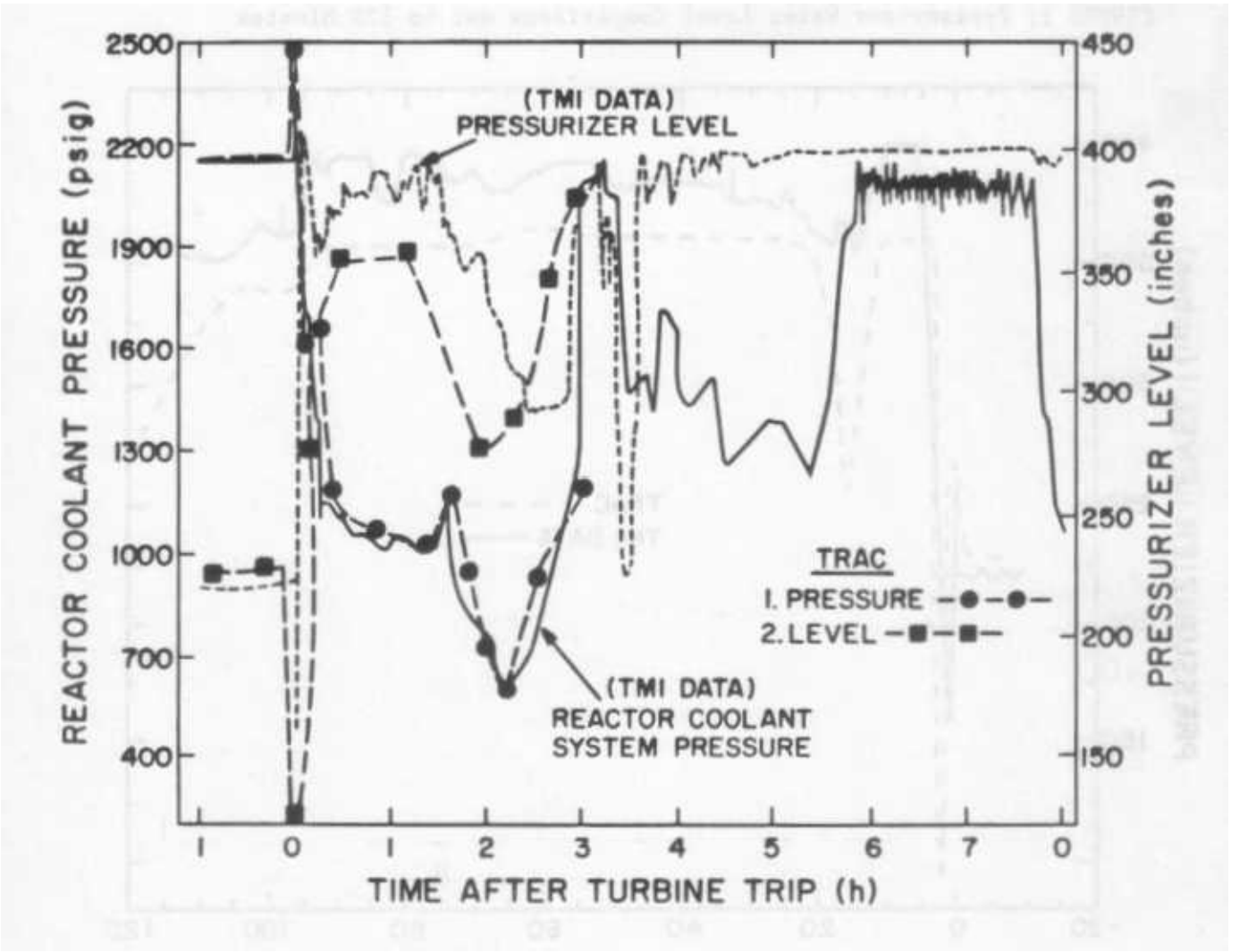
Some representative results are shown in Figures 1-4. The close match of observed and calculated pressurizer levels (Figures 1 and 2) shows that the inventories and void (or bubble) volumes must also be close. Both pressurizer level and reactor coolant pressure agree reasonably well with measured values out to 3 hours (Figure 2). The temperature calculations out to 3 hours (reference 1) in Figure 3 have been extended to 3-1/2 hours in Figure 4 (reference 2). A peak fuel rod temperature of 3,900 $^{\circ}$F was reached at 3 hours, 6 minutes after the accident started.

FIGURE 1: Pressurizer Water Level Comparisons out to 120 Minutes



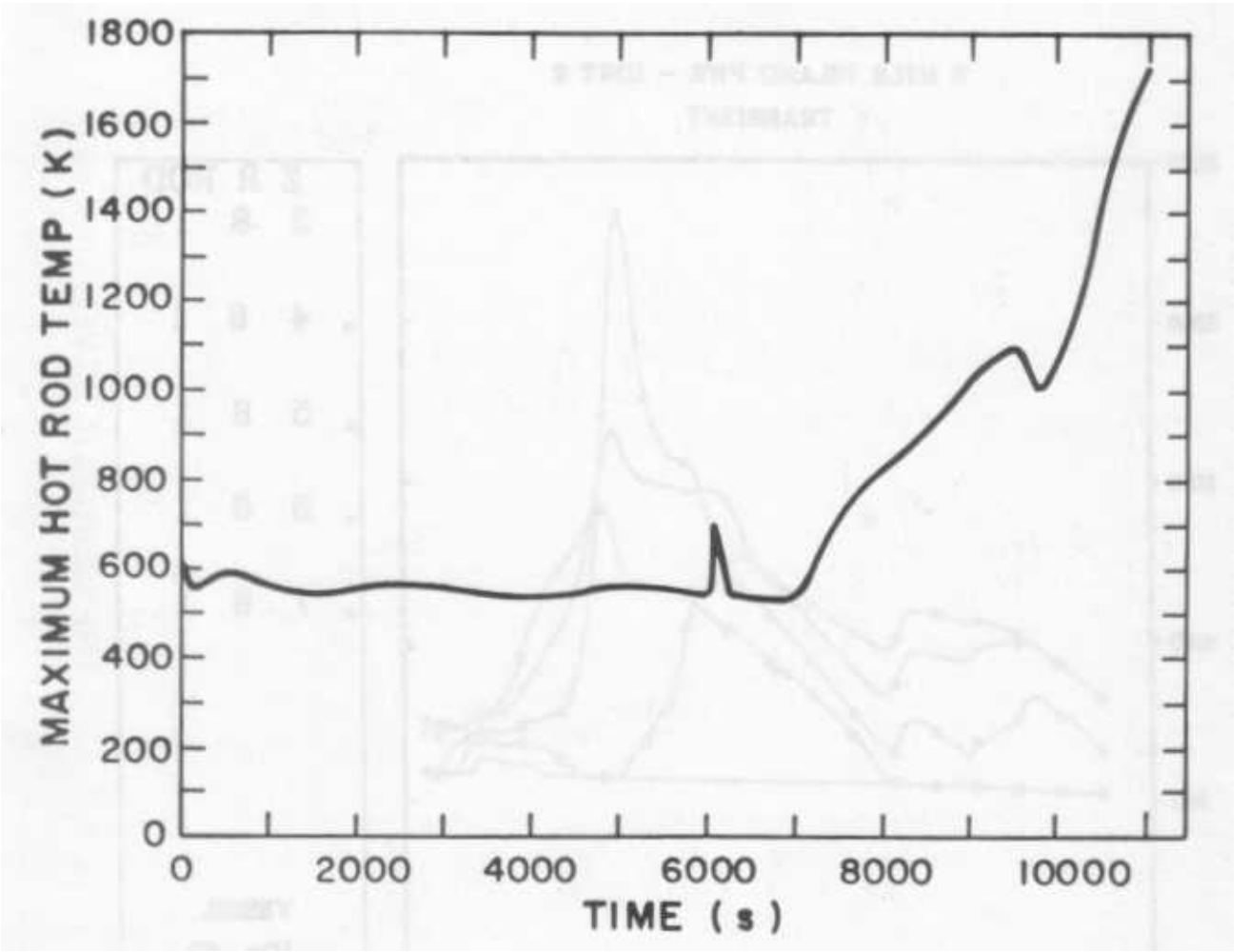
Source: Anon., "Preliminary Calculations Related to the Accident at Three Mile Island," Los Alamos Scientific Laboratory, LA-UR-79-2425, August 1979

FIGURE 2: TRAC Comparisons with TMI Data out to 3 Hours



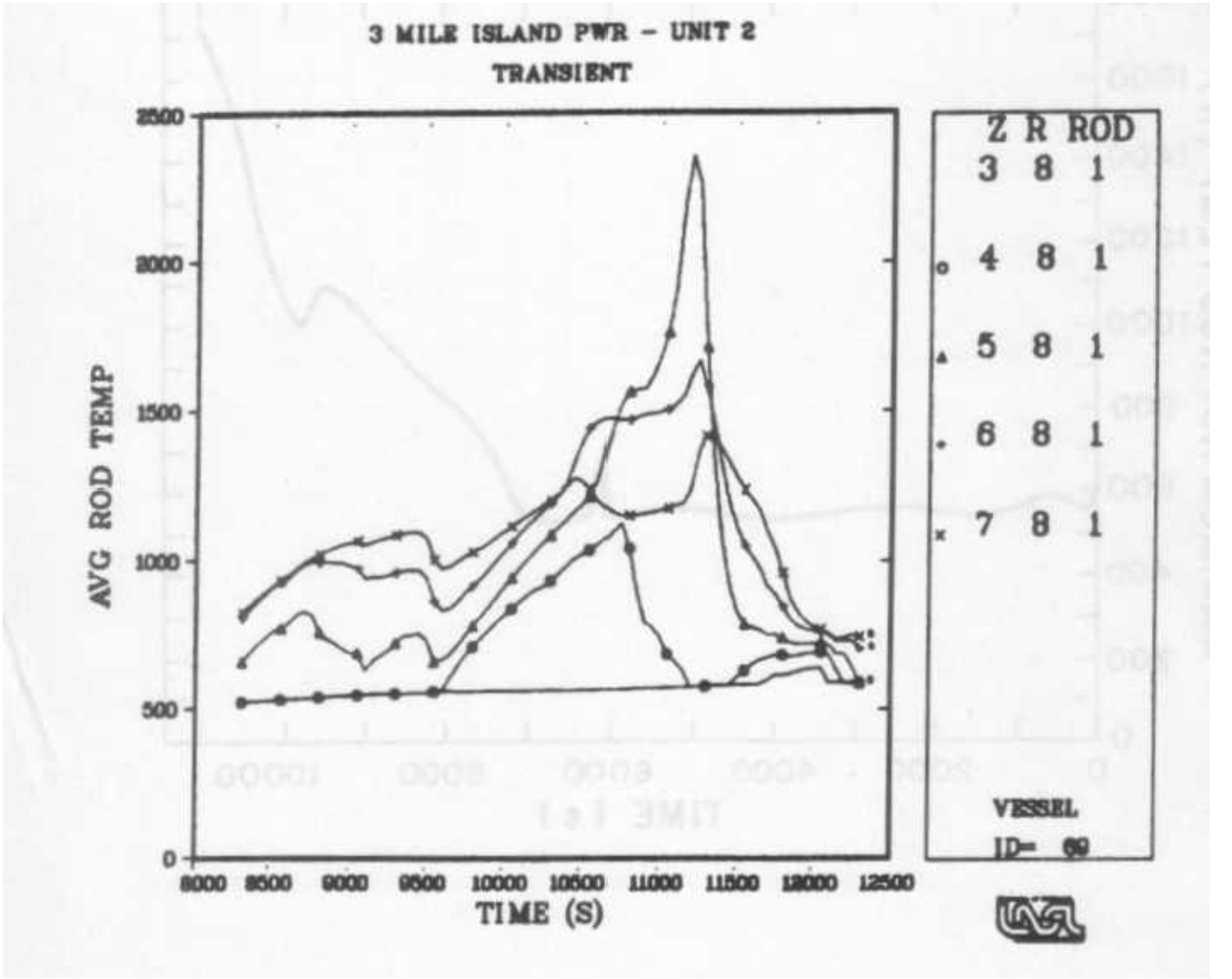
Source: Anon., "Preliminary Calculations Related to the Accident at Three Mile Island," Los Alamos Scientific Laboratory, LA-UR-79-2425, August 1979

FIGURE 3: Maximum Hot-Rod Cladding Temperature



Source: Anon., "Preliminary Calculations Related to the Accident at Three Mile Island," Los Alamos Scientific Laboratory, LA-UR-79-2425, August 1979.

FIGURE 4: Axial Temperature Distribution for the Fuel Rods, 8,000 to 12,500 Seconds



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Thermal hydraulic analysis of the TMI-2 reactor loops by means of the TRAC computer code fairly accurately reproduces the observed operating conditions over the first 3 hours. At about this time, a peak fuel rod temperature of about 3,900°F was reached.

STOPPING THE REACTOR COOLANT PUMPS

At about 74 minutes, the reactor coolant pumps in the B loop were shut down (references 7 and 9). At 101 minutes, the reactor coolant pumps in the A loop were also turned off. These two events were of considerable thermal hydraulic significance.

At 101 minutes, when all four pumps had been turned off, the reactor began a period of heating up that severely damaged it. At that time, the decay heat had declined to one percent of rated power and the heat fluxes in the reactor were accordingly low (reference 3). If the pumps had continued to operate and been able to circulate a two-phase flow through the reactor, even a froth, the water carried along would have kept the fuel elements wet and, in that way, would have kept them cool (reference 4).

Instead, the pumps were shut down because of vibration resulting from their inability to successfully operate on two-phase flow with high void fractions. If the pumps had tolerated continued operation on a mixture of water and steam of continually declining water content, damage to the reactor could have been delayed and perhaps avoided. Babcock & Wilcox Company (B&W) engineers stated that, to their knowledge, full-scale reactor pumps have not been tested in this full range from all water at one extreme to all steam at the other (reference 10).

Now consider the effect of turning off the B-loop pumps 27 minutes before the A-loop. The Electric Power Research Institute (EPRI) (reference 8, App. TH, pp. 49-50) postulates the following consequences of this sequence: As the A-loop pumps continued to pump a two-phase mixture into the reactor, some of this fluid would also flow from the reactor and into the idle B-loop pumps. In this region of essentially zero flow, liquid would drop to the bottom and vapor rise to the top of whatever space they occupied. Perhaps in this way, water accumulated in the idle steam generator B and completely filled its lower portion. In that event, the steam generator B might have contained considerably more water than steam generator A just after the A-loop coolant pumps were turned off at 101 minutes (see Figure TH 14, reference 8). During the operation of one set of pumps on two-phase flow, the idle steam generator is thus a reservoir in which a considerable portion of the liquid inventory might be trapped.

With both sets of coolant pumps turned off at 101 minutes, water was trapped in each steam generator. The geometry of the reactor loops prevented any of that trapped water from flowing back into the reactor. The staff has concluded that if the steam generators had been elevated relative to the reactor and the piping arranged to permit drainage of that water into the reactor, damage to the core could have been delayed and perhaps prevented.

FINDINGS

1. At 101 minutes after start of the accident, the inability of the reactor coolant pumps to pump a water-steam mixture having a very high proportion of steam made it necessary to turn off the pumps.

Stopping the pumps interrupted the reactor cooling provided by this two-phase mixture, and the reactor fuel elements rose in temperature to 3,500-4,000°F.

2. When the reactor coolant pumps were stopped, water was trapped in the lower portion of each steam generator. The geometry of the reactor loops prevented this water from draining into and cooling the reactor.

During the period from 101 to 175 minutes, no water was pumped through the reactor, the reactor heated up, and considerable damage occurred; throughout this period, several high-radiation alarms were recorded, indicative of release of fission products (reference 9). If effective natural circulation had been established at the beginning of this period, perhaps serious damage to the reactor could have been avoided. Reference 8 (App. TH, p. 54) makes the following judgment concerning the conditions following shutdown of the A-loop reactor coolant pumps, thereby producing natural circulation that was effective in cooling the reactor:

> The behavior of the primary coolant system following the trip of the loop A pumps indicates that steam occupied a substantial fraction of the system volume at that time. The behavior also illustrates that heat was effectively removed from the primary system by steam condensation on the primary side of the steam generators when the auxiliary feedwater was flowing.

But conditions were not maintained that would sustain this two-phase natural circulation.

Although the presence of steam would preclude the usual natural circulation in a water-filled loop, natural circulation with both water and steam can also provide effective reactor cooling if the proper operating conditions are established in loop design and in operation. In order that the reactor might be cooled by this two-phase natural circulation and that heat be transported to the steam generators, the following conditions must prevail:

(1) In order that there may be a temperature difference for producing heat transfer in the steam generator, the steam pressure in the secondary loop must be lower than in the primary, or reactor, loops. At TMI-2 the operators did not consistently maintain this condition.

(2) The steam in the reactor loops should be able to reach those portions of the steam-generator tubes having cold water on the secondary side. Either a liquid level higher on the primary side than on the secondary or noncondensible gas in the reactor loops could block the steam's path to the cold section of the tubes in the steam generators.

(3) Water condensed in the steam generators should be able to flow back to the reactor for reuse.

(4) For preservation of the reactor loops' inventory and, therefore, to permit continuation of the natural circulation, openings in the loops, such as the PORV, should be closed.

None of these conditions was fully met during the period 101-210 minutes. The PORV, of course, was open until about 140 minutes and the

operators did not always maintain secondary pressure lower than the primary.

Closed valves on the secondary side of the steam generator B prevented any heat removal by that steam generator. But feedwater was being supplied to steam generator A, and it therefore offered the potential for heat removal.

The crucial factor was that the steam generator A may have had only a small inventory of primary water after the period of operating the coolant pumps A on two-phase flow, as discussed above. In that event, condensate formed in the steam generator could flow to the bottom of the steam generator, but it would there be trapped and unable to flow back to the reactor. If the reactor loops had been designed so that the steam generators were elevated relative to the reactor and the piping arranged to permit drainage of that water into the reactor, damage to the core would have been delayed and perhaps prevented. Again, the design of the reactor loops appears to have impeded the process of cooling the reactors.

If the arrangement of the steam generators and their pipes had permitted condensate flow back to the reactor, perhaps noncondensible gases might still have precluded natural circulation. A large amount of noncondensible gas can always stop two-phase natural circulation, and a tiny amount will very likely always be tolerated. A key issue bearing on this question is the tolerance that the steam generator has of noncondensible gas. Consider that natural circulation was established at the time the pumps were stopped. If modest amounts of noncondensible gases were produced, they would be swept along by the flow of steam and then accumulate in the steam generator. In general, the gases would be carried toward the outlet end of the steam generator, where they would collect and effectively block the flow of steam to the cold end of the tubes. If only a small portion of the length of the steam generator tubes was immersed in the cold feedwater, then a modest amount of noncondensible gas could block the access of steam to that cold segment of tubing. So a crucial factor is the way in which the steam generator was operated.

Reference 9 on page 19 states that the water level was at 51 percent of the operating range, as required for establishing natural circulation. Figure OTSG-2 of reference 8 shows the full range, or height, of the steam-generator tubes as 600 inches and the operating and startup range as extending 250 inches from the bottom of the tubes. Noncondensible gases sufficient to fill the bottom 175 inches of the steam generator would block heat removal, if operated as described in reference 8.

Raising the water level in the steam generator to, say, the 525-inch level seems simple enough to do in a 600-inch steam generator, under the emergency conditions for which it would be required. This would at least triple the volume of noncondensible gas that could be tolerated. If the required water level had been, say, 525 inches, the system would have been more tolerant of noncondensible gas when striving for natural circulation on two-phase flow. Water-level indication on the secondary side for the full zero to 600-inch range is standard instrumentation for the steam generators (reference 8, App. OTSG, p. 2).

In reference 8, Figures TH4 to TH7 show that the water level in the steam generators never exceeded the 250-inch level at TMI-2.

B&W's once-through steam generator (OTSG) has a provision for spraying auxiliary feedwater onto the upper end of the tubes in the steam generator (Final Safety Analysis Report, Figure 5.5-3). This feature, if utilized, should improve natural circulation and permit the steam generator to tolerate a considerable quantity of noncondensible gas in spite of a low level of water on the secondary side of the steam generator.

Even in the absence of noncondensible gases, another factor must be considered in setting up natural circulation of two-phase flow: The liquid level of the condensate in the steam generator must be at least as high as the liquid level sought in the reactor vessel, usually above the top of the core. In Figure 5.1-5 of the FSAR, the 250-inch level in the steam generator appears to be at just about the same elevation as the pipes for flow to enter and leave the reactor. Because the condensate level in the steam generator will always be somewhat lower than the secondary-water level and because a goal should always be to keep the core covered, it appears that prudence dictates maintaining a water level above 250 inches for natural circulation with both water and steam in the reactor loops.

For natural circulation with water throughout the reactor loops, circulation and heat-removal capability will both be greatest if tall columns of cold water can be established in the steam generators. This factor also argues in favor of high feedwater levels in the steam generator for natural circulation. Elevation of the steam generator relative to the reactor would also improve natural circulation in a water-filled loop, as already incorporated into the design of the Davis-Besse power plant.

Perhaps the absence of such an approach as this prevented natural circulation from being effective and thereby contributed to the reactor damage at TMI-2.

Noncondensible gas in the reactor loops also interfered with effective cooling of the loops. During the period 150-210 minutes from start of the accident, a large amount of hydrogen was produced within the reactor. Remotely operated vent valves in the head of the reactor vessel and at the top of each candy cane could have vented this gas to the containment building and thereby have aided in maintaining natural circulation.

FINDINGS

1. Failure to always maintain a pressure lower on the secondary side of the steam generator than on the primary was one of several factors preventing natural circulation to cool the reactor during the period 100-150 minutes from the start of the accident.

2. The low elevation of the steam generators and the piping arrangement between the steam generators and the reactor trapped water in the steam generator rather than permitting it to flow back to the

reactor. This was one of several factors preventing natural circulation during the period 100-150 minutes from start of the accident.

3. Restriction of water level on the secondary side of the steam generators to levels not exceeding 250 inches impaired their capacity for natural circulation and made them sensitive to blockage of steam flow by noncondensible gases.

4. The design feature in the steam generators that permits spraying auxiliary feedwater onto the upper portion of the steam generator tubes improves natural circulation and improves their tolerance of noncondensible gases.

5. Following 150 minutes from the start of the accident, a large amount of hydrogen in the reactor loops prevented natural circulation from cooling the reactor. Remotely operated vents at the top of the candy canes may have permitted venting this gas to the containment building.

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REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CORE DAMAGE

BY

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October 1979
Washington, D.C.

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SUMMARY AND FINDINGS

In assessing core damage, three basic yardsticks were used:

1. An inventory of all the hydrogen released was used for estimating the degree of oxidation of the zirconium (reference 1).

2. The measured fission products released to the reactor coolant and to the atmosphere of the containment building were used to infer fuel peak temperatures reached (reference 1).

3. The thermal-hydraulics (reference 2) of the reactor loop over the first 3.5 hours were analyzed by means of the TRAC computer code (references 3 and 4). This produced a temperature history that would somewhat underestimate both the hydrogen generation and the quantity of gaseous fission products released.

References 5 and 6 are two early estimates of core damage by the Nuclear Regulatory Commission (NRC). Reference 7 describes the processes and extent of core damage and gives a physical picture of the nature and location of the various types of damage. More rigorous analysis (references 3 and 4) produces results in general agreement with the results presented in reference 7.

The following findings were reached:

1. Reference 17 specifies that power-generating nuclear reactors shall be designed to remain within the following limits: (a) Peak cladding temperature shall not exceed 2,200°F. (b) The zirconium cladding oxidized shall not exceed 17 percent of cladding thickness. (c) The total amount of hydrogen generated from the zirconium-water reaction shall not exceed one percent of that producible by oxidizing all of the fuel cladding. At TMI-2, all these limits were exceeded.

2. Ninety percent or more of the fuel rods have burst.

3. A hydrogen inventory indicates that 44-63 percent of all the zirconium in the reactor has been oxidized. The upper 60-70 percent of the zirconium clad is so embrittled that it has lost its structural integrity.

4. Fuel temperature exceeded 3,500°F throughout the upper 40-50 percent of the core. Fuel temperature may have exceeded 4,000°F throughout 30-40 percent of the core volume.

5. Some of the UO fuel may have become liquid at temperatures well below its melting temperature of 5,200°F. There is general agreement that some fuel was liquefied at TMI-2 but no general agreement on the extent of the liquefaction. Determination of the amount of fuel liquefied at TMI-2 must await actual inspection of the reactor fuel itself.

6. Continuing leaching of radioactive Cs, I, Sr, and Ba into the reactor's cooling water indicates that some of the fuel is in finely divided form.

7. From rapidly changed readings of both the in-core self-powered and ex-core source-range neutron detectors at 226 minutes after the start of the accident, the Electric Power Research Institute's (EPRI) Nuclear Safety Analysis Center surmises that a section of the core suddenly slumped downward as a consequence of earlier damage (reference 8). The staff agrees.

8. In the hottest portions of the core, sections of the control rods probably melted. Because the constituents of the rods are essentially insoluble in water, the neutron poisons contained in the rods are very likely still in the core (reference 5).

9. According to reference 9, slumping of the core (as in 6 above) increases core reactivity, but the core is not close to becoming critical. Even if the core slumps to its point of greatest reactivity and even if the neutron poisons in the control rods are somehow removed from the core, the core can be kept subcritical if its cooling water contains at least 3,180 parts of boron per million parts of water (by weight). At present, Metropolitan Edison (Met Ed) is maintaining 3,500 parts per million by adding boric acid (reference 9).

For core damage, a crucial period was the time interval of 101-210 minutes from the start of the accident. During this period, cooling of the reactor was interrupted for a protracted period, temperatures rose to high levels, and substantial damage to the core was the result.

Information on the extent of core damage stems from three principal sources, viz., (1) an inventory of the hydrogen produced by the reaction of hot zirconium with steam, (2) the fission products measured in the reactor coolant and in the atmosphere of the containment building, and (3) theoretical analysis of the thermal history of the reactor during this period. The information herein on those subjects is largely drawn from references 1 and 2.

Our purposes herein are as follows: (1) to assess the extent of core damage indicated by these three sources of information, and (2) to consider if further changes to the core could result in its becoming critical once more -- that is, reinitiating a nuclear chain reaction.

THE COURSE OF DAMAGE

When the reactor coolant pumps in the A loop were shut down 101 minutes after the start of the accident, boiling in this loop had already been under way for some time, and there was a considerable void (steam) volume in the loop. While they continued to run, the pumps, in addition to circulating water and steam throughout the loop, also acted as a homogenizer of the water-steam mixture. In this way water continued to flow through the reactor, to keep the fuel rods wet, and thereby to keep those rods cool. When the B-loop pumps were turned off at 73 minutes, phase separation took place within that loop. The water present in the primary side of the B steam generator settled to the bottom of that steam generator, and the steam rose to the top. That water was trapped in the bottom of the steam generator and unable to flow into the reactor because of the low elevation of the steam generator relative to the reactor.

The reactor coolant pumps in the A loop continued to circulate water and steam through both the reactor and the A steam generator until these pumps were also turned off at 101 minutes after start of the accident. Reference 8 (p. TH56) estimates 80 percent void (steam) in the reactor and the A steam generator at that time. If the void fraction throughout the A loop was that high, staff review indicates that the subsequent thermal history shows that the void fraction in the reactor itself was not that high; perhaps during their coastdown, the A-loop coolant pumps continued to pump water and steam into the reactor at a gradually diminishing rate, and phase separation within the reactor vessel covered 75-100 percent of the core with water.

At this point no heat was being extracted through the steam generators, and the pilot-operated relief valve (PORV) was still venting fluid to containment. High pressure injection (HPI) from the emergency core cooling system (ECCS) had also been throttled. The fuel rods were probably at a temperature close to 545 °F, the saturation temperature of the water at that time. Following that time, the decay heat from the portion of the fuel rods below the liquid surface all went into boiling water. Because initially the core was nearly covered with water, a large fraction of the total decay heat then being generated boiled water and produced steam. As time wore on and the water boiled away, a progressively smaller fraction of the fuel-rod height was covered by water and, accordingly, less water was boiled. For this reason, the water level dropped fairly rapidly at the beginning of this thermal transient, but the rate of fall slowed as less and less of the fuel-rod length was covered by water.

In turn, the rate of steam evolution was also high at the start of this thermal transient, but less and less steam was generated as the water level continued to fall. The steam flowing past the upper portion of the fuel rods cooled these rods and thereby the steam temperature was raised. The presence of this superheated steam was indicated by rising hot-leg temperatures at the reactor outlet. As time wore on and the water level in the reactor fell, less steam was available to cool the upper portion of the fuel rods, but more cooling was needed because a greater length of fuel rod was exposed to the steam. The results were

that the temperature of the upper portion of the fuel rods initially rose rather slowly, but the temperature rose more and more rapidly as progressively less steam was available for cooling an increasing fraction of fuel-rod length.

RUPTURE OF THE FUEL RODS

As the fuel-rod temperatures rose, the helium with which the rods had been filled during their manufacture also rose in pressure. With the rising temperatures, the strength of the fuel's zirconium cladding declined, and therefore its ability to resist the rising internal gas pressure diminished fairly rapidly. Because the zirconium had not yet been oxidized to a significant extent, the cladding was very ductile and therefore bulged under the conditions of rising internal pressure and falling clad strength. This bulging was probably greatest in the upper portion of the core where fuel-rod temperatures were high and clad strength low. When rod temperature reached about 1,500°F, the rods split probably in their hottest, upper portions, releasing to the reactor coolant both the helium with which the rods had been filled and gaseous fission products (such as xenon and krypton) that had escaped from the solid fuel matrix into the gas space. All the fuel rods reached temperatures above 1,500°F and therefore probably all ruptured in this way. The manner in which this ballooning and rupture could have occurred is detailed in reference 10.

Finding

All, or nearly all, of the fuel rods have burst.

Release of these gaseous fission products to the reactor coolant probably began shortly before the PORV was closed. During the period 134-180 minutes from the start of the accident, the following alarms gave indication of the release of these fission products: the particulate channel, gas-channel, and iodine-channel radiation monitors in the containment building. In that interval, the count rates on these channels all increased and went off-scale high.

OXIDATION OF THE ZIRCONIUM CLAD

As temperatures continued to rise in the upper portion of the core, the zirconium cladding began to oxidize, embrittling the clad. This oxidation also produced heat that resulted in more rapid temperature rise for the fuel and in accelerated oxidation. The Cathcart-Pawel relation for oxidation of the zirconium is given by the following (reference 12).

$$\frac{dT}{dt} = \frac{.01126}{T} \exp\frac{(-35890)}{(1.987T)} \qquad (1)$$

where

T oxidized thickness, cm
t time, s
T temperature, K

The rate of oxidation at any given moment was thus dependent not on temperature alone but also on the amount of oxidation t that had already taken place. Roughly, the oxidation rate was such that at 2,000°F the heat release from clad oxidation equaled the decay heat.

As reference 11 describes, zirconium tubing retains a fair amount of its strength if its local oxidation is less than 10 percent, but it becomes very brittle if the oxidation is 18 percent or greater. Although embrittlement by the hydrogen generated is a potential concern, reference 11 also points out that embrittlement of the clad by oxidation is the more severe problem.

Reference 1 constructs an inventory of the hydrogen generated and from this infers that from 44-63 percent of all the zirconium in the core was oxidized. Because oxidation proceeded most rapidly where temperature was high (equation 1), the oxidation was concentrated in the upper part of the core. The portion of the fuel rods that was continually immersed in water (perhaps the bottom 20-30 percent) was kept cool by that water and thus did not oxidize. Above the lowest water level, the oxidation is graded from slight to moderate to heavy. Inasmuch as even 18 percent oxidation produces severe embrittlement (reference 11), approximately the upper 60-70 percent of the fuel rods was severely embrittled and has likely lost its structural integrity.

Finding

A hydrogen inventory indicated that 44-63 percent of all zirconium in the reactor has been oxidized. The upper 60-70 percent of the zirconium clad is so embrittled that it has lost its structural integrity.

The upper perhaps 30-40 percent of the core had its zirconium completely oxidized, and this provides a clue concerning the temperatures attained. From equation 1, the complete oxidation of the 26.5-mil zirconium clad requires at least 40 minutes at 3,000°F or above or, at least 12 minutes at 3,500°F or above. These temperatures and times are, of course, required at the lower boundary of the zone of complete oxidation; higher temperatures and longer times would have prevailed within this zone of complete oxidation.

RELEASE OF FISSION PRODUCTS

The fission products released from the fuel also provide information on the extent of fuel damage and on the peak temperatures reached. In general, high fuel temperatures produce greater releases of fission products than low, expressed as proportions of the quantities of each given isotope produced in the fissioning process. At high fuel temperatures (above 3,500°F), comparable proportions of fission gases (such as xenon) and volatile fission products (such as iodine and cesium) are released by reactor fuels. However, the chemically active species (such as iodine and cesium) are more readily trapped by adsorption on surfaces or as precipitates, such as silver iodide. For this reason the fission gases are simpler to trace and more reliable. Herein they will be relied on exclusively; attention will focus on Xe-133.

Of the xenon produced by fissioning, 57 percent (references 1 and 13) to 60 percent (reference 14) was released from the fuel. Lorenz (reference 15) shows how the proportion of fission gas release depends on fuel temperature and time at that temperature. In periods of a few hours, very little of the fission gas would be released by fuel that did not exceed 1,600°C (2,900°F). During the thermal transient from 101-210 minutes after turbine trip, a portion of the core (perhaps 20-25 percent) remained so cold that it released very little of its fission gas. In turn, the portion of the fuel rods just above the water was effectively cooled by the steam produced, so perhaps one-third of the fuel-rod length remained cold enough that very little of its fission gases escaped from the fuel.

Release of 57 percent of the total inventory of xenon from two-thirds of the fuel requires an 85 percent release from that portion of the fuel. According to reference 15, a fuel temperature of the order of 2,400-2,500°C (4,350-4,500°F) would be required throughout that portion of the core.

Rest in attachment 4 to reference 6 shows that the proportion of fission gases released is sensitive to several factors. One of these is grain boundary separation. If the fuel has a great deal of fracturing along the grain boundaries, the fission-gas release is substantially higher than if the grain boundaries did not separate. At TMI-2, the fuel is probably highly fragmented from the successive thermal cycles through which it passed, expecially when water was introduced into the hot core; Cohen (reference 13), in particular, concludes that a portion of the fuel is finely divided. Rest also states that fuel power density (kW/ft) affects the size of gas bubbles in the fuel lattice which affects the ease with which the gases can escape from the fuel lattice. Because the release of fission gases depends on so many unknown and uncontrolled factors at TMI-2, the fuel temperatures of 4,350-4,500°F throughout the upper two-thirds of the core are only a rough indication of the temperature actually achieved.

Findings

1. The release of the gaseous fission product xenon from the fuel indicates that perhaps the upper two-thirds of the core exceeded 4,000°F.

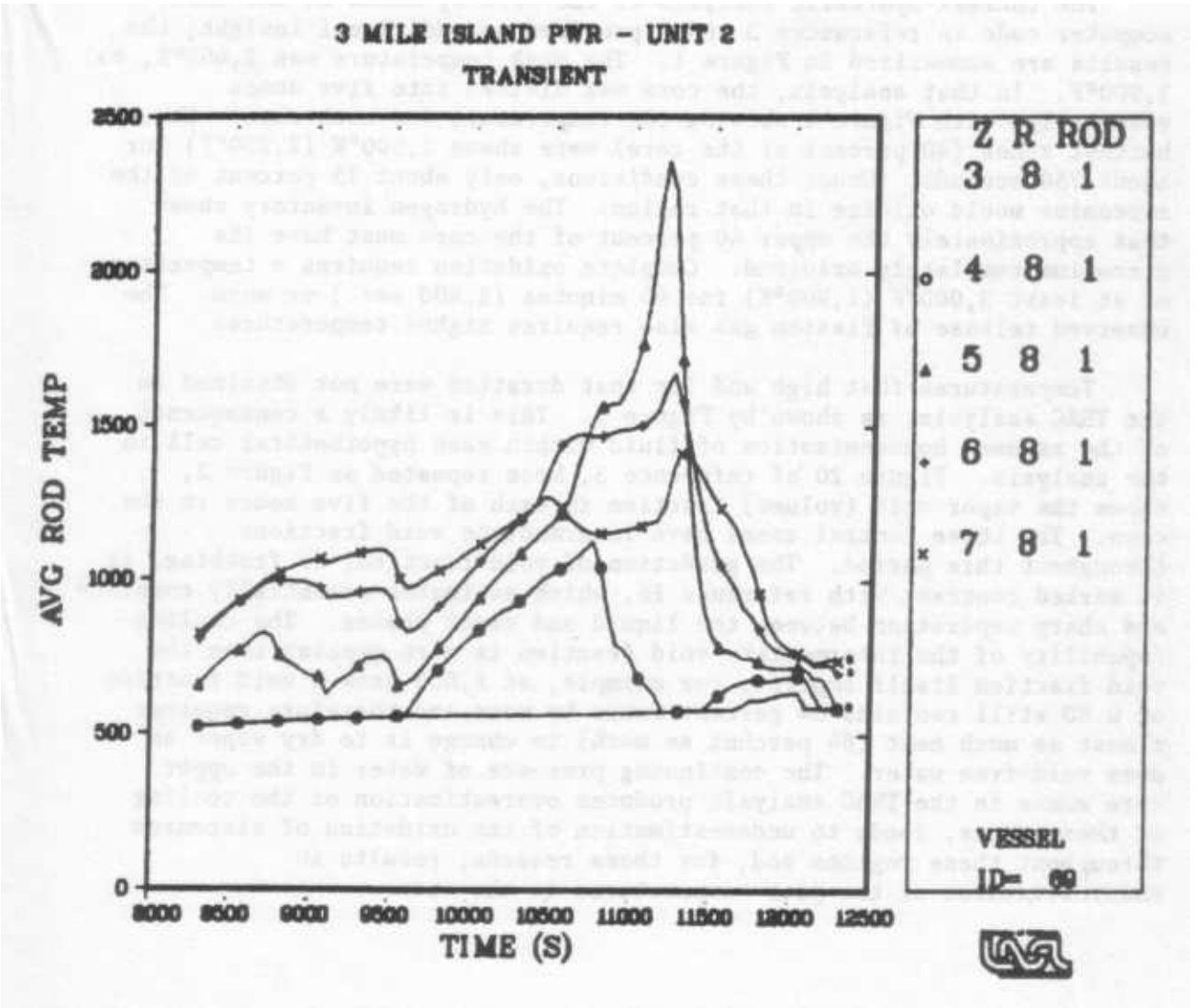
2. Continuing leaching of radioactive Cs, I, Sr, and Ba into the reactor's cooling water indicates that some of the fuel is in finely divided form.

THERMAL HISTORY OF THE CORE

The thermal-hydraulic analysis of the core by means of the TRAC computer code in references 3 and 4 provides an additional insight; the results are summarized in Figure 1. The peak temperature was 2,400°K, or 3,900°F. In that analysis, the core was divided into five zones vertically, with Figure 1 showing the temperature for each zone. The two hottest zones (40 percent of the core) were above 1,500°K (2,200°F) for about 750 seconds. Under these conditions, only about 15 percent of the zirconium would oxidize in that region. The hydrogen inventory shows that approximately the upper 40 percent of the core must have its zirconium completely oxidized. Complete oxidation requires a temperature of at least 3,000°F (1,900°K) for 40 minutes (2,400 sec.) or more. The observed release of fission gas also requires higher temperatures.

Temperatures that high and for that duration were not obtained in the TRAC analysis, as shown by Figure 1. This is likely a consequence of the assumed homogenization of fluid within each hypothetical cell in the analysis. Figure 20 of reference 3, here repeated as Figure 2, shows the vapor void (volume) fraction in each of the five zones in the core. The three central zones have intermediate void fractions throughout this period. The gradation of void fraction, or frothing, is in marked contrast with reference 16, which estimates essentially complete and sharp separation between the liquid and vapor phases. The cooling capability of the intermediate void fraction is much greater than the void fraction itself implies; for example, at 1,000 psia a void fraction of 0.80 still contains 84 percent water by mass and therefore requires almost as much heat (84 percent as much) to change it to dry vapor as does void-free water. The continuing presence of water in the upper core zones in the TRAC analysis produces overestimation of the cooling of these zones, leads to underestimation of the oxidation of zirconium throughout these regions and, for these reasons, results in underestimation of the peak temperatures in the core.

FIGURE 1: Axial Temperature Distribution for the Fuel Rod, 8,000 to 12,500 Seconds



Source: William R. Stratton, et al., "Alternative Event Sequences," Appendix B, Technical Assessment Task Force report to the President's Commission on the Accident at Three Mile Island, October 1979.

MOLTEN FUEL

The conditions leading to or preventing fuel melting are summarized by references 18-20. From such information, the staff has identified the various conditions under which the reactor's UO fuel could have been liquefied at TMI-2 below its melting point of ,840°C (5,150°F). The extent of fuel liquefaction throughout the core depends on several factors, as follows: the temperatures achieved, the degree of clad oxidation, the intimacy of contact between fuel and clad, and both the spatial and temporal variations in these parameters at TMI-2. In addition, the chemical kinetics of the fuel clad interactions are not well known.

There is general agreement that some fuel was liquefied at TMI-2 but no general agreement on the extent of the liquefaction because of the complexity of all the physical relationships involved. Determination of the amount of fuel liquefied at TMI-2 must await actual inspection of the reactor fuel itself.

