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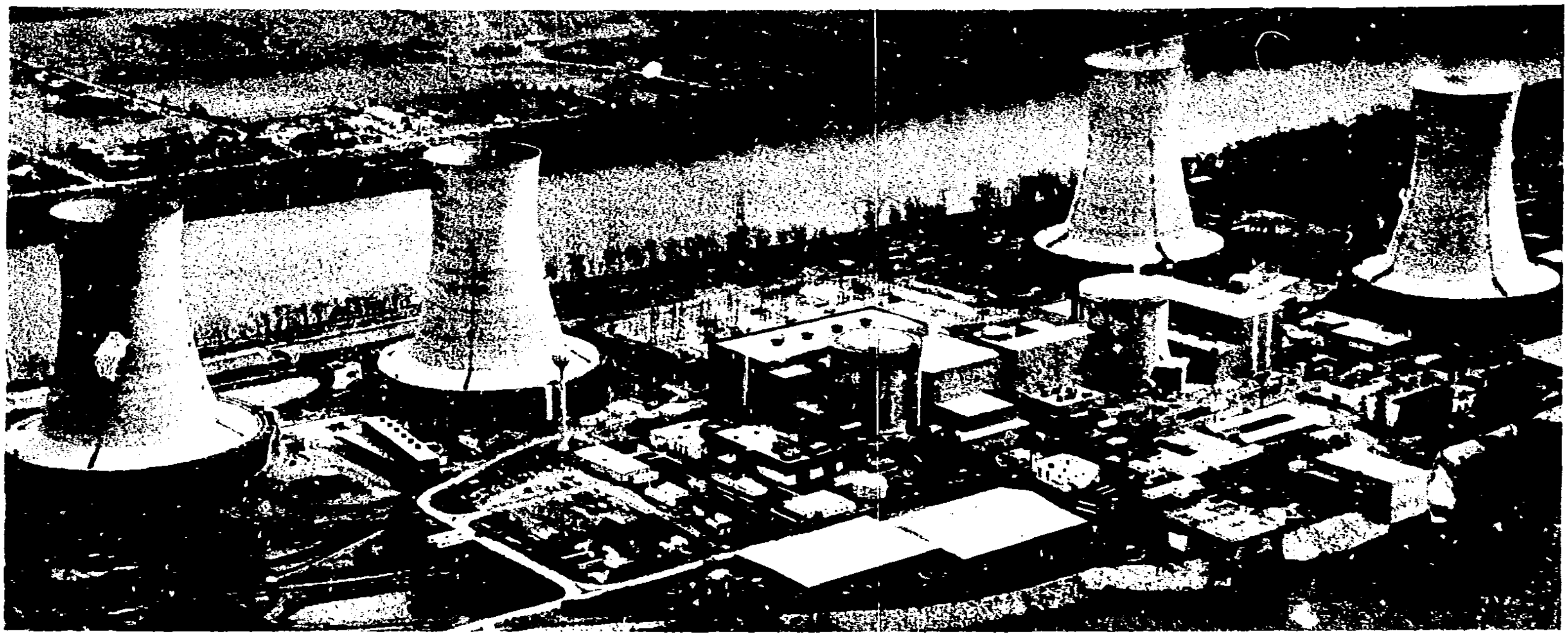

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Three Mile Island Unit-2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling

Douglas W. Croucher

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THREE MILE ISLAND UNIT-2 CORE STATUS SUMMARY: A BASIS FOR TOOL DEVELOPMENT FOR REACTOR DISASSEMBLY AND DEFUELING

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DISCLAIMER

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SUMMARY

The accident at Three Mile Island Unit-2 (TMI-2) on March 28, 1979 caused extensive damage to the core. A variety of analyses were performed using three general approaches to determine the extent of core damage. First, thermal-hydraulic events were reconstructed using available data, thermal-hydraulic principles, and computer analyses. Second, determinations of the hydrogen generated yielded estimates of the amount of zircaloy oxidized and embrittled. Third, the type and quantity of fission products released during the accident were used to estimate the location of core damage and the fuel temperatures which were achieved. Uncertainties exist in each type of determination due to the equivocal nature of the data. Thus, the purpose of this paper is to review and summarize the core damage assessments which have been made, identify the minimum and maximum bounds of damage, and establish a "reference" description for the current status of the core. The different degrees of damage present in the reference core will be considered during development of contingency tooling and procedures for inspection, sample acquisition, and defueling of the core.

From reconstruction of the thermal-hydraulic events, it was concluded that the core remained covered up to 100 minutes into the accident and that most of the damage occurred during the period from 100 to 210 minutes when the core is thought to have been uncovered. Damage to the core is a strong function of the time-dependent steam-water mixture level in the core which was greatly affected by the net makeup flow during this period. Cladding reached temperatures between 1030 K (1395°F) and 1150 K (1610°F) and failed by ballooning between 137 and 142 minutes. The cladding continued to increase in temperature, becoming oxidized and embrittled. The fuel reached peak temperatures varying between 2000 K (3140°F) and 2900 K (4760°F). At temperatures greater than 2175 K (3455°F), uranium dioxide fuel pellets in contact with molten cladding could have been dissolved by the zircaloy, forming a liquid phase of zirconium-uranium-oxygen termed "liquified fuel." The temperatures achieved are also high enough to melt

Inconel spacer grids, stainless steel core components and the silver-indium-cadmium poison material in the control rods. Estimated temperatures of the upper plenum structure range between 1500 K (2240°F) and 1800 K (2780°F), high enough to suggest that melting of control rod guide tube brazements and stainless steel components may have occurred. At 174 minutes, a sudden influx of water to the core is expected to have rapidly quenched the embrittled cladding and hot fuel, fracturing the fuel rods and forming a debris bed in the upper region of the core. Additional core damage, probably core slumping and densification of the debris bed, apparently occurred at about 225 minutes. Postaccident core flow resistance measurements indicated an effective core flow area blockage of approximately 90%. Since the peripheral fuel assemblies make up more than 20% of the core flow area, some blockage is expected at the periphery of the core.

Analyses of the hydrogen produced yielded estimates of the amount of zircaloy oxidized. Although oxidation of fuel and other core components could also generate hydrogen, the expected amount is small compared to that from oxidation of zircaloy. Estimates were based on the hydrogen and oxygen content of several postaccident containment building air samples, the pressure increase in the containment building signifying the amount of hydrogen burned, and estimates of the hydrogen present in the primary system, both free and in solution. About 50% of the zircaloy in the active core region is estimated to have been oxidized.

The assessment of core damage based on analyses of fission product release is not as precise as the first two methods, but it generally is confirmatory. Measured isotopic ratios of uranium and plutonium in the reactor building sump indicate that the central region of core, and perhaps the whole core, was uniformly damaged. The estimates of temperature based on fission products found in the coolant vary greatly, from gross core average temperatures below 2000 K (3140°F) to about 40% of the fuel greater than 2675 K (4355°F). There is general agreement, however, that the fuel remained below the melting point of UO_2 because very little strontium, tellurium, and ruthenium were present.

Factors of primary interest during reactor disassembly and removal of the core are the condition of the upper plenum, the amount of cladding oxidized, the presence of once molten materials such as liquified fuel and control rods, and the condition of the instrument and guide tubes. Some components of the upper plenum structure may have melted or fused together during the course of the accident necessitating the development of tooling and procedures for this contingency. It is evident that a bed of fragmented fuel and cladding has formed, perhaps extending to the core periphery. A few of the upper plenum components may rest on top of the debris. The amount of cladding oxidized, approximately 50%, is indicative of the fraction of the core which is brittle or fragmented. The presence of liquified fuel, or any once molten material, is enough to ensure that some areas of the debris will be fused together and that separation techniques and tools must be designed accordingly. The total weight of potential debris and embrittled cladding is 64,000 to 83,000 kg (140,000 to 184,000 lb).

CONTENTS

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1. INTRODUCTION

Following the accident at the Three Mile Island Unit-2 (TMI-2) Nuclear Power Station on March 28, 1979, examination planning groups were formed to make recommendations for core damage assessment and core removal, and for fuel experiments and examination. These are Planning Groups 7.2 and 7.4, respectively. In their recommendations contained in the GEND Planning Report,[1] each group recognized the need to thoroughly understand the nature of the present configuration of the TMI-2 core and related components. The planning reports addressed several types of examinations for a range of core damage, but only cursorily reported on the expected core damage. A review of the detailed core damage assessments is needed to help narrow the scope of the planning.