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REPORT OF THE
TECHNICAL ASESSSMENT TASK FORCE

ON

WASH 1400 -- REACTOR SAFETY STUDY

BY

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Washington, D.C.

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The WASH 1400 Reactor Safety Study (Rasmussen Report), published in 1975, contains descriptions of potential accidents in nuclear power plants and estimates of the probabilities of occurrence of accidents involving radioactivity release. Examination of WASH 1400 shows that it is relevant to the study of the Three Mile Island (TMI) accident because the sequence of failures in the accident are discussed in the report, and the occurrence of the accident is consistent with WASH 1400 predictions. WASH 1400 results, and lessons that should have been learned from the report, are discussed here.

WASH 1400 SUMMARY

WASH 1400 involved (1) compilation of a list of potential accidents in nuclear reactors, (2) estimation of the likelihood of accidents involving radioactivity release, (3) estimation of health effects associated with reactor accidents, and (4) comparison of nuclear accident risk with other accident risks in everyday life. The purpose was to provide information for the judgment of the acceptability of the risk associated with reactors.

The study determined that nuclear accident risk is small -- almost negligible -- compared to more common risks, including airplane accidents, fires, dam failures, chlorine spills, earthquakes, hurricanes, and tornadoes. Results show that risk is small because the more likely reactor accidents involve success of safety systems designed to accommodate them, and because accidents involving failure of safety systems are unlikely. These systems are provided in nuclear reactors to prevent core meltdown and to diminish radioactivity release.

The WASH 1400 risk assessment was reviewed by a Risk Assessment Review Group in 1977 (the Lewis Report) which concluded that "they were unable to determine whether the absolute probabilities of accident sequences in WASH 1400 are high or low, but believe that the error bounds on those estimates are, in general, greatly understated." However, they went on to say:

> WASH 1400 was largely successful in at least three ways; in making the study of reactor safety more rational, in establishing the topology of many accident sequences, and in delineating procedures through which quantitative estimates of the risk can be derived for those sequences for which a data base exists.
>
> Despite its shortcomings, WASH 1400 provides at this time the most complete single picture of accident probabilities associated with nuclear reactors. The fault tree/event tree approach coupled with an adequate data base is the best available tool with which to quantify these probabilities.
>
> WASH 1400 made clear the importance to reactor safety discussions of accident consequences other than early fatalities.1/

The Nuclear Regulatory Commission (NRC) accepted the findings of the Risk Assessment Review Group and issued a statement which said in part:

> The Commission accepts the Review Group Report's conclusion that absolute values of the risks presented by WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes and has taken and will continue to take steps to assure that any such use in the past will be corrected as appropriate. In particular, in light of the Review Group conclusions on accident probabilities, the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident.
>
> With respect to the component parts of the study, the Commission expects the staff to make use of them as appropriate, that is, where the data base is adequate and analytical techniques permit. Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking.2/

The Risk Assessment Review Group, while criticizing the actual numbers estimated in WASH 1400, commended the description of accident sequences and the "fault tree/event tree" approach as a valuable analytical tool for estimating probabilities of accidents.

THREE MILE ISLAND EVENTS

Two types of events are important in reactor accidents -- initiators and failures. Initiators are the causes of accidents. Sudden pipe breaks (such as loss-of-coolant accidents, or LOCAs) and transients are examples of initiators. A transient was the initiator at Three Mile Island. Transient is a general designation for all events causing interruption of normal operation and possibly requiring shut down of the reactor. Transients occur frequently in nuclear reactors and are routinely handled without incident. Failures refer to equipment malfunctions or operator errors in the response to an initiator event. If enough failures occur, a transient can result in an accident. Three Mile Island involved two significant failures -- a stuck relief valve and operator interruption of emergency core cooling.

WASH 1400 includes a list of 18 events that are likely transient initiators (Table 1). Included in the list is the Three Mile Island accident initiator -- loss of condensate pumps.3/

Condensate pump loss normally causes the main feedwater pumps to stop, requiring the auxiliary feedwater pumps to start up and remove heat from the reactor. Although these pumps started at Three Mile Island, the path for the cooling water to reach the steam generators was blocked by valves inadvertently left closed. Within 8 minutes, operators placed the valves in the correct position. WASH 1400 states, however, that auxiliary feedwater must be unavailable for a longer period of time before the event is regarded as a failure, because it takes some time for a shutdown reactor to generate sufficient heat before an alternate means of cooling the reactor is necessary. WASH 1400 suggests the delay could be one to 1-1/2 hours.4/ Analyses of the Three Mile Island accident also indicate that unavailability of auxiliary feedwater for 8 minutes was not significant.5/

The relief valve opened at Three Mile Island, as is normal in such transients, but it failed to close, causing the high pressure reactor cooling system to depressurize and spill radioactive water in the containment building. This failure is discussed in WASH 1400,6/ and its likelihood was predicted based on actual reactor experience with relief valves.7/

The normal response to the failure of a relief valve to close is actuation of emergency core cooling. This prevents excessive loss of water from the reactor. Emergency core cooling failed at Three Mile Island because the operators throttled the flow, causing the core to be damaged. Cooling was restored by the operators before core meltdown would have occurred. This failure event is discussed in WASH 1400. It states that failure to remove heat from the core could lead to core meltdown or damage,8/ and that operator action is required to prevent meltdown.9/

Other events discussed in connection with the accident involve hydrogen explosion, steam explosion, and collection of noncondensible gases in the reactor coolant system (RCS). WASH 1400 identified an accident involving core overheating with metal-water reaction (at TMI-2) as the important source of hydrogen. The report indicates that lack of oxygen prevents ignition of the hydrogen in the reactor vessel, but

TABLE 1: PWR

Likely Initiating Events	Unlikely Initiating Events
1. Turbine Trip	1. Rupture of High Energy Piping in Secondary Coolant System, a) Rupture of Main Feedwater Lines, b) Rupture of Lines in Main Steam System(a)
2. Spurious Signals from SICS	
3. Loss of Condenser Vacuum	
4. Inadvertent Closure of Main Steam Line Isolation Valves	2. Rupture of Steam Generator (See Preceding Discussions in section 4.1.5 for Coverage)
5. Loss of Main Station Generator with Failure to Relay Auxiliary Loads (e.g., Main Feedwater Pumps, Condensate Pumps) to AC Power Incoming from Offsite Network	3. Rupture of Control Rod Mechanism Housing on Reactor Vessel Leading to Small LOCA and Control Rod Ejection (See Preceding Discussions in sections 4.1.3 and 4.1.4 for Coverage)
6. Loss of Main Circulating Water pumps for Condenser Cooling	4. Abrupt Seizure of All Main RCS Recirculation Pumps
7. Loss of Main Feedwater Pumps	4. Startup of Inactive Reactor Coolant Loop with Abrupt Opening of Both Isolation Valves in One RCS Loop in PWR Plants Employing RCS Loop Isolation Valves
8. Loss of Condensate Pumps	(a) These ruptures are included somewhat arbitrarily with in the Unlikely Event Category. However, failures of lines in the PWR secondary coolant systems have occurred principally during plant testing and startup periods. These types of failures have included inadequate initial design of relief valve headers in the steam supply lines, discharge of secondary coolant from leaking feedwater valves, discharge of secondary coolant from cracks in main feedwater lines, etc. The RCS cooldown transients stemming from these failures would be less severe than those included under No. 12 of the Likely Event Category above. The potential impact of such high energy line failures in specific locations of the plant, since they might commonly interact with and affect availability of the plant ESFs, was considered as part of this study. Refer to Appendices II and IV.
9. Loss of AC Power Incoming from Offsite Network	
10. Inadvertent Opening of Steam Generator Power-Operated Relief Valves (10% Sudden Load Demand)	
11. Increase in Main Feedwater Flow; Malfunctions in Feedwater Flow Control	
12. Malfunctions of Control Resulting in Inadvertent Opening of all Turbine Steam Bypass Valves (40%) Sudden Load Demand)	
13. Uncontrolled Rod Withdrawal a) at Full power, b) At Startup	
14. Control Rod Assembly Drop	
15. Boron Dilution by Malfunctions in Chemical Volume and Control System	
16. Startup of Inactive Reactor Coolant Loop (in PWR with No RCS Loop Isolation Valves)	
17. Accidental Opening of Pressurizer Safety or Relief Valves	
18. Loss of RCS Coolant Flow (Main RCS Circulating Pump Malfunctions)	

Source: "Reactor Safety Study," An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix 1, NRC, Table 1-4.9, October 1975.

ignition in the atmosphere of the containment is likely after hydrogen leaks from the reactor. The conclusion in WASH 1400 is that the likelihood that hydrogen detonation will fail containment is negligible even in core meltdown accidents. This conclusion is supported by staff analysis.10/ Containment overpressurization failure due to hydrogen burning can occur in meltdown accidents if significant sources of hydrogen other than core zirconium are available or if additional equipment failures occur in containment cooling systems. 11/ In fact, a hydrogen explosion did occur at Three Mile Island, and did not threaten containment.12/

Steam explosions are postulated to occur in meltdown accidents when large quantities of molten fuel drop into large volumes of water. According to conservative calculations in WASH 1400, such explosions have the potential to rupture the reactor vessel and failure containment. However, sufficient uncertainties exist in the basic physical understanding of steam explosions that it is also possible that steam explosions pose no explosive threat to the reactor vessel. 13/ Experimental programs since WASH 1400 confirm that the WASH 1400 estimates are highly conservative. Significant steam explosions did not occur at TMI-2.

Noncondensible (hydrogen) gas prevented the cooling of the damaged Three Mile Island core for a period of time during the accident. 14/ This mechanism for interruption of core cooling was not considered in WASH 1400.

WASH 1400 ACCIDENTS VS. THREE MILE ISLAND ACCIDENT

The sequence of failures in the Three Mile Island accident can be considered in terms of safety functions performed and also in terms of specific systems required. The safety functions are common to all pressurized water reactors (PWRs), but the specific safety systems involved generally vary from reactor to reactor. The WASH 1400 general description of safety function performance corresponding to the Three Mile Island transient involves success of reactor shutdown and overpressurization protection, but failure to adequately cool the core. 15/ The result of this sequence, or combination of events, is eventual core melt if no operator action is taken. This is an accurate description of the Three Mile Island accident.

The more specific sequences, or combinations of system failures, listed in WASH 1400 cannot be compared directly to the Three Mile Island accident because WASH 1400 was based on a Westinghouse (W) pressurized water reactor (the Surry plant in Virginia) and the details of potential accidents differ from the accident in the Babcock & Wilcox (B&W) pressurized water reactor at Three Mile Island. For purposes of comparison, a compilation of potential accidents for transients in B&W reactors was prepared and is attached to this report.

Although WASH 1400 results are based on the designs of a small number of reactors, the risk estimates are intended to be valid for all reactors. To justify this, the study employs conservatism at many points in its analysis and argues that all reactors are within the WASH 1400 conservatism in terms of their accident potential.

WASH 1400 PREDICTIONS

The WASH 1400 estimates of radioactivity release accident probabilities are one per 2,000 years in PWRs and one per 7,750 years in boiling water reactors (BWRs). 16/ Due to uncertainties in the calculations, it was necessary to establish upper bounds on these probabilities, which are given in the report as one in 210 years for PWRs and one in 775 years for BWRs. Approximately 12 percent of the accidents involve core melt in PWRs, and 22 percent in BWRs. The more likely accidents involve no core melt and result in radioactivity release from spilled radioactive water in the containment building. Three Mile Island radioactivity release coincides with the severity of the WASH 1400 sequences without core melt, because while the radio-xenon release was of the magnitude expected in core melt accidents, the more important health hazard, radio-iodine, was effectively contained in the unmelted fuel and the containment building.

If WASH 1400 predictions of best estimate probabilities are valid, there was a 13 percent chance (about one in 8) of having had an accident at the time of the Three Mile Island accident or earlier (i.e., 223 reactor years of operating experience in PWRs, 187 in BWRs). Further, there was an 80 percent chance that an accident would have occurred at a PWR, rather than a BWR. There was also a 90 percent chance that an accident would not involve core melt or the high radioactivity releases associated with core melt. Hence, the fact that the Three Mile Island accident occurred when it did and that the consequences were limited in extent is consistent with WASH 1400 predictions.17/

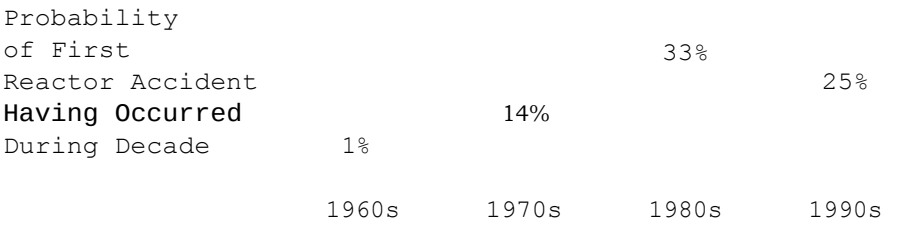
The WASH 1400 upper bound probabilities yield a predicted 80 percent chance of having had an accident this soon.

The available data -- the fact that one accident occurred after 223 PWR and 187 BWR years of experience -- are very limited and cannot be used to make a single estimate of accident probability. For example, the probability of a reactor accident involving radioactivity release could be any number between one in 100 reactor years and one in 8000 reactor years, based on the data. As time passes, with or without additional accidents, the range will change. The range is useful because it provides a basis for rejection of probability estimates outside of the range. Since the WASH 1400 estimates are within the range, they cannot be rejected and are consistent with the data.

Based on WASH 1400 probability estimates, there is a 40 percent chance of one or more accidents during the next decade. In the 10-year period, an accumulation of about 900 reactor years in PWRs and 400 in BWRs is expected. There is 50 percent chance during the 1990s, during which time 1,200 reactor years in PWRs and 500 reactor years in BWRs are expected. There is a 15 percent chance of a core melt accident by the end of the century, based on WASH 1400 best estimates and the expected number of reactors by that time (58 BWRs and 119 PWRs). The average of the expected consequences is less than one fatality for a core melt accident.18/

The following bar chart indicates the likelihood of the first nuclear accident having occurred in this decade, as opposed to another decade. These are based on WASH 1400 best estimates. (Note that the probability numbers associated with the decades-depend on how the

beginning and end points of the 10-year periods are chosen. However, shifting the decade definitions would not change the numbers enough to affect conclusions based on the bar chart as is.)



This chart indicates that the 1980s were the most likely time for the first reactor accident. The likelihood in the 1990s was lower than in the previous decade, because of the large probability that the first accident would occur before 1990.

LESSONS THAT SHOULD HAVE BEEN LEARNED FROM WASH 1400

WASH 1400 contains three important messages. These involve expected frequency of accidents, methods for improving reactor safety, and the most likely types of accidents. Perhaps it is a fault of the report that these messages were not emphasized, because the conclusion most often associated with WASH 1400 -- reactors are safe -- receives the primary emphasis in the report. Perhaps it is the fault of the Nuclear Regulatory Commission (NRC) that more effort was **dedicated to criticizing** WASH 1400 than was applied to understanding its messages. In fact, WASH 1400 predicted that accidents could happen, although most would present **little or no public hazard. One message of WASH 1400 is** that while nuclear accident risk is small compared to other societal risks, accidents similar to Three Mile Island should be expected. These accidents were not emphasized in WASH 1400, because they do not contribute as significantly to risk as the more severe core melt accidents.19/

The WASH 1400 study, in using the "event tree" and "fault tree" methodologies, borrowed from the aerospace industry, actually revealed a "weak link" in the safety of the Surry reactor. This led directly to a change in inspection procedures at Surry and reduced the probability of one major risk contributing accident 20/ by a factor of 20.21/ Thus, another message of WASH 1400 is that application of these methods to analysis of a specific reactor should be used to reveal "weak links" in safety. Recently, NRC officials have endorsed a plan to apply WASH 1400 techniques to the analysis of other existing reactors for this purpose. 22/ Since the accident at Three Mile Island, the NRC has applied reliability analysis to the study of auxiliary feedwater availability in all U.S. commercial reactors.23/

Reactor safety research, both before and after WASH 1400 was published, has concentrated on the double-ended pipe break, or large loss-of-coolant accident (LOCA). Safety systems were designed specifically to accommodate this type of accident. Yet, the WASH 1400 results published in 1975 indicated that reactor accident risk is dominated by small-break LOCAs and transient-initiated accidents, like Three Mile Island. 24/ A third message of WASH 1400 is that relative efforts in reactor safety research for large LOCAs, small LOCAs, and transient-initiated accidents should be consistent with priorities suggested by their relative risk contributions.

Generally, the NRC has based priorities on "good engineering judgement,"25/ although the Lewis Report and the NRC commissioners have recently endorsed the use of WASH 1400 techniques to carry out licensing more efficiently. In fact, the NRC staff has successfully applied the techniques to prioritize safety issues, study overpressurization of vessels, and optimize inspection time intervals.26/

It should be noted with regard to small-break LOCAs that it was thought by the NRC that safety systems designed to accommodate large LOCAs would necessarily be adequate to deal with small LOCAs. 27/ It should have been clear from WASH 1400 that such was not the case for transient-initiated pilot-operated relief valve (PORV) LOCAs.28/ Instead, WASH 1400 was taken by the NRC as an affirmation of its good regulatory work.29/

Further, practical considerations inhibit the application of WASH 1400 techniques. It is very difficult to apply the techniques properly, and few people are trained or experienced in such work. 30/ Also, the criticisms of WASH 1400 techniques by the NRC commissioners left the NRC staff unmotivated to develop ways to apply the techniques. Since the Lewis Report and the Three Mile Island accident, this trend appears to be reversing.

POTENTIAL ACCIDENTS RESULTING FROM TRANSIENTS IN B&W REACTORS

This section describes the TMI-2 accident and alternate courses the accident might have taken. Alternate sequences are discussed in more technical detail in another staff report, entitled "Alternative Event Sequences." "Event trees" are used here to illustrate graphically the results of possible combinations of failures in order to answer "what if" questions associated with TMI-2. The work draws on information in WASH 1400, which includes descriptions of possible failures and event trees relevant to TMI-2.

WASH 1400 contains a separate event tree for each different accident type. This is done because the sequence of possible system responses is different. The accident types for which separate trees are presented are (1) large-break LOCA (2) small-break LOCA requiring low pressure safety injection (HPSI) (3) small-break LOCA requiring HPSI (4) vessel rupture, (5) LOCA through boundaries separating the high pressure primary system from attached low pressure systems, or "interfacing systems LOCA," and (6) transients. The TMI-2 accident occurred during a transient. The TMI-2 sequence of events is related to WASH 1400 transient event trees in the present report.

GENERAL DESCRIPTION OF POSSIBLE OUTCOMES OF PWR TRANSIENTS

WASH 1400 includes a "functional event tree" for PWR transients (Figure 1) which describes system responses and possible failures which are common to all PWR designs -- those of Westinghouse, B&W, and Combustion-Engineering (C-E). It is the basis for the WASH 1400 PWR transient event tree (Figure 2) which translates the more general functional event tree into a more detailed event tree representing specifically the Westinghouse design. In the present report, the functional event tree from WASH 1400 is translated into a more detailed event tree representing specifically the B&W design (TMI-2)(Figure 3), since the tree for the Westinghouse design is not applicable for TMI-2.

The principal difference between the Westinghouse and B&W designs in regard to possible sequences of events following transient loss of main feedwater is that the pilot-operated relief valve (PORV) is normally required to operate in the B&W design, but not in the Westinghouse design. Westinghouse reactors have higher pressure settings. Hence, pressure in the primary coolant system does not rise rapidly enough to require operation of the PORV. B&W reactors had lower pressure settings prior to the TMI accident, but they have since been raised. The purpose of using this direct indication of a need for reactor scram is to minimize the number of unnecessary scrams of the reactor resulting from erroneous signals. In the B&W reactors, PORV operation was normally required to control system pressure until main or auxiliary feedwater supply to the steam generators becomes available to cool the reactor. Hence, PORV operation is not required in Westinghouse reactors unless auxiliary feedwater is not immediately available, and the combination of the success of auxiliary feedwater availability and failure of the PORV to close is not included in the transient event tree. In fact, WASH 1400 states that PORV failures to close (in Westinghouse reactors) are better

represented by the "small LOCA" event trees. 31/ (Accordingly, the TMI-2 accident sequence cannot be found on the WASH 1400 transient event tree, and is better described in LOCA event trees).

The functional event tree for PWR transients, Figure 1 attached to this report, includes a tracing of the path of the TMI-2 accident. More complete technical descriptions of the events are in the WASH 1400.32/ The events of TMI-2 are described as follows:

1. Transient loss of main feedwater supply constitutes event A. This is a normal occurrence at nuclear power plants. A list of possible transients is shown in Table 1.

2. Reactor scram was successful at TMI-2, which is represented by the path going up one level for event B. (Up normally represents the desirable outcome of an event).

3. Event C represents the failure to cool the core at TMI-2 through manually defeating the emergency core cooling. More generally, this involves failure to adequately cool the core even though main or auxiliary feedwater is available. This failure leads the TMI-2 path down one level in the functional event tree.

4. The TMI-2 path turns up one level at event D, because protection against overpressurization of the primary system was available at TMI-2. This involves opening of relief or safety valves on the primary system.

Hence, the TMI-2 sequence is designated "AC" on the functional event tree. Core status associated with sequence AC is that the core will eventually melt if no operator action is taken. At TMI-2, operator action to restart safety injection with the high pressure pumps prevented core melt. Further, WASH 1400 states that if adequate coolant inventory is not maintained, core damage and core melt may occur. 33/ At TMI-2, failure to keep the core flooded resulted in core damage.

OUTCOME OF TRANSIENTS IN WESTINGHOUSE PWRs

The PWR transient event tree for Westinghouse reactors is attached to this report (Figure 2). The TMI-2 path does not exist on this tree. If it were to be added to the tree, it would involve the following:

1. Transient loss of feedwater constitutes event T.

2. The path goes up for event K, due to successful scramming of the reactor.

3. The path continues up at event M, because auxiliary feedwater (unavailable for 8 minutes at TMI-2) did become available in time to constitute success of this cooling function.

4. A branch is required at event Q, to represent failure of the PORV

to close. This is not included in the tree, because PORV operation is not required in Westinghouse reactor transients which come to this point in the path. The path turns down with event Q.

5. Event U represents the failure to maintain coolant inventory in the reactor core.

Hence, a sequence designation of TQU represents the TMI-2 accident on the WASH 1400 event tree. (Further, the S2 sequence on the PWR small LOCA event tree in WASH 1400 can also be used to represent TMI-2, except that core damage must be included as a possible outcome of the sequence.)

OUTCOME OF TRANSIENTS IN B&W REACTORS

Transient events in B&W reactors are normally terminated without damage, but they can lead to loss of coolant through the PORV (either by PORV failing to close or by providing for an exit for water entering the reactor from the safety injection system) or core meltdown. The B&W event tree is attached to this report (Figure 3). It includes the path of the TMI-2 and Davis-Besse (D-B) transients. (A transient at Oconee-3 (1975) was similar to the Davis-Besse transient.) The sequence includes the following events:

1. The transient loss of main feedwater occurs (T).

2. The reactor scram system is available on demand (K). The TMI-2 and D-B transients follow the path up at this event, because scram on demand was achieved in both cases.

3. The path goes up at event P' if the PORV or the safety valves are not required to open in order to control primary system pressure and temperature. At TMI-2 and D-B they were required. The significance is that if any of these valves open, there is a chance they will fail to close, as at Three Mile Island. PORV operation is required in designs with low pressure settings for PORV actuation, and PORV or safety valve operation is always required if auxiliary feedwater is not immediately available. (NOTE: Pre-TMI-2, B&W designs included low PORV settings, so that the PORV was always required to operate in feedwater transients. Post-TMI-2 designs include higher settings so that PORV is not required unless auxiliary feedwater is not available. Thus, at TMI-2, the PORV would have been required due to auxiliary feedwater unavailability even if the PORV setting had been high.)

4. Event Q refers to failure of relief (or safety) valves to close. The TMI-2 and D-B paths turn down at event Q, due to PORV failures to close.

5. With rapid depressurization and coolant loss through the stuck-open PORV, HPSI is required to cool the core. The TMI-2 and D-B paths turn up at event U, because HPSI systems were available when needed.

6. Event U' represents failure to adequately cool the core, even though HPSI is available. This could occur if HPSI flow is interrupted, as at TMI-2, and results in core damage or meltdown. The D-B path turns up at event U', because the HPSI was successful at cooling the core until the loss of coolant through the PORV was stopped. (HPSI was interrupted by the operators at Davis-Besse but not for a long enough period to effect inadequate cooling and result in core damage.) The TMI-2 path turns down here, as HPSI interruption by the operator caused parts of the core to become uncovered and core damage resulted.

7. Event U" represents loss of core coolability leading to core meltdown. This can result from failure to recover HPSI flow before the core melts, collapse of a severely damaged core into an uncoolable geometry before HPSI flow is restored, or large quantities of noncondensible gases (hydrogen) blocking flow paths in the primary coolant system. HPSI recovery at TMI-2 constitutes success of event U".

Hence, the TMI-2 sequence is designated TP'QU', and represents the two failures in the accident -- PORV stuck open and interruption of HPSI flow. The D-B sequence is TP'Q. In both sequences, the TP' represents the normal situation of a transient in which the PORV is required to operate. Note that P' does not constitute failure unless PORV is not normally required to operate, but auxiliary feedwater is not immediately available.

The top sequence T and the sixth sequence TP' are the normal outcomes of transients. The difference between the two is in the requirement for PORV operation. Most other sequences result in loss of coolant through the PORV, either to accommodate HPSI flow through the system or due to failure to close.

These outcomes are designated PORV LOCAs, and involve core damage, if the sequence ends with U' (as at TMI-2). PORV LOCAs do not involve core damage if the sequence ends with L or Q (as at D-B). Core melt is the outcome in sequence TK and in sequences ending with U or U".

Differences between the WASH 1400 transient event tree (for Westinghouse reactors) and the B&W tree represent differences in the response to transients. Event P' is added to the B&W tree to separate situations in which PORV operation is required from those in which it is not. Event Q occurs earlier in the B&W tree, because its function is important earlier in B&W transients. WASH 1400 event M is combined with event L (prolonged auxiliary feedwater availability) in the B&W tree. Due to the high reliability of pressure relief and safety valves, WASH 1400 event P (failure to open) is not included in the B&W tree. WASH 1400 event U (HPSI availability) is divided into three events for B&W reactors, to represent HPSI availability, interruption resulting in core damage, and recovery in time to prevent core melt. The U' and U" events are added to the B&W tree because direct indication of PORV position is not available. The operator may not recognize the PORV LOCA, and may respond by turning HPSI off. WASH 1400 event W refers to final cooldown of the

reactor and availability of residual heat removal systems. Because of the long time available to put these systems into service (i.e., days), it is not included in the B&W tree.

SIGNIFICANT OUTCOME OF B&W TRANSIENTS

The event tree for B&W reactor transients (Figure 3) shows where deviations from the TMI-2 accident path would have significantly altered the course of the accident. The sequences one step different from that of TMI-2 (TP'QU') are discussed here:

(TK) If scram had not been successful, core melt down may have resulted. Due to the high reliability of scram systems and the high concentration of boron in the HPSI water, this was a very unlikely path.34/

(TP') The normal outcome of the transient loss of main feedwater is TP'. It requires successful operation of the PORV and availability of main or auxiliary feedwater within about one hour.35/

(TP'QU) Unavailability of HPSI may have led to core meltdown. The WASH 1400 estimate of HPSI reliability is 99 percent.36/

(TP'Q) The PORV LOCA sequence involving no core damage, as at D-B, would have been the outcome if HPSI had not been interrupted long enough to damage the core.37/

(TP'QU'U") Failure to restore HPSI flow may have resulted in core meltdown.38/

Sequence T and the four TL- sequences are not applicable, because PORV operation required in the TMI-2 transient involved delayed scram. If the PORV had not been required to operate, these sequences would have been applicable, and the most likely outcome would have been sequence T.

USE OF EVENT TREES IN SAFETY ANALYSIS

The accident probabilities in WASH 1400 are based on the event trees. Much of the work in WASH 1400 went into calculating probabilities for each of the events in the trees. The probability of any sequence was calculated by multiplying the event probabilities in the sequence together.

The results were used in WASH 1400 risk estimates and in identification of significant sequences which contribute more to the risk than others. It was found, for example, that one sequence (the "interfacing systems LOCA") contributed heavily to the risk associated with the Surry PWR, and a few similar reactors, and that a simple change in the inspection procedure for some valves could decrease the chance of the accident by a factor of 20. This led to NRC action to effect the change.

Further, WASH 1400 event tree results indicated that the accident types that dominate the risk are small LOCAs and transients (TMI-2 was a transient). It had previously been thought that large LOCAs (double-ended pipe breaks) were the major risk contributors and this thinking guided NRC safety research finding. WASH 1400 pointed out that consequences of small LOCAs and transients can be just as severe as those of large LOCAs, and their probabilities of occurrence are much higher.

Since TMI-2, changes have been implemented in B&W reactors to cause reactor scram upon turbine trip and raise the PORV pressure setpoint. At TMI-2, this would normally not have required the PORV to open, but since auxiliary feedwater was not immediately available, the PORV still would have opened. In this situation, the PORV would still have stuck open. The effect of the PORV setting change on the probabilities of outcomes of other transients in B&W reactors can be predicted by determining probabilities for the events in the B&W tree and using these to calculate probabilities for the sequences. This analysis is presented below.

The change affects event P' in the event tree (required PORV operation). With scram assumed to operate properly, only the probabilities of the events to the right of K in the tree need to be considered. WASH 1400 probabilities for the events are used when possible.

Event P' (Required PORV operation) - Probability is one for reactors with low pressure settings and about one in 100 for those with high settings (due to auxiliary feedwater availability)

Event Q (PORV stuck open) - WASH 1400 estimate for PORV stuck open is one in 100.

Event L (Secondary side cooling restored) - WASH 1400 estimate for not recovering auxiliary feedwater is approximately one in 10,000.

Event U (HPSI available) - WASH 1400 estimate for HPSI failure is approximately one in 100. Note, however, that with the higher PORV setting, HPSI ability to pump water into the core is diminished in transients with loss of main and auxiliary feedwater. In these

cases, operator action to open the PORV manually is required to lower primary pressure. The probability that the operator will not respond properly and HPSI will not be available is roughly estimated to be one in 10.

Event U' (HPSI interrupted) - In two TMI-2-like transients at D-B and Oconee-3, HPSI was interrupted by operators. The chance of HPSI interruption for the purposes of this analysis, is estimated to be one in 10.

Event U" (HPSI restored) - At TMI-2, HPSI was restored before core meltdown. The chance that HPSI will not be restored is estimated to be one in 10.

Using these values, the probability of the TMI-2 sequence having proceeded to core damage in sequence TP'QU' is:

(probability of Event Q) x (probability of Event U') _
(1 in 100) x (1 in 10) = 1 in 1,000.

Thus, for every 1,000 transients in B&W reactors involving delayed scram, one was expected to go the way of TMI-2.

The mathematics of probability combinations for other sequences is not included here, but the results are summarized as follows:

TABLE 2: Probability of Outcomes for B&W Transients

	Normal Termination	PORV LOCA	Core Damage	Core Meltdown
With Lower PORV Pressure Setting (Pre-TMI-2)	99%	1 in 100	1 in 1,000	2 in 10,000
With Higher PORV Pressure Setting (Post-TMI-2)	99.98%	2 in 10,000	2 in 100,000	1 in 100,000

This table indicates that both PORV, LOCAs, and core damage are 50 times less likely in the transients not requiring PORV operation than in those which do. Core melt is 20 times less likely. The reason is that the

critical component when the PORV is required is the PORV itself. For the transients in which the PORV is not required, the critical component is the more reliable auxiliary feedwater availability. The PORV failure rate is 100 times the product of auxiliary feedwater and PORV failure rates. Note, however, that the higher PORV pressure setting may degrade the reliability of the HPSI in transients with loss of main and auxiliary feedwater.

This analysis shows how event trees could have been used prior to TMI-2 to identify a simple change that could have greatly reduced the probability of the TMI-2 accident (and the D-B and Oconee-3 transients), just as they were used in WASH 1400 to reduce the probability of the "interfacing systems LOCA" in the Surry reactor. Note that this analysis was possible even with very rough estimates of the event probabilities. While the probability estimates presented here for outcomes of B&W reactor transients are not strictly reliable, the basis for comparison between the higher and lower PORV pressure settings is appropriate. This example positively demonstrates the utility and importance of applying WASH 1400 techniques.

FINDINGS FROM STUDY OF WASH 1400

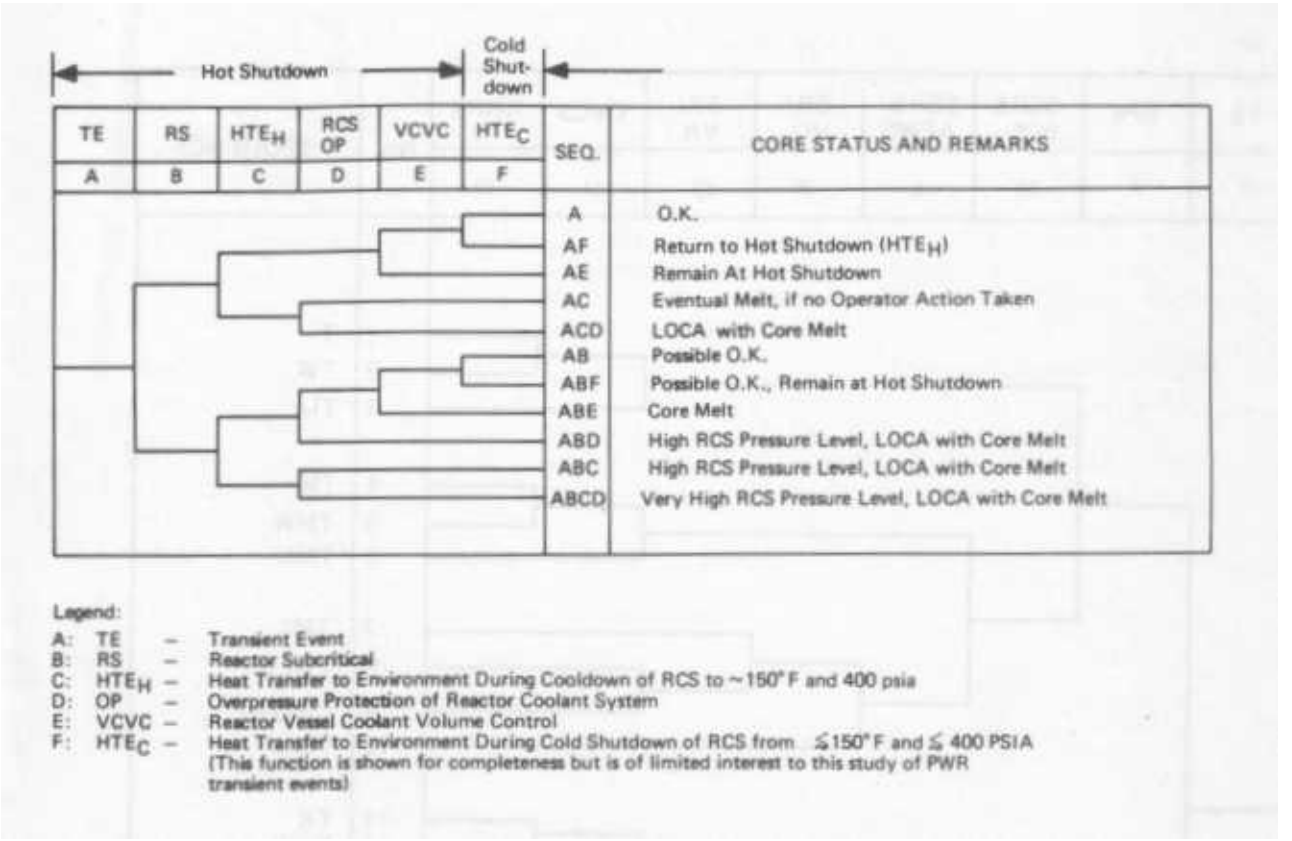
1. Predictions based on WASH 1400 accident probabilities indicated nearly one chance in six of a radioactivity release accident by the end of the 1970s, about one chance in two of one accident by the end of the 1980s. We should have anticipated TMI-2 and been prepared to deal with it.

2. Use of WASH 1400 analytical techniques can and have revealed "weak links" in the safety of nuclear reactors and have led to significant improvements in safety.

3. WASH 1400 is remarkably defined and accurate in its description of events corresponding to those which occurred at Three Mile Island.

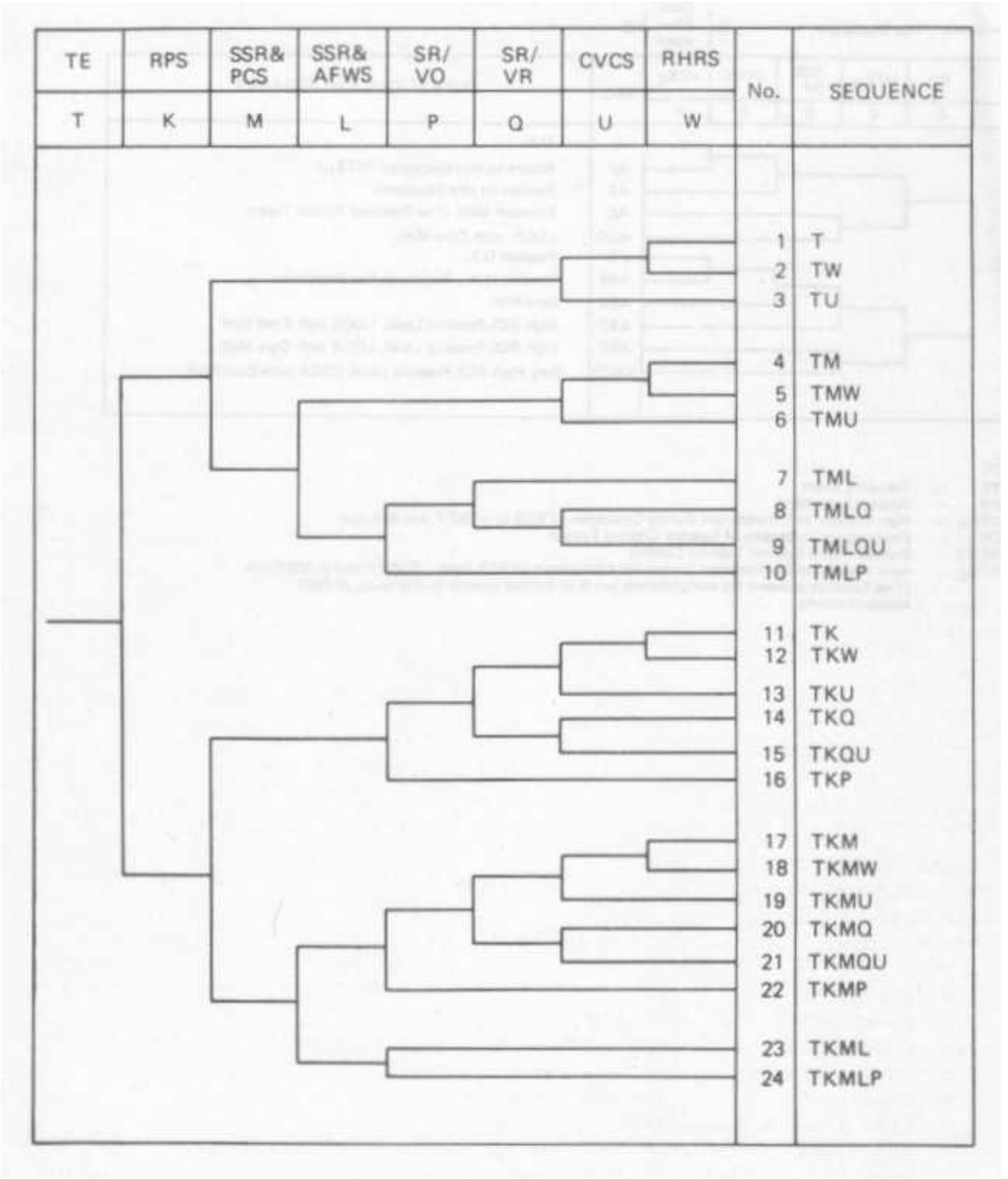
4. So much effort has been expended in criticism of WASH 1400 that little attention has been paid to the messages it contains.

FIGURE 1: Functional Event Tree -- PWR Transient Events



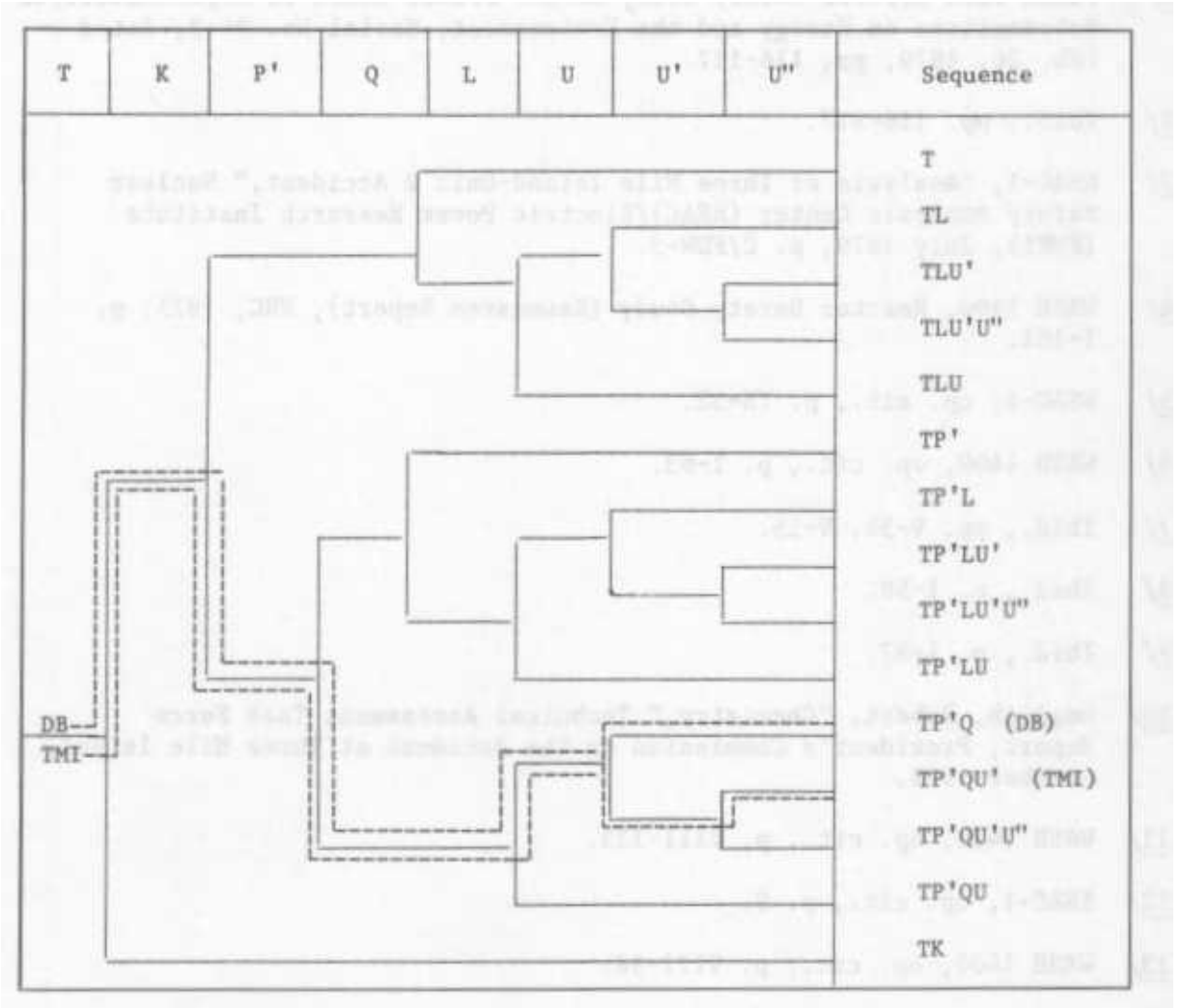
Source: WASH 1400, "Reactor Safety," An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix I, Figure 1-4.13, October 1975.

FIGURE 2: PWR Transient Event Tree



Source: WASH 1400, "Reactor Safety Study," An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix I, Figure 1-4.14, October 1975

FIGURE 3: Transient Event Tree for B&W Reactors



Normal Termination: T TP'
PORV LOCA (no damage): sequence ending in L or Q
Core damage: sequence ending in U'
Core meltdown: sequence ending in U, U", or K

T - Transient event
K - Scram availability (up/yes)
P' - PORV or safety valve operation required (down/yes)
Q - PORV or safety valve stuck open (down/yes)
L - Secondary side cooling restored (up/yes)
U - HPSI available (up/yes)
U' - HPSI interrupted and PORV open for sufficient period of time to cause core damage (down/yes)
U" - HPSI restored and PORV closed in time to avoid core meltdown (up/yes)

NOTES

1/ Taken from Reactor Safety Study Review before House of Representatives Subcommittee on Energy and the Environment, Serial No. 96-3, dated Feb. 26, 1979, pp. 116-117.

2/ Ibid., pp. 116-117.

3/ NSAC-1, "Analysis of Three Mile Island-Unit 2 Accident," Nuclear Safety Analysis Center (NSAC)/Electric Power Research Institute (EPRI), July 1979, p. C/FDW-3.

4/ WASH 1400, Reactor Safety Study (Rasmussen Report), NRC, 1975, p. 1-161.

5/ NSAC-1, op. cit., p. TH-32.

6/ WASH 1400, op. cit., p. 1-63.

7/ Ibid., pp. V-38, V-55.

8/ Ibid., p. 1-58.

9/ Ibid., p. 1-87.

10/ English, Robert, "Chemistry," Technical Assessment Task Force Report, President's Commission on the Accident at Three Mile Island, October 1979.

11/ WASH 1400, op. cit., p. VIII-123.

12/ NSAC-1, op. cit., p. 9.

13/ WASH 1400, op. cit., p. VIII-98.

14/ NSAC-1, op. cit., p. TH-5.

15/ WASH 1400, op. cit., p. 1-53.

16/ WASH 1400 op. cit., p. 79.

17/ Levine deposition, Aug. 8, 1979, p. 71; Rasmussen deposition, Sept. 15, 1979, p. 23.

18/ WASH 1400, op. cit., p. 83.

19/ Rasmussen op. cit., pp. 35-36.

20/ Ibid., pp. 26-29.

21/ WASH 1400, op. cit., p. 63.

22/ Levin deposition, Sept. 15, 1979, pp. 25-26.

23/ Rasmussen op. cit., p. 25.

24/ WASH 1400, op. cit., p. 63.

25/ Rasmussen, op. cit., pp. 56, 57.

26/ Ibid., pp. 58-59.

27/ Budnitz deposition, Aug. 27, 1979, pp. 28-30.

28/ Rasmussen, op. cit., pp. 14-15.

29/ Budnitz, op. cit., p. 33.

30/ Levine, op. cit., pp. 20-21.

31/ WASH 1400, op. cit., p 1-59.

32/ Ibid., p. 1-58, 1-59.

33/ Ibid., pp. 1-58.

34/ "Alternative Event Sequences," Technical Assessment Task Force, President's Commission on the Accident at Three Mile Island, October 1979, Chapter 4.

35/ Ibid., Case 2.

36/ Ibid., Case 9.

37/ Ibid., Case 3.

38/ Ibid., Case 9.

REPORT OF THE TECHNICAL ASSESSMENT TASK FORCE

ON

ALTERNATIVE EVENT SEQUENCES

BY

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INTRODUCTION, FINDINGS, AND SUMMARY

Very nearly all discussions of the TMI-2 accident touch upon the subject of various possible sequences of events or scenarios that might have developed, starting with the actual situation and leading one way or another from the actual situation to a variety of results -- some more, some less severe than the actual accident. These alternative scenarios can be thought of as being in one of two general classes: those that impose perturbations on the sequence of events that occurred during the development of the accident, and those that postulate somewhat different initial conditions at the time of the accident. These questions can range far and wide and can quickly lead to sequences of events that contain branches too numerous to investigate.

Recognizing both the value of examining these situations and the necessity to bound the number of cases considered, a study was made in which the actual sequence of events at TMI was followed (reference 1), but at significant times in the accident one more equipment malfunction was assumed or one additional operator action or nonaction was postulated. Also, five variations in plant conditions at the time of the accident were considered. Finally, the bounding case of fuel melting under a total absence of heat removal is presented.

Based on the approach outlined above, the development of the accident was examined to determine if it was ever close to a much more dangerous condition, and, if so, what would have been the potential consequences for the general public, the plant personnel, and the plant. Those operator actions or equipment "nonfailures" that would have improved the situation or degraded the situation are mentioned as appropriate.

The discussion is restricted to the design of the physical plant and environment at Three Mile Island (reference 2). Generalizations to other designs and other postulated accident conditions should be made with extreme caution.

FINDINGS FROM THE STUDY OF ALTERNATIVE EVENT SEQUENCES

Major findings:

A. The temperature of the hottest region of the fuel during the accident may have been as high as the melting temperature of uranium oxide (UO_2) (3,123°K = 5,162°F). Some small amount of fuel in the hottest zone may have melted. (See also reference 16.)

B. No single additional operator action or equipment failure that is tied to the actual sequence of events (reference 1) at TMI would have led unequivocally to large-scale fuel melting throughout the core or significantly larger release of fission products to the environment.

C. If the high pressure injection system (HPI) had not been turned on and if no heat sink were allowed, large-scale fuel-melting could occur throughout the core. This hypothetical situation was examined and bounded by postulating a fuel-melting accident under a total absence of heat removal from the reactor vessel. This study found that containment

would not be violated and opened to the environment by a steam explosion, by over-pressure, or by penetration of the reinforced concrete base of the containment (basemat) by the action of molten fuel. Because the containment integrity would not have been violated, the release of fission products would not be changed by a large factor over what actually occurred at TMI-2.

D. Essentially all of the radioactive iodine released from the fuel in the TMI-2 accident was retained in the water in the primary system, the containment building, and the auxiliary building. This is attributed to the chemical reducing conditions existing in the water near the fuel at the time of release of the iodine, to the high pH of the water, to the high chemical activity of iodine, and possibly to the presence of silver in the reactor vessel.

E. No radioactive cesium, strontium, barium, or lanthanum has been detected in the environment even though significant quantities of these materials were transported to the auxiliary building.

Findings relative to specific alternatives to the TMI-2 events:

1. Case 1: If the auxiliary feedwater had been available as designed, the accident would not have been changed except in minor detail.

2. Case 2: If the pilot-operated relief valve (PORV) had closed as designed, there would have been no accident. The 8-minute delay in auxiliary feedwater would have been a minor perturbation.

3. Case 3: If the HPI system had not been throttled, a stable condition would have been achieved with no damage to the core. Ultimate recovery would require that the operators recognize the open status of the PORV.

4. Case 4: If the containment had been isolated within a few minutes, and if the operators bypassed isolation by opening the let-down line (as was done at about 4-1/2 hours), the accident would have been unchanged.

5. Case 5: If the iodine filters had been in good condition, the release of radio-iodine to the environment would have been reduced from about 15 curies to less than one curie. Health effects of either of these amounts of radioactive iodine in the environment are insignificant.

6. Case 6: If auxiliary feedwater had remained unavailable, the reactor might have reached a high temperature sooner, i.e., the time scale might have been shorter with the quantity of fuel reaching melting temperatures before the HPI system was restarted being somewhat greater than may have occurred in the actual event.

7. Case 7: If the PORV had remained open after 2 hours, 22 minutes, the water remaining in the core would have boiled more vigorously, giving more cooling by flow of steam. It is uncertain, however, whether the core would have contained sufficient water to continue boiling until the HPI was turned on at 3 hours, 20 minutes. Further analyses could resolve this uncertainty.

8. Case 8: If the PORV had remained closed after 3 hours, 12-1/2 minutes, the quantity of fuel reaching melting temperatures near the center of the core would have been greater than may have occurred in the actual event. Some fuel melting might have occurred.

9. Case 9: If the high pressure injection system remained throttled (at 3 hours, 20 minutes), the quantity of fuel reaching melting temperatures near the center of the core would have been greater than may have occurred in the actual event. Some fuel might have become molten.

10. Case 10: If the containment sump pump had continued operating until the time of containment isolation, the release of radioactive iodine from the environment would have increased from 15 curies to about 100 curies. The health effect of either of these amounts of radioactive iodine in the environment is insignificant.

11. Case 11: If the containment had not been isolated, there would have been little change in the release of xenon and iodine because the operators had bypassed isolation by opening the let-down line.

12. Case 12: If the iodine filters had been in much poorer condition (or not in place), the radio-iodine released to the environment would have increased from 15 curies to about 125 curies. The increase could have been larger, except that most of the radio-iodine was retained in water and little actually reached the filter. The health effect of either of these amounts of radio-iodine is insignificant.

13. Case 13: If all the zirconium reacted with water, and if all the hydrogen gas generated were burned in the containment building, the building would remain intact. If all the hydrogen detonated, the pressure loads imposed are calculated to be somewhat less than the strength of the building.

14. Case 14: If an adequate hydrogen recombiner had been available, and used, the pressure pulse or detonation at about 10 hours would not have occurred. Because this event apparently did not affect the subsequent sequence of events, the presence of an adequate hydrogen recombiner would not have altered the consequences of the accident.

15. Case 15: If the local meteorology had been different (turbulent instead of nearly stagnant), the individual and population doses would have been reduced, depending on the assumed meteorology. (The meteorology at the time of the accident was unfavorable.)

16. Case 16: If the control rods and burnable poisons were removed and if the core geometry were changed to its most reactive configuration, the TMI-2 reactor would remain subcritical.

17. Case 17: If the reactor fuel had been at end-of-cycle instead of nearly new, the course of the accident would have been changed almost not at all.

Further findings of more general applicability:

18. The presence of silver, probably from the control rods, has been detected in the sump of the TMI-2 containment building. Silver that has been vaporized in a more severe accident could serve as a trap for iodine released from the fuel, and would not cause any adverse conditions in the reactor vessel or containment building.

19. Essentially all of the radio-iodine released from the fuel in the TMI-2 accident was retained in the water in the primary system, the containment building, and the auxiliary building. This is attributed to the chemical-reducing conditions existing in the water near the fuel at the time of release of the iodine. The radioactivity of the iodine has decayed by a factor of nearly 100 million after 7 months.

20. Failure of containment would be unlikely even in the event of a steam explosion developing out of a postulated fuel-melting accident.

21. Failure of containment to the atmosphere by penetration of the concrete basemat is unlikely even in the event of a postulated large-scale fuel-melting accident. Bedrock underneath the TMI plant is judged to be at least equivalent to concrete insofar as penetration by molten fuel is concerned.

22. The fission product decay heat load for a high burnup core is not significantly different at early times after shutdown from that of the TMI-2 core.

SUMMARY

Seventeen variations to the actual sequence of events have been considered in this-study, 12 of which relate to equipment or operator actions, and 5 to matters relating to conditions not tied to the sequence of events. The several cases may be classified as:

- resulting in no accident or no damage to the core (case 2,3);
- resulting in insignificant changes in the accident (cases 1, 4, 5, 10, 11, 12, 13, 14, 15, 16, 17); and
- resulting in potentially more serious consequences (cases 6, 7, 8, 9).

The cases resulting in no accident or only minor changes need little discussion; some of these terminate the accident, others create perturbations that damp out in time or reduce the consequences of the accident. Still others involve increased radioactive iodine release, but by amounts not significant for public health and safety.

Four possibly serious cases (6, 7, 8, and 9) require a more detailed study for definitive description than could be made in the time available. At best, the modified accident would have been changed only in detail; at worst, significant fuel melting in the hottest zone could have occurred. This last possibility is sufficiently uncertain and close enough to that of gross fuel melting that the consequences of a fuel-melting accident were investigated.

Such an extended accident was caused and bounded by assuming an adiabatic condition (no heat sink or water injection) at 3 hours, 20 minutes. The report documents a best-estimate analysis with detailed identification of possible errors, uncertainties, and alternative paths. Where realistic or best-estimate descriptions were not possible, a conservative path was chosen.

This portion of the study of an extended accident examined the physical and chemical effects associated with the melting of fuel and came to the following conclusions: Subsequent steam explosions would not be expected to threaten the containment. Collapse of the molten portions of fuel into an uncoolable geometry could have led to a **melting** penetration of the pressure vessel, but the subsequent pressure in containment would be less than that provided for in the design. However, the penetration of the containment concrete basemat by molten fuel is uncertain. If this should occur, the core material would be in a solidified form; also the containment rests on solid rock, thereby retarding fission product transport. It is unlikely that containment penetration to the atmosphere would have resulted, unless emergency systems designed to accommodate high temperatures and pressures in the containment were unavailable.

This extended fuel-melting accident is discussed below in the section on a hypothetical fuel-melting accident of this report and examined in more detail in the appendices.

CHRONOLOGY OF EVENTS

An abbreviated chronology of events is constructed from the more detailed chronology in the report from the staff to the Commission on the "Summary Sequence of Events" (reference 1) in order to make the discussion of alternative scenarios in subsequent sections easier to follow and relate to the events that actually occurred.

The alternative scenarios considered below are mentioned at the appropriate time in the chronology.

TABLE 1: Abbreviated Chronology of Events

Time (hrs:mins:secs)	Event or Condition	Comments
00:00:00	Feedwater pumps off, turbine trips off, emergency feedwater pumps start.	The valves that control flow of emergency feedwater were closed. All responses normal for feedwater transient. This bears on Case 1.
00:00:03	The PORV opened.	As designed.
00:00:08	The reactor scrammed.	As designed.
00:00:12	The primary pressure dropped below value at which the PORV should close.	Failure of equipment. This bears on Case 2.
00:01:45	Both steam generators boiled dry on the secondary side.	This bears on Case 1 in contributing to confusion in the control room.
00:02:01	High pressure injection pumps were actuated.	These require some seconds to reach full speed.
00:04:38	One HPI pump was throttled to "make up" flow rate, the second turned off.	This is an operator error, Case 3.
00:07:29	Reactor building sump pump was started automatically.	This bears on Case 10.
00:08:18	The emergency feedwater block valves were opened.	This restored flow to the secondary side of the steam generator. It bears on Case 1 and Case 6.

Table I - (Continued)

Time hrs:mns:secs	Event or Condition	Comments
00:10:19	The second sump pump started.	This bears on Case 10.
00:10:48	An alarm in the containment sump sounded, indicating high water level.	This bears on Case 4.
00:38:10	The reactor building sump pumps were turned off.	This stopped the pumping of water to the auxiliary building. It bears on Case 10.
01:13:29	One primary coolant pump turned off, loop B.	Flow of mixed water and steam maintained through the core by coolant pumps in loop A. Phases separated in loop B.
01:13:42	Second primary coolant pump turned off, loop B.	
01:40:37	Coolant pump 2A was turned off.	Forced flow through core was lost.
01:40:45	Coolant pump IA was turned off.	The hot-leg and cold-leg temperatures started diverging, indicating separation of steam and water.
02:22:00	The block valve (RC-V2) in series with the PORV was closed.	This action stopped the major leakage of water and steam out of the primary system. This bears on Cases 7 and 8.
02:54:09	The operator started reactor coolant pump 2B. Flow was indicated for only a few seconds then returned to zero.	A slug of water would have provided some cooling.
03:12:28	The block valve (RC-V2) was opened.	This action allowed greater steam flow and cooling to the fuel.
03:17:10	The block valve (RC-V2) was closed.	These two items bear on Case 7.

Table I - (Continued)

Time hrs:mns:secs	Event or Condition	Comments
03:19:45	The HPI pumps were started by operator action, injecting water into the primary system.	This action terminated the temperature transient.
03:37:00	One HPI turned off.	One pump remained running. These actions bear on the hypothetical fuel-melting section of this report.
03:40:00	The block valve (RC-V2) was opened.	Additional cooling.
03:55:39	The containment was isolated when the pressure reached 3.2 psig. The set point was 4.0 psi. HPI was initiated.	This bears on hypothetical fuel-melting accident. Two pumps running, and injecting 1,000 gpm.

SPECIFIC ALTERNATIVE EVENT SEQUENCES

The questions considered in this section include those items that have been identified as being recognizable equipment malfunctions or errors in operator action that occurred during the TMI accident. These items are examined to assess the real influence each exerted on the events as the accident developed (Cases 1-5).

Also, an analysis of the events in the TMI accident was conducted to ascertain the possible consequences of a single further equipment malfunction or operator error of omission or commission. Each postulated failure has been placed into the TMI scenario at the most serious time and followed to judge its eventual consequences (Cases 6-12).

Also, five variations in plant conditions are considered (Cases 13-17).

CASE 1 - EFFECT ON THE TMI ACCIDENT IF AUXILIARY FEEDWATER WAS AVAILABLE AS DESIGNED

The Babcock & Wilcox (B&W) primary system design has steam generators of relatively small secondary side volume, such as that given the circumstances of the TMI accident (loss of feedwater, turbine trip, and reactor trip in 8 seconds), the secondary side of the steam generator will boil dry in less than 2 minutes (this was observed in one minute and 45 seconds). This loss of heat sink could be a serious matter if not compensated for by other design feature(s). In the TMI-2 design, auxiliary heat sinks are provided by releasing coolant via the pilot-operated relief valve (PORV) (opens at about 2,250 psi) or the safety valves (open at about 2,450 psi) and by injection of compensating water (at a lower temperature) through actuation of the high pressure injection (HPI) system (reference 2).

The relief valve may have been regarded as an active part of the primary system heat sink; in fact, in Case 6 discussed below, the feedwater for the steam generator is assumed to be <u>not</u> available at any time in a postulated accident. For that rather extreme case, it is shown in reference 20 that, with a B&W design, the reactor core and primary system can survive intact without the main or auxiliary feedwater system.

Thus, the 8-minute delay in the availability of auxiliary or emergency feedwater added considerable confusion to the beginning of the transient but did not change the course of events in any significant way. Further discussions with those persons expert in thermal hydraulic problems (reference 3) lead to the conclusion that the two situations (auxiliary feedwater available on demand or after an 8-minute delay) would not be different except in detail. Pertinent variables (pressure, temperature) would converge to the same values.

CASE 2 - EFFECT ON THE TMI ACCIDENT OF PILOT-OPERATED RELIEF VALVE (PORV) CLOSING AS DESIGNED

If the relief valve had closed as designed at 2,050 pounds per square inch (psi), the incident would have been nothing more serious than a reportable event. Given a circulating, closed primary system

with no heat sink but with a heat source, the temperatures and pressures would again rise and open the PORV a second time, and it would attempt to close when the pressure dropped to 2,050 psi. We judge that this would happen in less than 8 minutes (time when the auxiliary feedwater valves were opened), but probably not more than two or three times in the interval. Even if operating within the design specifications (opening as pressure increases to about 2,250 psi and closing as pressure drops to 2,050 psi), the PORV would have provided an adequate heat sink during the first 8 minutes while auxiliary feedwater was valved off. The HPI would not actuate automatically because the pressure would not have dropped to the trigger level. The primary system would be short of a full supply of water but not by an amount of significance. The primary coolant would still be a single phase liquid with forced circulation providing good heat transfer from the core. There would be no damage to the core. When the auxiliary feedwater turned on after 8 minutes, the system would become stable indefinitely, allowing the operators to identify the cause of shutdown and restore the plant to an operable condition.

CASE 3 - EFFECT ON THE TMI ACCIDENT OF NOT THROTTLING HIGH PRESSURE INJECTION (HPI)

If the high pressure injection system had not been throttled, a stable heat balance would have been achieved without damage to the core.

For the case at hand, the PORV is assumed open and primary system circulation is continuing. At 2:01 minutes, the HPI system pumps started. These pumps have the capability to force open the relief valves. Thus, the fact that the auxiliary feedwater valves were closed for 8 minutes would cause only a perturbation on establishing an equilibrium condition. The equilibrium situation would be one in which the PORV would be open (occasional opening of relief valves might occur) and water makeup and high pressure would be provided by the high pressure injection system. Heat would be rejected via the PORV (and occasionally the safety valves) and, after 8 minutes, the steam generators. Ultimate recovery would require that the operators recognize the state of the PORV.

Analysis of the reactimeter data from TMI-2 (reference 1) shows that this mode of cooling was, in fact, realized during the period from about 2 minutes to 4.5 minutes, when with HPI activated and the PORV open, heat balance was established as indicated by a stable (constant) coolant temperature.

CASE 4 - EFFECT ON THE TMI ACCIDENT RADIOACTIVITY RELEASE IF CONTAINMENT ISOLATION OCCURRED WITHIN 10 MINUTES

The analysis of this situation requires that the major leakage paths from the primary system to the auxiliary building be identified. This has been completed and is discussed in a staff document submitted to the President's Commission (reference 4).

The conclusion reached is that the major leakage path for water containing fission products in solution was by way of the let-down line from the primary system. This let-down (and associated make-up) system

is the major control system for water chemistry, water conditioning, and maintenance of water inventory in the reactor coolant primary system. Control can be either manual or automatic. Importantly, during reactor operations or standby, this system is used to provide conditioned injection water for primary coolant pump seal, lubrication cooling, and make-up for leaking valve seals.

Apparently the decision was made by the operators during the accident that the operational value of the make up/let-down system was important enough that it must be kept in operation, even after the containment was isolated.*/ A positive indication is shown in the chronology of events about a half-hour after isolation that the necessary valves must have been operated to restore the let-down system to operation (reference 1). The evidence is a high temperature indication in the let-down cooler suggesting a high flow rate of water.

Given this action we assume that the same action would have been taken if the containment had isolated at 10 minutes, and there would have been no change of significance to the events of the TMI accident.

CASE 5 - EFFECT ON THE TMI ACCIDENT OF PROPERLY FUNCTIONING IODINE FILTERS

According to the water analyses received to date (discussed in Appendix A and E), most of the radio-iodine released from the fuel during the accident encountered chemical reducing conditions and entered into water solution. This condition was later enhanced by the addition of sodium thiosulfate into the primary system and in the auxiliary building and injection of sodium hydroxide into the containment atmosphere. Thus, most of the iodine was retained in water and, apparently, only a very small fraction was available to escape as a gas.

Analyses of some of the filter material show that about 110 curies of iodine-131 were deposited on the charcoal (references 5, 6, 7, 14). The estimated release to the environment was about 15 curies. If the filters had been in better condition, this release of 15 curies would have been reduced to less than one curie. The fractional change in release would have been large, but the environmental change would have been small.

CASE 6 - AUXILIARY FEEDWATER REMAINS UNAVAILABLE

The NRC's "Reactor Safety Study" published in 1975 (reference 8), examined this type of accident in which all feedwater was unavailable,

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*/ It should be noted that a crucial element in the ultimate restoration of cooling to the core was the availability of the primary coolant pumps 16 hours after the initiating event. The availability of conditioned injection water for the primary coolant pump seals from the let-down system may well have been basic to maintaining the operability of these pumps.

with the conclusion that fuel melting could be a consequence in the absence of proper operator intervention. That analysis did not consider a B&W design pressurized water reactor (PWR).

An answer to this question that is more specific to TMI-2 is available from a review of a similar situation analyzed with the TRAC code prior to the TMI-2 accident. These results are discussed in more detail in reference 20. A summary of relevant results is given here. In this prior analysis a condition is postulated (for a B&W reactor) in which: (a) all steam generator feedwater is lost, and (b) electrical power to the primary coolant pumps is temporarily lost, thus preventing circulation of water in the primary system.

In this analysis, the heat sinks provided are the safety valves rather than the PORV. Safety valves are designed to open at 2,450 psi and to remain open to lower pressure than does a properly functioning PORV. The question addressed is the extent of core damage, if any, and whether the plant can survive with the HPI supplying water (cold) and safety valves providing a heat sink. The answer is affirmative; the B&W design can survive complete loss of feedwater. One factor making this possible is the capability of the HPI pumps to inject water at pressures up to 3,000 psi.

A clear result of the study is that if auxiliary feedwater and primary circulation are <u>not</u> available, the HPI system must be operated at a rate sufficient to maintain inventory and force heat removal through the safety valves. Ultimate recovery would depend on control of relief valves and recovery of alternate heat sinks.

From the existing knowledge of the TMI-2 accident and this related study, we deduce that loss of feedwater beyond the actual 8-minute delay would have changed the time scale of the accident; that is, events might have occurred at earlier times, but the basic requirement that the HPI system must be initiated remains. Given initiation of the HPI system at a time equivalent in fuel temperatures, the consequences would be about the same. Delay of HPI operation would find the core suffering greater damage, the amount of damage unknown without careful study by a thermal hydraulic code such as TRAC or RELAP (references 9, 10, 11).

<u>CASE 7 - PORV REMAINS OPEN AT 142 MINUTES</u>

The pilot-operated relief valve (PORV) opened 3 seconds into the transient and remained open for 2 hours, 20 minutes. At that time a block valve in series with the PORV was closed, thus effectively closing the relief valve and reducing steam flow through the core somewhat. At this time, only the lower portion of the core was covered with saturated water, creating steam which passed through the upper part of the core. Fuel temperatures were rising and some steam interacted with the zircaloy, either to cool or (for high temperature zircaloy) to react chemically to create zirconium dioxide (ZrO_2) and hydrogen. This temperature transient was terminated when the HPI flow was restored at 3 hours, 19 minutes.

The question to be addressed is the difference between the sequence of events that occurred and those which would have occurred had the block valve not been closed at 2 hours, 20 minutes.

The difference between the actual situation and the postulated situation would be seen only by a careful calculation, probably by a thermal-hydraulics computer code such as TRAC (reference 10) or RELAP (reference 11). The open relief valve would allow a greater flux of steam past the fuel pins, and the resulting lower primary system pressure initially would allow more vigorous boiling of the water remaining in the core. Sufficient steam flow could hold the fuel cladding below the temperature of rapid oxidation, but the ultimate depletion of water supply (until the HPI were turned on) would eventually allow high core temperatures and significant core damage. It is quite possible that the extra steam cooling provided could have prevented some of the damage to the core.

Only a careful calculation can reveal whether adequate cooling to prevent fuel melting would be available until the time of restart of the HPI system. If cooling were inadequate an entry into the class of conditions discussed in the section on a hypothetical fuel-melting accident is possible.

Given full flow of the HPI system, the primary system would gradually refill, and pressure would be restored. At that time, the main circulation pumps could be turned on. A heat sink would then be provided by the steam generators and the PORV. Ultimate recovery from this situation would require that the operators recognize the open status of the PORV, and that circulation in the primary system be restored.

CASE 8 - PORV REMAINS CLOSED AT 192.5 MINUTES

The pilot-operated relief valve (PORV) opened 3 seconds into the transient and remained open for 142 minutes. At this time, a block valve in series with the PORV was closed, thus effectively closing the relief valve and reducing steam flow through the core. This valve remained closed until 192 minutes 28 seconds into the accident, when it was opened for 4.5 minutes. Three minutes after it was closed, the HPI system was restarted, which then quenched the temperature transient.

The question to be addressed is the difference between the sequence of events that occurred and those which would have occurred had the block valve not been open for nearly 5 minutes just before the HPI was restarted.

The fuel cladding temperatures used in this analysis are taken from the results of the TRAC code calculations (reference 9) and which are examined and extrapolated in the Appendix B to this report. One possibility not considered in Appendix B is developed below.

The temperature in the hottest region of the core began rising significantly at about 160 minutes and reached 2,100°K (3,321°F) at about 185 minutes, as is illustrated in Figure 1 of Appendix B. At this temperature (and steam flow used in the code), the rate of increase of temperature is calculated to approach 5°K/sec having increased from the fission product decay heat rate some 5 minutes earlier. Zircaloy melts at 2,100°K, and temperatures reached beyond this time depend in part on the continued presence of the molten metal in this region of the core.

If the molten cladding drains downward to a cooler region of the core, the temperature rise will be determined by the decay heat rate which leads to about 1 °K/sec (in the central part of the core) and possibly less because of steam and thermal radiation cooling. Assuming that the rate of 1°K/sec obtains (and the PORV is not opened), the temperature of the hot region would increase to 3,000°K (temperature change of 900 °K) in 15 minutes when the HPI was restarted. Since the melting temperature of UO is taken as 3,123°K for this discussion, some fuel melting would be expected, because the fuel near the surface of the pins would be hotter than the average.

The other extreme (not considered in Appendix B) is to assume that molten cladding does not move, that all the heat generated by the zirconium-water reaction remains in the cladding or UO and the temperature rises at the rate of 5°K/sec. This reaction rate is sufficiently high that all the zircaloy in the hot spot would have been consumed by 189 minutes, some 11 minutes before the HPI was turned on. The temperature at this time could be 2,940°K, only 180°K below the fuel-melting temperature and the heating rate would drop to that caused by fission product heating, about 1°K/sec. At this rate, the melting temperature (3,120°K) would be reached in an additional 3 minutes, the time when the PORV block valve was opened. If, as postulated, this was not done, the heat produced would cause fuel to melt until the HPI was turned on 7 minutes later. As much as 60 percent of the fuel at and near the hot spot could have become molten in this case.

Neither of these extremes is likely, and a reasonable estimate would fall between. It is unlikely that a molten metal cylinder would retain its configuration and position; some draining of molten cladding could be expected. Such flow to a cooler region and subsequent freezing could form blockages and impede the flow of steam. A reduced flow of steam would reduce the oxidation rate for the remaining cladding and hence the temperature rise might be less than the maximum. On the other hand, it is unlikely that all zircaloy would flow away from the hot region of the core and, indeed, some experiments suggest none would move before complete oxidation (reference 13). Certainly local blockages of zircaloy and ZrO_2 could form.

Thus, it is very likely that some fuel would melt in this alternative sequence of events. The amount is most uncertain but could be as much as a significant fraction of the fuel in the central, local, high-power density regions. As noted above, estimates can be high or low depending on assumptions about the position of the cladding. Additional uncertainties include thermal radiation and steam flow which could afford some cooling.

Finally, the possibility that an alloy consisting of UO_2, Zr, and ZrO might be formed must be noted. This alloy has been observed (reference 16) with melting points depending on the mixture proportions. Given the extreme conditions of this alternative sequence with both Zr and ZrO_2 possibly moving and in intimate contact with UO_2, some formation of this mixture would be expected. The amount, however, cannot be estimated.

CASE 9 - THE HPI REMAINS THROTTLED

The action by the operators at approximately 3 hours, 20 minutes, when the high pressure injection system was activated, flooded water into the primary system and terminated the first temperature transient of the fuel.

The postulated action at this time is that the operators did not restart the HPI. If this had been the case, fuel temperatures would have continued to rise as discussed in Appendix B, with some melting of fuel becoming a probable event. Automatic restart of the HPI by the containment isolation signal at 3 hours, 56 minutes and the operator action to open the PORV block valve at 3 hours, 40 minutes would arrest the melting. The PORV was then open for 1-3/4 hours while the HPI was on for 21 minutes. The latter could have quenched the temperature transient again, at least temporarily, unless fuel melting had progressed past a point of coolability and recovery. The PORV would allow a greater flux of steam past the fuel pins giving some cooling, but the supply of water in the core is questionable for long-term vigorous boiling.

A possible heat sink for the core could be steam condensation in the steam generators and reflux back to the reactor vessel. However, it is uncertain how much core cooling this action could have afforded (Appendix B).

Thus, it is uncertain as to whether or not substantial fuel melting could have occurred at TMI-2 had the operators not turned on the HPI system at 3 hours, 20 minutes. However, given uncertainties as to the state of the fuel, on the availability of make-up water, and the unpredictable response of the operators to this altered sequence, we are unable to show to our satisfaction that fuel melting would not have occurred. Therefore, as a bounding case, analyses were conducted to determine what impact on the environment would result assuming the worst cooling situations -- virtually no cooling at all (no heat sink, no injection water) -- following 3 hours, 20 minutes. This scenario is presented in the section on a hypothetical fuel-melting accident.

CASE 10 - CONTAINMENT SUMP PUMP OPERATION CONTINUES

The opening of the pilot-operated relief valve (PORV) allowed steam and water to escape from the primary system into the containment through the reactor coolant drain tank (reference 2). When water in the containment building sump reached the requisite level, the two sump pumps began operating automatically. The first responded to a signal at 7 minutes, 19 seconds, while the second started at 10 minutes, 19 seconds. These continued pumping automatically until the auxiliary operator turned them off at 38 minutes and 10 seconds (reference 1). During this half-hour period, we believe that the 8,400 gallons of water removed from containment contained no or very few fission products and that most of the fission products reaching the auxiliary building did so later by way of the let-down line (reference 4).

The immediate problem faced is addressed in reference 21, in which it is postulated that the sump pump continues in operation until containment isolation at 3 hours and 56 minutes. The assumptions as to the amount of water, timing of fuel failure, extent of fuel failure, and disposition of the contaminated water in the auxiliary building are discussed in detail. The conclusion reached is that the amount of iodine released to the environment would be about seven times as large, or a total of about 100 curies, compared to that believed to be released, which was 13-15 curies. Neither amount in the environment would be cause for a public health problem.

CASE 11 - CONTAINMENT NOT ISOLATED

The containment atmospheric pressure rose to 3.2 psi above ambient pressure at about 4 hours, causing the containment to be isolated or sealed closed. This action presumably blocked most if not all leakage paths from the containment to the auxiliary building. However, an analysis of the leakage paths shows that the let-down line accounted for most of the fission product activity that reached the auxiliary building (reference 4). The blockage of this line by the containment isolation action was defeated by the operators to provide flow for the primary loop pump seals, as discussed above in Case 4.

Thus, we conclude that with the let-down line the primary source of leakage, the isolation of containment had little effect on the leakage of fission products and had it not been isolated, the results would have been similar.

CASE 12 - GREATER DETERIORATION OF THE IODINE FILTERS IN THE AUXILIARY BUILDING

The iodine filters had been installed in the auxiliary building for some months before the time of the accident and had not been changed (reference 14). Based on measurements in the environment, it is estimated that 15 curies of iodine escaped from the plant during the accident.

Analyses of sections of the filters suggest that only 110 curies were deposited during the accident (reference 5, 6, 7). Thus, it appears that very little iodine actually reached the filters, primarily because of the holding ability of water in the auxiliary building, the primary system, and the containment. (This latter point is discussed in Appendix F.)

If all iodine reaching the filters had passed through the filters or if the filters had not been in place at all, the release to the atmosphere would have increased from about. 15 curies to about 125 curies with little change in consequences. The inventory of iodine-131 in the core at the time of the accident was about 65 million curies; the total inventory of all iodine isotopes, most of short half-lives, was about 10 times as large.

CASE 13 - ALL HYDROGEN FROM CORE DAMAGE BURNED IN THE CONTAINMENT BUILDING

At TMI-2, the reactor contained an estimated 49,700 pounds of zirconium (reference 15) which, if all reacted with water, could have produced 2,200 pounds of hydrogen. At TMI-2, about half this quantity of hydrogen (41-58 percent) was produced, and of that amount, 57-67 percent burned in the 2.1 million cubic foot containment building (reference 16). Consideration is given to the possible consequences of producing all this hydrogen and burning it in the containment building.

As TMI-2 demonstrated, ignition sources for this hydrogen are available. If the whole 2,200 pounds of hydrogen gradually were released to the containment atmosphere, several ignitions probably each would burn a portion of the entire amount. Since there are a large number of ignition sources available (relays, switches, etc.), the incremental burn off of the hydrogen in containment is considered most probable. In that case, the course of the accident would likely have been little changed.

The consequences of burning all the hydrogen at once are assessed in WASH 1400 (reference 8) and in letters from D. Rose (reference 17). Two modes of combustion were analyzed -- one an idealized constant-volume adiabatic combustion and the other a one-dimensional Chapman-Jouquet detonation. Before combustion, pressure was taken as one atmosphere, the relative humidity as 100 percent, and temperature as 120°F. The resulting instantaneous conditions are given below.

Constant-volume combustion:

Building volume, 10^6 ft.^{3}	2.1
Final pressure, psig.	79.0
Final temperature, °F	3668.0

Detonation:

Building volume, 10^6 ft.^{3}	2.1
Final pressure, psig.	166.0
Final temperature, °F	4042.0

The containment building at TMI-2 was designed for an internal pressure of 60 pounds per square inch gauge (psig) and has been proof-tested at 69 psig (reference 2). With its safety factor of 1.5, the building should withstand 90 psig static pressure without loss of its containment capability. The concrete shell might then have cracks that are visible, but the reinforcing steel should maintain the building's structural integrity. The steel plate that lines the inner surface of the walls and dome of the building should fulfill its role as a membrane that would prevent leakage of fission products even if the concrete were to crack. All this indicates that the building should successfully withstand the 79 psig gas pressure that would load the building shell for perhaps 5 seconds and then gradually decline when the water sprays inside the building to cool the air.

The detonation case presents a more difficult problem because of the dynamic interaction between the detonation's impulsive load and the elasticity of the building. WASH 1400 concludes that the containment buildings of the type approved by NRC should withstand such detonation. The Los Alamos Scientific Laboratory also evaluated this structural problem for TMI-2 by drawing on their background in explosions derived from the weapons program (reference 17). This evaluation showed that the force from the detonation would be imposed for a period much shorter than the building's periods of natural oscillations. As a result, the building's inertia, as well as its strength, is called upon to resist the detonation, and the maximum load computed in reference 17 for the structure at TMI-2 was below but close to the building's structural limit. Inasmuch as the analyses of the detonation and of the structural dynamics were each on a somewhat simplified basis, additional study is required before one could conclude with confidence that the containment building at TMI-2 could withstand such a detonation. (This matter is discussed in more detail in reference 16.)

Even if this quantity of hydrogen were to be released to the containment building, explosions of either of these magnitudes appear extremely unlikely because of the likehood that the hydrogen will be ignited before all of it enters the containment building. As TMI-2 demonstrates, the building does contain ignition sources, such as switches that arc. The peak pressure and the potential for damage would be reduced if the same amount of hydrogen were burned in several bursts, each individually smaller than the ultimate explosion. Reference 17 points out that the detonation of a lesser amount would present no containment problem.

The extent to which structural and electrical equipment could withstand the air temperature of such a severe hydrogen detonation has not been examined. However, it is known that hydrogen combustion occurred at TMI and, as stated in Case 14, had no apparent effect to the equipment.

CASE 14 - EFFECT ON THE TMI ACCIDENT IF AN ADEQUATE HYDROGEN RECOMBINER HAD BEEN AVAILABLE

Although TMI-2 had a recombiner on-site, it was not designed for this accident condition and was not put into service for several days; its capacity was low, only 0.044 gram-mole of hydrogen per second or 0.7 pound per hour (reference 2). At this reaction rate, 53-75 days would have been required in order to recombine all the hydrogen produced during the accident.