Thus, the purpose of this report is to review and summarize the core damage assessments which have been made, identify the minimum and maximum bounds of damage, and establish a "reference" description for the current status of the core. The different degrees of damage present in the reference core will be considered during planning of contingency tooling and procedures for reactor disassembly, fuel removal, core inspection, and fuel sample acquisition.

Possible damage to the reactor core has been analyzed. The core damage assessments were generally of three types:

1. Reconstruction of the transient thermal-hydraulic behavior of the system by using interpretations of data available from in-core and outside-of-core instrumentation and computer code analyses;

2. Analyses of the amount of hydrogen produced; and

3. Evaluation of the amount of fission products released.

Numerous uncertainties exist in each type of assessment due to the equivocal nature of the available data.

Section 2 of this report describes the thermal-hydraulic analyses of core damage. Analyses of the core damage using data on hydrogen generation and fission product release are discussed in Sections 3 and 4, respectively. Section 5 summarizes the best estimate damage to the core components and defines a "reference" core. Appendix A outlines some relevant design parameters of the TMI-2 reactor core. Appendix B specifies the equations to determine the amount of zircaloy oxidized from the total hydrogen produced. The inventory and distribution of zircaloy in the core are presented in Appendix C. Appendix D describes the method of constructing illustrations of the core damage limits.

2. THERMAL-HYDRAULIC EVENTS

Several investigators (see References 2 through 8) have attempted to reconstruct the sequence of thermal-hydraulic events in the TMI-2 core and primary system as one method of assessing core damage. They have used known events from log books and reactimeter data, information deduced from instrumentation that was often used for measuring parameters other than the ones originally intended (such as obtaining liquid level data from neutron detectors), and thermal-hydraulic principles and computer models to arrive at a consistent set of events.

The investigators agree that the core experienced no damage during the first 100 minutes into the accident. The reactor was shut down and produced energy at decay heat levels that was being removed by the primary coolant pumps. Although the core remained fully covered by water, the pilot operated relief valve (PORV) on the pressurizer discharge piping remained open, causing a decrease in pressure and a loss of coolant inventory from the primary coolant system. At 73 minutes into the accident, the primary coolant pumps in the B loop were stopped and at 100 minutes, the A loop pumps were stopped. As a result of the loss in primary coolant flow and the continued loss of coolant through the PORV, the core began to uncover. The core did not become covered again until about 210 minutes.

Most, if not all, of the damage to the core is believed to have occurred between 100 minutes and 210 minutes. This section briefly summarizes the results of the thermal-hydraulic studies as they relate to the behavior of the core materials during this period and to the core status at present.

2.1 Transient Water-Steam Mixture Level

During core uncovery, the two-phase steam and water mixture, which had been homogeneous during forced flow, separated. Steam collected in the high regions of the primary system. Below the water-steam mixture level, the coolant was near or at saturation, and heat transfer from the fuel rods to the coolant kept the rods near the saturated coolant temperatures. Relatively inefficient heat transfer occurred above the mixture level and fuel rod temperatures increased dramatically. Thus, the transient mixture level is significant in defining the extent of damage. Below an elevation that is a short distance above the mixture level, the zircaloy will remain relatively cool and retain its mechanical properties. At higher elevations, the zircaloy will become hot enough to react with steam, becoming oxidized and embrittled. Figure 1 summarizes the time-dependent water-steam mixture level in the core as determined by several investigators. Information obtained by these investigators is discussed in the following sections by organization.

2.1.1 Nuclear Safety Analysis Center

According to the Nuclear Safety Analysis Center (NSAC),[2,3] the initiation of core uncovery occurred at about 113 minutes. The calculated water-steam mixture level in the core during this period, shown in Figure 1, was derived from calculated heat and mass balances and a variety of data from such instruments as the outside-of-core source range detectors and in-core self-powered neutron detectors (SPNDs). Although the block valve on the PORV piping was closed at 140 minutes, the decrease in the core mixture level was still expected to continue. The mixture was expected to drop to the 1 metre (3.3 ft) elevation at 174 minutes and then begin to increase. However, after 174 minutes the mixture level was not well characterized.

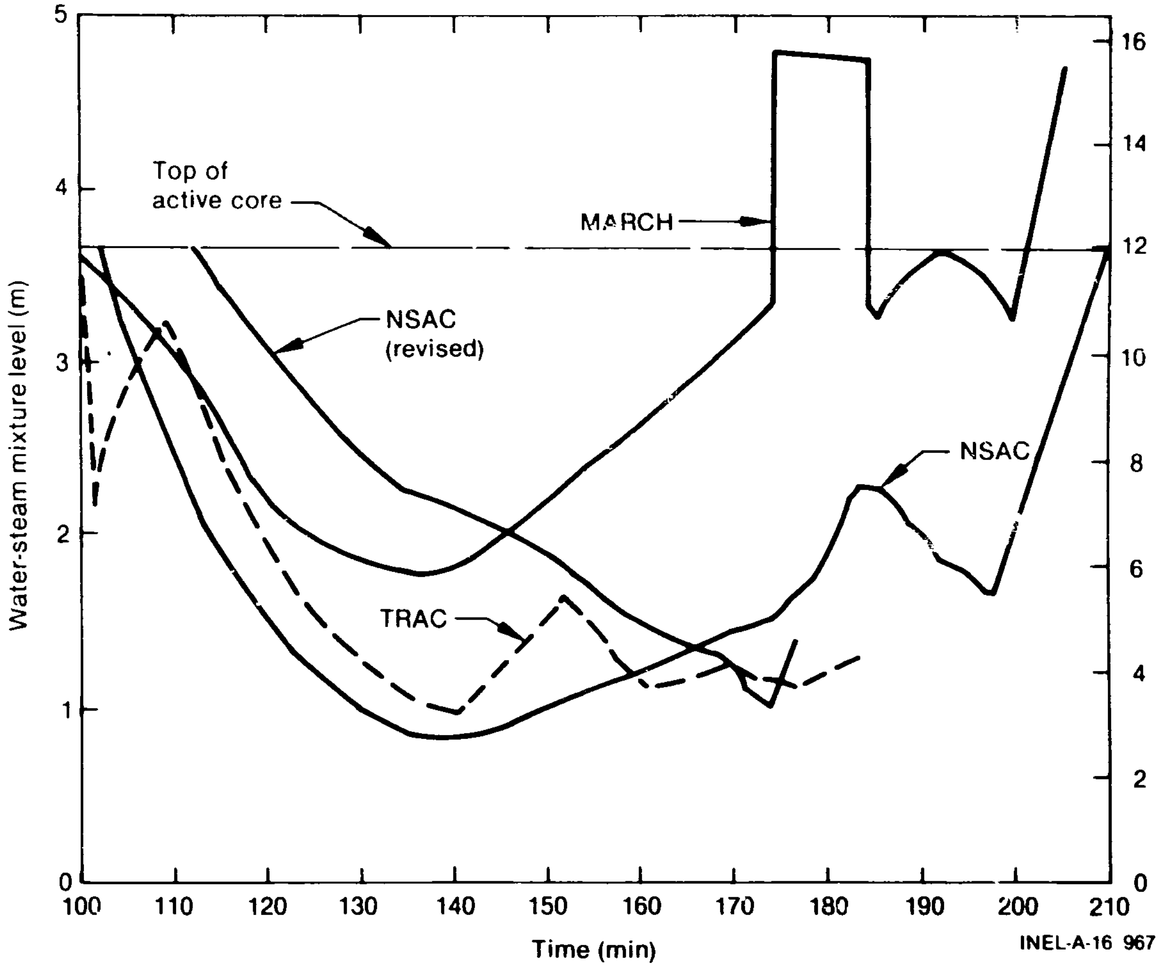

Figure 1. Water-steam and core-mixture levels during uncovery from 100 to 210 minutes.

2.1.2 Nuclear Regulatory Commission Special Inquiry Group

A slightly different mixture level history was outlined by the Nuclear Regulatory Commission (NRC) Special Inquiry Group,[4] although it is not specified in enough detail to be included in Figure 1. Core uncovery is believed to have started between 102 minutes and 112 minutes. An engineering code, TMIBOIL,[9] was used to calculate the fuel rod temperatures and core damage parametrically. The studies were performed varying the minimum core mixture level, boiloff rate, radial peaking factors, and conduction heat transfer coefficients. The damage calculated for each scenario was compared with the amount of hydrogen generated, fission product release, and maximum recorded temperatures. Based on these analyses, a scenario in which the core mixture level dropped to 1.2 ± 0.15 m (4.0 ± 0.5 ft) above the bottom of the core, and achieved a steady-state level after 120 minutes correlated well with the available data. At 130 minutes, the primary system pressure began to rise, and at about 140 minutes, the block valve to the open PORV was closed, stopping the loss of primary coolant. The primary system pressure continued to increase and the coolant mixture level in the core began to rise.