The effect on the course of the accident from having a larger recombiner already connected was, therefore, investigated. Conceivably such a recombiner could have reacted most of the hydrogen released to the containment atmosphere and thereby prevented the explosion in containment. This explosion occurred 9 hours and 50 minutes after the accident began and produced a pressure pulse in the containment building of about 28 psig.

An important factor is that this explosion had no discernible effect on the course of the accident. At the time of the explosion,

high pressure injection of cooling water into the reactor loop and spray of cooling water into the containment building both were turned on automatically, and the containment building was isolated (sealed). Twenty or 30 seconds later, the operators turned off high pressure injection and defeated containment isolation. After 6 minutes, the water spray into the building was also stopped (reference 1). At the time of the explosion, the operators apparently were unaware that one had actually occurred, and their responses were, therefore, the minor ones cited here.

Because the explosion had such a minor effect on the outcome of the accident, a recombiner that would have avoided the explosion would have had a correspondingly minor effect.

CASE 15 - EFFECT OF DIFFERENT METEOROLOGICAL CONDITIONS IN THE VINCINITY OF THE PLANT SITE

Meteorological conditions during the TMI-2 accident were not optimum for rapid mixing and dilution of the released radionuclides. The wind speed was, in fact, low and variable, and nonturbulent (low mixing) conditions existed. The effect of greater turbulence is to mix the radionuclides with a larger volume of air and, therefore, reduce the dose that any single individual might receive. The total population dose (person-rem) probably would be changed less, in spite of greater mixing and dilution, because the number of plume touch down points could increase.

A specific case was examined with the Oak Ridge National Laboratory computer program, known as AIRDOS-EPA. The wind speed and directions used were the average meteorological conditions for the months of March and April (west-northwest at 9.4 miles/hr) with average stability of turbulence conditions. Because of population density distribution, the collective dose in the sector centered about WNW would have increased slightly over what they actually were, but the total collective dose to the population within 50 miles of TMI would have decreased by more than 20 percent.

If the wind direction has been in a north-easterly direction, the nearby population dose would have been less because of a lower population density, but quantitative estimates for this and other cases could not be developed in the time available.

CASE 16 - CRITICALITY OF THE TMI-2 CORE

The present physical configuration of the TMI-2 core is not well-known, and for that reason, a study of the criticality of the core was performed to determine if there was any possibility that the reactor core could be close to a critical condition and actually become critical if the geometric configuration should change. Because the control rods may have been damaged or possibly partially melted during the accident, some calculations assumed that this poison was not in the core volume. The purpose of the study was to establish whether or not the system could become critical under any conceivable configuration of core material, control elements, and soluble boron. The detailed results of these studies are included in reference 22.

Results of these calculations are consistent with measurements made with the reactor core in its startup condition. Specifically, in the hot, zero-power configuration with the control rods inserted and 1,490 parts per million (ppm) of boron in the coolant, eff = 0.864. As a further reference, the as built cold shutdown core with control rods inserted and 3,180 ppm of boron in the coolant has $k_{eff} = 0.75$.

Three dimensional Monte Carlo-type computing codes were used for these surveys, thereby allowing detailed geometrical models to be simulated in the calculations. Various assumptions were made in modeling the disrupted core.

Calculations were made for various degrees of compaction of the core. In each case, as the core cylinder became shorter, the fuel density of the resulting core increased correspondingly, since the total fuel weight was maintained constant in the model. Results show that slumping of the core under these conditions does increase reactivity (a measure of the nearness to criticality). In fact the reactivity reaches its maximum value of 0.84 when the core is about two-thirds of its original height; however, the system is still not close to criticality even when the control rods are assumed to be completely removed, provided that the boron concentration in the water is not reduced (taken as 3,180 ppm by weight).

More sophisticated calculations were performed where an inverted bell-shaped section of the core, along its centerline, was allowed to slump, leaving a concave region at the top of the core and an increased density regip below, nearer the center. This resulted in a computed reactivity, eff = 0.862, higher than for the previous case but still very substantially subcritical. Assumed removal of the control rods in this more realistically modeled example indicates almost no resulting increase in reactivity.

These calculations demonstrate that the disrupted TMI core can be maintained safely in shutdown with present boron loaded coolant present, regardless of the condition of the core, control rods, and burnable poison rods. To our knowledge, there is no configuration that could be critical with the existing boron concentration in the water.

Separate and decades-old experimental and theoretical studies have shown that a homogeneous mass of enriched uranium or uranium dioxide, without water or other moderators, cannot be made critical unless the enrichment is more than about 6 percent U-235 (reference 18). A moderator (e.g., water) can be present as a reflector with no change in the accuracy of the statement. The TMI core enrichment is 2.5 percent.

CASE 17 - THE EIIECT ON THE TMI ACCIDENT IF THE CORE FUEL WERE IN EQUILIBRIUM AT END-OF-CYCLE

A question has been raised that relates to the fission product inventory in the core at the time of the accident. Specifically, if the core had been an end-of-cycle equilibrium core rather than fuel only 3 months old, would the accident have been significantly changed? The answer is no. The data to address this question is contained in reference 19.

In this study, the calculation of the generation of fission power and decay heat is accomplished by use of computer codes that have been tested against known experimental data. Two calculations are included: the case for the actual power history prior to the accident on March 28, 1979, and the case for an end-of-cycle equilibrium core.

The end-of-cycle fission product inventory would be very similar to that in TMI-2 for times soon after shutdown. The production of heat early after reactor shutdown (hours and days) is dominated by short-lived isotopes (half-lives of seconds to days) while the long-term decay heat (months) is dominated by isotopes whose half-life ranges between one and 30 years. Another factor that must be considered is the fissioning of plutonium in the fuel. Toward end-of-cycle, a significant fraction of the fission products derive from fission of plutonium-239 rather than from uranium-235. The distribution of fission products from plutonium is somewhat different and the abundance of short-lived isotopes is slightly lower, to the extent that for the first few seconds, the decay heat from an equilibrium, end-of-life core is actually less than in the TMI-2 core. Cooling times and the ratio of decay heat of the equilibrium core to the TMI-2 core are presented below out to one year. (These data are taken from Table VII of reference 19.)

Cooling Time	TMI-2 MW	Eq. Core MW	Ratio
1 sec	168	163	0.97
4 sec	148	145	0.98
10 sec	130	128	0.985
40 sec	103	103	1.0
100 sec	86	87	1.01
400 sec	65.2	67	1.03
1,000 sec	52.8	54.6	1.034
1 hr	35.6	37.3	1.047
2 hr	28.4	30.3	1.067
5 hr	21.4	23.8	1.112
10 hr	17.4	19.9	1.144
20 hr	13.9	16.5	1.187
50 hr	8.93	11.5	1.288
100 hr = 4.17 days	6.59	8.90	1.350
200 hr = 8.3 days	4.55	6.57	1.444
500 hr = 20.8 days	2.59	11.31	1.664
1,000 hr = 1.39 months	1.56	3.02	1.94
2,000 hr = 2.78 months	0.88	2.03	2.31
5,000 hr = 6.9 months	0.32	1.02	3.19
8,760 hr = 1 year	0.14	0.609	4.35

From this table, it can be seen that the decay heat rates would be the same at 40 seconds, but thereafter the equilibrium core has the higher value, with the ratio increasing with time. For the first hours and days, the differences probably are not significant and are smaller than the uncertainties in the fraction of heat-producing isotopes that might be boiled out of a very hot fuel.

We conclude that the course of the accident would have been little altered had the fuel been end-of-cycle instead of relatively new. However, there would have been a substantial increase in the long-lived activity to be coped with in the cleanup operation.

Sufficient uncertainty remains in the analyses of alternative sequences that the possibility of the melting of a significant fraction of the fuel in the hottest zone cannot be precluded. Because this possibility is sufficiently close to that of gross fuel melting, and because additional inappropriate actions were possible, it was decided that the consequences of such an accident should be investigated. A hypothetical accident was therefore defined by assuming conditions that can be thought of as an extension of the more serious of the alternative sequences discussed above. Such an extended accident can be derived, for example, from Case 8, in which the PORV is assumed to be not closed at 192 minutes; should HPI not be turned on, fuel conditions would progressively worsen. In Case 9, if the HPI is not turned on at 200 minutes and if further actions, manual or automatic,' are not allowed, it is clear that fuel melting would be initiated. This last case is used as the descriptive entry point into fuel melting.

The postulate to initiate this sequence of events is that the operators do not restart the HPI at 3 hours, 20 minutes and do not allow any cooling or injection of water after this time. The sequence involved will be discussed briefly, and then the principal uncertainities in following consequences will be identified for each branch of significance. Details are discussed in the appendices.

Time to Reach Melting Temperatures. At 200 minutes, the HPI is not restarted; fuel temperatures are rising and will reach melting temperatures at some point in the core in less than an hour, possibly only a few minutes, depending on the assumptions.

Fuel Melting. Some fraction of the core is assumed to melt, to reach the lower plenum in the reactor vessel, and to release heat to the water. The time for melting could take as little as a hour but could be much longer, depending on steam flow and efficacy of thermal radiative cooling. No proof is offered that a large fraction of the core melts; the fraction might be small.

Steam Explosions in the Reactor Vessel. There is enough energy stored in molten fuel such that if highly efficient transfer of this energy to water were to occur upon contact, it'is conceivable that the explosive force of rapidly generated steam could rupture the pressure vessel and threaten the containment. As a practical matter, however, it is difficult to postulate physical mechanisms that could permit highly efficient energy transfer from fuel to water. Further, it is difficult to imagine how large quantities of fuel and water could be brought to

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^/ The TMI-2 core was flooded with water by operator initiation of the HPI system at 200 minutes. To continue to deny any remedial measures by the operators is very conservative and unrealistic. Such measures would be possible well into this hypothetical fuel-melting accident, but these were denied in order to investigate consequences to the extent possible in the time available.

interact simultaneously, since a sufficient quantity of fuel would very likely not melt at the same instant, nor would a sufficient quantity of molten fuel all contact the water at the same instant. In addition, recent reactor safety experiments directed at resolution of the steam explosion potential indicate that mechanisms for efficient interactions are not found. The conclusion is reached that a steam explosion in the pressure vessel would not rupture vessel or containment.

Debris Bed Cooling. When the molten fuel falls in the water in the lower plenum, cooling and fragmentation is expected. Debris beds of certain particle sizes can be cooled by water under high pressure conditions. Large or very small particle size debris beds cannot be cooled. Because no believable predictive model is known, the assumption is taken that the fuel will form a molten pool.

Penetration of Reactor Vessel. Given the existence of a large amount of molten fuel in the vessel, penetration is predicted to occur in a relatively short time, some tens of minutes. Proof of a large amount of molten fuel in the vessel is not offered.

Containment Pressure. Given failure of the reactor vessel and release of steam and hot fuel, the pressure in the containment is evaluated. All of the latent and sensible heat in a mass equal to that of the whole core is placed into the vaporization of water and added to the pressure already present. The total by this conservative method is less than that in the design-basis accident.

Fuel Reaching the Cavity Below the Reactor Vessel. The matter of steam explosions is considered a second time. Because cavity pressure venting leakage paths exists, consequences of the interactions between fuel and water in this area are less serious than in the vessel. The amount of water in the cavity at this time may be insufficient for the purpose of producing significant steam explosions.

Penetration of the Containment Basemat. The action of hot fuel on the containment floor is to melt, erode, and disintegrate the reinforced concrete. Steel reinforcing and metallic oxides would dissolve in the molten uranium dioxide. Water vapor and carbon dioxide would be liberated in the reaction and reach the containment air space either through or around the molten material. Penetration of the basemat is predicted to occur in no less than 3 days but maybe never. The mixture is calculated to have solidified in one or 2 days, well before the minimum predicted time for penetration of the basemat. Should the basemat be penetrated, the material would encounter solid siltstone, which is essentially impervious to water. The containment air pressure will be above atmospheric because of the addition of water vapor and gases from the decomposition of concrete, but the design pressure will not be exceeded.

Containment Failure. The three mechanisms that might cause failure of containment -- projectiles from a steam explosion, overpressure, and penetration of the basemat -- have been examined. The conclusion is reached that containment would not fail and result in an uncontrolled release of fission products to the atmosphere, and the release generally would be less than that in the design basis.

Fission Product Behavior. The fission product inventory in the containment is estimated by use of assumptions depending in part on WASH 1400, and also by dependence on the actual experience gained at TMI-2 during and after the accident. The two methods give different results for iodine, depending on the assumptions relative to formation of methyl and other organic iodides, WASH 1400 being the more conservative. Essentially, all of the noble gases must be assumed to be released to the containment atmosphere, in consonance with other studies and the design bases.

SOME UNCERTAINTIES IN**THE ANALYSIS.**

Some of the areas of uncertainty involved in the fuel-melting sequence of events are listed below. Many of these topics are matters of active research by the NRC* and will continue to be so treated for some time to come. For example, the topic of steam explosions has been investigated extensively throughout the international reactor safety community for more than a decade, but it is clear that there will never be controlled experimental data on the large masses of molten fuel that could possibly be involved in the melting of a fraction of the fuel in a reactor. Hence, while such events can be bounded, the sequence of events can never be predicted with absolute certainty. The existence of uncertainties in specific parts of the overall fuel-melting scenario does not mean, however, that reasonable estimates of the consequences cannot be made; information is available now to make such estimates. These estimates will continually be better refined as further research provides more understanding of each phenomenon.

Time to Reach Melting Temperatures

- Cooling water (possibly 50 gallons/minute) was not considered in the TRAC computer code analysis after 2 hours, 20 minutes.
- Thermal radiation can provide cooling of fuel pins, possibly bundles, nearest the vessel wall.
- The amount of water in the reactor vessel.

Fuel Melting

- Cooling water (possibly 50 gallons/minute) was not considered in the TRAC computer code analysis after 2 hours, 20 minutes.
- Thermal radiation can provide cooling of fuel pins, possibly bundles, nearest the vessel wall.
- The amount of water in the reactor vessel.
- The time at which fuel melting could be arrested.

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* No attempt has been made by this staff to address the question of the adequacy or priority of these research programs.

Steam Explosion

- Manner in which fuel enters water in the lower plenum, i.e., a stream or small masses which would create steam that provides cooling.
- The mechanisms for disintegration of large masses of molten fuel and its distribution in the coolant.
- The fuel particle size sufficient for rapid heat transfer to water.
- The mechanism for initiation of rapid heat transfer.
- Coherency (time and space) required to achieve pressure adequate to create a projectile.
- The necessary conditions for a projectile to penetrate both the vessel and containment.

Penetration of the Reactor Vessel

- The coolability of a debris bed or a bed of molten fuel.

Fuel Reaching the Cavity Below the Vessel

- Steam explosions.

Penetration of the Containment Basemat

- Detailed physics and chemistry involved in fuel-concrete interactions.

Fission Product Inventory

- The generality of the high-partition function of iodine in water as observed in the TMI-2 accident.
- The chemistry of iodine under accident conditions.
- The amount of methyl iodine created.

Operator Action

- PORV and block valve operation.
- HPI operation.
- Make-up/let-down system operation.

This portion of the study concludes that it is conceivable that alternate sequences could have led to significant melting in the hottest region of the core. Some additional degradation would be necessary to cause large-scale fuel melting. Should this occur, the following could be expected: Steam explosions that might occur within the reactor vessel would not be expected to threaten containment. Collapse of the molten portions of fuel into an uncoolable geometry could lead to a melting penetration of the pressure vessel. However, the penetration of the containment concrete basemat by molten fuel is uncertain. If this would occur the core material would be in a solidified form, thereby retarding fission product transport. It is unlikely that containment penetration to the atmosphere would have resulted unless emergency systems designed to accommodate high temperatures and pressures in the containment were unavailable.

The several appendices discuss the relevent issues in fuel-melting accidents in greater detail.

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APPENDIX A

CONSEQUENCES OF A HYPOTHETICAL FUEL-MELTING ACCIDENT AT TMI-2

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APPENDIX A

CONSEQUENCES OF A HYPOTHETICAL FUEL-MELTING ACCIDENT AT TMI-2

The generic problem of fuel melting is investigated under the conditions that the reactor primary cooling system is maintained in an adiabatic condition beginning at 200 minutes after turbine trip.

The adiabatic condition after 200 minutes implies that, after that time:

1 there was no HPI water injection;

2 the PORV was not vented;

3 there was no make-up water flow; and

4 there was no effective reflux of steam to the steam generators.

A considerable body of experimental data and results of computational analysis have been developed since the publication of WASH-1400 (reference 28) with regard to the calculated consequences of accidents during which a large amount of fuel melts. However, despite the additional data presently available, the required base of knowledge for answers for all potential alternatives in this postulated extension of the TMI accident is incomplete. To assess the expected consequences based on available data, a variety of studies of core behavior under conditions of fuel melting, slumping, and gross motion have been examined. These investigations indicate that the most likely off-site consequences of a gross fuel-melting accident at TMI would have been a slow leakage from the reactor containment building (RCB) at a rate no greater than and probably less than the building leakage design rate. The fission product inventory within containment is estimated in Appendix F.

There appears to be little chance that a breach of containment could have occurred with a resulting drastic increase in the severity of a postulated molten fuel accident. The bases for this assessment are outlined in this appendix, in which potential events following the start of fuel melting are described. This reasoning is developed in greater detail in the following appendices.

In some cases estimates, assumptions, and judgments based on limited experimental information or theoretical studies or knowledge of the existence of such work were required. Those areas where more experimental or theoretical information is needed are identified, or are obvious from the discussion.

PLANT CONDITION

It was assumed that the conditions existing at the time were as follows: The pilot operated relief valve (PORV) was closed at 142 minutes, and then opened for 4-1/2 minutes beginning at 192-1/2 minutes (3 hours, 12 minutes, 28 seconds). Attempts had been made to start the loop 2B

pump, but they were not effective because of vapor binding. Nevertheless, as much as 1,000 gallons of water may have been injected into the reactor vessel at this time from the stagnant B loop.

Refill of the secondary side of loop A had been initiated at 135 minutes, when the core mixture level was estimated to be about 3 feet above the bottom of the active core; makeup into the primary system was at a rate of 40-50 gallons per minute (gpm) and water loss through the let-down system was occurring at a low but unspecified rate. (This make-up rate was disallowed for purposes of this special study.)

At 3 hours, the pressure was 2,050 psig, and a mixture of steam and hydrogen filled the upper half (approximately) of the core and most of the primary side of the steam generator in both loops A and B. The lower parts of the loop seals were partially filled with water. The primary pumps were not functioning because of vapor locking by the hydrogen-steam mixture.

ESTIMATED FUEL TEMPERATURES

The Transient Reactor Accident Code (TRAC) results (Appendix B) provide a calculated record of temperature up to the time of initiation of the HPI system that is consistent with the available recorded behavior of the reactor core. These TRAC code results will be sketched briefly to outline the conditions in the core as the crucial time of HPI initiation approached and continued, assuming failure to start HPI.

The calculated fuel temperature reached a peak of 2,100°K just above the center of the core at 185 minutes. The rise in temperature at a rate of approximately 1°K/sec until 175 minutes was caused by decay heat. The rise in temperature then increased to near 5°K/sec between 175 minutes and 187 minutes due to heat available from the zirconium-water reaction -- the temperature had reached what might be called a "threshold" temperature, even though the reaction rate is a continuous function of this variable. As the cladding reached 2,100°K, the zircaloy melted and may have, in part, flowed to a cooler region where it would freeze. This may have removed the major source of heat from the hottest zone. No fuel melting to this time was indicated by TRAC. The program results indicated a rapid decrease in the computed temperature after 187 minutes, which apparently is due to a calculated enhanced steam cooling brought about by a flow path in the primary loop due to partial voidage in the lower loop seals. This may not be a real effect, but rather may be an artifact of the TRAC simulation. (See Appendix B for further details.) If this cooling path is not real, or if the zircaloy had not moved, further heating is expected until the PORV is opened and acts as a heat sink. In this case, some fuel melting might have occurred in the hottest region near the center of the core.

When the PORV was opened at 192 minutes, the additional boiling and steam flow cooled the fuel and the peak temperature decreased rapidly to about 1200°K until the PORV was again closed at 197 minutes; after this action, the fuel temperature rose again at the adiabatic rate of about 1°K/sec.

This rise in temperature continued, and if the HPI had not been started, the fuel temperatures would have risen again to the melting point. The factors affecting the time when this might have occurred are discussed in Appendix B. The best estimate is between five and fifty minutes after the time at which the HPI was actually initiated.

It is thought that some settling or slumping of parts of the core may have occurred at about 3 hours, 46 minutes, as also suggested by the large and widely varying core exit temperatures measured later (reference 24). This movement may have been triggered by the shock of HPI initiation to relatively brittle and hot ZrO_2 and to partially molten fuel pins near the center of the core.

DISCUSSION OF FUEL MELTING

From the discussion above and elsewhere in this report, it is apparent that at least some limited melting may have taken place in the high temperature zone near the center of the core. In order to investigate the consequences of a fuel-melting accident, it is now postulated that no injection water or cooling is allowed after 200 minutes; fuel melting would certainly take place.

Two modes of slumping of molten fuel were considered in "WASH 1400" (1975), and a clear choice between these two still has not been established.

In the first mode, molten fuel flows downward through the core support grid plate directly into the pool of water in the lower head region. Experience with molten fuel at Argonne National Laboratory indicates that fuel can flow for long distances without refreezing (reference 1). Conflicting experience at Sandia Laboratories in Albuquerque, N. Mex., and at the Karlsruke Nuclear Center in West Germany (reference 25), however, shows less mobility and greater tendency for refreezing of fuel, which implies greater difficulty for molten fuel to flow directly to the lower pool.

In the second mode, molten fuel may refreeze above the grid plate, or solid fuel may fall into spaces above the grid plate. A molten pool may eventually build up above the grid plate, which may later fall bodily into the pool below when the grid plate is sufficiently weakened.

According to calculations at Battelle Columbus Laboratories (reference 8), the rate of fuel melting is slower for the first mode than for the second, partly because additional steam is produced as fuel moves into the lower pool, which provides additional cooling of the core. For the second mode, the calculated time from beginning of fuel melting to the point of a substantial fraction of the core becoming molten is about one-half hour (reference 4, 5, 21, 29).

The Battelle calculations include both axial and radial thermal radiation cooling, which causes large temperature gradients near the outer boundaries of the core. A quantitative evaluation of this effect and estimates of the fraction of the core that would not melt or do so late in time is not available for this report.

It is likely that the real situation would involve both a relatively slow streaming of some of the molten fuel into the lower vessel, followed by a more rapid slumping of a portion of the core into the water. Peripheral fuel elements would slump much later if at all.

THE POTENTIAL FOR CONTAINMENT RUPTURE FROM STEAM EXPLOSIONS

Should a large mass of hot molten fuel drop into water at the bottom of the reactor pressure vessel, it can be postulated that the result might be comparable to some industrial accidents and research reactor accidents that have occurred when molten metals have interacted with water. The accidents involve a rapid and large exchange of heat from metal to water, production of steam and violent discharge of water (and sometimes metal) from the zone of interaction. In some of the industrial accidents, damage to surrounding structures has been caused.

The violent appearance of the interaction has come to be known as a "steam explosion." The physical damage from steam explosions has led to the following concerns: Can a steam explosion occur in the reactor vessel; if so, can its magnitude be large enough to threaten the integrity of the vessel and the containment building? With regard to possible mechanisms that cause damage from a steam explosion, two questions are frequently raised. First, can the violence of the interaction generate sufficiently high pressure shock waves to rupture the vessel in the sense, for example, that blasting powder is used to excavate for roads or foundations for buildings? Second, can the energy in the steam generated somehow be contained and directed so as to create in effect a projectile (e.g., a slug of water) that is moving with sufficient momentum that it could rupture the vessel?

The matter of the high pressure shock wave will be addressed first. A steam explosion, even with its violent appearance, does not have the characteristics of a chemical explosion. Peak transient pressures in a steam explosion are limited to a few hundred atmospheres compared to millions of atmospheres in a TNT explosion. Moreover, pressure pulse rise times for a steam explosion are of the order of 100 times longer than rise times for shock waves caused by high explosives. The severe damage caused by chemical explosives derives from both the fast pressure rise times and the high pressure itself (reference 5, 27). Thus, in terms of causing damage from shock waves, for a given amount of thermal energy released, a chemical explosion is much more severe than a steam explosion.

Preliminary studies of vessel failure by this mechanism have been completed (reference 6). The results suggest that a steam explosion would not cause vessel failure.

This leaves the second mechanism for damage from a steam explosion -- a contained, directed expansion of the steam generated. This is the mechanism that was considered in WASH 1400, and may be the only mechanism that requires serious consideration for the generation of a projectile that might break the reactor vessel and reactor containment building (RCB). The scenario is that expanding steam from the steam explosion accelerates a piston-like steam-free slug of water upward until it

impacts the upper vessel head. If the water slug is allocated sufficient energy, or if a simplistic model of vessel response is assumed, or both, a projectile may be developed that can impact the roof of the RCB. In WASH 1400, this projectile is suggested to be the vessel head and the shield cover. Given enough energy, failure of the RCB can be postulated.

This second failure mechanism for a steam explosion in this extended accident is not considered likely because there are a number of physical effects that could cause incoherencies which would weaken the rapid energy release so that no high energy projectiles would be generated. Some of them include: (1) the mode of fuel melting may allow fuel to stream into the water over a period of time and thus be quenched before any massive fuel addition occurs (references 25,29); (2) should a large mass of fuel fall into the water, its fragmentation and mixing may not occur rapidly enough to create a large coherent steam explosion before substantial solidification occurs (reference 2); (3) the high pressure within the vessel would retard the triggering of the interaction (references 11, 12, 13, 17, 22, 23); (4) the water above the fuel-coolant mixture may not behave as a coherent piston-like slug because steam content could cushion any impact with the reactor vessel (reference 2); and (5) the internal structure remaining within the vessel and the vessel itself which remains at high temperature would absorb a portion of the slug kinetic energy (reference 6). Any of the above would inhibit the coupling of released energy to process mechanisms which would compromise the vessel or containment.

IN-VESSEL COOLING

Given the conclusion that a coherent, directed steam explosion is not likely to occur in the reactor vessel but that fuel fragmentation would occur, the next sequence of events to consider is the cooling of molten and solid fuel in a fragmented form within the reactor vessel. As the molten fuel begins to resolidify (within about 10 seconds), energy will be transferred into vaporizing the water in the lower plenum and heating and melting the inner reactor vessel wall. The composition of the molten fuel mixture will be altered as steel from the vessel is melted and mixed with the molten fuel constituents. The fraction of the core in the lower plenum cannot be estimated short of a careful, detailed study, but enough could be present to make an analysis of vessel failure appropriate.

It is not clear that a deep (4.18 feet. thick) bed of debris can be cooled by the overlying water (references 14, 15, 18). Some data does exist that are directly relevant to the TMI case. These data and extensions of correlations, developed as part of the breeder reactor program, indicate that deep particulate beds of heat generating materials have coolability limits that are extremely sensitive to the particle diameters of the fragmented debris. Therefore, if the debris particles are large (one mm or greater), the bed is predicted to be coolable provided sufficient water supply and an ultimate heat sink are available. However, if the debris particles are smaller, correlations predict vapor blanketing of the bed. Thus, there is a possibility that a molten fuel debris pool may be formed. This is assumed to be the case.

It is very likely that some fraction of the fuel debris would be cooled and if make-up water were available, this cooling could extend the residence time in this region.

MOLTEN**FUEL VESSEL**PENETRATION

If cooling of the vessel or fuel were not applied and effective, the vessel could fail about 10-15 minutes after massive (whole core) molten fuel contact, but could take much longer depending on the reaction of the core involved and specific local conditions (references 21, 25). For a fuel mass in the form of a molten pool, surrounded by a frozen crust, the greatest heat transfer rate would be upward and to the sides of the pool (reference 1). Thus, the sides of the vessel would be most weakened. Additionally, the lower head is penetrated by more than 50-3/4 inch diameter instrumentation probes. It is possible that some of these would open up, venting some of the fuel mixture and possibly steam into the cavity below the vessel and decreasing the pressure in the vessel. Failure of the reactor vessel then may occur at either of these two locations.

The internal pressure of the vessel (1,000-2,300 psi) would be reduced quickly when the molten core mass failed the lower plenum and would fall into the reactor cavity below it. The short-term behavior at this point depends on the extent to which water exists in the sump area. For the TMI case, we assume (conservatively) that the water depths at time of vessel melt-through would be 6 feet. Other estimates suggest as little as two feet (reference 24). Evaluation of the blow-down forces on the vessel due to steam ejection showed that the vessel and attendant piping would remain in place (reference 20).

At this point, the possibility of steam explosions must again be assessed. However, the initial conditions of the molten fuel-coolant have changed such that steam explosions are more likely to occur than at high pressure, but the previously mentioned time and space coherency discussions above are still relevant and even more applicable than before. The amount of water may be insufficient to permit the generation of a steam explosion.

Additionally, the confinement constraints are less stringent in this region, as steam can flow through the holes in the reactor vessel skirt and into spaces adjoining the cavity. A side effect of the interaction of fuel and water would be to spread the fuel materials so as to improve the subsequent post-accident decay heat removal situation.

At about this time, pressures would have dropped below the limiting value for the reactor vessel flood tanks and the low pressure high volume water injection system. Both of these would drive water toward the reactor cavity, some of which would reach the neighborhood of the fuel.

Again, as in the discussion above on the fraction of the core in the lower plenum, a smaller fraction might be expected to reach the floor of the reactor cavity. Some fuel might remain behind in the lower part of the vessel, and some fuel bundles on the periphery of the core would remain intact, largely due to radiant heat transfer.

The pressure in the containment building that might obtain at this time was evaluated. The sources of high pressure steam include the remaining water and steam in the primary system and containment water that could take heat from the hot fuel. The pressure was evaluated at less than 50 psi assuming that the entire core was molten, and this heat was transferred to the optimum amount of water. If the fuel were quenched in a large volume of water, the pressure could be as low as 20 psi.

MOLTEN FUEL-CONCRETE INTERACTIONS

If molten core debris falls into the concrete reactor cavity of the reactor, a vigorous interaction involving gas generation, aerosol formation, and concrete erosion by melting and decomposition can begin.

The time required to penetrate a 13-foot-thick reactor basemat was estimated in WASH 1400 to be 18 (+ 10, -5) hours. Since WASH 1400, more sophisticated studies of the time for penetration have been much longer. Calculations made as part of the German Reactor Safety Program (reference 25) have yielded 13.5 days to penetrate a 20-foot-thick basemat. Marchese et al. have predicted that molten core debris would not penetrate a 13-foot basemat if the fuel melting commenced about 3 weeks after reactor shutdown (reference 19). Fontana (reference 9) has calculated only very limited erosion of concrete by molten fuel debris. He also noted that the extensive gas evolution during molten fuel-concrete interactions would cool the fuel and serve to sparge a significant amount of the fission products, reducing the heat source in contact with the concrete. The INTER code, developed at Sandia Laboratories as part of their experimental investigation of molten fuel-concrete interactions, has been used to make estimates of the penetration time. The TMI concrete was calculated to erode in 4 days, with a minimum estimate of 3 days and a maximum of never penetrating (reference 26). The large error bands associated with these estimates are due to uncertainties in the specific processes involved and their relative roles during the interaction of molten fuel and concrete. The INTER model predicts that the molten core materials resolidify prior to penetrating the basemat. This solidification is predicted 1-2 days after the beginning of the interactions, but this phase change does not necessarily halt the erosion of concrete. The mobility of remaining fission products is, however, greatly reduced, and the insulating effect of gases released and remaining between the two materials would be more effective.

Should the more pessimistic of the above estimates be accurate, the core debris would emerge from the concrete basemat and enter the bedrock below the reactor. As with all Class 1 structures at TMI, the reactor containment building is built on red siltstone bedrock (FSAR, TMI-2, Section II).

The bottom of the RCB foundation mat is at an elevation of 268 feet above mean sea level and is excavated 5 to 8 feet into the bedrock. The local water table is defined by the river around TMI, but this water does not penetrate the bedrock. There is evidence of some artesian water below the siltstone in adjacent parts of the same geologic province and there are about 20 wells in the vicinity of Olmsted State Airport

which tap such reservoirs at depths of 300-800 feet. Core drill holes to a depth of 150 feet below the RCB have found no such artesian water.

Emergence of the debris into the bedrock formation should not produce events greatly different than those of the concrete interactions. No significant and vigorous interaction with groundwater would occur. Even if water were present in the bedrock below the RCB, work by Peehs (reference 25) has shown that thermal conduction ahead of the hot debris would dry the ground prior to emergence of the debris from the concrete basemat.

During molten fuel-concrete interactions, hydrates and carbonates are decomposed to yield CO_2 and H 0, some of which are chemically reduced to CO and H by metallic components of the core debris. Reliable estimates of the extent of gas generation during this decomposition have been difficult to make because of uncertainties in the phenomenology of energy and mass transport within concrete, but most models assume a proportionality between concrete erosion rate and gas generation rate (reference 26).

Gas generation was predicted in WASH 1400 to overpressurize the containment building only when containment sprays and similar emergency systems failed. Marchese et al (reference 19) also found for TMI that excessive pressurization due to steam generation would occur only if decay heat removal capabilities were lost. INTER model calculations indicate that molten fuel-concrete interactions would pressurize the RCB by about 15 psi in 2 days if steam were condensed, hydrogen were burned, and the containment atmosphere temperature were controlled. Battelle (reference 21) Punq that CO_2 generation would increase the RCB pressure by 25 psi if 10 ft of limestone concrete were decomposed.

Hydrogen produced by melt/concrete interactions and steam-metal interactions would be expected to burn near its point of creation or when its concentrations within containment exceeded 4 percent by volume. The assumptions for this study allow sprays, coolers, and flowing systems to operate unless events prevent such operation. We have not found reasons to assume failure.

Molten fuel-concrete interactions also induce intense aerosol formation. Aerosol formation during fuel melting and interaction with concrete is an area of limited understanding and active investigation. Experimental programs to identify the nature of the aerosol source term and the behavior of aerosols within the RCB are underway both in the USA and the Federal Republic of Germany.

FISSION PRODUCT INVENTORY WITHIN THE REACTOR CONTAINMENT BUILDING

The postulated fuel melting accident, particularly if it progresses through the stage where the molten mass is sparged by gases arising from the decomposition of concrete, could release all but the most refractory fission products. However, for the conditions existing at TMI, the more impgrtant radionuclides would be in the water. On the order of 10^{-6} to 10^{-} of the halogens might be in the RCB gas phase and available for leakage to the atmosphere (see Appendix E). Factors that enhance retention

of the halogens in the liquid phase are high pH, NaOH in the water, and chemical reducing conditions. The existence of large amounts of silver in the core might trap iodine as silver iodide, but the magnitude of this possible effect cannot be estimated without experimental data.

The persistent form of iodine, methyl iodide, is less likely to be found initially in the TMI primary system because of the high temperatures, water vapor, radiation, and absence of organic compounds.

Largely due to the operation of the RCB spray sygtem, the airborne particulate fission products could range from 4 x 10 (for the tellurium group) to 5 x 10 (for the lanthanum group) of the core inventory at one day after start of fuel melting.

Because only slow leakage of the RCB would be expected and because of factors discussed in Appendix E, the radiological consequences of the extended TMI fuel-melting accident are believed to be substantially less than for a design-basis accident. The major and very important radiological lessons learned from TMI are those relating to iodine and to cesium. Both of these isotopes were, to our knowledge at this time, largely caught and retained in the water in the primary system, the containment sump, and the auxiliary building. Because of this retention, these isotopes were not available for escape to the environment. As discussed in this report, the same conditions should exist for the extended fuel-melting accident considered here.

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APPENDIX B

FUEL DAMAGE ESTIMATES WITH THE TRANSIENT REACTOR ANALYSIS CODE (TRAC)

BY

TECHNICAL ASSESSMENT TASK FORCE

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APPENDIX B

FUEL DAMAGE ESTIMATES WITH THE TRANSIENT REACTOR ANALYSIS CODE (TRAC)

A best-estimate analysis of the first 3 hours of the TMI-2 accident has been performed with the TRAC code and is reported in detail in reference 1. The analysis out to that point indicated that while the zircaloy cladding sustained considerable damage (rupture, oxidation, and embrittlement), no melting of fuel occurred (excluding eutectic formation). Beyond that point in the accident sequence, there exists considerable uncertainty in the actual thermal hydraulic conditions (make-up flow versus let-down flow). In addition, certain modeling deficiencies in the TRAC code are severe enough (lack of treatment of noncondensible gases in the vapor field equations) to raise questions about the quantitative validity of the TRAC calculation beyond this point.

Therefore, to make an estimate of possible core damage during the time from 180 minutes to 210 minutes in the accident, we have used the results of the continuation of the base case TRAC calculation out to 210 minutes, supplemented by engineering judgment on the possible effects of TRAC modeling deficiencies and system mass balance uncertainties. In addition to the base case accident sequence, we have also analyzed two alternative scenarios to the nominal accident sequence involving operation of the pilot operated relief valve (PORV) and high pressure injection (HPI) system.

The TRAC base case is discussed in detail in references 1 and 2. Briefly, the important features of the assumed accident sequence are:

1. The closure of the PORV block valve at 138 minutes (8,280 sec).

2. The assumption of zero net make-up flow from the time the PORV is closed and later.

3. The opening of the PORV block valve from 192 minutes to 197 minutes.

4. The initiation of full HPI at 200 minutes.

The principal modeling deficiencies in TRAC that could alter the calculated sequence of events are:

1. The effect of noncondensible hydrogen on the system pressure via its reduction in condensation rates of steam and its accumulation in high elevation regions throughout the primary system.

2. The effect of hydrogen on heat transfer phenomena and possible starvation of the zirconium-steam oxidation reaction.

3. The effect of disrupted core geometry (clad ballooning, rupture, melting, and fuel-clad motion) on the coolability of the core.

4. The effect of phase separation on blocking flow paths. TRAC is an Eulerian-model, finite-difference, fluid-mechanics code.

5. Local temperature increase at the steam-liquid interfaces in the primary side loops and its self-limiting effect on vapor condensation would be underestimated because of the node size.

The principal uncertainty in the hydraulic boundary condition is the magnitude of the HPI, make-up, and let-down flows. Recent estimates put the net make-up flow during the period after the PORV block valve is closed to be as high as 50 gpm (reference 3). This may have a significant effect on the core thermal-mechanical response.

To give a best estimate of core damage, we have provided a damage estimate for:

1. The base case TRAC calculation out to 210 minutes.

2. The base case modified by estimates of the effects of TRAC modeling deficiencies.

3. The preceding (2) plus an estimate of the effect of increased make-up flow.

Case 3, assuming the data on make-up flow to be correct, represents our best engineering judgment, pending code modification as needed prior to running this case with TRAC.

In addition to the base case scenario previously described, we have also extended our best-estimate case (3) to include two alternative sequences. These variations are:

a. The PORV block valve is not opened from 192 minutes to 197 minutes.

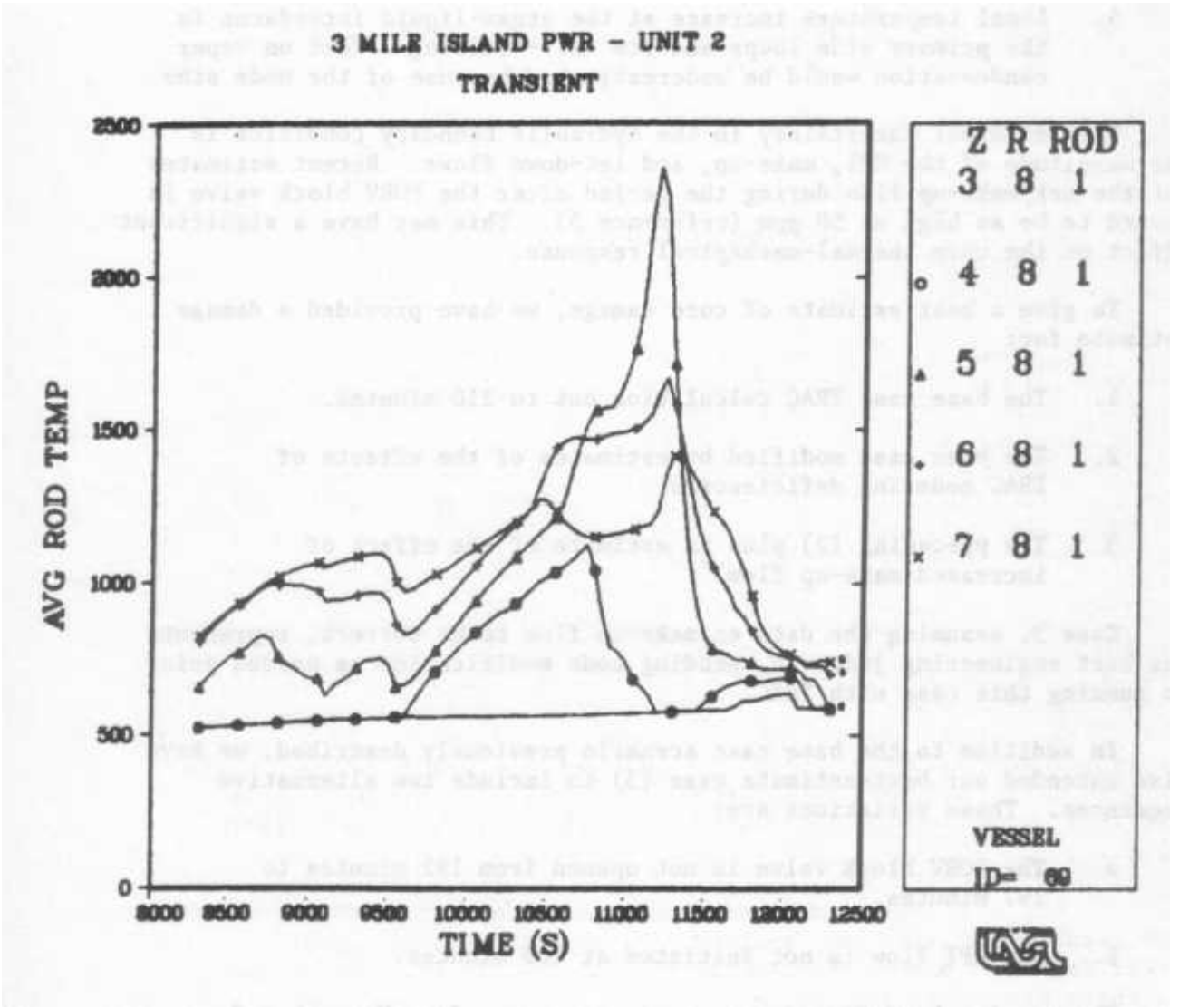
b. The HPI flow is not initiated at 200 minutes.

Thus, a total of five damage estimates are made. Note that for reasons stated above these estimates are somewhat speculative.

1. ANALYSIS OF TRAC BASE CASE

The extension in time (reference 2) of the base case results as reported in reference 1 indicates that the calculated cladding temperature transient is terminated prior to the opening of the PORV block valve at 192 minutes. Figure 1 illustrates the axial cladding temperatures (in five zones) for the average rod as a function of time. The temperature excursion is calculated to be terminated and temperatures decrease rapidly at about 186 minutes (11,160 sec). The temperatures are reduced due to higher steam velocities in the core region, caused by increased flow through the primary loops. This flow path was formed by voiding in the pump loop seal piping and the vessel lower plenum, which allowed steam to circulate through the loops (see above discussion on TRAC modeling). This flow path may not be real; intuitively one would expect

FIGURE 1: Axial Fuel Rod Temperature Profiles
(Levels 3-7 are in the Core Region)



the liquid phase in the pump loop seals to close off any flow path of steam as long as the liquid level is higher than the pipe diameter.

After the peak temperature of about 2,400°K is reached at 186 minutes and temperatures are decreasing, the cooldown rate is further enhanced by the opening of the PORV block valve at 192 minutes (1,1520 sec).

This creates an additional heat sink. Closure of the PORV block valve at 197 minutes (11,820 sec) has little effect on the core temperatures because the natural circulation path for steam is still available. Initiation of the HPI at 200 minutes (12,000 sec) rapidly reduces the core temperatures to saturation range values (about 600°K).

Core damage in this case is primarily limited to cladding damage. Due to high peak cladding temperatures, about 0.5 meter length of cladding just above the core mid plane (3 or 5 of Figure 1) experiences melting. Above that region, another 0.5 m length of cladding is severely embrittled due to oxidation. This region is suspected to have undergone brittle fracture fragmentation when the HPI flow was restored at 200 minutes. No fuel melting is calculated to occur. (UO_2-Zr-ZrO_2 mixture formation has not been considered in this analysis.)

The temperature rise prior to 186 minutes is almost adiabatic and is due to fission product decay heat and the zircaloy-steam exothermic reaction. From 160 minutes until 175 minutes, the adiabatic heatup is from decay heat and is approximately 1°K/sec. This is consistent with the TMI-2 adiabatic heatup rates calculated in Annex I. From 175 minutes to 186 minutes, the heatup rate is much higher and approaches 5°K/sec. This is due to the addition of heat produced by the zircaloy-steam reaction, which becomes the dominant energy source at these higher cladding temperatures.

It has been suggested that use of the Baker-Just correlation for zirconium-steam oxidation overpredicts the actual rates possible because steam depletion may be a factor in situations like TMI-2 (reference 3). To check whether this might be the case for the TRAC base case calculation, we have calculated maximum oxidation rates as a function of steam flow through the core. The possible heat generation from this reaction (in Kw/m) is given in Annex-II. Using the TRAC calculated core steam velocities of about 0.1 m/sec (about 0.15 g/sec for each fuel rod) and assuming oxidation to occur over 1 m length (consistent with TRAC results), a linear power rate of 4 Kw/m is calculated. This is equivalent to an adiabatic heatup rate of about 11°K/sec. The TRAC calculated temperature rise is 5°K/sec, indicating steam depletion is unlikely. Also the TRAC calculated heat generation rate is also much less than 4 Kw/m. Hence, it can be concluded that in TMI-2, the oxidation rate was limited due to chemical reaction rate kinetics, not steam starvation.

2. QUALITATIVE ANALYSIS OF THE BASE CASE WITH AN ESTIMATE OF THE EFFECTS OF TRAC MODELING DEFICIENCIES

A. Effect of Cladding Melting

The zircaloy cladding will melt at about 2,100°K in the hottest core region and probably will flow down due to gravity to a cooler location in the core. When this occurs, the heat source from the cladding is lost, leaving only the decay heat in the fuel to cause further fuel heatup. Thus, the temperature rise rate will not be as rapid as in the base case, and the peak cladding temperature will probably be lower. This effect is graphically illustrated in Figure 2 (the sharp temperature peak is truncated).

For this case, the peak clad temperature would be about 2,300°K and the temperature would begin to decrease shortly thereafter as in the base case. No fuel melting would have occurred.

B. Effect of Cladding Melting Coupled with Noncondensable Hydrogen

The hydrogen that is produced from the metal-water reaction above 1,273°K is not accounted for in the TRAC hydrodynamics, nor is the effect of cladding melting. In reality, the hydrogen that is produced will have at least three major effects:

1. steam flow blockage in candy-cane region of the hot legs;

2. reduced steam condensation rates; and

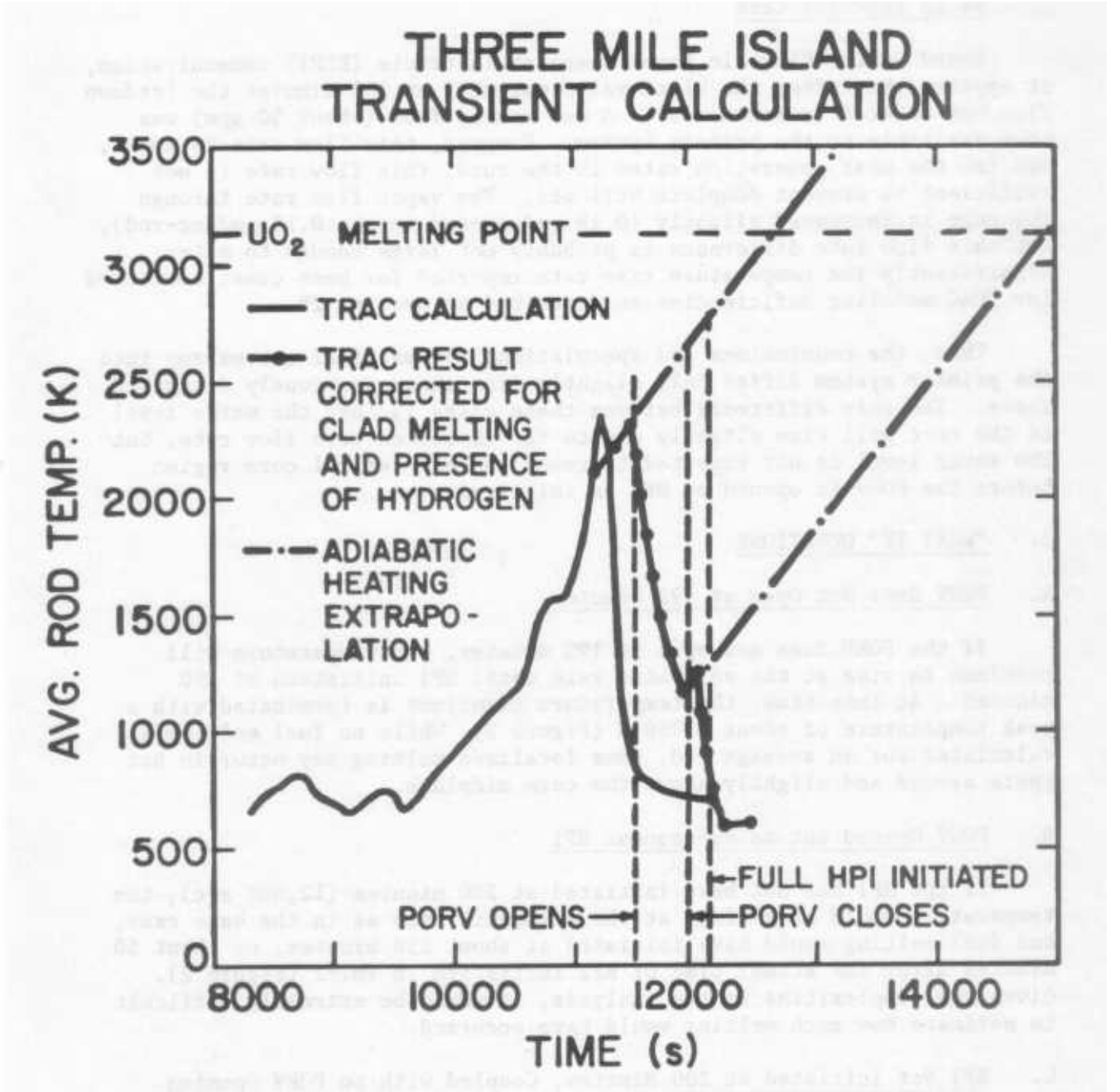
3. higher system pressure (thus altering system thermodynamics and heat transfer characteristics).

As discussed previously, if cladding melting effects are considered, the peak temperature will be lower than in the base case by about 200°K. The results of coupling the effects of both hydrogen and cladding melting are shown graphically in Figure 2.

The effect of cladding melting causes the slope of the temperature rise to change as discussed previously (adiabatic heatup of the fuel only). The main effect of the hydrogen that is produced is to block the steam flow path through the system in the loop seals. Thus, the temperature will not begin to decrease as in the base case. The temperature continues to rise until the PORV opens, at which time the temperature begins to decrease, and continues to decrease to about 1,200°K, when the PORV is shut. It should be pointed out that we are assuming that no significant flow blockages occur in the core due to cladding melting and subsequent solidification in the cooler regions below.

When the PORV is closed, the temperature again rises at about the adiabatic rate, but decreases when the HPI is initated. The temperature decreases to the coolant saturation temperature when the core refloods. The peak clad temperature for this case would be about 2,300°K and no fuel melting would have occurred.

FIGURE 2: Temperature Estimates at Core Midplane



C. **Effect of Having Net Makeup into the System Rather than as in the Base Case**

Based on the Electric Power Research Institute (EPRI) communication, it appears that after the block valve was shut at 138 minutes the letdown flow was reduced (reference 3). A net makeup flow (about 50 gpm) was then available to the primary system. However, this flow rate is small, and for the heat generation rates in the core, this flow rate is not sufficient to prevent complete boil off. The vapor flow rate through the core is increased slightly (0.18 gm/sec-rod versus 0.15 gm/sec-rod), but this flow rate difference is probably not large enough to alter significantly the temperature rise rate reported for base case, corrected for TRAC modeling deficiencies as described in section 2B.

Thus, the conclusions and speculations for marginal net makeup into the primary system differ only slightly from those previously discussed above. The only difference between these cases is that the water level in the core will rise slightly due to the increased core flow rate, but the water level is not expected to reach the hot central core region before the PORV is opened or HPI is initiated.

3. **"WHAT IF" QUESTIONS**

A. **PORV Does Not Open at 192 Minutes**

If the PORV does not open at 192 minutes, the temperature will continue to rise at the adiabatic rate until HPI initiation at 200 minutes. At this time, the temperature transient is terminated with a peak temperature of about 2,750°K (Figure 2). While no fuel melting is calculated for an average rod, some localized melting may occur in hot spots around and slightly about the core midplane.

B. **PORV Opened but no Subsequent HPI**

If the HPI had not been initiated at 200 minutes (12,000 sec), the temperature would have risen at the adiabatic rate as in the base case, and fuel melting would have initiated at about 250 minutes, or about 50 minutes after the actual time of HPI initiation in TMI-2 (Figure 2). Given the complexities in the analysis, it would be extremely difficult to estimate how much melting would have occurred.

C. **HPI Not Initiated at 200 Minutes, Coupled with no PORV Opening**

If the HPI were not initiated at 200 minutes, the temperature will continue to rise at the adiabatic rate. The initation of fuel melting would occur at about 205 minutes, or roughly 5 minutes after the time that HPI was initiated at TMI-2 (Figure 2).

4. **CONCLUSIONS**

In conclusion, the best-estimate base case (refer to section 2B), which includes the effects of cladding melting and noncondensible gas, shows that the peak cladding temperature would have been about 2,300°K, with no significant fuel melting. (Note that fuel melting might occur

in localized regions where the cladding melted, moved, and solidified to create local flow blockages.) The rods cool down prior to the initiation of HPI due to enhanced vapor flows through the core when the PORV is opened at 192 minutes. The rods are then quenched after HPI initiation at 200 minutes, and temperatures cool down to saturation values. These conclusions would not differ significantly under the condition of a net make-up flow (50 gpm as discussed in section 2C).

If the PORV had not opened, the temperature would rise at close to the adiabatic rate until HPI initiation. The peak cladding temperature for this case would have been about 2,750°K, with no substantial fuel melting. If the HPI had not been initiated at 200 minutes, fuel melting would have begun at about 50 minutes after the actual time of HPI initiation in TMI-2.

If the HPI had not been initiated at 200 minutes (as well as no PORV opening), the temperature would rise at close to the adiabatic rate, and fuel melting would have begun at about 5 minutes after the actual time of HPI initiation in TMI-2. Thus, if HPI flow had not been reestablished at 200 minutes, significant fuel melting, 10-20 percent of the total core at the central midplane region and up to a meter above, would have occurred within the next hour. Assessment of the consequences of large-scale fuel melting are given in the other appendices of this document.

APPENDIX B REFERENCES

1. "Preliminary Calculations Related to the Accident at Three Mile Island," Los Alamos Scientific Laboratory Report LA-UR-79-2425, August 1979.

2. Letter to W.R. Stratton, from Walter L. Kirchner, Sept. 10, 1979.

3. EPRI communication to TMI Commission, September 1979.

ANNEX I -- ADIABATIC CORE HEATUP RATE

The adiabatic heatup rate of a fuel rod may be calculated as follows:

$$\nabla k \nabla T + q''' = \rho c_p \frac{\partial T}{\partial t} \qquad (1)$$

where

$\nabla k \nabla T = 0$, for adiabatic conditions
q''' internal volumetric heat generation rate (W/m^3)
T temperature (K)
t time (s)
ρ density (kg/m^3)
c specific heat (J/kg K)

The above formula yields, assuming volume weighted cladding and fuel properties and independent of length:

$$\frac{\partial T}{\partial t} = \frac{q'}{(\rho c_p)_f A_f + (\rho c_p)_c A_c} \times 10^3 \qquad (2)$$

where $A_f = \pi R_f^2 = 6.8134 \times 10^{-5} m^2$

$A_c = \pi(R_o^2 - R_i^2) = 2.1725 \times 10^{-5} m^2$

q' = linear power generation rate, KW/m

and $\rho_f = 1.0422 \times 10^4$ kg/m³
$\rho_c = 6.551 \times 10^3$ kg/m³

and tables of specific heats,

c_{pf} (J/kg K)	T(K)
314	1,000
330	1,500
388	2,000

c_{pc} (J/kg K)	T(K)
281	300
302	400
331	640
375	1,090
502	1,093
360	1,248

For two average fuel rod temperatures of interest, this simplifies to:

$$\frac{\partial T}{\partial t} = 3.682q' \text{ at } 1{,}000°K$$

or

$$3.060q' \text{ at } 2{,}000°K$$

Below is a table of adiabatic heatup rates (K/s) based on representative linear heat generation rates and decay heat power fractions for TMI-2.

TABLE 1: Adiabatic Heatup Rate (K/s)

		Linear Heat Rate (Kw/m)*					
		T_{avg} = 1,000°K			T_{avg} = 2,000°K		
Time(s)	Power Fraction	12	24	36	12	14	36
0	1.0	43.5	87.1	130.6	36.7	73.4	110.2
2	0.1475	6.42	12.8	19.3	5.4	10.8	16.25
120	0.0369	1.61	3.21	4.82	1.35	2.71	4.06
200	0.0252	1.10	2.19	3.29	0.93	1.85	2.78
500	0.0224	0.98	1.95	2.93	0.82	1.65	2.47
1,000	0.0184	0.80	1.60	2.40	0.68	1.35	2.03
5,000	0.0117	0.51	1.02	1.53	0.43	0.86	1.29
10,000	0.0096	0.42	0.84	1.25	0.35	0.71	1.06
20,000	0.0079	0.34	0.69	1.03	0.29	0.58	0.87

*Excluding the zircaloy-steam reaction heat input.

ANNEX II -- LIMITING OXIDATION HEAT RATES

This appendix outlines a bounding estimate on the rate at which the zirconium-steam oxidation reaction can produce heat. The estimate assumes that the reaction is limited only by the steam available in the fuel assembly (this discounts any kinetic effects that limit the rate at which the zirconiumoxide front propagates into the cladding and its temperature dependence). The maximum linear heat generation rate due to oxidation becomes:

$$q' = q' = \dot{m}Q/L \cdot m_{Zr}/m_{H2O} \quad (1)$$

where

q' = linear heat generation rate, KW/m
Q = exothermic reaction energy, J/kg
L = length, m
m_{Zr} = 91.22 kg Zr reacted
m_{H_2O} = 32.0 kg H_2O (steam) reacted
$\dot{m}$ = steam flow rate per rod (Kg/s)

This yields:

$$q' = 1.857 \times 10^{4} \dot{m}/L \text{ (kw/m)} \quad (2)$$

Given the system ressure, steam temperature, and flow area per fuel rod (A^{a} 1.1 x 10^{-4} m), the linear heat rate can be further reduced to function of steam velocity:

$q' = 21.1\ V_g/L$ (6.8 MPa, 1,500°K)
$q' = 27.1\ V_g/L$ (10.2 MPa, 1,500° K)
$q' = 36.2\ V_g/L$ (13.6 MPa, 1,500° K)

For typical TRAC conditions predicted for TMI-2, steam velocities are quite low (V_g < 0.1 m/s). However, except at the hot spot, the calculation would indicate very little steam starvation of the oxidation rate. If the heat calculated by TRAC were to exceed the value from equation (2), it should be limited to that value. Normally, the values calculated by TRAC will be limited by kinetic effects, not the availability of steam, resulting in heat rates smaller than equation (2) would indicate. Note also, that steam velocities greater than about 0.1 m/s in equation (2) would predict oxidation rates far in excess of physically realistic values.

APPENDIX C

POTENTIAL FOR DAMAGE TO REACTOR VESSEL OR CONTAINMENT DUE TO STEAM EXPLOSIONS ASSOCIATED WITH FUEL-MELTING ACCIDENTS

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APPENDIX C
POTENTIAL FOR DAMAGE TO THE REACTOR VESSEL OR CONTAINMENT DUE TO STEAM EXPLOSIONS ASSOCIATED WITH FUEL-MELTING ACCIDENTS

If extensive fuel melting occurs, core materials will eventually slump and fall into the pool of water below the core. This raises the question of whether a steam explosion can occur, and if so, can it result in the rupture of the reactor vessel and the reactor containment building. A steam explosion refers to the creation and rapid expansion of steam produced when molten material comes in contact with and rapidly transfers heat to water.

Vessel or containment rupture due to a steam explosion is not judged likely in TMI, even if a significant amount of the fuel had melted. This judgment is based on steam explosion experiments, together with current understanding of the phenomena that would be required to produce vessel or containment rupture.

STEAM EXPLOSION IN THE REACTOR VESSEL

Two mechanisms have been considered as possible causes for vessel damage resulting from a steam explosion. The first mechanism is the shock wave associated with the steam explosion. The second is the slower expansion of a two-phase water-steam mixture that might drive a piston-like liquid slug against the upper vessel head. As was explained in Appendix A and further amplified below, vessel rupture from a steam explosion seems possible only from the second mechanism. Shock waves from a steam explosion are not of sufficient magnitude to cause a failure of the vessel.

Two aspects of shock waves were studied to determine the potential for vessel failure (reference 5). The first is by propagation of a shock wave through the vessel wall up to the head of the vessel. If the rise time of a shock wave is short enough, reflection at a free surface could cause the vessel surface material to spall or break off and a projectile to be formed. The results of the analyses demonstrated that a steam explosion is a much slower phenomenon (with rise times in the 10 millisecond range) than a chemical explosion (for which rise times are of the order of one microsecond or less), and vessel failure by this mechanisim is precluded. As reinforcement to this conclusion, it is noted that shock pressures from steam explosions (limited to hundreds of atmospheres) are many orders of magnitude lower than the shock pressures from chemical explosions (which are millions of atmospheres).

The second aspect of shock waves is the possibility of local failure of the instrumentation penetrations in the bottom of the vessel. This method of failure cannot be ruled out, although projectiles generated in this way would be directed downward into the reactor cavity and consequently would pose no threat to containment integrity. The main consequences of this failure would be depressurization of the vessel and discharge of molten material to the reactor cavity and would not threaten containment.

To pursue the point even further, the strength of the TMI vessel against gross failure by shock waves can be bounded by considering the more severe case of a chemical explosion. Experiments have been performed with large masses of high explosive (hundreds of pounds) in spherical steel vessels as large as several feet in diameter (reference 13). Computer programs have been developed to analyze these experiments in order to extrapolate to different situations (reference 13). These programs have been used to analyze the response of a sphere of radius and wall thickness equal to those of the bottom of the TMI reactor vessel. This region has the thinnest walls, and restraining the imagined explosion to this sphere is regarded as conservative in evaluating the strength of the vessel. The calculated mass of high explosive that this spherical vessel could contain is 400 pounds of TNT -- probably a factor of 10 or more larger than any steam explosion that might occur within the TMI reactor vessel. Clearly, this analysis indicates the implausibility of a rupture of the TMI vessel from shock waves from a steam explosion.

The second mechanisim for damage from a steam explosion is the potential acceleration of a liquid slug of water and its subsequent impact against the upper vessel head. In order for this mechanism to generate a projectile that might rupture both the reactor vessel and the containment, each of the following conditions must be met:

1. A sizeable fraction of the core, e.g., greater than 10 percent, must provide its energy to the steam explosion.

2. This large mass must mix coherently both timewise and spacewise with the pool of water below the core before the explosion is triggered.

3. The vaporized steam must accelerate a slug of liquid water of diameter equal to the inside diameter of the vessel (i.e., 14 feet) upward to the vessel head.

4. The water slug must impart enough energy to the reactor vessel head to cause it to rupture and generate a projectile.

5. There must be enough energy transferred to the projectile to propel it upward 95 feet to the containment roof and then rupture the 3.5-foot-thick containment upon impact.

Each of these steps is considered in turn below.

1. For a steam explosion to rupture the containment in TMI, a molten mass of fuel greater than 10 percent of the core would have to have participated in the explosion. Ten percent of the TMI fuel is 10,000 kg. The largest masses for which steam explosions have been observed in controlled experiments are of the order of 10 kg: Fe-Al_2O_3 in water (reference 3) and Fe-UO_2 in water (reference 2). In these experiments, the conversion of thermal energy to mechanical energy (i.e., energy that could lead to projectile generation) varies from near zero to a value of about one percent. If 10 percent of the TMI core drove a coherent steam explosion at one percent energy conversion, the resulting mechanical energy could be of the order of 200 megajoules (MJ)

(about 100 lbs of low-grade TNT), a value that would, in our judgment, be insufficient to cause vessel rupture. However, the difference between the 10,000 kg of fuel (10 percent of the core) and the 10 kg used in the largest experiments is a factor of 1,000; hence, large uncertainties exist in extrapolation to the reactor case, but it is important to recognize that requiring larger amounts of material to mix thoroughly prior to an explosion imposes progressively lower efficiencies in energy availability.

In the case of TMI, the pressure in the vessel at the time that fuel might have fallen into the core would have been high, i.e., of the order of 1,000 psi. Experiments have shown that high ambient pressure tends to make the occurrence of a steam explosion more unlikely even if coherent mixing takes place (references 6, 7, 8, 9, 11). It is significant that the only observed cases of steam explosions in nuclear reactors (SL-I and SPERT) occurred near atmospheric pressure.

2. During fuel melting, which occurs over a time duration of minutes to hours, it is possible that some of the fuel will stream down into the pool (reference 1). Small amounts of fuel falling into the pool over an extended time will lead to boiling and may lead to small steam explosions, too small to cause mechanical damage. The resulting steam generation should provide cooling and slow the fuel-melting process. However, experimental evidence cannot rule out the possibility that, later in the melting process, a larger pool of molten material could form above the grid plate and suddenly drop into the water below (reference 12), and the consequences of this step must be examined.

In a steam explosion, all of the molten material that participates in the interaction must break up into relatively small pieces and mix with the cold liquid during a time of the order of one second or less (this rapid disintegration and mixing is called "coherent" mixing). Present estimates on the extent of mixing that can occur vary over a wide range of molten fuel fragment sizes and fuel-water mass ratios. During the disintegration and mixing, explosions may occur randomly in time at the periphery of the large mass that would disperse the water and further prevent large-scale coherent mixing. Thus, coherence in both time and space is difficult to conceive. It should be recalled that the volume of 10,000 kg of fuel (i.e., 10 percent of the core) is greater than one cubic meter -- the volume of a moderate-sized refrigerator. Coherent mixing of masses of this magnitude has not been observed experimentally, and no defensible mechanistic models have been developed to show how this can occur. On the other hand, it is easy to understand the mixing of metallic fuel and water prior to steam explosions in the SL-1 and SPERT research reactors. The geometry for mixing in both cases was much more favorable than for fuel dropping into a pool, because the thin fuel element plates were separated by narrow cooling channels of water prior to the beginning of the reactor transients.

3. Molten fuel and hot metal falling into water would boil in the film boiling mode, i.e., a layer of steam would surround the molten mass, separating it from the liquid pool. Steam leaving this film would rise through the pool above the molten mass. Were a steam explosion to occur, the resulting steam pressure would accelerate material upward

above the explosion zone. The material accelerated could do significant damage only if it were in the form of a uniform slug, containing little void (or steam) space, and having a diameter covering the entire 14-foot diameter vessel. Since steam would be streaming up through the pool and with the added possibility of localized steam explosions occurring, it is difficult to imagine how a uniform liquid slug without steam voids could be accelerated by the expanding steam.

4. Where an energetic uniform water slug is generated, it must traverse the volume occupied by the large amount of structure between the lower plenum and the upper head. The slug must then crush a large amount of material in the upper portion of the vessel and must compress considerable amounts of gas and vapor in this region. These effects would reduce the impact loading of any slug on the upper head even further.

After the slug contacts the head, additional energy would be dissipated in failing the vessel and causing some portion of the head to be thrown as a projectile. Preliminary analysis of vessel failure (reference 5) suggests that the energy required to fail the vessel head is of the order of hundreds of megajoules of energy. In addition, the failure would most likely be asymmetric with the vessel failing in an isolated location. This would allow coolant to be ejected from the vessel, with projectile generation being an unlikely result. The other source of a projectile would be a control rod drive assembly ejected as a result of the water slug impact. The possiblity of this event cannot be precluded, although an analysis of the buckling loads on the control rod drive assemblies would indicate that it is not likely (reference 5), since the impact of the coolant slug would first cause buckling of the control rod assembly as it is being torn loose from the vessel.

5. Following the events described above, the **excess** energy would go into kinetic energy of the projectile. To penetrate the containment, it must first overcome gravity and reach the containment roof structure. Experimental correlations exist for projectile penetration of concrete structures, although agreement between correlations is only fair (e.g., agreement within a factor of three). Using the correlations that predict the lowest energy required for penetration, an energy equivalent to about 10 percent of the core participating in a steam explosion at one percent thermal-to-mechanical energy conversion would be required just to propel a projectile against gravity and penetrate the containment. When the dissipative mechanisms described above are added to this, the fraction of the core required to supply sufficient energy rises dramatically.

STEAM EXPLOSION IN THE REACTOR CAVITY

If the fuel melts through the reactor vessel, it could fall into a pool of water on the concrete basemat in the reactor cavity. At this level, the reactor cavity is a cylindrical room 12.5 feet in diameter. Two doors connect passageways to the reactor cavity, which could provide vent paths if a steam explosion occurred.

A water slug accelerated by a steam explosion in this region would enter the previously melted lower head of the reactor vessel and then follow the same path that was described above for a steam explosion within the vessel and also vent into regions around the vessel. While the atmospheric pressure in the reactor cavity provides a more favorable environment for a steam explosion than the higher pressure in the vessel, the difficulty of achieving coherent mixing of a large fraction of the core with water is similar to the earlier described situation within the vessel. Moveover, the vent paths due to the passageways to the reactor cavity, plus the twelve 9.25-inch diameter holes around the vessel support skirt, would decrease the driving pressure. A steam explosion in the reactor cavity is not expected to have the potential for generating a projectile that would threaten the containment.

SUMMARY

Recognizing (1) the long timescale required for fuel melting relative to required mixing times for coherent steam explosions; (2) the inherent phenomena mitigating against coherency for the large molten masses required for projectile generation; and (3) with all of the dissipative mechanisms between the pool in the vessel or the reactor cavity and the containment building, it is difficult to conceive of a scenario in which enough molten material could mix coherently with a pool to generate a steam explosion that would rupture the reactor vessel or the containment building.

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APPENDIX D

PENETRATION OF THE CONCRETE BASEMAT

BY

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APPENDIX D

PENETRATION OF THE CONCRETE BASEMAT

If debris consisting of molten fuel and structure falls from the reactor pressure vessel onto the concrete floor or basemat of the containment building, an intense interaction would be initiated. The high-temperature molten fuel would begin to erode the concrete by initiating its melting along with thermal decomposition of hydrates and carbonates in the concrete, leading to the generation of large quantities of gases consisting of water vapor, CO, and CO . These gases can move through the molten mixture producing aerosols, cooling the mixture, carrying heat to the containment building, and sparging nonrefractory fission products from the molten mixture. The gases can chemically interact with metallic components (steel from the reactor pressure vessel and the reactor core) to form hydrogen.

The concerns with respect to safety during the molten fuel/concrete interaction phase of a fuel-melting accident are:

1 Will containment be failed by erosion of the concrete?

2 Will containment be failed by over-pressurization of the reactor containment building?

3 Will extensive release of fission products sparged from the molten fuel as aerosols be possible and a serious problem?

4 Will hydrogen buildup lead to explosions that could threaten the containment building?

Molten fuel/concrete interactions and their implications to the safety of nuclear reactors are areas of current active research. A significant number of uncertainties exist, but bounding estimates of the magnitudes of the interactions can be made.

Erosion of concrete by molten reactor core materials has been shown to proceed by a melting process (reference 1). Spallation or thermal shock as mechanisms for erosion of concrete are initially spectacular, but ultimately contribute little to the overall erosion rate. Several models have been used to estimate the extent of concrete erosion during a reactor accident (references 2,3). One of the first estimates of the time for penetration of the basemat of a reactor was made in WASH 1400 (reference 4). This estimate, made without the benefit of experimental information, was 18 (-5, + 10) hours and was based on a series of assumptions, not the least of which was that spallation would be significant. Later work based on experimental studies of molten fuel/concrete interactions has produced much longer estimates of the time for penetration of the foundation mat. Peehs et al. (reference 3) predicted that a 20-foot-thick mat of basaltic concrete would erode after 13.5 days. Recent

calculations with the INTER* model of molten fuel-concrete interactions (reference 2) have indicated that the TMI reactor foundation might be eroded in 4 days, with a minimum time of 3 days, and possibly not at all. The INTER calculations also indicated that the core debris would solidify in 1 or 2 days after the onset of molten fuel/concrete interactions. The large uncertainty band associated with the estimated time of foundation penetration from the INTER code is due to uncertainties in the manner and rate at which solid, hot fuel-steel-concrete debris erodes concrete.

The INTER calculations cited above were made for the postulated bounding case of all the core melting and dissolving the lower head of the reactor pressure vessel, with the interaction beginning 5 hours after shutdown of the reactor. The decay power used for the calculations is taken from reference 10 and was modified by assuming that radioactive noble gases and halogens were not present. Reference 1 to this report discusses in detail the matter of fission product escape from molten fuel and Fontana (reference 5) has observed that gases liberated from the concrete might sparge more of the fission product decay heat sources from the mixture. Loss of heat sources from the molten fuel would further slow the rate at which the mixture could erode the concrete.

If penetration of the reactor foundation does in fact occur, the debris -- most probably in the solid state -- would attack the bedrock below the reactor. As with all Class 1 structures at the TMI site, the reactor containment building is built on red siltstone bedrock (reference 6). The bottom of the reactor foundation is 5 to 8 feet in the bedrock at a level of 268 feet above mean sea level (MSL). The local water table level is at 280 feet above MSL, but the bedrock is impervious to water. Underground artesian water is found below the siltstone formulation in the same geologic province as is TMI at depths of 300-800 feet. But core holes up to 150 feet deep have found no such ground water below the reactor containment building.

No ground water is then available to interact with the core debris should it penetrate the foundation. Further, experimental investigations by Peehs and Hassman (reference 7) have shown that thermal conduction ahead of the debris mixture would dry the ground prior to emergence of the debris from the concrete.

Gases produced during the molten mixture/concrete interaction stem from thermal decomposition reactions of hydrates and carbonates in the

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INTER is a calculational program designed to investigate the parameters of importance in the interactions between molten fuel and concrete and how these parameters affect the penetration into concrete. More advanced programs are under development. These are known as WEXEL (in West Germany) and CORCON (in the United States, Sandia Corp.). Results from these latter programs are not yet available for the TMI-2 design.

concrete. The decomposition reactions occur in three temperature regimes (reference 8):

1. loss of "free" water	30-230°C
2. loss of chemically bound water Ca(OH) , Al (OH) (CA (OHJ$_2$, Ai (OHJ$_6$, etc.)	380-550°C
3. loss of carbon dioxide from limestone	685-1,000°C

On a weight basis the concrete at TMI-2 contains about 2-3 percent free water, 2-3 percent chemically bound water, and 22 percent carbon dioxide.

Models to predict the extent of gas generation during melt/concrete interactions have been difficult to formulate because of uncertainties concerning matters such as the movement in concrete of water vapor away from the heat source; the rate then becomes important (reference 9). Most models, such as INTER, assume a proportionality between the extent of concrete erosion and the extent of gas generation. As such, the estimates from these codes do not include gas generated within the concrete due to thermal conduction into the concrete.

INTER calculations for the TMI-2 reactor indicated gas generation during a hypothetical meltdown accident would consist of CO, CO , H , and steam. Conversion of these amounts of gas into a pressure within the containment building depends on the assumptions taken: For example, if steam is condensed, hydrogen is burned and the mean temperature within the containment is 298°K, then pressure within containment would be increased by 12.5 psi after one day of fuel/concrete interaction. If steam does not condense but the mean atmosphere temperature is still only 298°K, the pressure increase would be about 17.8 psi.

The concentration of hydrogen in the gas liberated during fuel/concrete interaction has been estimated from INTER calculations. Throughout the first 2 days of interaction the gas mixture is ignitable as it escapes from the mixture. Although the gas may be cooled sufficiently as it escapes into the containment building so that spontaneous ignition may not occur, a number of ignition sources are present in the containment building. It is possible then that hydrogen will burn as it is generated and will not accumulate to pose a detonation hazard.

Gases released during the debris/concrete interactions can sparge aerosols from the mixture. Aerosol concentrations as high as 100 grams/cubic meter have been observed in some debris/concrete experiments (reference 8). However, most of the aerosols come from nonradioactive sources, such as the concrete and the steel. The behavior of these aerosols within the containment building is not fully understood. Active experimental investigations of aerosol behavior are underway in both the USA and the Federal Republic of Germany. It is currently believed that the high concentrations of aerosols produced by the debris/concrete interactions would lead to rapid agglomeration and sedimentation of the particulate material, and sprays in the containment building would further sweep aerosols from the atmosphere and prevent accumulation of significant amounts of airborne, radioactive particulates.

In summary, published analyses of core/concrete interactions suggest that core debris would be resolidified prior to penetration of the containment concrete basemat. The motion of the debris is then drastically slowed and the mobility of the remaining fission products is greatly reduced. It is not certain that core debris will penetrate the concrete basemat. Even so, at TMI, since the bedrock is similar to the concrete, further penetration would be limited to a few feet at most.

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APPENDIX E

FISSION PRODUCTS WITHIN THE REACTOR CONTAINMENT BUILDING AS **A** CONSEQUENCE OF THE HYPOTHETICAL FUEL-MELTING ACCIDENT

BY

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FISSION PRODUCTS WITHIN THE REACTOR CONTAINMENT BUILDING AS A CONSEQUENCE OF THE HYPOTHETICAL FUEL-MELTING ACCIDENT

In the hypothetical fuel-melting accident that violates the primary system boundaries, a large fraction of some fission products would be released to the reactor containment building (RCB) atmosphere. The RCB is designed to withstand the conditions ensuing from a loss-of-coolant accident (LOCA) resulting from a double-ended break of a main coolant pipe, and normally is capable of withstanding the severe pressure loads and of containing the activity released from fuel for the design-base LOCA case. The airborne activity for this postulated accident is generally taken to be 25 percent of the core inventory of the halogens, 100 percent of the noble gases, and generally, one percent of the nonvolatile fission products. The airborne activity would be different from this in the extended fuel-melting accident under consideration. This inventory is estimated in this appendix.

For most postulated releases of activity, leakage of gas and vapor provides the most rapid transport of biologically important radionuclides. Thus, it is important to estimate the distribution of radionuclides in the potentially mobile gas state and in the relatively immobile liquid state.

For early times after shutdown, iodine represents the major contributor to the hazard presented by the fission product inventory in the core. Estimates indicate that for situations involving fuel melting, essentially all of the iodine would be released from the core (reference 1). At TMI, about one-third was released, and there was significant contact of iodine with water, as would be the case for the postulated fuel-melting accident addressed in the present analysis. Several effects are considered in evaluating the fraction of iodine in the gas state in the reactor containment building (reference 2): 1) The chemical conditions existing in the primary system during the time between release from the fuel and release from the primary system; 2) the chemical conditions in the RCB; 3) the partition coefficient, which relates the concentration in the gas state to that in the liquid state; 4) the availability of silver (to form insoluble AgI); and 5) the propensity to form organic compounds, such as methyl iodide.

These factors are discussed in Appendix F. The combination of high pH, low absolute iodine concentration in the liquid, and reducing conditions (partly aided by radiation) tend toward a high partition coefficient, P., defined as the concentration, per unit volume, in the liquid divided by the concentration in the gas. Conditions in the TMI RCB indicate a partition coefficient of about 2 x 10 , which is not unreasonable when compared to theoretical predictions (reference 3).

Gas-borne concentrations with containment spray operation have been calculated by A. Postma and R. Hilliard (refere~ce 4) and are reproduced in Appendix G. Assuming an initial value of 10 for the partition coefficient, obtained from WASH 1400 (reference 5), they computed the gas-borne amount of iodine, as a fraction of the total core inventory,

as shown in Figure 2 of Appendix H, and in Table 1 (Case 1) of this appendix. Organic iodides are more difficult to remove by reagent sprays. Postma and Hilliard, adapting WASH 1400, assumed that 0.2 percent of the initial iodine present in the core would be in the organic state.

Unfortunately, little can be done in this discussion to predict quantitatively the amount of organic iodides that would be present. However, the methyl iodide fraction given in WASH 1400 is that assumed for the LOCA case, in which the iodine quickly goes to the RCB gas phase and not for the type of accident experienced at TMI. Conditions existing during the TMI accident would tend to inhibit the production of organic iodides in the core and primary system. These conditions are high temperatures, presence of hydrogen, and radiation in the gas space in the primary system. A realistic treatment of the organic fraction would properly set the initial value of methyl iodide in the containment to zero and allow the concentration to climb to an equilibrium value based on a physical model for its creation, taking into account release paths, chemical state during release from the core, release rate of iodine from water, and containment sprays. Thus the estimates for methyl iodine may well be very conservative; the matter certainly deserves more attention. Time did not permit the development of such a model for this report.

After release from the primary system, conditions exist in the RCB that tend to reduce the concentration of volatile organic iodides in the containment airspace. The high temperatures and moist environment in the RCB would tend to hydrolize methyl iodide to methyl alcohol and HOI (reference 6). This process is relatively slow. Any higher organic iodides (than MeI) that might be formed by contact of iodine with oils and greases would be essentially nonvolatile (reference 6).

Although difficult to quantify, it appears that the iodine concentration in the gas phase available for transport is a smaller fraction of the total than that used in WASH 1400 and that required for licensing calculations. To illustrate the effect of different assessments of methyl iodide fractions, Table 1 shows the airborne fractions of meythl iodide for two cases in addition to Case 1 mentioned above. In Case 2, it was assumed that the methyl iodide fraction initially was 0.4 percert of the initial airborne fraction, which was assumed to be 3 percent of the core inventory (reference 4); subsequent reduction due to spray operation was assumed to be similar to Case 1. In Case 3, it was assumed that the methyl iodide fraction was initially 0.4 percent of the initial airborne fraction, whic9 was calculated to be 0.3 percent of the core inventory, assuming P = 10 , the volume of liquid as 50,000 gal, and the volume of gas as 1.5×10^{7} gal (reference 9).