2.1.3 Los Alamos Scientific Laboratory

The Los Alamos Scientific Laboratory (LASL) performed Transient Reactor Analysis Code (TRAC) analyses of the thermal-hydraulic events.[5,8] Estimated values for high pressure injection, letdown, and PORV flows were input to the code. Heat from fission product decay and the oxidation of zirconium were modeled. TRAC calculated core uncovery at 100 minutes. The coolant mixture level dropped to 1 m (3.3 ft) at 140 minutes, the time at which the block valve to the PORV was closed. The mixture level gradually stabilized at about 1.2 m (4.0 ft) where it remained until about 185 minutes, the time at which the calculation was terminated.

2.1.4 Battelle Columbus Laboratories

The TMI-2 accident was also analyzed by Battelle Columbus Laboratories using MARCH, a code which predicts thermal and hydraulic conditions in the primary coolant system and containment building for hypothetical core melt accident senarios.[6] Boundary conditions input to MARCH were varied within their uncertainties to yield approximately 20% of the core at temperatures greater than 2550 K (4130°F). This represents an upper bound on the fuel temperatures based on examination of the data on radioactivity release to the containment, and maximizes the sensitivity of the core response to variations in alternative event sequences. Initiation of core uncovery was calculated to occur at 101 minutes. At about 135 minutes, a minimum collapsed liquid level of 1.7 m (5.5 ft) was calculated. The mixture level decreased at a rate approximately 50% faster than expected by boiloff due to decay heat because steam condensation in the steam generator was depressurizing the primary system and causing additional flashing. After 142 minutes, corresponding to block valve closure, the mixture level increased, rising to 2.4 m (8 ft) at 160 minutes.

2.1.5 Westinghouse-Nuclear Energy Systems

Westinghouse Nuclear Energy Systems (W-NES) performed calculations of the TMI-2 accident for the President's Commission to evaluate degraded core cooling for four different degrees of core damage.[7] One set of analyses treated an originally intact core representative of TMI-2. Core and system conditions at 100 minutes were obtained from NSAC.[10] A best-estimate, net makeup flow of 2.6 kg/s (41 gpm) was assumed. For the purposes of the analyses, closure of the block valve, operation of the high-pressure injection system after 100 minutes, and restart of the primary coolant pump after 100 minutes were assumed not to occur, although all three of these assumptions were inconsistent with the actual TMI-2 transient. The calculated time-dependent core mixture level was in close agreement with the original NSAC estimate;[10] the predicted minimum level was about 0.9 m (3 ft).

2.1.6 Summary

The variablility of the water-steam mixture level results summarized in Figure 1 may, in part, be explained by parametric analyses using MARCH which indicated that the calculated results were sensitive to the net makeup/letdown flow rate due to uncertainties in the high pressure injection and letdown line flow rates. The net inlet flow rates used in some of the analyses and the fuel rod behavior parameters which are affected by the flow rate are summarized in Table 1. As might be expected, as the net inlet flow decreases, the minimum mixture level generally decreases, fuel temperatures increase, and the amount of zircaloy oxidized increases.

2.2 Core Damage

During the period of core uncovery from about 100 minutes to 174 minutes, the fuel rods were heating up. As cladding temperatures reached a range of about 1030 K (1395°F) to 1150 K (1610°F), rupture of the rods began to occur.[2,4] Since virtually all the rods reached temperatures of this magnitude, greater than 90% of the rods are expected to have failed.[2,4] The best estimate of the time of failure ranges from 137 to 142 minutes after the start of the accident.[2] This coincides well with an estimated three-minute transport time of the fission products to the containment radiation monitors which responded at 145 minutes.[2]

The cladding continued to heat up, becoming oxidized and embrittled. This exothermic reaction contributed to the rapid heatup of the core. Hot zircaloy in the upper regions of the core may have become fully oxidized. As the heat source from oxidation decreased, the oxidized cladding would have cooled. Steam rising from the lower regions of the core carried energy from the peak power locations to the upper region of the core, thus smearing the fuel rod temperatures and the axial extent of cladding oxidation.[11] Approximately the upper half of the core was embrittled. Detailed estimates of the amount of zircaloy oxidized are discussed in Section 3.

TABLE 1. VARIATION IN ESTIMATED NET INLET FLOW RATES BETWEEN 100 AND 200 MIN.

	Net Inlet Flow Rate (kg/s)[a]	Minimum Mixture Level (m)[b]	Fuel Rod Peak Temperature (K)	Fraction of Core >2550 K (>4130°F)	Fraction of Cladding Oxidized
BCL-MARCH[6] (high ECC)	6.9	--	~2550	0.01	0.02
BCL-MARCH[6] (base case)	5.7	1.8	>2550	0.24	0.15
BCL-MARCH[6] (low ECC)	4.4	--	>2550	0.42	0.24
W-NES[7]	2.6	~0.9[c]	2880	--	>0.33[d]
NRC	1.6[e]	--	--	--	--
NSAC[2]	f	0.97	--	--	--
LASL-TRAC[5,12]	-12.8[g], -4.5[g], 0[g]	~1	~1800,[h] 2500[h]	--	<0.13[i]

a. 1 kg/s = 15.87 gpm, assuming that the density of water is 1 g/mL (8.35 lb/gal).

b. 1 m = 3.28 ft.

c. Based on close agreement with Reference 10.

d. Top 1.2 m (4 ft) of core was 100% oxidized.

e. Letter, L. S. Tong to M. Cunningham, Attachment III, "Best Estimate Initial and Boundary Conditions for TMI-2 Analyses," USNRC, July 26, 1979 (see Reference 6).

f. May be negative (see Reference 6 pp. 4-2).

g. Assumed net inlet flow rate for 101 min < t < 120 min = -12.8 kg/s (-203 gpm); 120 min < t < 138 min = -4.5 kg/s (-71.4 gpm); t > 138 min = 0.

h. About 1800 K (2780°F) was calculated by TRAC at 180 min; extrapolation of the TRAC results to 200 min yields about 2500 K (4040°F) in the hottest core regions.

i. Calculated maximum oxidation at hottest axial location extrapolated to the total core yields 130 kg (287 lbm) of hydrogen at 185 min, which is equivalent to oxidizing about 13% of the cladding (Reference 26).

A range of peak fuel rod temperatures has been estimated. Based on TRAC calculations, peak temperatures were estimated to be 2400 K[8] (3860°F) to 2600 K[12] (4220°F) before resumption of high pressure injection flow. Fuel rod plenum temperatures were estimated to be about 1700 K[12] (2600°F). NRC's Special Inquiry Group and W-NES estimated slightly higher peak fuel temperatures of 2700 K[4] (4400°F) and 2900 K[7] (4760°F), respectively. The President's Commission concluded that the fuel temperatures may have exceeded 2475 K (4000°F) throughout 30 to 40% of the core volume, and 2200K (3500°F) throughout the upper 40 to 50% of the core.[13] These estimates are higher than the estimate by NSAC[14] that the gross core average temperatures did not exceed 2000 K (3140°F). However, a quantitative comparison cannot specifically be made since NSAC specified neither the size of the damaged region nor the peak temperatures.

At temperatures above 2175 K (3455°F) , it has been experimentally observed[15] that the UO_2 fuel pellets, in contact with the cladding, can be dissolved by the zircaloy, forming a liquid phase of Zr-U-O termed "liquified fuel." Essentially all investigators expected that liquified fuel would have been produced in small quantities, but that little or no fuel melting occurred.[13,14] NRC's evaluation is somewhat more pessimistic, suggesting that no less than 32% of the fuel assemblies have fingers of previously liquified fuel extending below the region of primary damage.[4]

Estimates of the damage to other core components, whose materials and melting points are tabulated in Appendix A, can be made from these temperatures. Calculations performed using radiative and convective heat transfer between fuel rods, steam, and nonfueled rods such as control rod guide tubes and burnable poison rods indicate that the temperature of the non-fueled rods may only have been about 10 K (18°F) less than that of the fuel rods.[16] However, for the period up to 174 minutes in the accident, the NRC Special Inquiry Group[17] suggested that "percolation" may have occurred in the instrument and control rod guide tubes:

"Instrumentation tubes and control rod guide tubes survived longer than the neighboring fuel rods because they were not significant heat sources and because they served as 'percolator tubes' during depressurization, in which steam bubbles, formed in the annuli, caused liquid water to percolate above the average level in the core to reach higher temperature regions before evaporating. The net effect was to produce a much higher mass flow of steam, as well as velocity of steam flow, through the annuli between the guide tubes and the control rods (and in the double annuli of the instrumentation tubes) than occurred in the subchannels between neighboring fuel rods. Thus, the guide tubes, control rods, and instrumentation tubes stayed much cooler than otherwise expected during depressurization and, consequently, lagged significantly in temperature rise compared with their surroundings. Their heatup started later, and the heat absorbed by them was transferred by radiation from neighboring fuel rods and by conduction-convection interaction with the steam in the fuel subchannels."