The resultant activity, in curies, corresponding to the cases discussed above are given in Table 1. Values of total core inventory used in tables in this appendix were obtained from reference 7.

Campbell (reference 2) has identified additional factors that would influence the amount of iodine in the gaseous state. It appears that some of the silver in the control rods (which contain about 5,000 lbs.) was released in the TMI accident, and subsequently solidified to a finely divided form; currently, the amount is unknown. From subsequent

Oak Ridge National Laboratory (ORNL) analyses of precipitates at TMI, it is suspected that much of the iodine reacted with the silver to form insoluble AgI. The thermodynamic equilibria favors the formation of AgI for the conditions existing in the containment building (reference 6), but without experimental data this effect cannot be evaluated quantitatively. Irres8ective$_8$of AgI formation, partition coefficients could be as high as 10 to 10 at the potentially attainable NaOH concentrations (pH 8 to 10). If this were so, any iodine transport to the environment would be completely dominated by the assumed organic iodide fractions and by iodine that might be absorbed on aerosols. Also, there is the case that boiling of the water will not cause iodine evolution. At pH 8 to 10 and at high temperature (100°C), iodine is more likely to form compounds in the liquid state than at low temperatures, resulting in larger apparent partition coefficients.

After iodine, the most important radionuclides can be placed in the following groups: (1) Cs and Rb; (2) Te, Se, and Sb; (3) Sr and Ba; (4) Ru, Mo, Pd, Rb, and Tc; and (5) La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb. These are grouped as in WASH 1400, Appendix VII, Table 6 (reference 5). Postma and Hilliard (reference 4) computed the airborne aerosol concentration of these groups, assuming a puff release at the time of core penetration of the pressure vessel (assumed by them to occur at 240 minutes after start of fuel melting), and a continuous release after that, with a release half-life of 30 minutes, as given in WASH 1400. Their results are shown in Figure 3-7 of Appendix G. The asymptotic lower level shown *derives* from resuspension due to spray system operation. The resultant activities in curies in the RCB atmosphere are shown in Table 2.

The iodine-132 resulting from decay of tellurium-132 was addressed. It was assumed that the iodine would be in the RCB atmosphere in the same traction as the tellurium parent; application of a partition function of 10 to the iodine as created then led to an increment to the airborne activity negligible to that addressed previously.

For completeness, the noble gas inventory is shown in Table 3; this is the same as that used for the design-basis accident; for these isotopes, as opposed to the halogens, the historic assumptions appear to match information derived from TMI-2.

It is clear from the above discussion that the amount of radionuclides in the RCB gas space is considerably less than the total inventory, and the amount of activity available for release depends on the time after start of fuel failure that the containment is assumed to fail, and on the made of containment failure.

The hypothetical modes by which radionuclides could be released from the RCB are as follows:

1. RCB failure due to projectile generation at the time that hot fuel falls into the vessel lower head.

2. RCB failure due to projectile generation at the time the molten materials fall into the cavity below the reactor vessel.

3. RCB failure due to overpressurization from hydrogen and noncondensible gases arising from interactions among core and structural materials, water, and concrete.

4. RCB failure due to overpressurization that, for example, could be caused by loss of decay heat removal capability.

5. RCB failure due to penetration of the containment building by molten fuel.

6. RCB violation by open flow paths due to poor operation or poor design such as lack of containment isolation or coolant let-down system communication with noncontained areas.

7. "Normal" leakage.

Modes 1 and 2 are very unlikely because of the arguments enumerated in Appendix C. Mode 3 is unlikely because of the insufficient gas generation rate, as reported in Appendix D and because of the tolerance of the RCB to hydrogen combustion.

Mode 4 -- the possibility that fuel melting could cripple the decay heat removal system (DHRS) because of interference with the sump -- debris generation, and concrete pad undercutting were investigated. It was found that the sump for the DHRS and of the containment spray system is sufficiently separated from the reactor cavity to preclude direct damage, and that the communication path from the reactor cavity to the sump is sufficiently torturous to minimize interference by debris arising from molten fuel (reference 8). However, the potential effects of DHRS failure are such that these factors should be examined in more depth.

Mode 5, where containment is failed by penetration of the concrete basemat by molten fuel, is unlikely. Appendix D indicates, by a conservative treatment, that penetration of the basemat is possible, but after core mass solidification and after a long time. The rock below the TMI-2 site is competent siltstone, and judged to be at least as resistant as reinforced concrete.

Mode 6 (poor operation or design) was shown to be operative at TMI. The main activity release of noble gas (2.5-15 million curies) and iodine (15 curies) is believed to have been from the primary system via the let-down line into the auxiliary building.

Mode 7, "normal" leakage, would be the expected path from containment to the environment for the extended accident and assuming that containment isolation was effective.

If the containment were to fail early in the accident, such as in Modes 1 and 2, the amount of iodine released to the environment would depend on the chemical conditions existing at the time of release of the iodine, the pH of the containment water, the availability of silver in the water, the length of time the sprays (with NaOH) had been operating, and the amount of primary system water that might escape the containment. The data in Table 1 may be near-upper limits. The experience at TMI-2

(Appendix F) supports the assumption of chemically reducing conditions, a high pH, some availability of silver, and operative containment sprays with NaOH. Also, it has been suggested that formation of methyl iodide in the primary system before vessel failure is most unlikely (reference 6). Hence, the fractions of iodPe airborne in the RCB might be very low, perhaps of the order of 10 to 10 . Water ejection that might carry radionuclides from the containment is speculative, depending on the assumptions of the failure mode, time, temperature, and water content. The water remaining in the containment would continue to release iodine if the gas phase concentration is depleted, assuming no remedial measures. This would be a relatively slow effect, as seen to be the case at TMI-2. However, if the iodine is tied up as AgI, which is insoluble, essentially none would be released.

If the RCB were to fail due to overpressurization, as in Modes 3 and 4, cracks would open up in the reinforced concrete and possibly the steel liner and a subsequent release of the gas would occur, the rate depending on the crack size. However, no source of pressure is forseen that could fail containment.

In the event of penetration of the concrete basemat (Mode 5), fission products would have to leak through the siltstone rock and along the concrete-siltstone interface. It is unlikely that significant amounts of radionuclide would reach the atmosphere by this path. Because of ion-exchange characteristics of most soil components, transport through the geologic structure would be quite slow.

The violation of containment due to flow paths arising from poor design or poor operator action was the only path of radionuclides to the environment to occur at TMI. This was due, to a small extent, to containment sump operation at pressure below 4 psi, and, to a greater extent, to continuation of let-down flow from the primary system. It is recognized that the primary system pump seals obtain their cooling water from the let-down system, and therefore, this flow probably was essential for subsequent operability of the pumps. The design fault is that parts of the let-down system were in uncontained areas. It is apparent from measurements at TMI that the liquid transport paths resulted in very little release to the atmosphere.

The remaining leakage path from containment is called "normal", defined for TMI-2 as 0.13 percent of the RCB volume per day at the pressure and temperature conditions existing after a major LOCA. This rate is sensitive to internal pressure and could be much less for the extended accident in hand. However, for the purpose of conservative calculations, the dose to the surrounding population could be computed on this basis using the RCB concentrations given in the tables at the end of this appendix.

We conclude that, for the hypothetical fuel-melting accident under conditions extrapolated from those existing at TMI, the release of radioactive materials to the atmosphere would be larger than at TMI-2 but smaller than that often predicted for the design-basis accident, especially for the iodine isotopes, and might be a great deal smaller.

In summary, the evaluation of the consequences of reactor accidents have, in the past, been dominated by the iodine (thyroid) doses. TMI-2 demonstrates that in this type accident, at least, those estimates have been grossly and conservatively pessimistic. It is apparent that study and evaluation of the behavior, movement, and final place of residence of the iodine isotopes during an accident deserve far more attention by an independent body of experts than was permitted by time and manpower for this study.

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TABLE 1: Halogen Activity in Reactor Containment Building Atmosphere

Time[a]	Total Core Inventory[b]	Fraction Airborne				Curies Airborne		
hrs	Curies	Inor&anic P-10 (c)	Organic[d] Case(1)	Organic[e] Case(2)	Organic[f] Case(3)	Case A[g]	Case B[h]	Case C[i]
5	3.9(8)	0.0	0.0	0.0	0.0	0.0	0.0	0.0
10	3.2(8)	1.8(-3)	1.8(-3)	1.1(-4)	1.1(-5)	1.1(6)	6.1(5)	5.7(5)
20	2.4(8)	5.0(-4)	1.4(-3)	8.4(-5)	8.4(-6)	4.1(5)	1.4(5)	1.2(5)
50	1.5(8)	1.2(-4)	5.6(-4)	3.3(-5)	3.3(-6)	1.0(5)	2.2(4)	1.8(4)
100	9.2(7)	3.3(-5)	1.2(-4)	7.5(-6)	7.5(-7)	1.4(4)	3.7(3)	3.1(3)
200	4.9(7)	2.8(-5)	7.0(-6)	4.2(-7)	4.2(-8)	1.6(3)	1.4(3)	1.3(3)
500	1.2(7)	2.8(-5)	4.3(-6)	2.6(-7)	2.6(-8)	3.9(2)	3.4(2)	3.3(2)

[a]Time after shutdown (fuel melting is assumed to start 3 hours after shutdown; reactor vessel penetration assumed to occur at 4 hours after start of fuel melting).

[b]Iodines and Bromine, from England & Wilson (reference 7).

Starts at P - 10^{4} and increasing to P - 10^{6}, based on Wash 1400, from Hilliard & Postma (reference 4). Assumes credit for AgI formation.

[d]Case 1 - 0.2% of total core inventory, based on Wash 1400, as modified by Hilliard & Postma (reference 4).

[e]Case 2 - 0.4% of initial airborne released at time of vessel failure. Assumed to be 3% of core inventory by Hilliard and Postma (reference 4).

[f]Case 3 - 0.4% of initial airborne concentration, which is 0.3% of core inventory, based on P = 10^{5}, volume of liquid - 50,000 gal, vol of gas - 1.5 x 10 gal [from Campbell (reference 6)].

[g]Inorganic + Organic Case 1.

[h]Inorganic + Organic Case 2.

i Inorganic + Organic Case 3.

TABLE 2: Nonvolatile Fission Product Activity Airborne in Reactor Containment Building Atmosphere

	Cs + Rb		
Time[a] (hours)	Total Activity (curies)	Fraction Airborne	Airborne Activity in RCB (curies)
5	2.6(7)	0.0	0.0
10	8.3(6)	5.0(-3)	4.0(4)
20	2.0(6)	4.5(-5)	9.0(1)
50	1.4(6)	1.0(-6)	1.4(0)
100	1.4(6)	1.0(-6)	1.4(0)
200	1.3(6)	1.0(-6)	1.4(0)

	Sr + Ba		
Time[a] (hours)	Total Activity (curies)	Fraction Airborne	Airborne Activity in RCB (curies)
5	3.2(8)	0.0	0.0
10	2.6(8)	3.0(-4)	8.0(4)
20	2.1(8)	2.3(-6)	5.0(2)
50	1.8(8)	6.0(-8)	1.0(1)
100	1.6(8)	6.0(-8)	1.0(1)
200	1.4(8)	6.0(-8)	1.0(1)

Table 2 (Continued)

Time[s] (hours)	Te + Sb + Se Total Activity (curies)	Fraction Airborne	Airborne Activity in RCB (curies)
5	1.3(8)	0.0	0.0
10	1.2(8)	3.0(-2)	4.0(6)
20	1.0(8)	2.0(-4)	2.0(4)
50	7.6(7)	5.0(-6)	4.0(2)
100	5.0(7)	5.0(-6)	2.0(2)
200	2.3(7)	5.0(-6)	1.0(1)

Time[s] (hours)	Ru, etc.[b] Total Activity (curies)	Fraction Airborne	Airborne Activity in RCB (curies)
5	4.3(8)	0.0	0.0
10	4.0(8)	2.0(-3)	8.0(5)
20	3.7(8)	1.0(-5)	3.7(3)
50	2.9(8)	3.0(-7)	8.7(1)
100	2.1(8)	3.0(-7)	6.3(1)
200	1.4(8)	3.0(-7)	4.2(1)
500	8.7(7)	3.0(-7)	2.6(1)

Table 2 (Continued)

Time[a] (hours)	La, etc.[c] Total Activity (curies)	Fraction Airborne	Airborne Activity in RCB (curies)
5	2.7(9)	0.0	0.0
10	2.4(9)	3.0(-5)	7.2(4)
20	2.0(9)	2.0(-7)	4.0(2)
50	1.5(9)	6.0(-9)	9.0(0)
100	1.0(9)	6.0(-9)	6.0(0)
200	6.6(8)	6.0(-9)	3.4(0)
500	4.0(8)	6.0(-9)	2.4(0)

[a]Time after shutdown. (Fuel melting is assumed to start 3 hours after shutdown; melt through is assumed to occur at 4 hours after start of fuel melting.)

[b]Ru, etc., includes Ru, Mo, Pd, Ph, and Tc.

[c]La, etc., includes La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

TABLES 3: Noble Gas Activity in Reactor Containment Building Atmosphere

Time (min)	(hours)	(days)	Activity[b] (curies)
300	5	0.2	2.7(8)
600	10	0.4	2.4(8)
1,200	20	0.8	2.0(8)
3,000	50	2.1	1.4(8)
6,000	100	4.2	9.3(7)
12,000	200	8.3	6.3(7)
30,000	500	20.8	1.2(7)

[a] 100% of noble gases are assumed to be airborne.

[b] Obtained from reference 7.

APPENDIX F

BEHAVIOR OF IODINE UNDER ACCIDENT CONDITIONS AT THREE MILE ISLAND

BY

David 0. Campbell

October 26, 1979
Washington, D.C.

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

September 7, 1979

To: R. E. Brooksbank

From: D. O. Campbell

Subject: Supplement to Memo of August 16, which was a Response to Item (2) of the Letter of August 2 from the President's Commission on the Accident at Three Mile Island, Regarding Iodine Evolution

Bill Stratton has requested further information regarding iodine behavior, and particularly more quantitative descriptions of why iodine did what it did. The comments here supplement my earlier response to this question. Unfortunately, I am unable to provide anything approaching a quantitative description of the behavior of iodine at TMI. The more I have looked into the problem the more doubt I have accumulated with respect to providing such a treatment. There are simply too many uncertainties and loose ends, both in iodine chemistry and in the exact chemical conditions during the sequence of events during and following the accident.

This response is quite subjective and represents, in many cases, my personal opinion about iodine behavior. If I have concluded anything, it is that anyone who thinks he thoroughly understands why iodine did what it did during the accident is following a simplistic approach based on lack of knowledge of the literature. The most he can hope for is a framework to provide a qualitative understanding about iodine behavior. In general, iodine behavior was not contrary to anything we know about it, but prediction of its behavior in a quantitative way is simply not possible with the information available. If more than this is needed, then I believe that much more time and effort is going to be required than I can expend now.

The interpretation of the behavior of iodine is complicated by two major factors in addition to the limited knowledge about the accident itself; these are (1) the complex chemistry of iodine, involving at least four and possibly more than eight different simultaneous equilibria, depending on conditions, and involving probably eight or ten iodine species, and (2) gaps in the body of knowledge about these many reactions, especially for temperatures very far from room temperature. In addition, there is the whole field of radiation chemistry and reactions with radiolysis products of water. As a result, it is generally not possible to provide a quantitative prediction, a priori, about how iodine will behave during an accident sequence; on the other hand, the observed behavior can be interpreted, in a general way, in terms of the published information. Thus, theory is of very limited value for predictive purposes, but it provides a qualitative framework for understanding what happened after the fact.

A good example of the situation is provided by the most recent review of this problem, by Vinson in France.1 This is a long review, listing some 247 references, and concluding with a note of helplessness, if not despair. He actually concludes that we know a great deal but can predict almost nothing, that more information is needed in a number of areas including both basic chemical data and large pilot system tests, and that interpretation of even that data could be very difficult. He also concluded that under normal conditions in an operating reactor, when the iodine concentration is lower by perhaps 10^4, predominant species should be HOI and iodate ion. In contrast, the people I have talked to expect that, under TMI conditions, the predominant form was probably iodide ion.

There is agreement that a number of species of iodine can exist simultaneously. Many of these are highly reactive transient or intermediate species present in very low concentration; but they can, over a period of time, cause iodine to react in ways not expected for the dominant species (for example, with paint or organics). It is agreed that molecular or elemental iodine is the main form leading to volatility; but there is great disagreement about whether or not one or more other species may be volatile, particularly HOI. There has been argument about the volatility of HOI for over a decade, and no conclusive evidence has been produced, so it seems likely that it is rather nonvolatile. In that case, volatility relates primarily to the extent to which iodine exists as molecular iodine. Complications can arise from additional problem areas, however, if organic material is present; and this results from formation of methyl iodide and possibly other organic iodides. Thermodynamic data suggest that in a moist environment, especially at elevated temperatures, methyl iodide should hydrolyze and not exist to an appreciable extent. In other environments, however, such as on a charcoal filter that also contains organic material in flowing air, methyl iodide may form, and it could be a significant source of iodine release from charcoal.

I am suggesting, in Fig. 1, a flowchart of the pathways by which iodine was transported around the TMI system and out of it. I have very limited data, so the discussion based on this chart will be rather qualitative. With more complete information, a better description of the events at TMI should be possible. However, I suggest that the use of such a chart may assist in both understanding and explaining what happened during and following the accident.

A large fraction of the iodine in the fuel was released when the fuel heated up to a very high temperature, and at that time the fuel must have been surrounded by a gas phase consisting of superheated steam and hydrogen (arrow 1). The hydrogen originated from the zirconium-steam reaction that would be going on at the same time because of the high temperature. Iodine is presumed to be released from fuel as iodide (RbI or CsI, for example) or molecular iodine. At very high temperatures these compounds might dissociate, but during the cool-down the iodine should be converted to iodide because of the strong reducing properties of high temperature hydrogen. Thus, the iodine at this point should be RbI, CsI, and/or HI.

The gas containing the iodine cools and contacts primary coolant within the primary loop (arrow 2), still in the presence of excess hydrogen and intense radiation. These conditions are strongly reducing and favor the existence of iodine as iodide ion which is quite soluble in water. If the pH is reasonably high the volatilization of iodine is very slight. At this point the path branches several ways.

First, consider the containment building. This was not a source of release at TMI, but it is about the only situation where the observed behavior can be compared to theory. The gas "bubble" was vented into the containment building (arrow 3), but this gas probably contained very little iodine. A large volume of water from the primary coolant loop was discharged into the containment building (arrow 4), and this carried a large amount of iodine, probably most of that released from the fuel. This was then mixed with the gas, the atmosphere in the containment building, and other quantities of water (spray system, leakage from coolers, etc.)(arrows 5 and 6). By this time the containment building was sealed, so the iodine is still there. It is, then, possible to compare the volatility of that iodine with theory.

The common approach is to treat the partition coefficient. P, of iodine:

$$P = (\text{concentration in liquid})/(\text{concentration in gas}) \quad (1)$$

where the concentrations are in the same units, such as grams per liter (g/1). The ratio of the quantity of iodine in the liquid to that in the gas is P times the appropriate ratio of volumes:

$$\frac{\text{(iodine in liquid)}}{\text{(iodine in gas)}} = P \frac{\text{(volume of liquid)}}{\text{(volume of gas)}} \quad (2)$$

The estimates I have seen place about 0.007% of the iodine (core inventory) in the containment building atmosphere. The fraction in the liquid phase is around 50% of the core inventory (within a factor of 2). Thus, the left side of Eq. (2) is 7×10^{3}. The volume ratio is about 500,000 gallons of water to 1.5×10^{7} gallons of gas or 0.033. Thus, the value of P is about 7000/0.033, or 2×10^{5}.

One of the most understandable explanations of iodine behavior is the paper by Eggleton,[2] which, although fairly old, is still a standard reference. A graph of partition coefficients from Eggleton's report is given in Fig. 2. This is data calculated for 25°C, close to the containment building temperature. Variables are iodine concentration, pH, and oxidation-reduction potential. If the water does contain half the iodine in about 500,000 gallons, the concentration is about 0.0004 g/l. The pH is about 8. Figure 2 did not include a curve for these conditions, but I have estimated the position of such a curve by interpolation from the curves given. The estimated curve is drawn in red and further indicated by a series of **x's**.

The minimum value of P is about 10^{3}, occurring for the case of the average iodine valence being zero. The values increase by large amounts if the solution is either reduced or oxidized from this value, so the redox potential

is really the critical variable. I have guestimated that the iodine is strongly reduced to begin with (right side of curve), and the observed value of P at 2×10^5 corresponds to a potential less than 0.6 volts. On the other hand, there is oxygen in the containment building, and at pH 8 the slow reaction to produce iodate ion could occur in a time short compared to the cooling time. The iodate reaction causes P to be much larger than indicated in Fig. 2 for the minimum value (from about 10^3 to 10^6). Thus, about all that can be said is that the observed iodine volatilization is not unreasonable, but it certainly could not have been forecast without a lot more information than we have, particularly about the oxidation-reduction potential.

It is apparent that there are several ways to achieve a larger value of P (lower iodine **volatilization)** by changing conditions, namely (1) a lower iodine concentration, (2) a higher pH, or (3) either strong oxidation or reduction. The concentration could be decreased by dilution (not really practical) or by precipitation of part of the iodine (as AgI, for example); in fact, there is indication that the sludge in the bottom of the containment building contains some iodine, possibly because such precipitation occurred with silver or some other metal in the system. The pH could certainly be increased further, by adding sodium hydroxide, but this would make subsequent cleanup of the water more difficult. Finally, the solution could be oxidized or reduced by adding appropriate chemicals. Quite a bit of attention has been given to the addition of thiosulfate, to reduce iodine to iodide, along with the sodium hydroxide and boric acid in the containment spray solutions. Of interest in this respect is the April 1971 issue of Nuclear Technology, which published the papers from a Symposium on Reactor Containment Spray System Technology.

The actual release of iodine from the TMI plant occurred through pathways different from this, however, involving the transfer of primary coolant water into the auxiliary building, and then volatilization of iodine from that water into the atmosphere. I have no detailed information on these events, so the discussion is qualitative. Some water was transferred into tanks in the auxiliary building (arrow 7), such as bleed tanks and parts of the primary letdown and makeup system. The air in these tanks was vented to the building as the tanks were filled (arrow 8). This represents release of a small volume of air (comparable to the liquid volume), and that air was probably reasonably well equilibrated with iodine. In the worst case P would be about 10^2 (see Fig. 2) and the volume ratio would be about 1, so Eq. (2) gives a maximum release of about 1% of the contained iodine. I have heard statements that, when low-level solutions were transferred from one tank to another, in some cases a quantity of iodine approximating 1% of that contained was in fact released.

This iodine can be retained in the plant by holding the solution within the tanks until the iodine decays, or by changing the chemical conditions as discussed above (higher pH, reducing agents, etc.). I also understand that, a few days after the accident, certain tanks were vented back to the containment building (dashed arrow) to prevent the iodine being vented to the auxiliary building and released. Better provision for this sort of option might be a useful backfit to other plants. The important point is that the iodine-containing gas in tanks should not be vented to a large-volume gas stream, whether this is accomplished by simply not transferring solutions unnecessarily or by modifying the tank vent systems.

The other pathway for iodine involved the overflow and leakage of contaminated water to the auxiliary building floor(arrow 9). This water was then contacted by a relatively very large air volume flowing through the building, and there would be significant transfer of iodine to the air stream (arrow 10) even if P is quite large. It is also possible that the contact with excess air, over a period of time, would oxidize the reduced iodine species, thereby causing a decrease in P and greater volatility. After a few days attempts were made to reduce iodine volatilization from this water and the wet floor by adding sodium hydroxide and thiosulfate to reduce iodine to iodide, and still later by covering the floor with plastic to isolate it from the flowing air; these actions probably achieved a fair measure of success. Iodine containment is clearly much easier, however, when the iodine-containing solutions are confined in tanks.

Very rough estimates of the relative importance of these two paths through the auxiliary building might be made along the following lines. Suppose that 99% of the approximately 2×10^6 Ci of 1-131 that ended up in the auxiliary building followed arrow (7) and 1% followed arrow (9). Supose further that in the tanks P was 2×10^5 (as in the ***containment*** **building) and that V (liquid)/V (gas) was** 1. Then, the amount of iodine released to the atmosphere of the auxiliary building (arrow 8)) would be, from Eq. (2), 2 x 106/ 2×10^5 - 10 Ci. Presumably, the total iodine released to the auxiliary building atmosphere (plant release plus the amount on charcoal filters) was an order of magnitude larger than this. In contrast, nearly all of the xenon in the water following this path would have been released.

The 1% assumed to follow arrow (9) would contain 2×10^4 Ci. Even if P was 10^5 (it may well have been much smaller because of oxidation of iodide to iodine), the volume ratio, V(liquid)/V(gas), was very small. For example, if 100 Ci was released via arrow (10) and P was 10^5, the effective volume ratio would be $2 \times 10^4/(10^2 \times 10^5)$, or 2×10^{-3}. This would not be the true volume ratio since equilibrium would not be reached with the flowing air stream; the process would probably be diffusion controlled.

It is my understanding that the iodine release rate decreased appreciably when the caustic thiosulfate was put on the auxiliary building floor. This action would substantially increase P for steps indicated by arrow (9) and (10), decreasing the iodine release via that path. This observation confirms that the water outside tanks was probably the primary source of iodine release. The situation with respect to xenon is quite different, with the dominant source probably being gas vented from tanks.

The final barrier to iodine release was the charcoal filter system (arrow 11), and the charcoal was not particularly effective for the first few days. Thus, the low iodine release from the plant was only to a small extent the result of the charcoal air filters. It was primarily caused by (1) the quite small volume of air that contacted the larger amounts of iodine contained in various tanks (venting of this air probably was a minor contributor to the total iodine release), and (2) the relatively much smaller amount of iodine

that was in the water not contained in tanks (a larger fraction of this iodine volatilized and was probably the major contributor to the release).

References

1. J. M. Vinson, "Hydrolysis and Transport of Iodine in the Primary Circuit of a Light Water Reactor, CEA Department de Surete Nucleaire, FRRSR-131, Feb. 1978.

2. A. E. J. Eggleton, UKAEA Report AERE-R-4887 (1967).

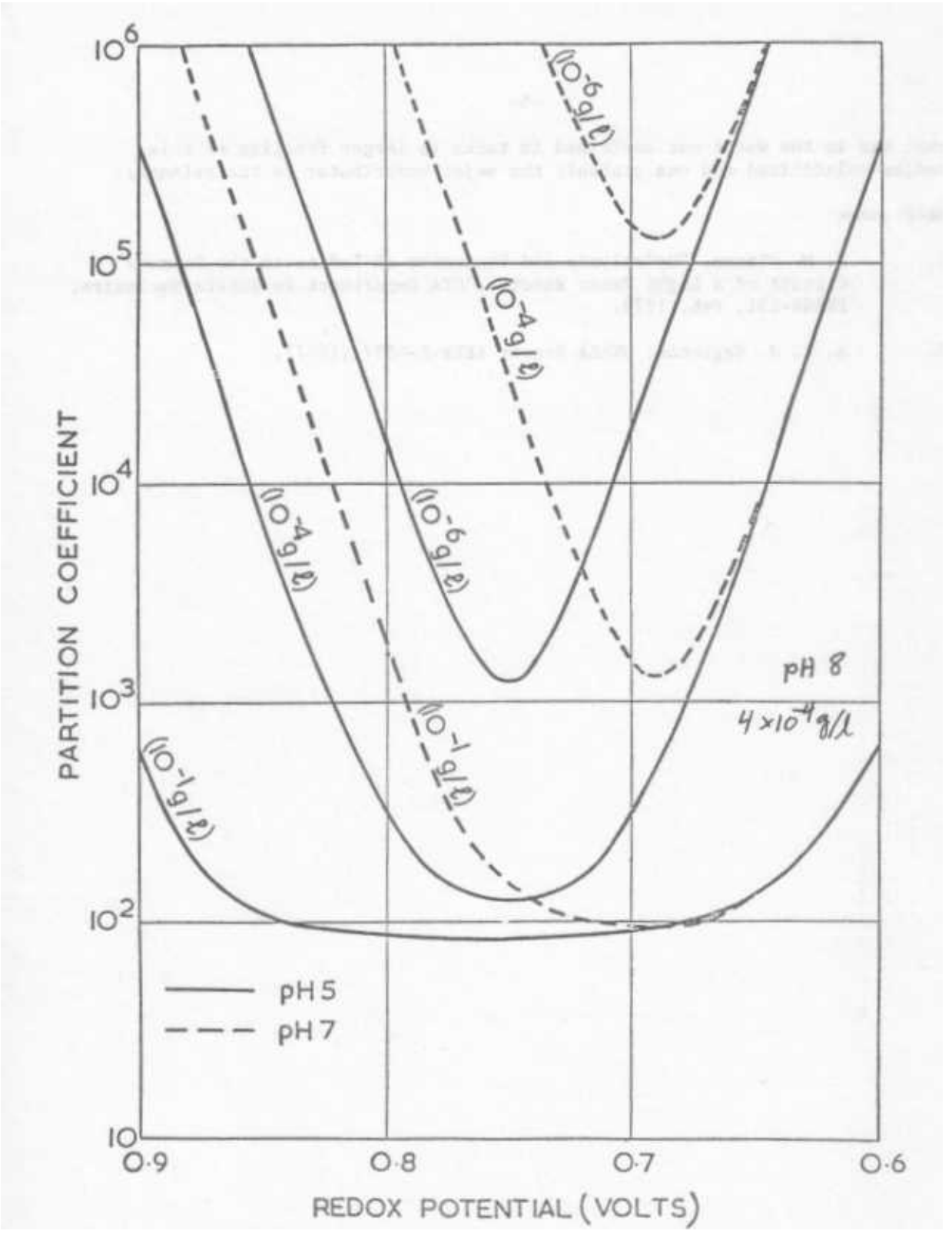

FIG.2. PARTITION OF IODINE BETWEEN GAS PHASE AND WATER AS A FUNCTION OF REDOX POTENTIAL
(iodine concentration of aqueous phase in brackets)

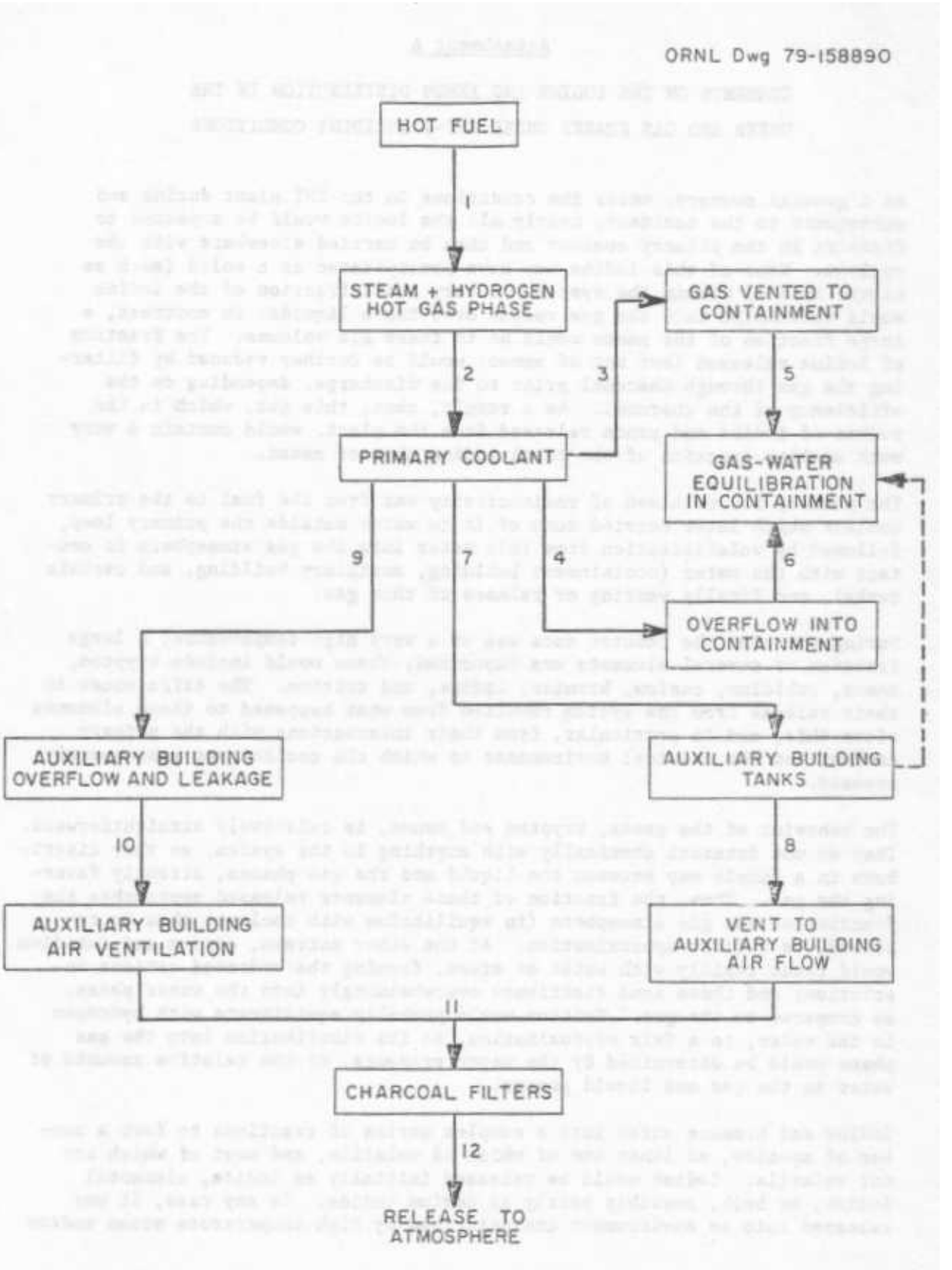

Fig. I. Iodine transport paths

Attachment A

COMMENTS ON THE IODINE AND XENON DISTRIBUTION IN THE WATER AND GAS PHASES UNDER TMI-2 ACCIDENT CONDITIONS

As a general summary, under the conditions in the TMI plant during and subsequent to the accident, nearly all the iodine would be expected to dissolve in the primary coolant and then be carried elsewhere with the coolant. Some of this iodine may have precipitated as a solid (such as silver iodide) within the system. A very small fraction of the iodine would volatilize into the gas volume over these liquids; in contrast, a large fraction of the xenon would be in these gas volumes. The fraction of iodine released (but not of xenon) would be further reduced by filtering the gas through charcoal prior to its discharge, depending on the efficiency of the charcoal. As a result, then, this gas, which is the source of iodine and xenon released from the plant, would contain a very much smaller fraction of the total iodine than of xenon.

The pathway for released of radioactivity was from the fuel to the primary coolant which later carried some of it to water outside the primary loop, followed by volatilization from this water into the gas atmosphere in contact with the water (containment building, auxiliary building, and certain tanks), and finally venting or release of this gas.

During the time the reactor core was at a very high temperature, a large fraction of several elements was vaporized;-these would include krypton, xenon, rubidium, cesium, bromine, iodine, and tritium. The differences in their release from the system resulted from what happened to these elements after this, and. in particular, from their interactions with the primary coolant and the chemical environment to *which* the coolant was subsequently exposed.

The behavior of the gases, krypton and xenon, is relatively straightforward. They do not interact chemically with anything in the system, so they distribute in a simple way between the liquid and the gas phases, strongly favoring the gas. Thus, the fraction of these elements released approaches the fraction of the gas atmosphere (in equilibrium with coolant) that is released, to a first approximation. At the other extreme, cesium and rubidium would react rapidly with water or steam, forming the hydrated cations in solution; and these ions distribute overwhelmingly into the water phase, as compared to the gas. Tritium would probably equilibrate with hydrogen in the water, to a fair approximation, so its distribution into the gas phase would be determined by the vapor pressure, or the relative amounts of water in the gas and liquid phases.

Iodine and bromine enter into a complex series of reactions to form a number of species, at least one of which is volatile, and most of which are not volatile. Iodine would be released initially as iodide, elemental iodine, or both, possibly partly as cesium iodide. In any case, it was released into an environment characterized by high temperature steam and/or

water, and hydrogen. Under these conditions iodine, whatever its initial form, would be reduced very rapidly to iodide, and the iodide would exist in the primary coolant as iodide ion. Volatile species are molecular iodide and, it has been suggested but not proven, hypoiodous acid (HOI); these species would not exist to any appreciable extent in the presence of the large amount of water and the reducing conditions resulting from hydrogen and radiation.

Thus, the iodine would quickly dissolve in the reactor coolant and be carried wherever the coolant went. The distribution of iodine between this liquid phase and a gas phase would strongly favor the liquid phase, and would depend on the oxidation-reduction potential and pH, among other variables. Both the reducing potential (because of hydrogen) and the relatively high pH of the coolant following the accident favor a very low volatility for the iodine. In the containment building the high pH (from the borate buffer and the sodium hydroxide introduced via sprays) similarly stabilized iodine in solution.

There is no actual data for the exact conditions that existed during the accident, especially for high temperatures. However, there have been a number of studies from which qualitative estimates of iodine behavior can be projected. The work of A. E. J. Eggleton at Harwell, UK (AERE-R-4887), for example, would suggest a partition coefficient (cocentration in the water divided by concentration in the gas phase) for iodine in excess of 104. Then, for a given fraction of the gas-released, which would contain a comparable fraction of the xenon, there would be a small fraction of the iodine by this large factor - more than 104. This is in general agreement with V. L. Johnson's observation which prompted the question.

The gas phase associated with the reactor system, then, contained a large fraction of the xenon and a very small fraction of the iodine. This gas normally passed through charcoal filters before being released from the plant; such filters remove iodine from the gas with an efficiency varying from very poor to very high, depending on the condition of the charcoal. In contrast, they do not remove xenon. This provides a further reduction in iodine release, relative to xenon. The charcoal filters on the auxiliary building off-gas were apparently rather inefficient shortly after the accident, but the system was subsequently uprgraded to provide effective iodine retention.

As to where the iodine is now, nearly all has decayed, with only about 200 curies remaining. The daughters of 1-131 are Xe-131m and stable Xe-131. Of the 1-131 remaining, most is still in the water and a small fraction is on charcoal filters. In addition, some of the iodine may also exist as solid silver iodide, formed by reaction of the iodide ion with silver probably from the control rods. It is not clear when or if this compound was formed, but it would further reduce the release of volatile iodine species. Such a solid would have settled out in quiescent regions of the primary system or in the containment building if it was transported out of the primary loop.

The partition coefficient for iodine could be substantially smaller (thereby leading to greater volatilization of iodine into a gas phase) if the pd is decreased or the oxidation potential changed. Because of the possibility of an inadvertent change of conditions that could lead to increased iodine release if the water is manipulated or processed, the safest course was to delay such operations until 1-131 largely decayed away.

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September 19, 1979

Dr. W. R. Stratton
President's Commission on the
Accident at Three Mile Island
2100 M Street, NW
Washington, D.C. 20037

Dear Bill:

Subject: Consideration of Iodine Release Problem in the Event of a More Severe Accident Sequence at TMI, Leading to a Partial Core Meltdown

Reference: Previous Memos of August 16 and September 7, 1979.

I was asked to look into the problem of iodine release from the TMI reactor in the case that something else went wrong leading to a meltdown that eventually penetrated the core vessel but not the containment building. I have no detailed information about the sequence of events postulated, so the contents of this memo are based on generalizations in that respect. The hypothetical accident will be compared to the results of the actual accident, insofar as they are known.

The comparison of the two accidents may be summarized as follows:

1. The total amount of iodine released from the fuel would increase by a factor of probably about 2, and possibly as much as 5.

2. Part of this iodine would precipitate with silver originating from the control rods, and it would thereby be removed from the water, and from consideration as a source for release. This probably happened to some extent in the actual accident; silver is present in the insoluble material in the bottom of the containment building. I would expect it to be much more significant in the hypothetical case, possibly reducing iodine in the water to a very small fraction of its concentration in the real accident.

3. The partition coefficient would be much larger in the hypothetical case following the spray injection of larger amounts of sodium hydroxide into the containment building. Such injection is one of the expected responses to a threatened meltdown. With reasonable addition of sodium hydroxide, the partition coefficient would probably increase to the range of 10^6 to 10^8, leading to a reduction in iodine volatility by a factor of 10 to 10^3.

Because of the effectiveness of these mechanisms in preventing iodine volatilization, the primary release route would probably be something different, namely dispersion of a mist, aerosol, or fine particles in the air or transfer of solution containing iodine out of the containment building and into a different chemical environment. The latter route was, in fact, the primary source of iodine release in the actual accident.

Reevaluation of iodine partition in TMI containment building. The analysis of containment building water indicates a lower concentration of dissolved iodine than was estimated in my memo of September 7. The measurements are approximately 1.4×10^{-4} g/l based on ^{131}I determined by gamma scanning and 1.0×10^{-4} g/l based on ^{129}I determined by neutron activation analysis. Because of decay since the accident, the radioactivity of ^{131}I is very low, so accuracy is not expected to be very good. The ^{129}I determination involves some chemical separations during which small losses could lead to a value somewhat low. Accordingly, the average, 1.2×10^{-4}, seems to be as good an estimate as any.

There is a substantial amount of precipitate in the sample from near the bottom of the containment building. This precipitate contains a significant amount of iodine, and it also contains silver. The actual amounts are not known because we do not know the total amount of precipitate or the degree to which the sample is representative. However, precipitation of silver iodide would seem reasonable. Other data suggests that some iodine precipitated; for example, the I/Cs ratio in containment building water is lower by about a factor of two than it was in three separate samples of primary loop water analyzed earlier. This suggests that some (perhaps half) of the iodine released precipitated after the primary water was transferred into the containment building. Significant amounts of a number of radioisotopes are also found in the precipitate.

The volume of water in the containment building is now about 6×10^5 gallons, so with 1.2×10^{-4} g/l of iodine, there would be about 270 g of iodine or 17% of the core inventory. It is still presumed that 0.007% of the iodine is in the gas phase in containment (although I do not know whether this number applies now or was derived early in the accident when there was less water in containment). Using these numbers in equation (2) of the previous memo, the left side of the equation is 2.5×10^3 instead of 7 x 103, the volume ratio is 4×10^{-2}, and P comes out to be 6×10^4, about one-third the previous value. Probably 10^5 is as good a number to use as any.

The red curve on Fig. 2 of the prior memo of September 7 would be displaced upwards slightly because of the lower iodine concentration present (1.2 x 10^{-4} *instead of 4 x* 10^{-4} *g/l assumed before.) A partition coefficient of* 10^{5} *still puts us pretty far up one side of the curve. I will point out,* again, that the partition coefficient is increased if the pH increases, except when iodine is very strongly reduced to iodide, in which case it is little affected.

Evaluation of iodine distribution under meltdown hypothesis. In the new situation you are considering in which a core meltdown occurs, the sequence of events culminating in iodine release would change in several respects. These are considered in the following discussion:

(a) How much more iodine would be released from a meltdown? More iodine would be released, but it cannot be a large factor because quite a bit was released anyway. At most, all of it would be released, and this would amount to an increase by a factor of 2 from estimates that half the iodine was released to a factor of 4 or 5 from the fraction that can be accounted for based on analyses of the liquid inventory. As stated above, some is precipitated and not included in these analyses.

(b) What about silver from the control rods? There would be more silver released, possibly much more, In the hypothetical accident. However, the amount of silver entering the water cannot be estimated with much confidence. At some temperature between the melting points of the control rod alloy and of its cladding, the silver, cadmium, and indium would fall to the bottom of the core vessel, and a small amount might dissolve. This probably happened to some extent in the actual accident. When melted fuel reaches the bottom of the core vessel, the very high temperature would cause volatilization of the silver. The vapor pressure of silver Is about 0.1 atmosphere at 2000°K, much higher than that of barium or tin, for example. The silver vapor would condense in the water as very fine metal particles or, if the temperature is low enough, it might form silver oxide or hydroxide. Probably, iodine would be released before silver, while the fuel is heating up, and silver would not be vaporized until after the fuel melts.

In the presence of oxygen, even at very low concentration, silver metal will react with iodide to form silver iodide. At pH 8 to 10, the reaction goes very strongly to form AgI, leaving an iodide concentration of the order of 10^{-20} *M. In the presence of excess hydrogen, this reaction would be inhibi-*ted, but there is excess air in the containment building. Thus, it is conceivable that AgI would not form in the primary loop but would form once the primary water escapes into the containment building where there is an excess of oxygen. In addition, the high radiation level would tend to strongly favor the reaction even If oxygen, per se, is absent. (I consulted with C. F. Saes, Jr. about the thermodynamics of this reaction.)

(c) What does silver iodide precipitation do to the iodine solubility? There is a very large quantity of silver in a reactor, several thousand

pounds, and dissolution of only a very small fraction--the order of a gram--will suppress the iodine solubility. The solubility product of AgI is 1.5 x 10-16 at 25°C. In the existing containment water, the Cilver concentration is 1.1 x 10-6 M, So

$$Ksp = (AG+)\ (C) = (Ag)\ (1.1 \times 10\text{-}6) = 1.5 \times 10\text{-}16.$$

So, the silver concentration in equilibrium with this iodine concentration is 1.35 x 10-10 M, which translates into 33 mgs in 600,000 gallons. The point is, very little silver has to react. The solubility product increases with temperature, and, by analogy to AgCl, it might be 10 to 20 times larger at 100°C. This would require less than a gram, so it is apparent that only a minute fraction of the silver had to dissolve to exceed the solubility product of AgI in the actual accident.

In the hypothetical accident, I would expect that nearly all the silver would vaporize in the core vessel, and it probably would form very fine or colloidal particles in the water. This eventually would reach the water in the containment building, either by venting or through a rupture in the core vessel. At some point, the silver and iodide would react, and some silver might also react to form oxide at fairly low temperature. In any case, only a few grams of silver dissolved in several hundred thousand gallons would precipitate iodine to a concentration ten times lower than that now observed in the containment building.

At this point, we have bracketed the iodine concentration in containment water between a maximum of 4 to 5 times greater than that actually observed, and a minimum that is very much less than that observed. The former estimate is based on rather unrealistic assumptions. The latter is based on some reaction with silver, which is supported by the presence of both silver and iodine in the precipitate in the containment building now. My best guess is that the iodine concentration actually in the water, following a meltdown scenario, would be similar to (or possibly much lower than) the iodine concentration now in the water at TMI.

(d) Finally, what would the partition coefficient be, whatever the iodine concentration in the water should happen to be? As pointed out in Section 1, the partition coefficient now is about 105 (factor of 2). In the event of a hypothesized meltdown, certain actions would be taken which would change the events substantially from those which actually occurred. Most important, there would be much greater use of the spray system, and more water and sodium hydroxide would be introduced into containment. This would increase the pH, and the higher pH would increase P, thereby decreasing iodine volatility.

At this point, I will return to Eggleton's study, referred to before. Figure 1 gives values for P as a function of iodine concentration at several pH's. The solid curves represent the worst possible case--they represent

the minima of the family of curves represented on Fig. 2 of the previous memo. Thus, if there is any net oxidation or reduction (either one), P will be larger than indicated' by the curves of Fig. 1. For our actual case, the iodine concentration is 1.2×10^{-4} g/l (red line). For the hypothetical case, it could (inconceivably) be as large as 5×10^{-4}, but more likely it would be less than 10^{-4} ~to the right). At 10^{-4} g/l and pH 8, P is about 10^{4}; at pH 9, it is 10^{5}, and it increases one order of magnitude with each pH unit. At lower iodine concentrations, P increases in inverse proportion to the iodine concentration.

In the actual accident, the water ended up at a Ph slightly above 8, and it contains about 2000 ppm boron and 1200 ppm sodium. If the boron was added as boric acid, its neutralization with sodium hydroxide (based on experimental titration curves we ran some months ago) would require 6000 kg of NaOH to reach pH 9 and 12,100 kg to reach pH 10 (and less if the smaller volume existing shortly after the accident is used instead of the 600,000 gallons existing now). These quantities are approximately 4000 and 8000 gallons of 30% sodium hydroxide solution, respectively, which is no big deal. Some sodium hydroxide has already been added, but the important point is that it would be quite easy to increase the pH to 10 or higher by spraying in a reasonable volume of sodium hydroxide solution, an action that probably would have been taken in any case. This would increase P to 10^{6} or greater. Thus, reasonable actions that should be taken in case of a pending or actual meltdown, namely spray injection of considerable quantities of sodium hydroxide, will bring about a larger value for the partition coefficient and a lower volatility of iodine than that attained at TMI.

In Figure 2, similar curves are shown for water at 100°C, and the hypothetical situation would be somewhere between these two cases, possibly approaching Figure 2. Note that P is considerably larger at conditions of interest--10^{-4} g/l iodine and pH 9, for example, giving $P = 3 \times 10^{6}$. Thus, contrary to first expectations, iodine evolution is much lower under pertinent conditions if the water is hot, although this is not the case at low pH and high iodine concentration.

The dashed curves on Figure 1 are worth mentioning. They are calculated for the case of iodate being formed. This reaction (to form iodate) is very slow at low pH, but at pH 9 or above it probably occurs in times of the order of one minute at room temperature. There is no data on the rate or equilibrium constant for higher temperatures, but it would presumably be faster. If the pH is 7 or greater, P is increased by a large factor, and at pH 9 it is clear off the graph. Thus, if this reaction is significant, P would be larger, and it could well be very large--10^{7} or more.

It is concluded that, under the meltdown situation, the partition coefficient would be substantially larger than the value actually existing at

TMI now. This is primarily the result of the addition of larger quantities of sodium hydroxide, thereby increasing the pH over that presently existing.

Very truly yours,

D. O. Campbell
Chemical Development Section
Chemical Technology Division

DOC:scr

Attachments (2)

cc: R. E. Brooksbank
D. E. Ferguson
A. P. Malinauskas
R. G. Wymer

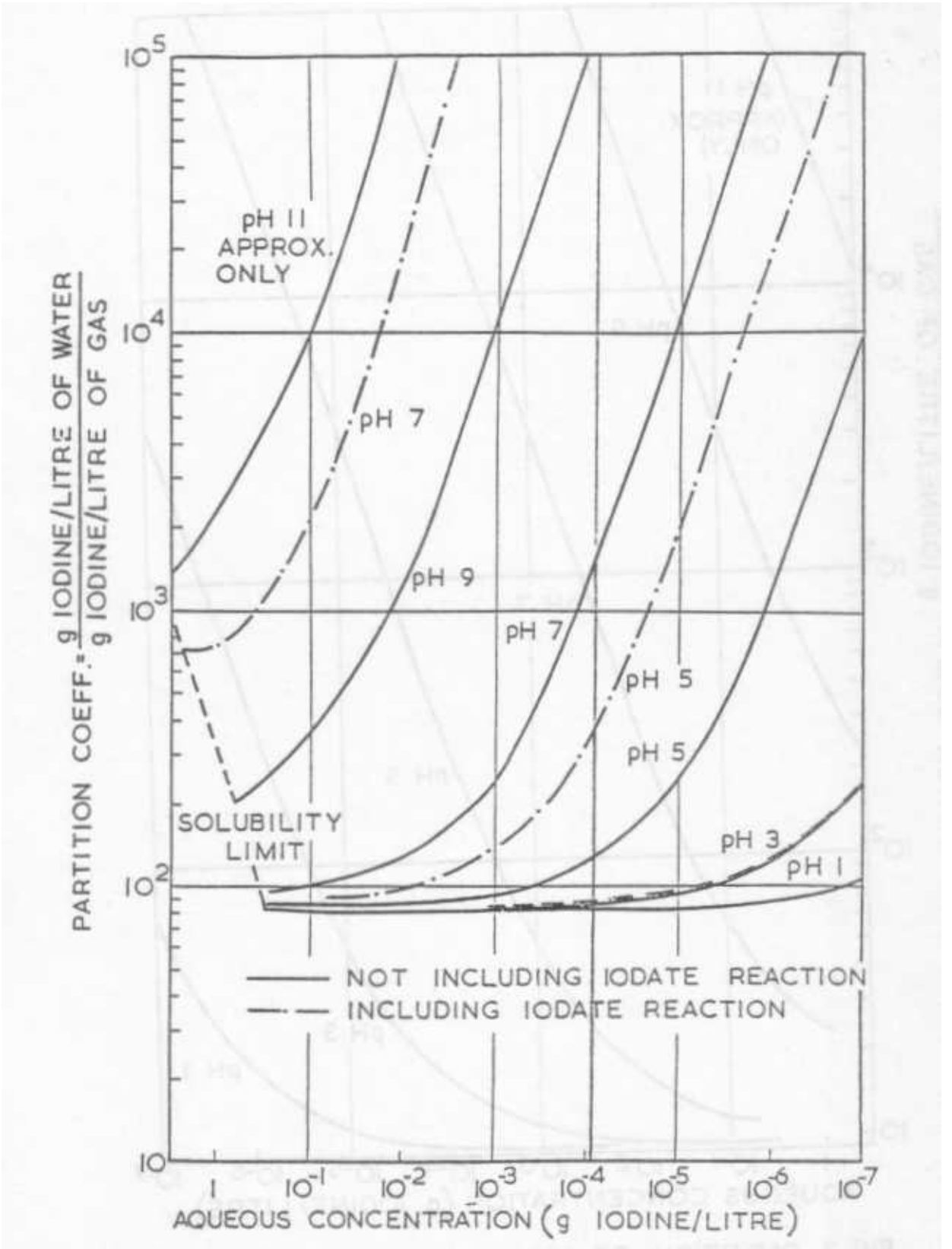

FIG. I. PARTITION OF IODINE BETWEEN GAS PHASE *AND WATER AT 25 ^{o}C*

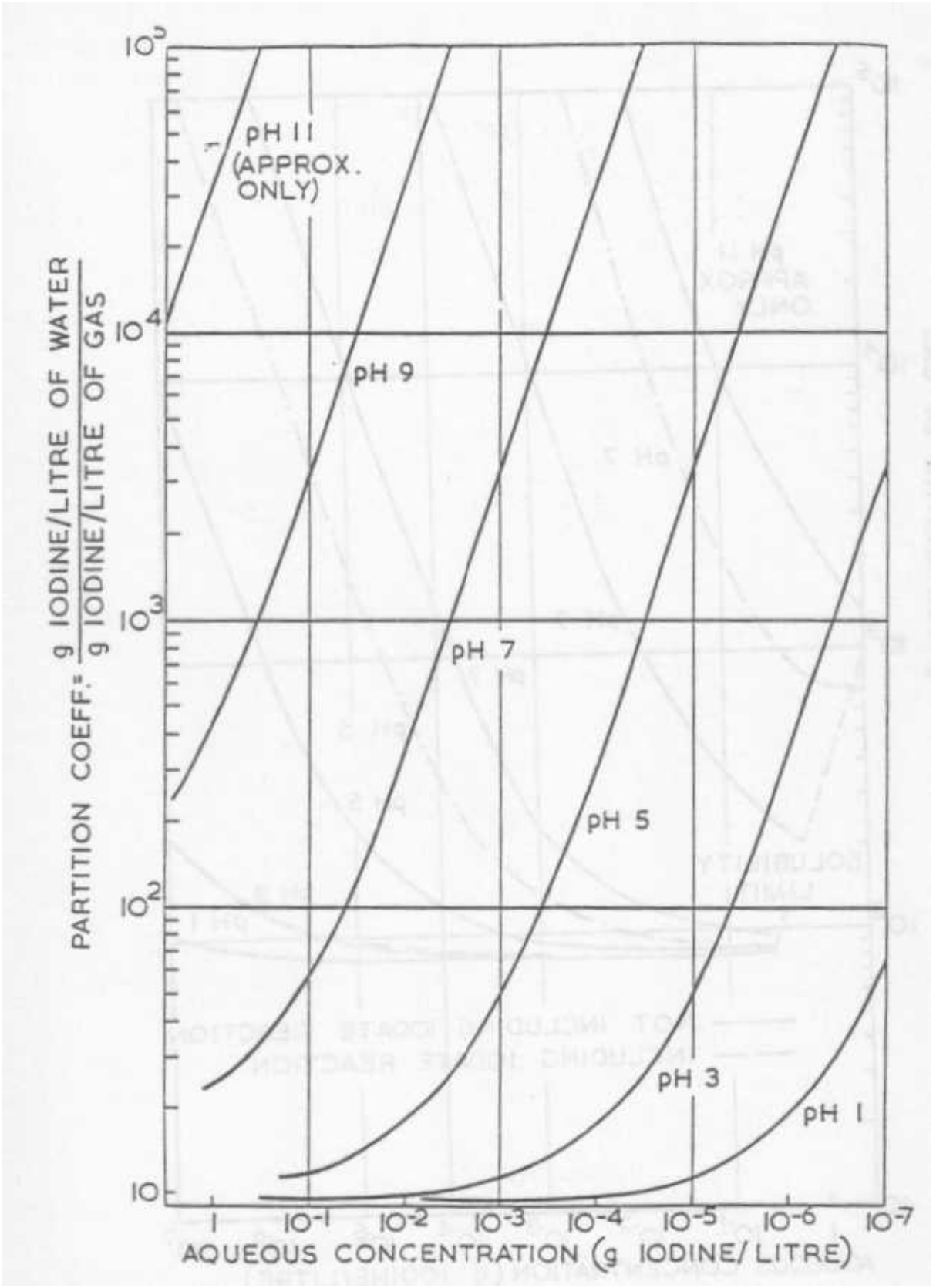

FIG.2. PARTITION OF IODINE BETWEEN GAS PHASE AND WATER AT IOO° C

APPENDIX G

AN ESTIMATE OF AIRBORNE FISSION PRODUCTS IN THE TMI-2 CONTAINMENT ATMOSPHERE FOR POSTULATED FUEL MELTING

BY

R. K. Hilliard
A. K. Postma

October 26, 1979
Washington, D.C.

Attachment to HEDL Letter 7953693

AN ESTIMATE OF AIRBORNE FISSION PRODUCTS IN THE TPII-2 CONTAINMENT ATMOSPHERE FOR POSTULATED FUEL MELTING

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September 20, 1979

D R A F T

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I. INTRODUCTION

Dr. W. R. Stratton, working for the President's Commission on the Accident at Three Mile Island, requested that HEDL compute the fraction of the available fission product (FP) inventory in the fuel which would be airborne in the reactor containment building (RCB) atmosphere given the assumption of fuel meltdown and breach of the pressure vessel. The calculation takes into consideration the various release and removal mechanisms which would be operative under the TMI-2 conditions.

Release mechanisms are assumed to be those discussed in the Reactor Safety [1] for clad failure, fuel melting and vaporization. There would be no significant FP escape to the RCB unless the primary reactor vessel (PRV) pressure boundary was violated by over-pressure or melt-through. The latter is assumed for the purpose of these calculations. The validity of this assumption or of the assumed melt-through time, is not addressed by this document. Removal mechanisms include reaction, plateout, and disposition in the primary coolant and on structural surfaces, and removal by the containment spray and the low pressure high volume injection (LPHVI) system.

The fission products are divided into seven groups according to volatilities, as was done in Reference (1). The noble gases are assumed to be completely released, without significant removal. Halogens are the only other gaseous FP, and these are treated separately from the remaining fission products which must exist as aerosol particles.

The following calculations are believed to be a realistic assessment of the FP behavior given the specified assumptions. The calculations were made by hand and results are presented as tables and graphs of airborne fraction as a function of time.

II. SUMMARY AND CONCLUSIONS

Hand calculations were made of the fraction of FP inventory airborne in the RCB atmosphere for postulated fuel melting in conjunction with the TMI-2 accident. The results are presented as tables and graphs of fraction airborne versus time after start of fuel melting. The chief conclusions are as follows:

(1) A puff release would occur to the RCB of airborne fission products in the primary system atmosphere at the time of assumed primary vessel melt-through.

(2) The RCB spray would remove inorganic halides from the RCB atmosphere with an initial half-time of 1.2 min., organic halides with a half-time of [4 hr., and particulate fission products with a half-time of 80 minutes.

(3) Essentially all the noble gases would be released to the RCB and remain in the atmosphere without significant removal.

(4) A continuous source of particulate fission products would release the materials to the RCB atmosphere at a decreasing rate over a several hour period. Essentially all the halogens would be released during the initial puff.

(5) The halogen airborne concentration would peak at a value of 0.032 of the fuel inventory at the instant of PRV melt-through. This value consists of 0.030 due to inorganic halogens and 0.002 due to organic halides.

(6) The inorganic halogens would quickly become equilibrated between gas and liquid phases and be removed from the gas phase with time as chemical reactions occurred. The inorganic halogen concentration would be decreased by a factor of ten in 100 minutes a factor of 100 in 24 hours.

(7) the organic halide concentration would decrease only slowly with time (14-hr half-time) so that within a few hours, the total airborne halogens would be nearly all organic.

(8) The concentration of particulate fission products would increase with time after the initial puff, due to the continuous source term. The peak concentration would occur 70 minutes after PRV melt-through for all particulate FPs and decrease thereafter until a low equilibrium valve of 10^{-4} of the peak value was reached.

(9) The Cs and Rb group would peak at 1.1 x 10^{-2} and decrease to 1.1 x 10^{-6} within one day.

(10) The Te group would peak at 4.7 x 10^{-2} and decrease to 4.7 x 10^{-6} within one day.

(11) The Sr, Ba group would peak at 6.1 x 10^{-11} and decrease to 6.1 x 10^{-8} within one day.

(12) The Ru group would peak at 2.8 x 10^{-3} and decrease to 2.8 x 10^{-7} within one day.

(13) The La group would peak at 5.7 x 10^{-7} and decrease to 5.7 x 10^{-11} within one day.

III. ASSUMED ACCIDENT CONDITIONS

The TMI-2 plant is equipped with a low leakage containment building. The containment atmosphere would have been scrubbed and cooled by water made basic by sodium hydroxide. Therefore, gaseous and particulate contaminants would be removed from the containment atmosphere, endinq up in the sump water.

Aspects of the containment system which impact the removal of airborne containments are shown schematically in Figure 1.

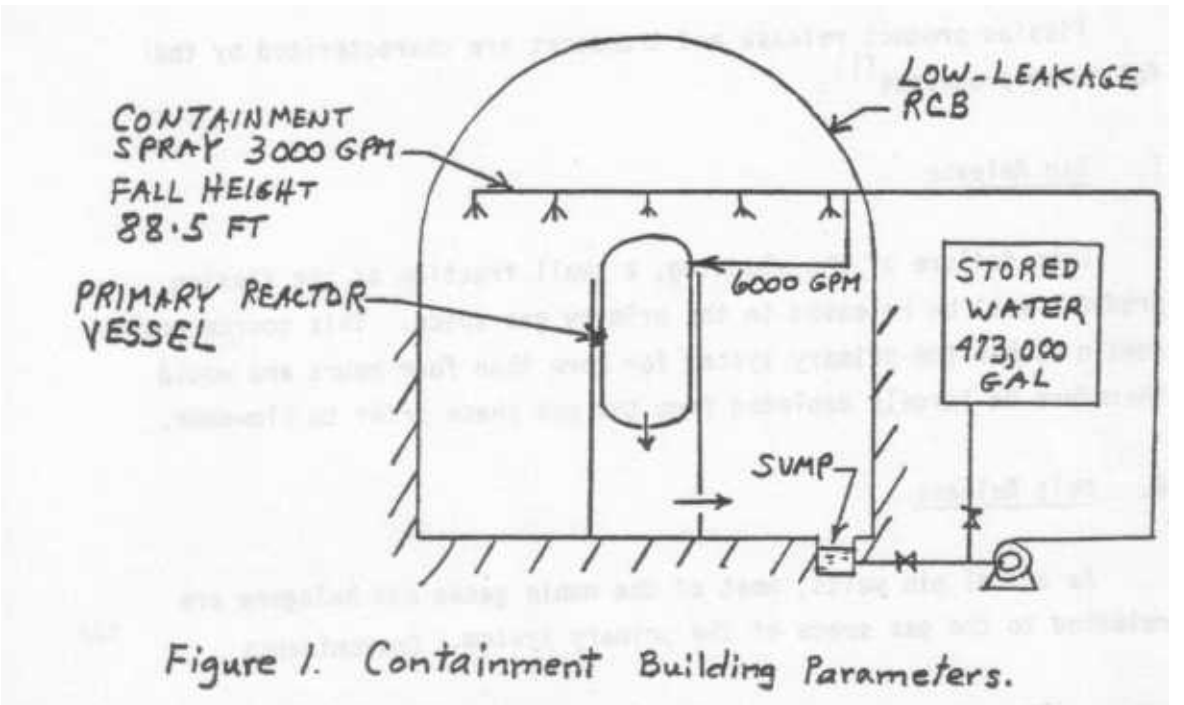

Figure 1. Containment Building Parameters.

Events which affect the transport of contaminants released from the core were postulated on the assumption that core cooling water was not started. The following scenario was formulated from information discussed in Appendix VII of Wash-1400 [1], accounting for specific features and conditions applicable to TMI-2.

Time	Description of Event(sI
0	beginning of fuel melting
0-3 hr	fission products released from fuel, linear with time, following core melting progress
3 hr	fuel mostly molten, resting on grid support plate
3 hr	molten fuel drops onto bottom head of reactor vessel
3-4 hr	molten core melts bottom head of reactor vessel
4 hr	core perforates reactor vessel and falls into cavity
4 hr	primary water blows down into containment vessel through cavity
4 hr	containment sprays and low pressure high volume injection pumps start
>4 hr	vaporization release begins as melt attacks concrete, and sprays and LPHVI continue to operate

Fission product release and transport are characterized by the following processes [l]:

1. Gap Release

Upon failure of the cladding, a small fraction of the fission products will be released to the primary gas space. This source would remain within the primary system for more than four hours and would therefore be largely depleted from the gas phase prior to blowdown.

2. Melt Release

As a fuel pin melts, most of the noble gases and halogens are released to the gas space of the primary system. Contaminants

released by melting would remain in the primary for times of 1 to 4 hours, and except for noble gases, would be largely depleted from the gas phase by deposition and absorption.

3. Vaporization Release

After the molten core falls into the reactor cavity, some additional release will be caused by a sparging process caused by the decomposition of concrete. The vaporization release is approximately exponential in time, having a half-time of approximately 30 minutes [1].

4. Release of Contaminants to Containment Building

Substantially all contaminants will remain within the primary system until the reactor vessel is perforated by the molten core (at 4 hr). When the reactor vessel fails, the primary water will blow-down to the containment building, and carry much of the released fission product mass with it. Most of the fission product mass would remain in suspension in the water, but a fraction of the iodine might be evolved to the gas space according to gas-liquid equilibrium constraints.

5. Washout by Containment Sprays

Contaminants released to the containment atmosphere would be continually scrubbed by water containing sodium hydroxide. Rapid washout would occur early, and then when equilibrium was approached, the washout rate would slow.

6. Organic Iodide Formation

The high temperatures and radiation levels in the core would preclude the existence of appreciable quantities of organic iodides in the primary system. However, when the primary water is blown into the containment building, lower temperatures would allow a small fraction of the halogens to be converted to organic forms, such as CH_3I. In this study the fractional conversion is estimated to be 0.2% of the total halogen released to the RCB in gas and water phases. This is equivalent to 7% conversion of the maximum gas phase halogen release. This estimate, though considered to be realistic, may be too high for TMI-2 because the presence of silver in the core has not been explicitly accounted for. Silver ions would lead to the formation of solid AgI, and such chemically bound iodine (or bromine) would not be susceptible to conversion to organic iodides. A more detailed analysis of organic halide formation might thus lead to a smaller fractional conversion which would in turn lead to lower halogen concentrations in the containment building atmosphere.

IV. AIRBORNE HALOGENS IN THE CONTAINMENT ATMOSPHERE

A. Gap Release

The gap release will have already occurred prior to melting. In order to simplify the analysis, the gap release will be added in with the melt release. From Table 6 of reference (1), 1.7% of the halogens would be released when the cladding failed.

B. Melt Release

During the fuel melting period, iodine will be released with noble gases to the gas phase in the reactor vessel. This iodine will remain in primary system until vessel melt-thru occurs. Therefore, iodine will be in contact with solid surfaces and with water for approximately 4 hr. Release from fuel may be obtained from Table 6 of reference (1) as 88.3% of the inventory. Adding the gap release to this indicates that 90% of the iodine inventory will have escaped from the fuel by the time the core is molten.

Most of the iodine will be absorbed by primary water. Retention of iodine by primary water is estimated using the following assumptions:

(1) No deposition on solid surfaces.

(2) Gas-liquid equilibrium with liquid volume equal to gas volume and an equilibrium coefficient, H, of 10^4 (Reference 1) .

From a mass balance under equilibrium conditions, the fraction of iodine in the gas phase is:

$$\frac{\text{mass of I in gas phase}}{\text{total mass of I}} = \frac{1}{\frac{V_L H}{V_g} + 1} \qquad (1)$$

Where V_L = Volume of liquid, ft^3

V_g = Volume of gas, ft^3

H = gas-liquid equilibrium coefficient.

For the TMI-2 conditions, the fraction of iodine in the gas phase is 10^{-4}

Thus, only 0.01% of the halogen inventory will be airborne when the primary vessel melt-through occurs. When the primary water enters the containment vessel, a new equilibrium will be established between the gas and liquid. Because the containment building gas volume is much larger than that of the primary system, some of the source iodine would be re-evolved into the containment atmosphere.

For the containment building, the equilibrium partitioning of iodine between gas and liquid may be computed from Eq. (1), allowing for time variations in H and V,:

$$\text{Fract. Airborne} = \frac{1}{1 + \frac{V_{L1}H_1}{V_g} + \frac{V_{L2}H_2}{V_g}} \qquad (2)$$

Where V_{L1} = Volume of primary water at 4 hr,

H_1 = H for primary water,

V_{L2} = Volume of fresh water in containment,

H_2 = 2 H for fresh water.

The variation of H with time for a caustic spray solution, taken from reference (1) is shown in Figure 1. H increases with time from 5000 at zero time to a maximum of 10^6 at 7000 minutes.

H_1 was evaluated for an average contact time of 2 hr, and from Figure 2 was given a value of 1.6×10^4. V_{L1} was fixed at 30,000 gal by estimate.

V_{L2} increases with time at a rate of 9000 gal/min until 473,000 gal of emergency water has been injected (52.5 min). Thereafter, V_{L2} was taken as constant at 473,000 gal.

Equilibrium values of airborne iodine, calculated from Eq. (2) are listed in Table I.

TABLE I

EQUILIBRIUM FOR INORGANIC IODINE IN CONTAINMENT

Time, min *	V_{L1}, Ft^3	H_1	V_{L2}, Ft^3	H_2	Fraction Airborne
0	4010	1.6×10^4	0	5.0×10^3	2.96×10^{-2}
20	4010	1.6×10^4	2.41 x10''	5.0×10^3	1.03×10^{-2}
40	4010	1.6×10^4	4.81×10^4	5.0×10^3	**6.2410^{-3}**
52.54	4010	1.6×10^4	6.32×10^4	5.0×10^3	5.00×10^{-3}
100	4010	1.6×10^4	6.32×10^4	9.1×10^3	2.97×10^{-3}
500	4010	4.0×10^4	6.32×10^4	4.0×10^4	7.07×10^{-4}
1000	4010	7.0×10^4	6.32×10^4	7.0×10^4	**4.0410^{-4}**
3000	4010	**2.8×10^5**	6.32×10^4	2.8×10^5	1.01×10^{-4}
5000	4010	2.5×10^5	6.32×10^4	7.5×10^5	3.82×10^{-5}
7000	4010	1.0×10^6	6.32×10^4	1.0×10^6	$2.83\times10^{-}$ '

*Zero time is when primary coolant blowdown occurs.

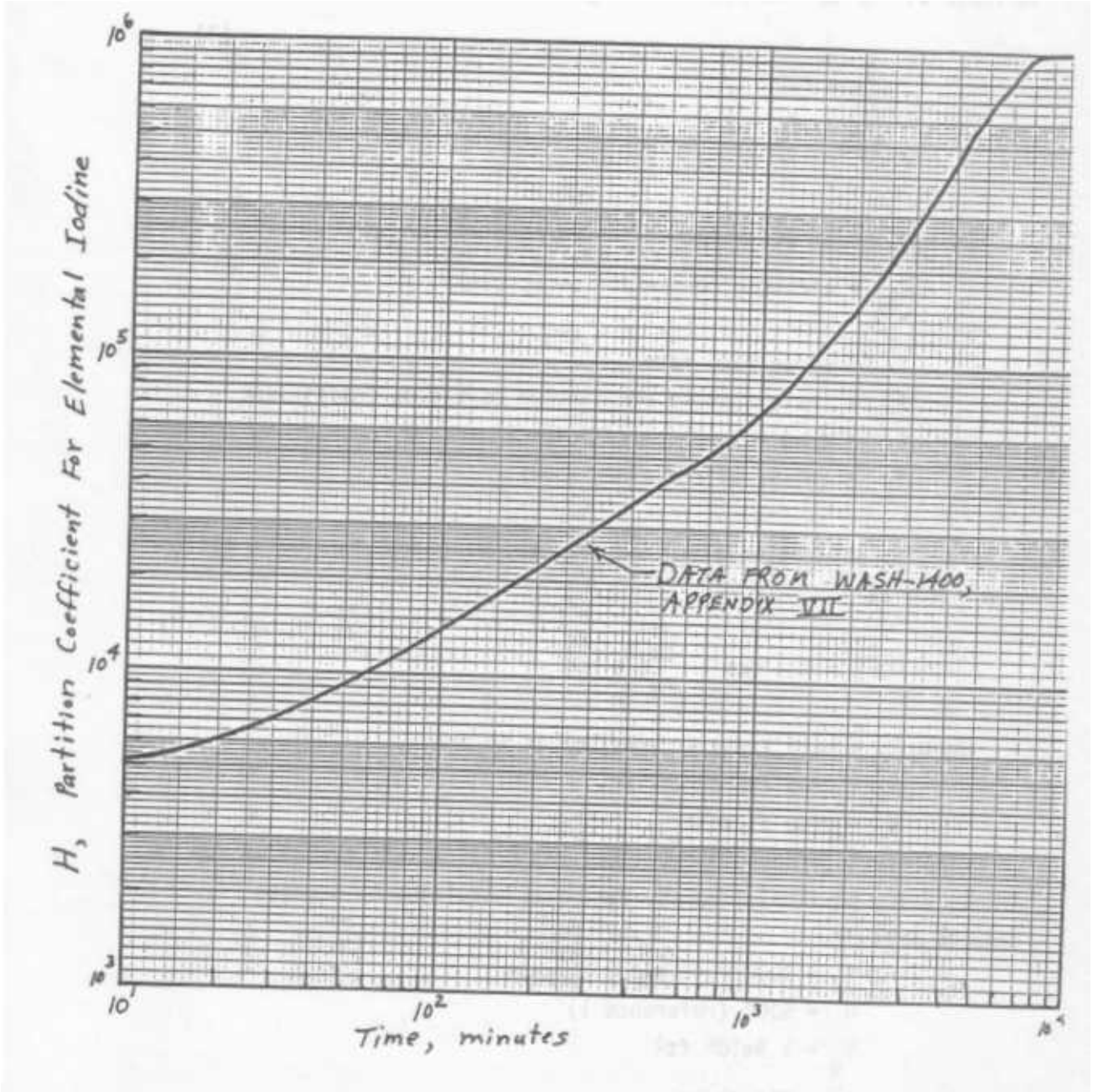

Figure 1. Equilibrium Partition Coefficient For Elemental Halogens as a Function of Time.

Equilibrium between gas and liquid would be attained only if spray washout is fast compared to the airborne removal rate implicit in Table I. Spray washout rate is given by [2]:

$$\frac{C_g}{(C_g)_o} = \frac{FHEt}{V_g} \qquad (3)$$

Where C_g = airborne concentration at time, t,

$(C_g)_o$ = airborne concentration at time o,

F = spray flow rate,

H = instantaneous equilibrium partition coefficient,

E = drop absorption efficiency,

V_g = volume of contained gases.

The drop absorption efficiency, E, may be calculated for a well-mixed drop as [2]:

$$E = 1 - \exp\left[-\frac{6k_g t_e}{d\,H}\right]$$

Where k_g = mass transfer coefficient to drop,

t_o = drop exposure time,

d = drop diameter

Parameters defined in Equations (3) and (4) were evaluated for TMI-2, using physical property estimates tabulated by Knudsen [3]. Results are:

$F = 3000/7.48 = 400\ ft^3/min$

H = 5000 (reference 1)

$V_g = 1.9 \times 10^6\ ft^3$

$k_g = 813$ ft/hr

$t_o = 2.6 \times 10^{-3}$ hr

d = 995 um (surface mean diameter)

When these values are used in Eq. (3), the removal rate, expressed as a washout half-time, is computed to be 1.2 minutes. This is a very fast removal rate compared to that limited by equilibrium, so equilibria constraints will dominate. Therefore, the inorganic iodine concentration will follow the equilibrium concentration history given in Table 1.

c. Vaporization Release

Based on the 30-min half-time for the vaporization release [1], most of the iodine released would enter the containment during the first hour after melt-through. It would be contacted with the LPHVI water and spray, and would be in equilibrium with water. Therefore, the vaporization release can be added to the melt release.

Organic Halide Formation

In reference (1), organic halides were estimated to form to the extent of 0.4% of halogens released to the containment building. This fractional conversion is the sum of that due to non-radiolytic and radiolytic processes. For the accident scenario, less then 3% of the core inventory of halogens is expected to become airborne in the containment atmosphere. This low airborne fraction is in contrast with the scenario envisioned in reference (1) where nearly all of the halogen inventory was postulated to enter the containment atmosphere in a gaseous state. Because radiolysis in the gas phase was the dominant process (accounting of 0.3%±0.2p), the low airborne halogen fraction would minimize organic halide formation due to radiolysis. Therefore, the lower limit estimate for radiolysis given in reference (1), 0.1%, appears to be a realistic figure. Adding the non-radiolytic formation (0.1%), the total estimate for organic halide conversion is 0.2%.

Organic iodides would be depleted slowly by spray washout. Based on containment tests described in references (4) and (5) the removal half-time is estimated to be 24 hr.

The airborne organic halide fraction is shown as a function of time in Table II.

TABLE II

ORGANIC HALIDES AIRBORNE

Time, Min	Fraction of Core Halogen Inventory Airborne in RCB As Organic Halide
240	2.00 (-3) [a]
400	1.85 (-3)
800	1.53 (-3)
1,000	1.38 (-3)
1,500	1.09 (-3)
2,000	0.86 (-3)
4,000	3.28 (-4)
6,000	1.25 (-4)
8,000	4.78 (-5)
10,000	1.83 (-5)
12,000	7.00 (-6)
13,000	4.30 (-6)

(a) 200 (-3) means 2.00×10^{-3}

The total airborne halogen concentration, expressed as a fraction of the core inventory airborne in the whole containment vessel, is summarized in Figure 2 and in Table III.

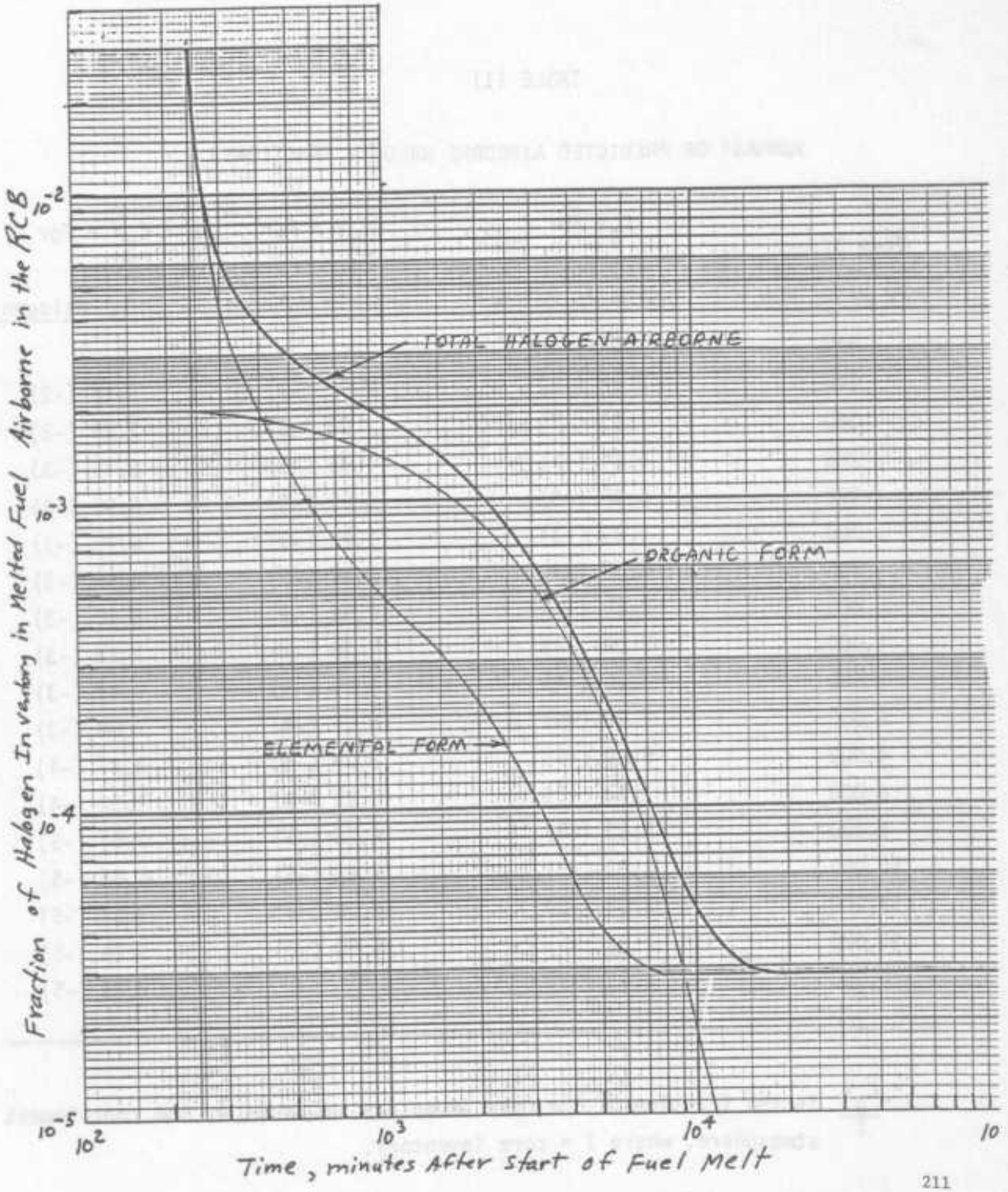

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Figure 2. Halogen Concentration in the RCB Atmosphere.