Recent evidence suggests that the instrument tubes may not have survived. Figure 2 is a cross section of the instrument string comprised of seven SPNDs, one thermocouple, and one background detector. The Inconel oversheath is the primary pressure boundary and the Inconel center tube is the secondary pressure boundary. A swagelock fitting on the instrument string acts as a third pressure boundary. After removing the swagelock fitting on one instrument string, a traveling in-core probe (TIP) was inserted into the central hole of the instrument string that was located in the center of the N-8 assembly at about the mid-radius of the core, four rows from the center assembly as shown in Figure 3 (based on Reference 3). The TIP was partially inserted, perhaps as high as 1 m (3.3 ft) above the bottom of the core, with great difficulty before being withdrawn.[18,19] During removal, the detector stuck several feet below the core and is currently immovable. Magnesium oxide insulation in the TIP is known to swell when in contact with water. The inner tube of the instrument string is the last pressure boundary for the assembly with the TIP inserted. White boric acid crystals are present on the concrete floor where the in-core detectors originate. Although the boric acid crystals cannot be confirmed as being deposited from water emanating from the instrument string with the TIP, this observation is consistent with the postulation

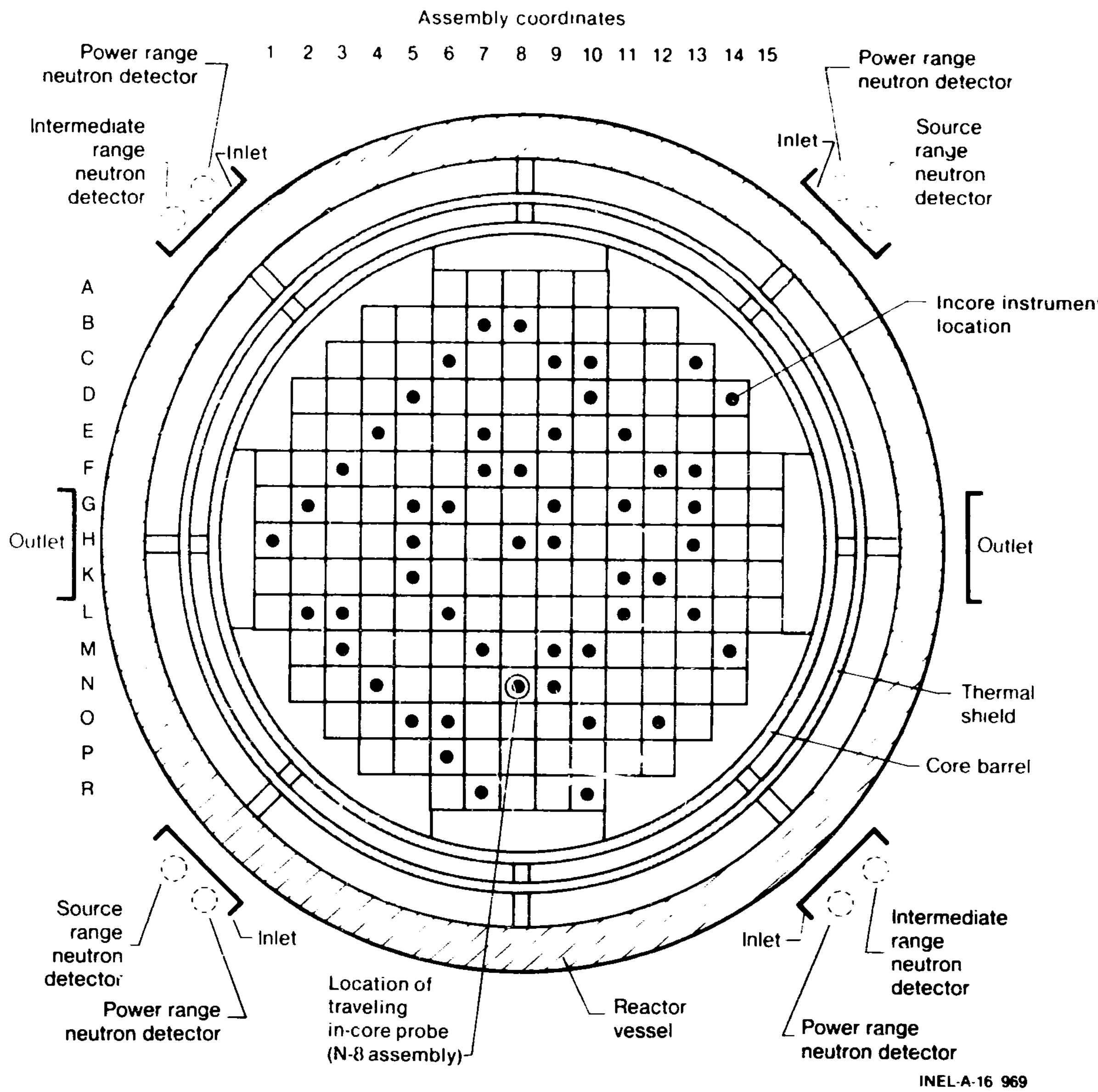

Figure 2. In-core instrumentation position.