TABLE III

SUMMARY OF PREDICTED AIRBORNE HALOGEN FRACTIONS

Time After Beginning of Core Melt, Min.	$C_g V$ / I For * Inorganic Halogen	$C_g V$ / I * For Organic Halogen	$C_g V$ / I * For Total Halogen
240	0	0	0
240	**2.96 (-2)**	2.00 (-3)	3.16 (-2)
260	1.03 (-2)	1.98 (-3)	1.23 (-2)
280	**6.24 (-3)**	**1.96 (-3)**	8.19 (-3)
293	5.00 (-3)	1.95 (-3)	**6.96 (-3)**
300	**4.30 (-3)**	1.92 (-3)	**6.22 (-3)**
400	2.00 (-3)	1.85 (-3)	3.85 (-3)
600	9.20 (-4)	1.68 (-3)	2.60 (-3)
1,000	5.00 (-4)	1.38 (-3)	1.88 (-3)
1,500	3.20 (-4)	1.09 (-3)	1.41 (-3)
2,000	**2.20 (-4)**	8.55 (-4)	1.08 (-3)
4,000	5.90 (-5)	3.28 (-4)	3.87 (-4)
6,000	3.30 (-5)	1.25 (-4)	1.58 (-4)
8,000	2.83 (-5)	4.78 (-5)	7.61 (-5)
10,000	2.83 (-5)	1.82 (-5)	4.65 (-5)
12,000	2.83 (-5)	7.00 (-6)	3.53 (-5)
13,000	2.83 (-5)	4.30 (-6)	3.26 (-5)
>20,000	**2.83 (-5)**		2.83 (-5)

* $\frac{C_g V}{I}$ is the fraction of the core inventory suspended in the containment atmosphere, where I = core inventory.

V. AIRBORNE PARTICULATE FISSION PRODUCTS CONTAINMENT ATMOSPHERE

Note: See Table IX for summary of airborne fraction for all particulate fission products.

A. Equations

$$\text{Fraction of core inventory airborne in RCB} = \frac{\left(C_g\right)_T V_g}{I}$$

Where $\left(C_g\right)_T$ = Total airborne concentration, Ci/ft^3

V_g = gas volume, ft^3

I = Core inventory, Ci

Note: Not corrected for radioactive decay.

$$\left(C_g\right)_T = \left(C_g\right)_{puff} + \left(C_g\right)_{cont.} \quad (5)$$

Where $\left(C_g\right)_{puff}$ = concentration resulting from instantaneous release at time of primary vessel melt through.

$\left(C_g\right)_{cont.}$ = concentration resulting from continuous release starting at time of PV melt through.

$$\left(C_g\right)_{puff} = \left[\frac{\left(f_{gap} + f_{melt}\right) f_1 I}{V_g}\right] e^{-\lambda_s t} \quad (6)$$

Where f_{gap} = gap fraction release from fuel, Table 6, Ref. (1)

f_{melt} = meltdown release fraction, Table 6, Ref. (1)

f_1 = fraction of material released from fuel which escapes to RCB as a puff

λ_s = removal rate constant due to washout by spray, min^{-1}

t = time after PV melt through, min

Source term for continuouse release = S

Assume release decreases exponentially with a 30-min. half-time, Table 6, Ref. (1).

$$\therefore S = W_o e^{-\alpha t} \qquad (7)$$

Ref. (6), p. 35

Where W_o = Total release = $(f_{gap})(f_2)(I)$

$$\alpha = \frac{\ln 2}{30 \text{ min}} = \frac{0.693}{30} = 0.0231 \text{ min}^{-1}$$

$$\left(C_g\right)_{cont.} = \left(\frac{f_{vap} \; f_2 \; I}{V_g}\right)\left(\frac{\alpha}{\lambda_s - \alpha}\right)\left[e^{-\alpha t} - e^{-\lambda_s t}\right] \qquad (8)$$

Egn. (4) from Ref. (6)

Where f_{vap} = Vaporization release fraction, Table 6, Ref. (1)

f_2 = fraction of vaporized release which escapes to RCB

α = exponential source term for vaporization release to RCB, min^{-1}

$$\lambda_s = \frac{3 \; F \; h \; E}{2 \; V_g \; d} \qquad (9)$$

Egn. (5) from Ref. (1), p. J-10

Where F = spray flow rate, ft^3/min

h = average spray drop fall height, ft

E = drop collection efficiency

d = mean spray drop size, ft

TMI-2 parameters: (Telecon, Wm. Stratton to RK Hilliard, 9-18-79)

F = 3000 gal/min = 400 ft^3/min

h = 88.5 ft

d = 1080 μm MMD σ_g = 1.5

$\frac{E}{d} = 0.01 \text{ cm}^{-1} = 0.305 \text{ ft}^{-1}$
(taken from Ref. (2), p. 136

$$\therefore \lambda_s = \frac{3\,(400)\,(88.5)}{(2)\,(1.9 \times 10^5)}\,(0.305) = \underline{8.52 \times 10^{-3} \text{ min}^{-1}}$$

spray removal half-time = $\frac{0.693}{\lambda} = \underline{81 \text{ min}}$

B. ASSUMPTIONS

1. All FPs except noble gases and halogens vaporize, nucleate and condense as small aerosol particles in the primary vessel atmosphere during the melting process.

2. All aerosol particles co-agglomerate.

3. Plateout and deposition occurs on the primary vessel, piping, pressurizer, steam generator surfaces and in the coolant. Turbulance, thermophoretic, diffusio phoretic,Brownian diffusion and gravity effects all contribute to rapid removal from the primary atmosphere. Since there is no steam or gas sweep out of the PV until vessel meltdown occurs, a large attenuation can be expected. It is assumed that $f_1 = 10^{-3}$.

4. At the time of PV melt through, the FP particles remaining airborne are swept rapidly to the RCB and mix instantly to a uniform concentration.

5. Releases from the fuel are based on Ref. (1), Table 6.

6. The RCB leakage is low (negligible removal rate).

7. The FP materials released by vaporization (see Ref. (1), Table 6) provides a continuous source term to the RCB. However, attenuation mechanisms continue, as in assumption (3). f2= 10^{-1}.

8. A lower limit to the airborne fraction suspended in the RCB atmosphere occurs, due to mechanical resuspension from the sprays. The lower equilibrium concentration is 0.01% of the peak concentration.

C. CALCULATIONS - Cs and Rb

$f_{gap} = 0.05$	Ref. (1), Table 6
$f_{melt} = 0.76$	Ref. (1), Table 6
$f_1 = 1 \times 10^{-3}$	Assumed
$f_2 = 0.1$	Assumed
$f_{vap.} = 0.19$	Ref (1), Table 6

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Equation (6) becomes: $\frac{(C_g)_{puff} V_g}{} = (0.05 + 0.76)(10^{-3})\, e^{-0.00852\,t} = (8.1 \times 10^{-4}) e^{-0.00852\,t}$

Eqn. (8) becomes:

$$\frac{\left(C_g\right)_{cont}}{I} V_g = (0.19)\,(0.1)\left[\frac{0.0231}{0.00852-0.0231}\right]\left[e^{-0.0231t} - e^{-0.00852t}\right]$$

$$= -3.01 \times 10^{-2}\left[e^{-0.0231t} - e^{-0.00852t}\right]$$

These two equations are solved for various times, added to obtain the total airborne fraction, and tabulated in Table IV. The data are plotted in Figure 3.

D. CALCULATIONS - Te, Se, Sb

$f_{gap} = 10^{-4}$	Ref. (1), Table 6
$f_{melt} = 0.15$	Ref. (1), Table 6
$f_1 = 10^{-3}$	Assumed, this work
$f_2 = 10^{-1}$	Assumed, this work
$f_{vap} = 0.85$	Ref (1), Table 6

Eqn. (6) becomes:

$$\frac{\left(C_g\right)_{puff}}{I} V_g = (10^{-4} + 0.15)\,(10^{-3})\, e^{-0.00852t}$$

$$= 1.5 \times 10^{-4}\ e^{-0.00852\,t}$$

Eqn. (8) becomes:

$$\frac{\left(C_g\right)_{cont}}{I} V_g = (0.85)\,(0.1)\ \frac{0.0231}{0.00852-0.0231}\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

$$= 0.1347\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

See Table V and Figure 4.

E. CALCULATIONS - Sr and Ba

$f_{gap} = 10^{-6}$	Ref. (1), Table 6
$f_{melt} = 10^{-1}$	Ref. (1), Table 6
$f_{vapor} = 10^{-2}$	Ref. (1), Table 6
$f_1 = 10^{-3}$	Assumed, this work
$f_2 = 10^{-1}$	Assumed, this work

TABLE IV

AIRBORNE FRACTION - Cs & Rb

Minutes after start fuel melt	Minutes after Primary blow-down	a-xt	a-at	$\int_I C_g \, V_g$ .uff	$(C_l V_g)$ Cont.	$\sim C_g V_g$ * (a)
240	0	1.0 (b)	1.0	8.10 (.4)	0	8.10 (4)
245	5	9.58 (1)	8.91 (1)	7.76 (11)	2.03 (3)	2.80 (-3)
250	10	9.18 (1)	7.94 (1)	7.44 E4)	3.90 (3)	4.64 (3)
255	15	8.80 (i)	7.07 (1)	7.13 (~4)	5.20 (3)	5.91 (.3)
260	20	8.43 *Fi)*	6.30 (1)	6.83 E4)	6.42 (3)	7.10 (-3)
270	30	7.74 (1)	5.00 (1)	6.27 E4)	8.26 *O)*	'8.89 (-3)
280	40	7.11 *FI)*	3.97 (1)	5.76 (4)	9.46 (-3)	1.00 (-2)
290	50	6.53 (1)	3.15 (1)	5.29 *E4)*	1.02 (-2)	1.07 (-2)
310	70	5.51 (1)	1.98 F1}	4.46 E4)	1.06 (-2)	1.11 F2)
340	100	4.27 (1)	9.93 E2)	3.46 (-4)	9.85 (3)	1.02 (-2)
390	150	2.79 (1)	3.13 E2)	2.26 E4)	7.44 *O)*	7.67 F3)
440	200	1.82 (1)	9.85 (3)	1.47 (4)	5.18 (3)	5.33 (3)
540	300	7.76 ¢2)	9.78 F4)	6.29 (S)	2.31 F3)	2.37 (3)
640	400	3.31 (2)	9.71 E5)	2.68 F5)	9.94 *O)*	1.02 (.3)
840	600	6.02 (3)	9.56 (-7)	5.06 *E6)*	1.81 (4)	1.86 E4)
1040	800	1.10 (3)	9.42 (9)	8.88 F7)	3.30 F5)	3.39 (5)
1240	1000	1.99 F4)	9.29 (11)	1.62 F7)	6.00 E6)	6.17 (.6)
1440	1200	3.63 (5)	9.15 (-13)	2.94 F8)	1.09 (.6)	1.12 (-6)
1640	1400	6.60 (6)	9.01 (15)	5.35 E9)	1.99 F7)	2.04 (7)

(a) $\frac{*(C_g)_F V_g}{I}$ = Fraction of FP inventory in the melted fuel which is airborne in the RCB.

(b) 9.58 (-1) means 9.58×10^{-1}

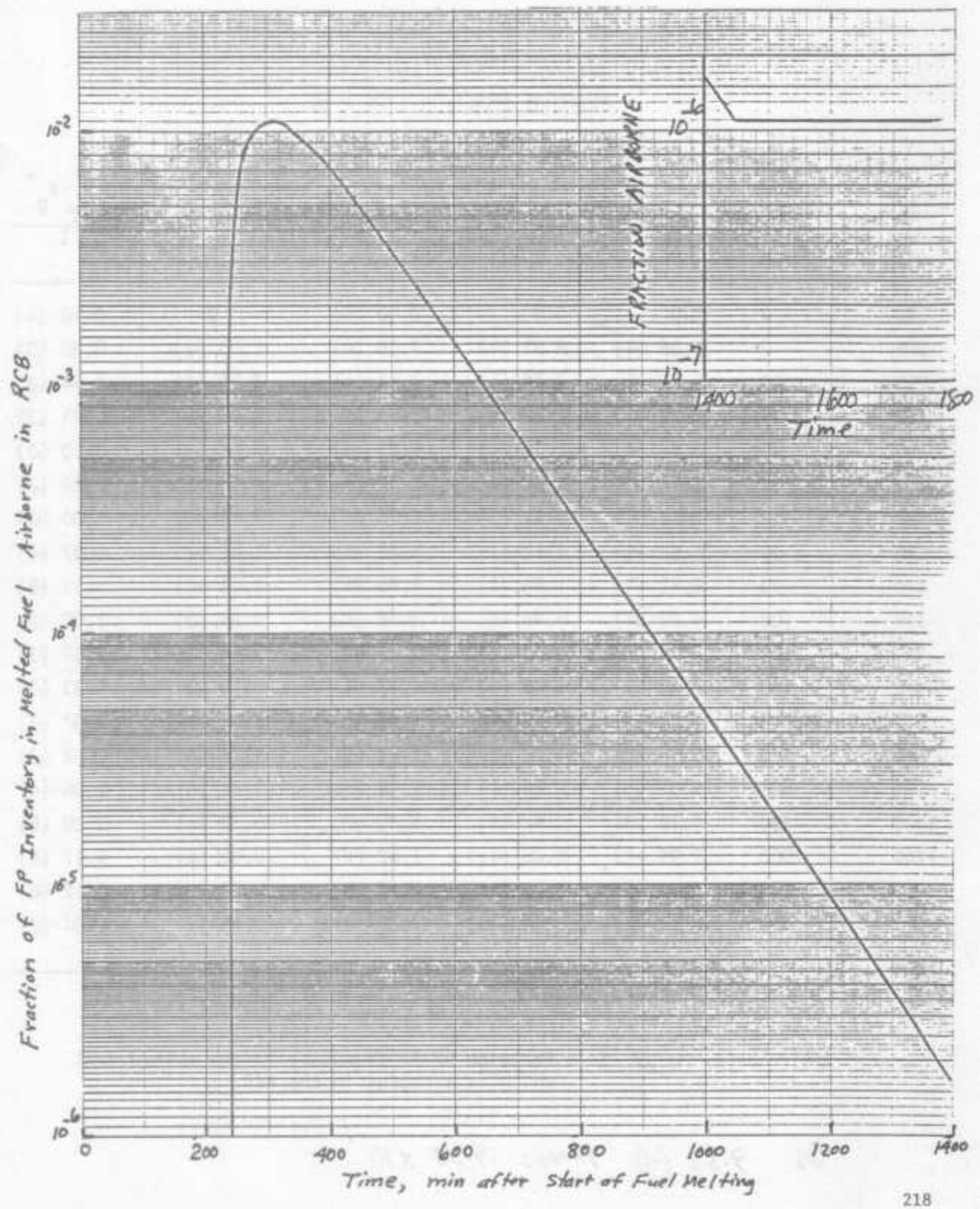

Figure 3. Airborne Cs and Rb.

TABLE V - Te, Se, Sb

Minutes after start fuel melt	t minutes after PV melt	$-a_s t$ a	$-at$ a	1C9) Duff V 9	(9) cont. V9 I	$C_9 ?V_9$) I
240	0	1.0	1.0	1.50 (4)	0	1.50 (4)
250	10	0.92	0.79	1.38 (4)	1.68 F2)	1.69 F2)
260	20	0.84	0.63	1.27 E4)	2.87 F2)	2.89 (2)
270	30	0.77	0.50	1.16 F4)	3:70 E2)	3.71 (-2)
280	40	0.71	0.40	1.07 E4)	4.23 (2)	4.24 (2)
300	60	0.60	0.25	9.00 (.5)	4.71 (2)	4.70 F2)
320	80	0.51	0.16	7.59 F5)	4.69 (-2)	*4.70 E2)*
340	100	**0.43**	9.93 E2)	6.40 (.5)	4.41 F2)	4.42 (2)
360	120	0.36	6.25 (.2)	5.40 E5)	4.00 (2)	4.01 F2)
400	160	2.56 (1)	2.48 (2)	3.84 E5)	3.11 (2)	3.12 *R)*
500	260	1.09 F1)	**2.46** F3)	1.64 E5)	1.44 (2)	1.44 (.2)
600	360	4.66 (,2)	2.45 (4)	6.98 E6)	6.24 (-3)	6.24 (.3)
800	560	8.47 (.3)	2.41 (6)	1.27 (.6)	1.14 (.3)	1.14 (3)
1000	760	1.54 0)	2.37 (-8)	2.32 (-7)	2.08 F4)	2.08 F4)
1200	960	2.80 E4)	2.34 (-10)	4.21 (.8)	3.78 (-5)	3.78 (5)
1400	1160	5.10 (.5)	2.31 (-12)	7.65 E9)	6.87 0)	6.88 (.6)
1500	1260	2.18 E5)	2.29 (.13)		2.93 ('6)	2.94 (6)

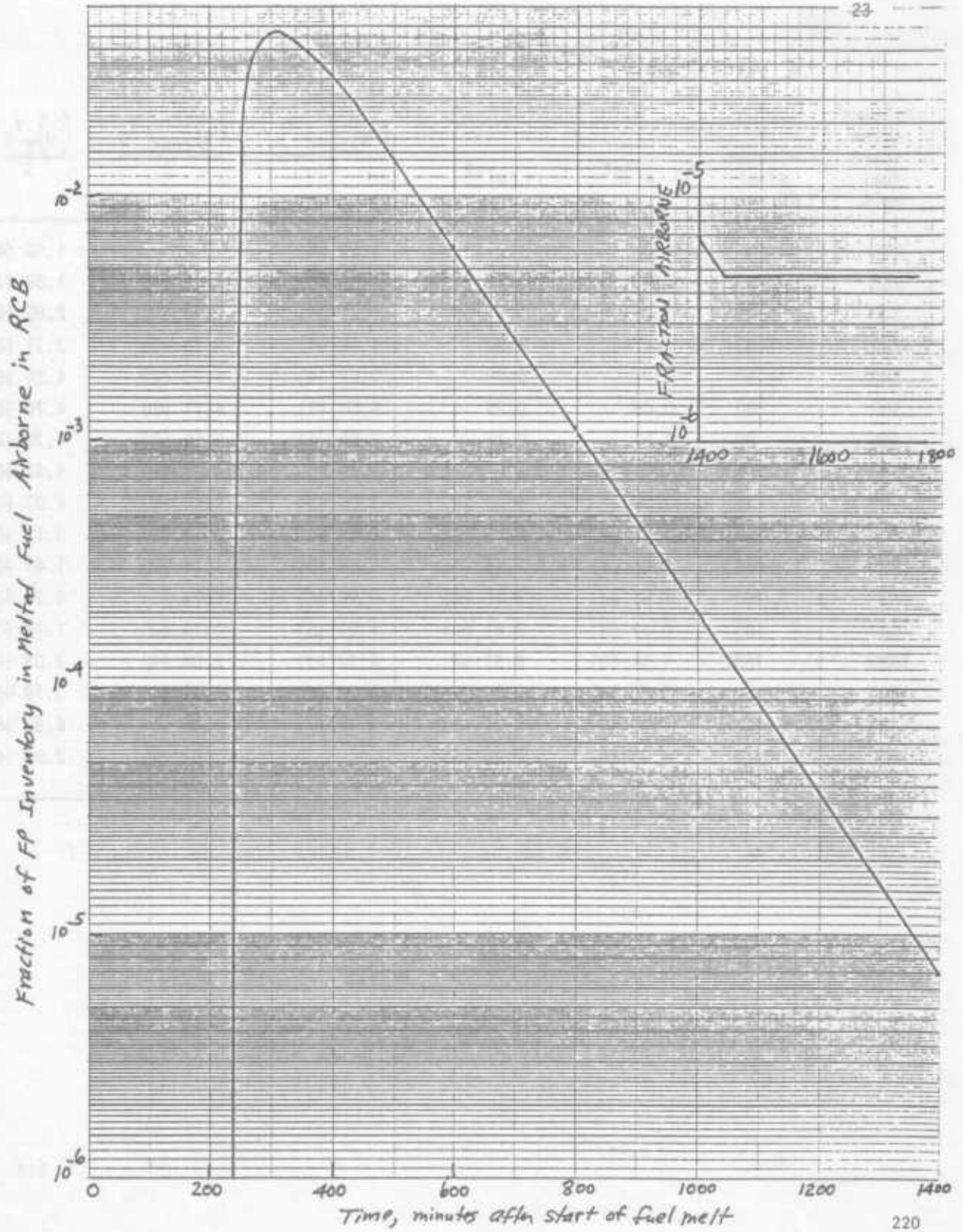

Figure 4. Airborne Te, Se, Sb.

TABLE VI

AIRBORNE - Sr & Ba

Minutes after start fuel melt	t, minutes after PV melt	$e^{-\lambda_s t}$	$e^{-\alpha t}$	$\frac{(C_g)_{puff} V_g}{I}$	$\frac{(C_g)_{cont} V_g}{I}$	$\frac{(C_g)_T V_g}{I}$
240	0	1.0	1.0	1.00 (-4)	0	1.00 (-4)
250	10	0.92	0.79	9.18 (-5)	1.37	2.89 (-4)
260	20	0.84	0.63	8.43 (-5)	3.37	4.21 (-4)
270	30	0.77	0.50	7.74 (-5)	4.34	5.11 (-4)
280	40	0.71	0.40	7.11 (-5)	4.96	5.67 (-4)
300	60	0.60	0.25	6.00 (-5)	5.52	6.12 (-4)
320	80	0.51	0.16	5.06 (-5)	5.50	6.01 (-4)
340	100	0.43	9.93 (-2)	4.26 (-5)	5.17	5.60 (-4)
360	120	0.36	6.25 (-2)	3.60 (-5)	4.70	5.06 (-4)
400	160	0.26	2.48 (-2)	2.56 (-5)	3.65	3.91 (-4)
500	260	0.11	2.46 (-3)	1.09 (-5)	1.69	1.79 (-4)
600	360	4.66 (-2)	2.45 (-4)	4.65 (-6)	7.32	7.78 (-6)
800	560	8.47 (-3)	2.41 (-6)	8.47 (-7)	1.34	1.42 (-5)
1000	760	1.54 (-3)	2.37 (-8)	1.54 (-7)	2.43	2.59 (-6)
1200	960	2.80 (-4)	2.34 (-10)	2.80 (-8)	4.43	4.71 (-7)
1400	1160	5.10 (-5)	2.31 (-12)	5.10 (-9)	8.06	8.57 (-8)
1500	1260	2.18 (-5)	2.29 (-13)	2.18 (-9)	3.44	3.66 (-8)

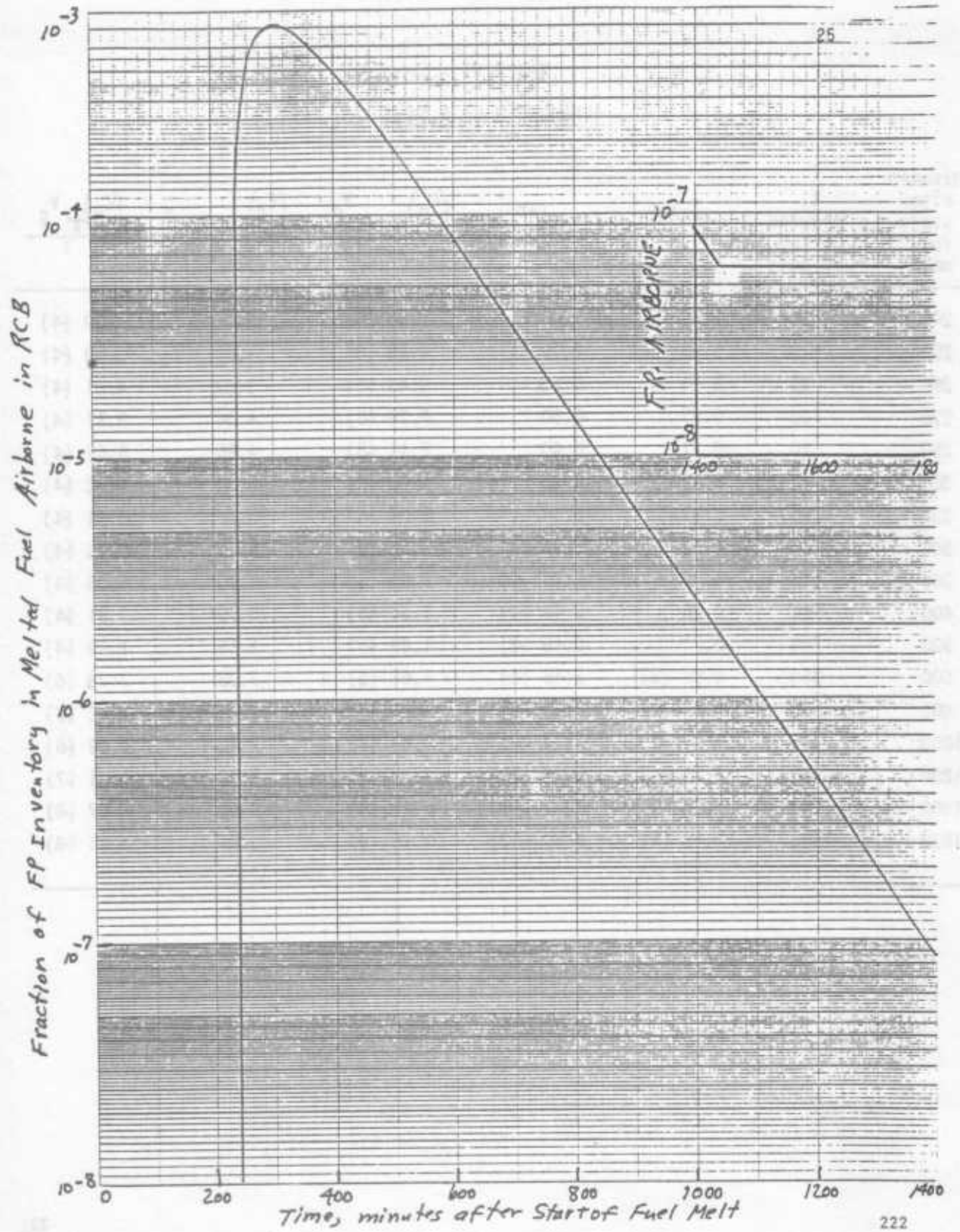

Figure 5. Airborne Sr, Ba.

Eqn. (6) becomes:

$$\frac{(C_g)_{puff}\, V_g}{I} = (10^{-6} + 10^{-1})\,(10^{-3})\; e^{-0.00852t}$$

$$= 1 \times 10^{-4}\; e^{-0.00852t}$$

Eqn. (8) becomes:

$$\frac{(C_g)_{cont.}\, V_g}{I} = (10^{-2})\,(0.1)\left(\frac{0.0231}{0.00852-0.0231}\right)\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

$$= 1.58 \times 10^{-3}\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

See Table VI and Figure 5.

F. CALCULATIONS - Ru, Mo, Pd, Rh, Tc

$f_{gap} = 0$	Reference (1)
$f_{melt} = 0.03$	Reference (1)
$f_{vap.} = 0.05$	Reference (1)
$f_1 = 10^{-3}$	Assumed, this work
$f_2 = 10^{-1}$	Assumed, this work

Eqn. (6) becomes:

$$\frac{(C_g)_{puff}\, V_g}{I} = (0.03)\,(10^{-3})\; e^{-\lambda_s t}$$

$$= 3 \times 10^{-5}\; e^{-0.00852t}$$

Eqn. (8) becomes:

$$\frac{(C_g)_{cont.}\, V_g}{I} = (0.05)\,(10^{-1})\left(\frac{0.0231}{0.00852-0.0231}\right)\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

$$= 7.92 \times 10^{-3}\left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

See Table VII and Figure 6.

TABLE VII

AIRBORNE Ru, Mo, Pd, Rh, Te

Minutes after start fuel melt	t Minutes	$e^{-\lambda_s t}$	e^{-at}	$\frac{(C_g)_{puff} V_g}{I}$	$\frac{(C_g)_{cont.} V_g}{I}$	$\frac{(C_g)_t V_g}{I}$
240	0	1.0	1.0	3.00 (-5)	0	3.00 (-5)
250	10			2.75 (-5)	9.87 (-4)	1.01 (-3)
260	20			2.53 (-5)	1.69 (-3)	1.72 (-3)
270	30			2.32 (-5)	2.17 (-3)	2.20 (-3)
280	40			2.13 (-5)	2.49 (-3)	2.51 (-3)
300	60			1.80 (-5)	2.77 (-3)	2.79 (-3)
320	80			1.52 (-5)	2.76 (-3)	2.77 (-3)
340	100			1.28 (-5)	2.59 (-3)	2.60 (-3)
360	120			1.08 (-5)	2.37 (-3)	2.38 (-3)
400	160			7.68 (-6)	1.83 (-3)	1.84 (-3)
500	260			3.27 (-6)	8.45 (-4)	8.48 (-4)
600	360				3.67 (-4)	3.68 (-4)
800	560				6.71 (-5)	6.74 (-5)
1000	760				1.22 (-5)	1.23 (-5)
1200	960				2.22 (-6)	2.23 (-6)
1400	1160				4.04 (-7)	4.05 (-7)
1500	1260				1.72 (-7)	1.73 (-7)

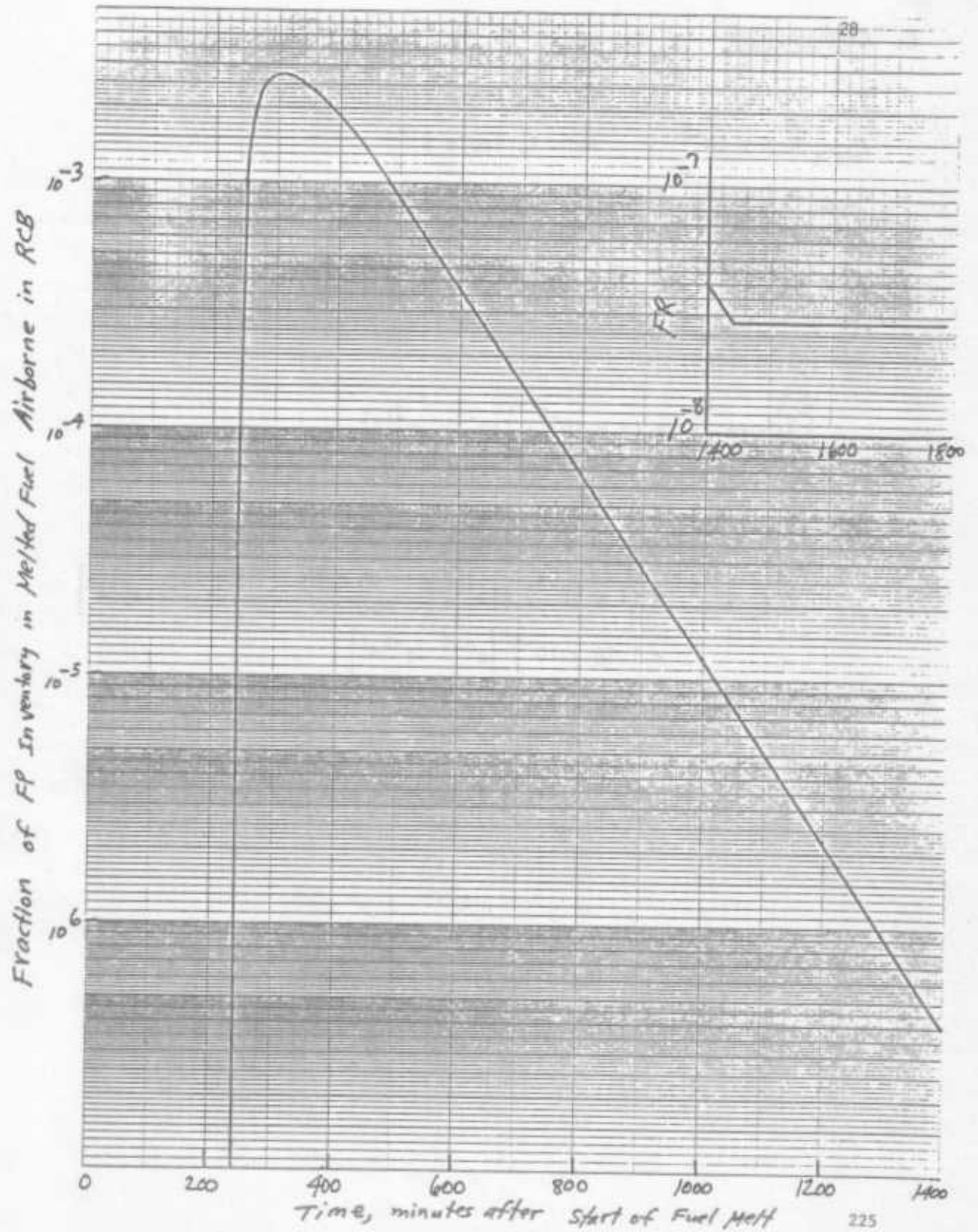

Figure 6. Airborne Ru, Mo, Pd, Rh, Tc

G. CALCULATIONS - La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

$f_{gap} = 0$ Ref. 1, Table 6

$f_{melt} = 3 \times 10^{-3}$ Ref. 1, Table 6

$f_{vap.} = 1 \times 10^{-3}$ Ref. 1, Table 6

$f_1 = 10^{-3}$ Assumed, this work

$f_2 = 10^{-1}$ Assumed, this work

Eqn. (6) becomes:

$$\frac{(C_g)_{puff} V_g}{I} = (3 \times 10^{-3})(10^{-3}) e^{-\lambda_s t} = 3 \times 10^{-6} e^{-\lambda_s t}$$

Eqn. (8) becomes:

$$\frac{(C_g)_{cont.} V_g}{I} = (1 \times 10^{-3})(10^{-1}) \left(\frac{0.0231}{0.00852-0.0231}\right) \left[e^{-\alpha t} - e^{-\lambda_s t}\right] = 1.58 \times 10^{-4} \left[e^{-\alpha t} - e^{-\lambda_s t}\right]$$

See Table VIII and Figure 7.

TABLE VIII

AIRBORNE - La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

Minutes after start fuel melt	t Minutes	$e^{-\lambda t}$	$e^{-\alpha t}$	$\left(C_g\right)_{puff} V_g / I$	$\left(C_g\right)_{cont.} V_g / I$	$\left(C_g\right)_T V_g / I$
240	0	1.0	1.0	3×10^{-6}	0	3.00 (-6)
250	10			2.75 (-6)	1.97 (-5)	2.24 (-5)
260	20			2.53 (-6)	3.37 (-5)	3.62 (-5)
270	30			2.32 (-6)	4.33 (-5)	4.57 (-5)
280	40			2.13 (-6)	4.96 (-5)	5.18 (-5)
300	60			1.80 (-6)	5.53 (-5)	5.71 (-5)
320	80			1.52 (-6)	5.50 (-5)	5.65 (-5)
340	100			1.28 (-6)	5.17 (-5)	5.30 (-5)
360	120			1.08 (-6)	4.70 (-5)	4.80 (-5)
400	160			7.68 (-7)	3.65 (-5)	3.73 (-5)
500	260			3.27 (-7)	1.69 (-5)	1.72 (-5)
600	360				7.32 (-6)	7.42 (-6)
800	560			2.54 (-8)	1.34 (-6)	1.36 (-6)
1000	760				2.44 (-7)	2.48 (-7)
1200	960				4.43 (-8)	4.45 (-8)
1400	1160				8.06 (-9)	8.10 (-9)
1500	1260				3.44 (-9)	3.48 (-9)

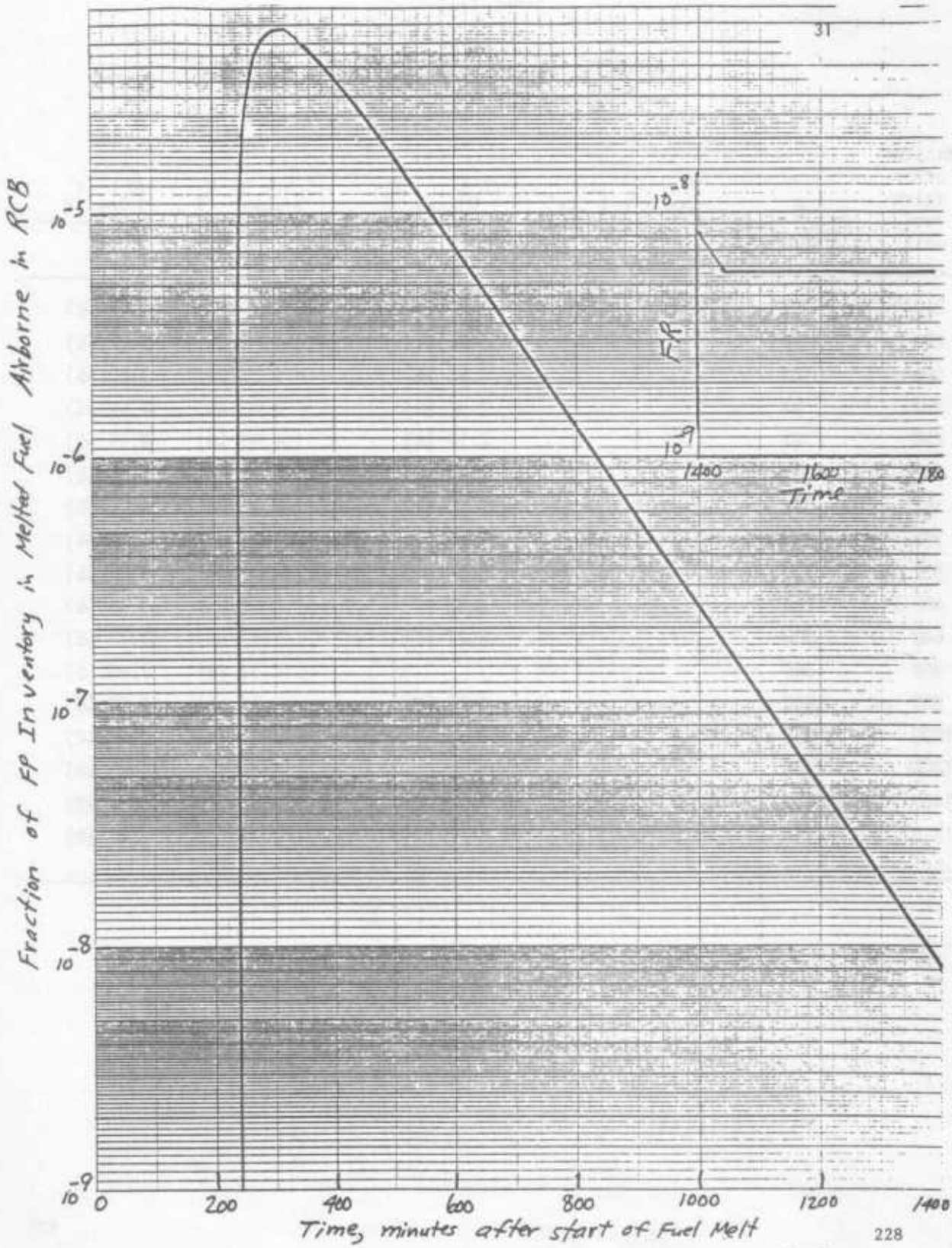

Figure 7. Airborne La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

TABLE IX

SUMMARY - PARTICULATE FISSION PRODUCTS -

AIRBORNE FRACTION OF CORE INVENTORY IN RCB

Fraction of FP Inventory in Melted Fuel Airborne in RCB

Min. After Start of Fuel Melt	Cs, Rb	Te, Se, Sb	Sr, Ba	Ru, Mo, Pd, Rh, Tc	La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb
0	0	0	0	0	0
239	0	0	0	0	0
240	8.1 (4)	1.5 (-4)	1.0 E4)	3.0 F5)	3.0 (6)
250	4.6 (.3)	1.7 F2)	2.9 (4)	1.0 (-3)	2.2 E5)
260	7.1 (-3)	2.9 (-2)	4.2 (4)	1.7 F3)	3.6 (-5)
280	1.0 (-2)	4.2 F2)	5.7 (4)	2.5 E3)	5.2 F5)
300	1.1 E2)	4.7 (.2)	6.1 (.4)	2.8 O)	5.7 O)
320	1.1 (-2)	4.7 (2)	6.0 O)	2.8 F3)	5.7 (-5)
340	1.0 E2)	4.4 (-2)	5.6 (~4)	2.6 (.3)	5.3 (.5)
360	9.0 (,3)	4.0 (2)	5.1 F4)	2.4 F3)	4.8 (-5)
400	7.2 (-3)	3.1 (.2)	3.9 (4)	1.8 F3)	3.7 (-5)
500	3.6 (.3)	1.4 ('2)	1.8 E4)	8.5 (4)	1.7 (5)
exponential d&creasc					
1200	8.7 (6)	3.8 F5)	4.7 F7)	2.2 (6)	4.5 E8)
Lower Limit (-.1440)	1.1 (-6)	4.7 E6)	6.1 F8)	2.8 (2)	5.7 ('9)

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Staff Reports To

THE PRESIDENT'S COMMISSION ON
THE ACCIDENT AT
THREE MILE ISLAND

The Nuclear Regulatory Commission, Report of the Office of Chief Counsel

The Role of the Managing Utility and Its Suppliers, Report of the Office of Chief Counsel

Emergency Preparedness, Emergency Response, Reports of the Office of Chief Counsel

Reports of the Technical Assessment Task Force, Vol. I
- "Technical Staff Analysis Reports Summary"
- "Summary Sequence of Events"

Reports of the Technical Assessment Task Force, Vol. II
- "Chemistry"
- "Thermal Hydraulics"
- "Core Damage"
- "WASH 1400 -- Reactor Safety Study"
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Report of the Emergency Preparedness and Response Task Force

Report of the Public's Right to Information Task Force