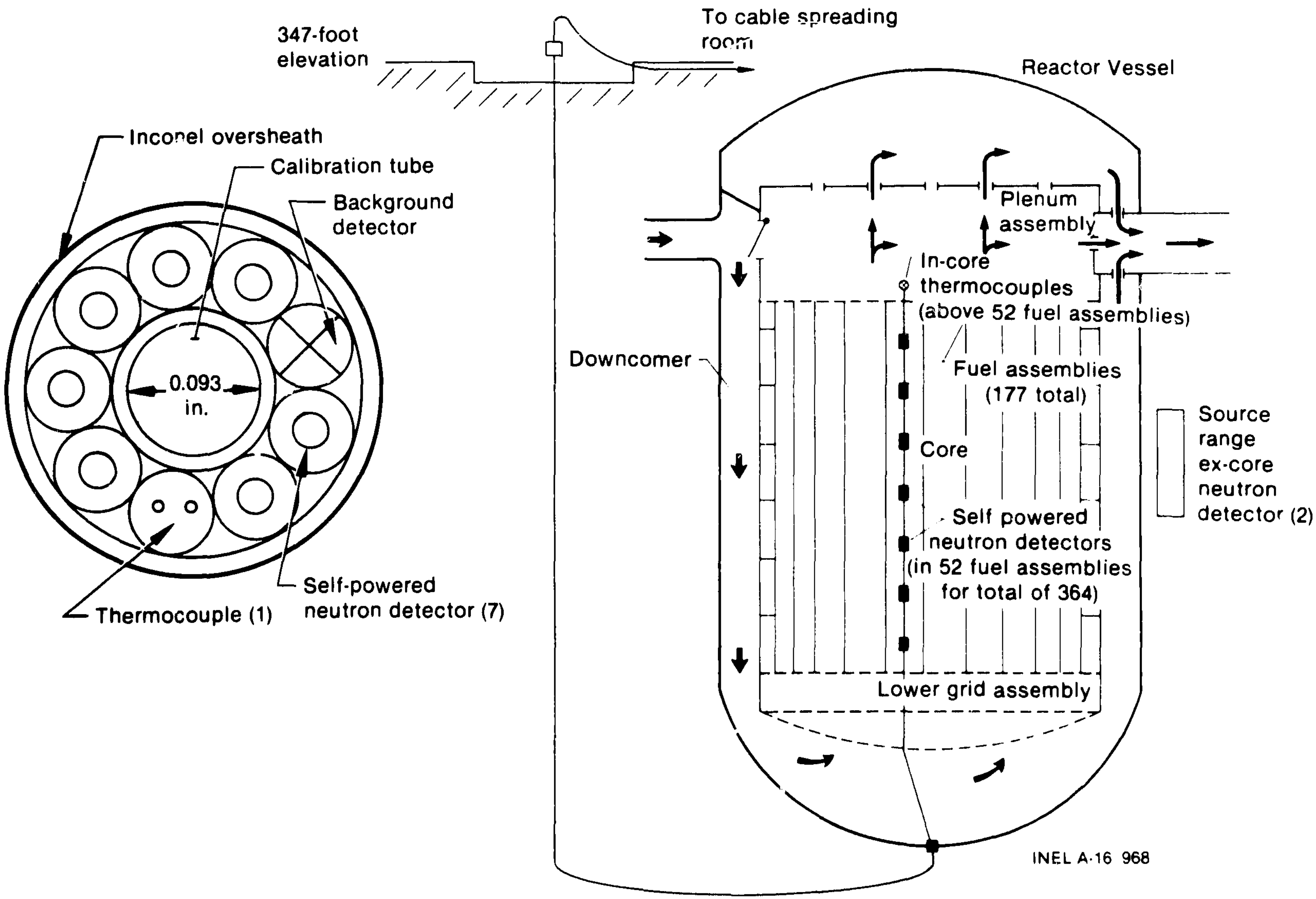

Figure 3. A top view of the reactor core, illustrating location of the traveling in-core probe.

that that particular instrument string has failed. If this is in fact the case for an assembly at the midradius of the core, then it is very possible that a large majority of the instrument strings in the central region of the core have failed.

The axial power shaping rods, control rods, and burnable poison rods were also damaged. Since the melting points of the silver-indium-cadmium alloy and the 304 stainless steel are about 1075 K (1475°F) and 1675 K (2555°F), respectively, these rods melted over much of the same volume of the core in which the fuel rods were oxidized. The Ag-In-Cd alloy probably remained in the core region since it is insoluble in water. Both materials contributed to formation of the debris bed and fusing of portions of the rubble. The zircaloy cladding of the burnable poison rods oxidized over the same region as the cladding on the fuel rods so these rods are in the same fragmented condition. The rods are probably in place, but the boron absorber is known to leach out in the presence of water in a radiation environment. Gadolinia burnable poison rods (UO_2-Gd_2O_3) in two test assemblies are also oxidized and fragmented. Since the melting point of Inconel 718 is about 1550 K (2330°F), the grids would have melted over most or all of the region of the core in which the fuel rods were oxidized. The zircaloy guide tubes may have oxidized over a region somewhat smaller than that of the fuel rod cladding due to the early percolation effect, however, they are expected to have contributed to the material in the debris bed.

Temperatures of the upper plenum assembly were calculated by TRAC[5] to have reached 1100 K (1520°F) at 185 minutes into the transient, a time when peak fuel temperatures were calculated by TRAC to be 1800 K (2780°F).[5,12] Calculated fuel rod plenum temperatures were 1200 K (1700°F).[12] The fuel rod plenum and peak fuel temperatures were extrapolated to about 1900 K (2960°F) and 2600 K (4220°F),[12] respectively, before resumption of high pressure injection flow at about 200 minutes. Although these investigators did not extrapolate their analyses of the upper plenum assembly, in view of their estimates for the other temperatures, it is possible that the temperatures of the upper plenum could have risen to between 1500 K (2240°F) and 1800 K (2780°F).

Temperatures in this range would imply, first, that brazements of Beryllium-Nickel 5 that hold the control rod guides to structural support plates in the upper plenum would have melted since they have a melting range of 1365 to 1420 K (2000 to 2100°F). Second, stainless steel components, such as the fuel assembly upper end fittings which have a melting range of 1670 to 1695 K (2550 to 2590°F), would have melted or fused at their contact points with the plenum. Spider failure and leadscrew distortion would be likely.

At 174 minutes, with the coolant mixture level at about 1.5 m (4.9 ft) above the bottom of the core, one primary coolant pump in the B loop was turned on for 19 minutes. This produced a sudden influx of water to the core from the once-through steam generator (OTSG) B. Since the cladding was embrittled due to oxidation, the entering water would have produced a thermal shock to the cladding, causing fragmentation of the ZrO_2 and UO_2.[20] This would have either formed a debris bed above the axial midplane of the core or increased the size of one already present. Although substantial quenching of the rods occurred, the debris bed itself remained hot and in steam.

The next major change in core condition occurred between 222 minutes and 226 minutes into the accident. The source range monitors showed a sharp increase in activity, the primary system pressure increased even though the block valve was open, and the cold leg temperatures of both the A and B loops increased. Temperature estimates from thermocouple and SPND data indicate that temperatures of 800 K (980°F) were reached at locations 25 to 75 cm (10 to 30 in.) above the bottom of the fuel rods.

Although it is educated speculation, additional core damage apparently occurred during this time. Given the oxidized and embrittled cladding before about 225 minutes, it is possible that "unstable thermal-hydraulic conditions"[3] developed to fracture additional cladding, resulting in some additional slumping of the core and densification of the debris bed.[3,4] A steam blanket may have formed below a crust in the bed, blocking coolant and permitting additional zircaloy oxidation and hydrogen formation. Based on available instrumentation, no apparent change in the condition of the core occurred after the event at about 226 minutes.

Independent assessments of the core flow resistance following the accident were made by Babcock and Wilcox (B&W) and Battelle Pacific Northwest Laboratories (PNL).[19] The assessments indicated that a large portion of the core was blocked. B&W made two estimates by comparing reactor coolant system flow meter readings, with one pump operation, before and after the accident and performing a core heat balance after the accident. The estimated decay heat and measured core coolant temperature change determined the flow. An effective blockage area of about 90% was indicated. PNL performed COBRA calculations to reproduce the TMI-2 core exit temperature distribution during single loop operation, as well as a simple heat balance using the average of measurements from the core exit thermocouples located at the top of the instrument tubes. They estimated blockages of 60 to 80%, with local blockages of 95%. An effective core blockage area of 90% was also estimated by performing a simple core heat balance using the average core exit thermocouple readings on the periphery of the core. From these three assessments, it was concluded that an effective core flow area blockage of approximately 90% had occurred. Temperatures in the peripheral assemblies indicated that the minimum blockage occurred in the peripheral assemblies. However, since more than 20% of the core flow area is made up by the peripheral assemblies, some blockage at the edge of the core is expected.

Damage parameters from the three primary assessments are summarized in Table 2. With the exception of the estimated fuel temperature, the expected damage is quite consistent.

Based on the foregoing review, the core condition can be described in the following manner. A debris bed of fractured, oxidized zircaloy cladding and fragmented fuel pellets rests on fuel rod stubs and Inconel spacer grids. The debris bed extends downward to between 0.9 and 1.8 m

TABLE 2. SUMMARY OF CORE DAMAGE ESTIMATES BASED ON RECONSTRUCTION OF THE ACCIDENT

Damage Parameter	NSAC[14]	President's Commission[13]	NRC Special Inquiry Group[4]	LASL[12]
Number of failed rods (%)	--	>90	100	100
Fuel temperature	Gross average in damaged region <2000 K (<3140°F)	Distribution: a. >2200 K (> 3500°F) in 40 to 50% of core b. >2475 K (>3995°F) in 30 to 40% of core	Peak ∿2700 K (∿4400°F)	Peak ∿2600 K (∿4220°F)
Liquified fuel	Locally possible	Yes	Yes, over most of core radius	Yes, over most of core radius
Molten fuel	Locally possible, very limited	--	--	--
Core slumping at 225 min	Probable	Probable	Yes	--
Fuel rod fragmentation	Yes	Yes	Yes	Yes
Damage level	Embrittlement extends down to 0.9 to 1.8 m (3.0 to 5.9 ft) from bottom of core at centerline. Peripheral rods relatively intact.	Debris extends down to 1.4 to 1.5 m (4.6 to 4.9ft) from bottom of core at centerline.	Embrittlement extends down to 1.2 to 1.5 m (3.9 to 4.9 ft) from bottom of core at centerline.	Embrittlement extends down to 1.0 m (3.3 ft) from bottom of core at centerline.

(3 and 6 ft) above the bottom of the core at its center. The debris boundary extends outward and upward from its lowest point near the core centerline, encompassing a volume in the shape of an inverted bell. Damage to the rods near the periphery ranges from moderately intact (not fully embrittled) to partially liquified and oxidized. Liquified fuel formed, fusing core components and debris in several areas. Some of the upper plenum components could have partially melted or fused together at their contact points.

3. HYDROGEN GENERATION

A number of mechanisms in light water reactors may result in the generation of hydrogen. One mechanism is radiolysis, in which water decomposes into hydrogen and oxygen as a result of the absorption of energy from ionizing radiation. Oxidation of UO_2 fuel and structures within the core is also a potential source. However, for the TMI-2 accident, the generation of hydrogen by these processes is expected to have been small when compared with the volume of hydrogen that was produced by oxidation of the zirconium in the fuel rod cladding by the steam in the reactor vessel.[4] Assessment of the amount of hydrogen generated yields an additional estimate of the amount of zircaloy oxidized, and hence embrittled, in the reactor core. This section dicusses the hydrogen inventory analyses, zircaloy oxidation, and computer code calculations of this aspect of the TMI-2 accident.

3.1 Hydrogen Inventory

Material balances were used by several investigators to determine the amount of zircaloy oxidized based on the amount of hydrogen produced. These balances are summarized in Table 3. EPRI[21] indicated that the material balances were performed using the reactor building as the boundary. Any hydrogen escaping the building would not be included. Since a maximum of 8%[22] to 10%[21] of the inventory of noble gases escaped from the reactor building during the first few days of the accident, at least 8 to 10% of the hydrogen might also be expected to have escaped. This would make estimates of the amount of zircaloy oxidized too low by the same amount.[21] However, if oxygen were depleted by oxidation of materials other than zircaloy, less oxidation of the zircaloy would result.

Table 4[21,23,24] identifies the components of the containment atmosphere which was sampled twice on March 31, 1979 at about 0600 hours. The amount of hydrogen in the second sample is in question. As noted in the table, two different compositions were used by the various

TABLE 3. HYDROGEN AND CLADDING INVENTORIES AND CLADDING OXIDATION

A. Hydrogen Inventory (kg)	Preburn (Ref. 21)	March 31 (Ref. 21)	March 31 (Ref. 6)	March 31 (Ref. 22)	March 31[a] (Ref. 4)	April 1-2 (Ref. 21)	June 1 (Ref. 21)	Aug. 2 (Ref. 21)	Average[b]
1. Total produced (2 + 5)	475	548	--	454 ± 20 to 582 ± 36	450	< 304	513	550	510
2. Released to containment (3 + 4)	308	437	--	339 to 467	350	262	513	550	--
3. Burned	0	372[c]	256[d]-469[c]	267[d] to 395 ± 30[c]	270	181[c]	490[c]	526[c]	391
4. Remaining in containment	308	65	--	72 ± 4	80	81	23	24	--
5. Remaining in reactor coolant system (6 + 7)	167	111	--	115	100[d]	< 42	< 1[e]	< 1[e]	-
6. In solution	31	33	--	33	26[f]	33	< 1	< 1	--
7. In bubble	136	78	--	82±20	74[f]	< 99	0	0	--
B. Cladding Inventory (kg)	24 040	24 040	--	22 585	24 040	24 040	24 040	24 040	--
Cladding Oxidized (kg)	10 818	12 261	--	9937 to 14 282	~12020	6731	11780	12501	11943
(%)	45	51	--	44 to 63	~50	28	49	52	50.6
C. Cladding Oxidized[h] (kg)	11 053	12 752	--	10 565 to 13 543	10 472	7074	11 938	12 799	11 876
(%)	46	53	--	44 to 57	44	30	50	54	49.6

a. Investigators[4] judged these values to be "most likely" after reviewing calculations from several sources, including Reference 23.

b. Does not include results from April 1-2 measurements.

c. Based on oxygen depletion.

d. Based on hydrogen burn.

e. Hydrogen recombiners had been operating.

f. At 16 hr.

g. Bubble virtually gone by this time.

h. Results from this work based on 23 922 kg (52 740 lb) of zircaloy in the core (see Appendix [illegible]).

TABLE 4. CONTAINMENT ATMOSPHERE SAMPLES, MARCH 31, 1979 AT 0600 HOURS

	Atmosphere Composition (at. %)		
Component	Sample 1	Sample 2a	Sample 2b
H_2	1.7[a,b]	1.7[b]	1.9[a]
O_2	15.7[b,c]	16.5[b,c]	16.5[b,c]
N_2	82.6	81.8	81.6

a. Reference 24.

b. Letter, Lavallee to Zebroski, NSAC, June 5, 1979 (in Reference 21).

c. Personal communication from A. D. Miller, NSAC, undated, to R. E. English, President's Commission (in Reference 23).

Note: In analyses of the containment atmosphere, the following were used--References 3, 5, and 21: Average of Samples 1 and 2a; Reference 23: Average of Samples 1 and 2b.

investigators. Since the variability in hydrogen content of subsequent samples taken in April[21] was of the magnitude of these two samples, 0.2%, these results were judged to be the same.[25] These and other containment atmosphere samples were used to calculate the amount of hydrogen remaining in the containment building and the amount burned by considering the oxygen depletion.

The preburn hydrogen inventory shown in Table 3 was calculated using the 0.19 MPag (28 psig) pressure pulse in the containment building shown in Figure 4[21] and the containment atmosphere composition. The amount of free hydrogen in the primary system was calculated based on estimates of bubble size by Metropolitan Edison,[24] and the temperature, pressure, and free volume of the containment building. Hydrogen in solution in the primary system was estimated based on the primary coolant temperature and the hydrogen overpressure.

Table 3, Section A, shows that, except for the calculated inventory based on the April 1 and 2 samples, estimates of the total amount of hydrogen produced are fairly consistent, averaging 510 kg (1125 lbm) and ranging between 450 and 582 kg (992 and 1283 lbm). The individual values comprising the total amount of hydrogen on April 1 and 2 that are low relative to the March 31 data are the hydrogen content in the bubble and amount of hydrogen burned. These low values result in a comparatively lower estimate of the percent of cladding oxidized. Note that a large uncertainty arises due to the methods of calculating the amount of hydrogen burned, namely, by using the remaining hydrogen concentration, or the oxygen depleted.

Section B in Table 3 summarizes the cladding inventory and the amount of cladding oxidized as reported by the investigators. To place these results on a consistent basis in this work, the kilograms of zircaloy oxidized were calculated directly from the total amount of hydrogen produced using the equation in Appendix B. The percent of zircaloy oxidation was obtained by dividing the amount oxidized by the inventory, 23 922 kg (52 470 lbm), from Appendix C.

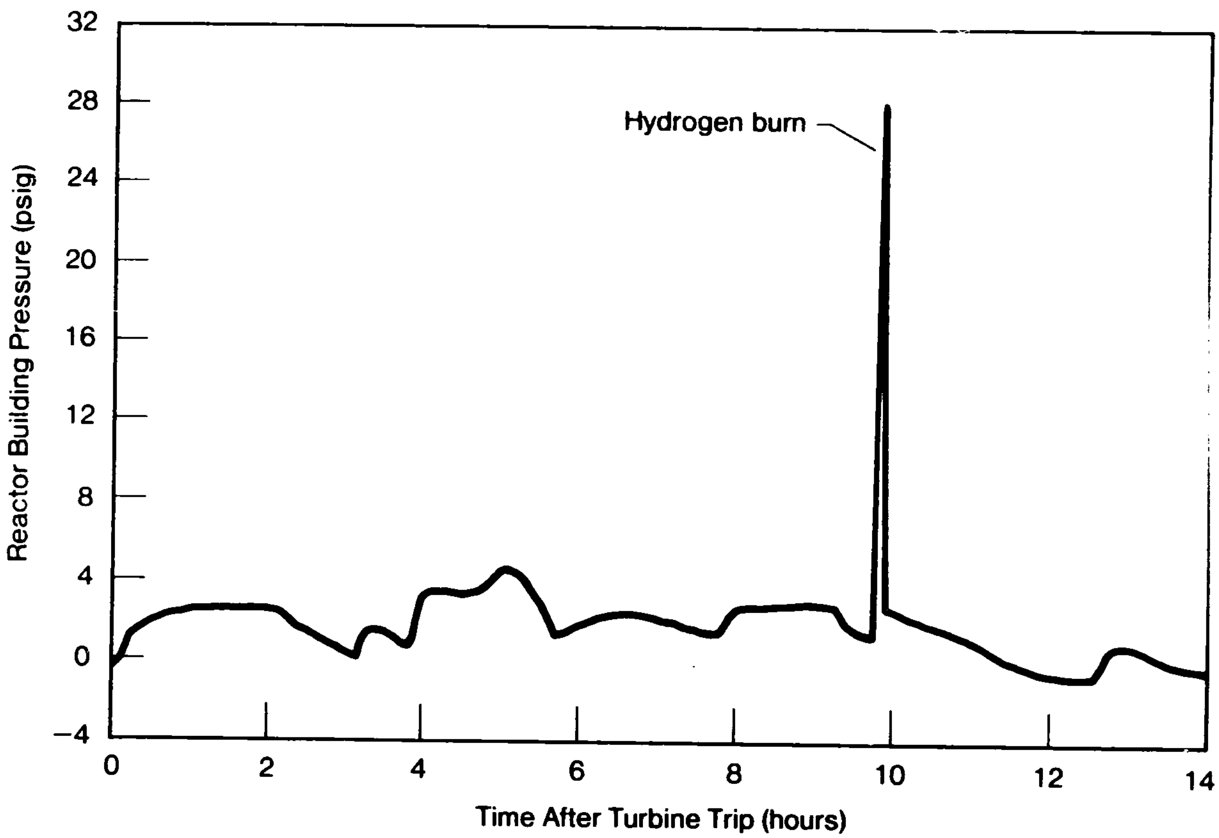

Figure 4. Pressure versus time in the reactor building after turbine trip.

Since Sections B and C of Table 3 are not significantly different, it is concluded that about 50% of the zircaloy, 11 961 kg (26 235 lbm), in the core is oxidized. About 10% of the zircaloy inventory, 2288 kg (4767 lbm), is in the plenum region of the fuel rods, as calculated in Appendix C. If it is assumed that none of the zircaloy in the plenum was oxidized, the fraction of the zircaloy in the active region of the core that was oxidized is determined by dividing 11 961 kg (26 235 lbm) by 21 634 kg (47 703 lbm). Thus, a maximum of 55% of the zircaloy in the active core region was oxidized. Since the rod plenum was estimated to have sustained temperatures up to 1900 K (2600°F)[12] some oxidation is expected; thus, the actual amount of zircaloy oxidation in the active region of the core probably lies between 50% and 55%. The accuracy of this value is estimated to be ±10% of the inventory.

3.2 Computer Code Calculations

The amount of hydrogen generated was also calculated as part of several thermal-hydraulic analyses performed using TRAC[5] and MARCH.[6] Although the estimates of the hydrogen generated were generally well below the range of values in Table 3, they are presented for completeness.

TRAC determined that melting of unoxidized cladding would begin to occur at 185 minutes into the transient, so the calculation was stopped. At that time, the outer third of the cladding thickness at the hottest axial node, located at a fractional core height of 0.75, had been oxidized. If this amount of oxidation was extended on a corewide basis, 130 kg of hydrogen would have been generated.[26] Without extending this value on a corewide basis, TRAC predicted generation of approximately 40 kg of hydrogen.[27] Hydrogen generation continued during the accident at least until the core was recovered by coolant at around 210 minutes.

Since the MARCH code is designed to calculate thermal-hydraulic conditions in the primary system and containment building during a core meltdown accident, it was able to continue running for the full course of the accident. One assumption of the calculation was that 20% of the core was at a temperature greater than 2550 K (4130°F). At 185 minutes, the total hydrogen generated was calculated to be about 90 kg compared to 40 kg with TRAC. During the core uncovery period from 101 to 210 minutes, a total hydrogen production of 160 kg was calculated, corresponding to oxidation of 15.4% of the cladding.

The hydrogen production calculated by TRAC and MARCH is not considered to be accurate in view of the consistency of the hydrogen production deduced from the experimental measurements discussed in Section 3.1. One possible reason is that there is little or no data on zircaloy oxidation rates above 1700 to 1800 K (2600 to 2780°F). This would impact the hydrogen generation rate.

4. FISSION PRODUCT RELEASE

Assessments of the damage to the TMI-2 core based on analyses of fission product release are not as precise as those determined from thermal hydraulic calculations or from analyses based on hydrogen assays. However, the fission product release analyses are significant since they generally confirm the findings of the two other damage assessment methods.

The foundation of the fission product release damage assessments rests on: (a) calculated inventories of the various fission products and actinides; (b) samples of the primary coolant; (c) a sample of the containment building atmosphere; and, (d) a sample of water from the containment building sump. Analyses of the fission product inventory were performed by LASL[28] using CINDER and EPRI-CINDER, and by B&W[22] using their version of ORIGEN. Samples of primary coolant were taken from the letdown line on March 29 and April 10, 1979 and sent to Bettis Atomic Power Laboratories (BAPL), Savannah River Laboratory (SRL), Oak Ridge National Laboratory (ORNL), and B&W for analysis; the results are presented in Table 5.[23] In addition, a gas sample was obtained from the containment building atmosphere on March 31, 1979; the results from BAPL's analysis are in Table 6.[29] A more complete accounting of the radioactivity inventory was achieved when a water sample from the containment building sump was obtained on August 28, 1979.

This section discusses the estimates derived from the above information regarding the location of core damage, the range of fuel temperatures achieved during the accident, the occurrence of UO_2 fuel melting, and fuel particle size distributions.

4.1 Location of Core Damage

The location of core damage may be estimated by comparing the isotopic ratios of uranium and plutonium observed in the reactor sump to those expected for various regions of the core. Three ^{235}U fuel enrichments

TABLE 5. FISSION PRODUCTS IN THE REACTOR COOLANT FROM SAMPLES TAKEN ON MARCH 29 AND APRIL 10, 1979

		First Primary Coolant Sample (taken 3-29-79; analyses corrected to 3/30/79)		Second Primary Coolant Sample (taken 4-10-79; analyses corrected to 4/11/79)							
		BAPL		SRL		ORNL		BAPL		B&W	
Nuclide	Half Life	Coolant Concentration ($\mu Ci/cm)^3$	Fraction[a] of Core Inventory in Primary Coolant	Coolant Concentration ($\mu Ci/cm^3$)	Fraction[a] of Core Inventory in Primary Coolant	Coolant Concentration ($\mu Ci/cm^3$)	Fraction[a] of Core Inventory in Primary Coolant	Coolant Concentration ($\mu Ci/cm^3$)	Fraction[a] of Core Inventory in Primary Coolant	Coolant Concentration ($\mu Ci/cm^3$)	Fraction[a] of Core Inventory in Primary Coolant
^{131}I	8 days	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
^{131}I	20.8 hr	6.8×10^3	0.083	--	--	--	--	--	--	--	--
^{134}Cs	2 yr	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
^{136}Cs	13 days	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
^{137}Cs	30 yr	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
^{89}Sr	50 days	{5.4	{0.000 031	{1.5×10^3	{0.009	6.0×10^2	0.004 2	{7.3×10^2	{0.004	{ --	{ --
^{90}Sr	29 yr					5.0×10^1	0.022				
^{106}Ru	368 days	3.6×10^{-1}	0.000 039	--	--	--	--	--	--	--	--
^{140}Ba	12.8 days	2.1×10^1	0.000 066	1.7×10^2	0.001 0	2.9×10^2	0.001 8	2.2×10^2	0.001 4	1.5×10^2	0.000 94
^{140}La	40 hr	--	--	1.4×10^2	0.000 75	1.6×10^2	0.000 86	1.4×10^2	0.000 75	--	--
^{99}Mo	66 hr	--	--	1.3×10^2	0.012	1.8×10^2	0.017	--	--	1.3×10^2	0.012
^{132}Te	78 hr	2.0×10^2	0.001 2	--	--	--	--	--	--	--	--
^{141}Ce	32 days	--	--	1.05×10^2	0.000 51	--	--	--	--	--	--
^{136m}Ba daughter of ^{136}Cs	0.35s	--	--	9.0×10^1	0.61	--	--	--	--	--	--
Gross α		3.6×10^{-4}	1×10^{-10}	$<1 \times 10^{-3}$	$<1 \times 10^{-8}$	$<4.5 \times 10^{-4}$	4.6×10^{-9}	1.3×10^{-3}	1.3×10^{-8}	--	--
				<1 ppb	--	<20 ppb	--	--	--	--	--
U		--	--	--	--	1.2	0.1	--	--	--	--
^3H	12yr	--	--	7.0	--	8.0	--	7.7	--	8.4	--
pH		--	--								

a. Based on a primary coolant volume of 3.8×10^8 mL. If the makeup water (9×10^8 mL) from the Borated Water Storage Tank 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.

TABLE 6. RADIOACTIVITY OF GAS SAMPLE FROM CONTAINMENT BUILDING AIR, MARCH 31, 1979

Isotope	Concentration (μCi/mL)
^{133}Xe	676.0
^{133m}Xe	16.0
^{135}Xe	8.1
^{131}I	0.063
^{133}I	<0.03

are present in the TMI-2 core, 1.98%, 2.64%, and 2.96%. The calculated isotopic ratios of uranium and plutonium for these three enrichments are compared with the measured ratios in the reactor sump in Table 7.[30] The measured ratios compare favorably with the average isotopic ratios for the 1.98% and 2.64% ^{235}U-enriched fuel and the core average ratios. From this comparison, it was concluded by D. A. Powers[30] that the central region of the core containing the two lowest ^{235}U enrichments was certainly damaged, and that the observed ratios were generally indicative of a core uniformly damaged across its cross section. However, a cursory review of the data in Table 7 would suggest that the measured isotopic ratios are in better agreement with the core average ratios, indicating uniform damage.

4.2 Fuel Temperatures

Estimates of the fuel temperatures can be made based on the types of isotopes released and on their release fractions. Isotopes can be grouped as a function of their volatility as shown in Table 8, a synthesis of tables in References 31 and 32. Under normal operation, the noble gases and halogens are released from the fuel matrix to the fuel-cladding gap. At higher temperatures, such as those that might be sustained under accident conditions, isotopes of progressively lower volatility are released. The longer the fuel remains at high temperature, the greater is the release fraction of that isotope.

Isotopic release fractions are summarized in Table 9. All of the data shown represent a nearly complete accounting of the radioisotope inventory following August 28, 1979 when a sample of the containment building sump was obtained. In general, about 60 to 70% of the noble gases, and 50% to 60% of the iodines and cesiums were released to the coolant. The increase in the strontium release fraction from March to August is consistent with its leaching rate from fuel exposed to water. Under the conditions that were calculated for TMI-2, the NRC Special Inquiry Group cautiously concluded that between 40% and 60% of the core inventory of

TABLE 7. COMPARISON OF CALCULATED AND OBSERVED ISOTOPIC RATIOS OF URANIUM AND PLUTONIUM[30]

	Amount (%) of Isotope Relative to Total Elemental Abundance in:					
Species	1.98% Enriched Fuel	2.64% Enriched Fuel	2.96% Enriched Fuel	Observed in Sump	Average of 1.98 and 2.64% Enriched Fuel	Core Average
^{235}U	1.605	2.254	2.572	2.207	1.943	2.156
^{236}U	0.074	0.081	0.083	0.064	0.078	0.080
^{238}U	98.32	97.665	97.345	97.71	97.98	97.763
^{239}Pu	87.916	90.274	91.098	90.6	89.145[a]	89.807
^{240}Pu	9.684	7.97	7.341	7.84	8.790	8.299
^{241}Pu	2.292	1.697	1.497	1.46	1.982	1.818

a. The original table in Reference 30 had 87.145, a typographical error.

TABLE 8. VOLATILITY GROUPINGS OF FISSION PRODUCT ISOTOPES[a]

				Range of Boiling Points	
Group	Classification	Volatility	Isotopes	(K)	(°F)
I	Noble gases	High	Kr, Xe	120 to 165	(-243 to -162)
II	Halogens	High	Br, I	332 to 456	(138 to 361)
III	Alkali metals	Intermediate	Cs, Rb	958 to 973	(1265 to 1292)
IV	Tellurium	Intermediate	Te	1260	(1809)
V	Alkaline earths	Low	Sr, Ba	1639 to 1908	(2491 to 2975)
VI	Noble metals	Volatile under highly oxidizing conditions	Ru, Rh, Pd, Tc, Mo	2475 to 5073	(3996 to 8672)
VII	Rare earths, actinides	Low	Sm, Ce, La	1875 to 3750	(2916 to 6290)
VIII	Refractory oxides	Low	Zr, Nb	4600 to 5200	(7820 to 8900)

a. From References 31 and 32.

TABLE 9. SUMMARY OF FISSION PRODUCT RELEASE FRACTIONS

	Release Fraction (%)				
	March 28, 1979	August 28, 1979			
Isotope	Reference 22	Reference 33	Reference 4	Reference 23	Average
^{85}Kr	71.0	60.0[a]	--	--	65
^{131m}Xe	70.0	60.0[a]	--	--	65
^{133}Xe	68.0	60.0[a]	46	57[b] - 60[c]	58
^{131}I	59.0[d,e]	29.0[f]	39[g]	--	56[h]
^{134}Cs	76.0	39.0	44	--	53
^{136}Cs	57.0[e]	--	--	--	57
^{137}Cs	60.0	49.0	63	--	57
^{89}Sr	<0.01	1.5	--	--	i
^{90}Sr	<0.07	1.7	--	--	i

a. Reference refers to noble gas release fraction of 60% without distinguishing between Kr and Xe.

b. P. Cohen, "Fission Product Release from the Core, Three Mile Island-2," July 20, 1979 (see Reference 23).

c. H. R. Denton, Letter from NRC to V. L. Johnson, Director, Technical Staff, President's Commission on the Accident at Three Mile Island, September 28, 1979 (see Reference 23).

d. The release fractions of ^{131}I, ^{136}Cs, and ^{137}Cs should be quite close since reactor coolant samples showed that the fractions of the core inventory of these nuclides in the coolant were close (0.124, 0.120, and 0.126, respectively); therefore, the release fractions of ^{131}I and ^{136}Cs were estimated by multiplying the ^{137}Cs release fraction by the ratio of the fractions in the reactor coolant.

e. The iodine release based on literal acceptance of the analytial results is 42%, but based on its chemical behavior and fission product release experiments,[34] the iodine release fraction should be close to the cesium release fraction.

f. The iodine and cesium release fractions are expected to be similar; thus, it is anticipated that about another 20% of the iodine will be found in the reactor purification demineralizer, deposited on reactor control rod material (silver), or plated on the reactor containment cooling coils (copper).

g. This work is considered to be about 20% too low as noted in Footnote f.

h. Average includes an additional 20% above the August 28, 1979 measurement as noted in Footnotes f and g.

i. Amount released from March 28 to August 28, 1979 is consistent with leaching rate from fuel exposed to water.

Groups I, II, and III in Table 8 was released to the coolant.[4] A small amount of Group IV and a minute amount of the remaining groups were released. The average values shown in Table 9 would tend to support the higher end of this range.

A variety of conclusions regarding fuel temperatures can be drawn from these data. Based on analyses of the water sample March 29, 1979, BAPL[35] concluded that: (a) most of the volatile fission products were released to the coolant; and (b) 2 to 12% of the fuel reached 1900 to 2500 K (2960 to 4040°F). From the air sample on March 31, 1979, BAPL[35] concluded that: (a) the cladding of about 90% of the 36816 fuel rods ruptured; and (b) about 30% of the fuel exceeded 2200 K (3500°F).

Considering only the ^{133}Xe release fraction of 57%,[36] the technical staff of the President's Commission speculated on the fuel temperatures.[13,23] Lorenz[37] stated that over a period of a few hours, very little of the fission gas would be expected to be released from fuel at temperatures up to 1875 K (2915°F). During the thermal transient between 101 and 210 minutes after the turbine tripped, perhaps the lower one-quarter of the fuel rods remained covered with water. Near the water/steam interface the rods were cooled by steam. Thus, the staff considered that about one-third of the rod length remained cool enough to retain the fission gas within the fuel matrix. A ^{133}Xe release fraction of 57% from the whole core implies that 85% must be released from the upper two-thirds of the core. Lorenz[37] stated that a fuel temperature of 2675 to 2775 K (4355 to 4535°F) would be required. On the basis of these considerations, the staff concluded that 50%[23] to 66%[13] of the core exceeded temperatures of 2475 K (3995°F), and that more than 90% of the fuel rods ruptured.[23] The number of cladding ruptures is consistent with BAPL's analyses, but the fuel rod temperatures are somewhat higher.

A substantially different estimate of the fuel rod temperatures has been obtained by J. Rest and C. E. Johnson.[38] Their analysis of essentially the same fission product release data described above indicates that most of the severely damaged regions of the core remained below 2000 K (3140°F).[4,22,33]

4.3 Fuel Melting

By the time fuel rod temperatures achieved high enough values to cause liquefaction or fuel melting, many fission products would have volatilized and diffused out of the fuel. Upon liquefaction or melting, other fission products, such as tellurium and ruthenium, may be released. Many factors control the release of Te and Ru, which may also occur prior to fuel melting. So, although the presence of Te and Ru does not always mean that melting has occurred, their absence generally means that melting has not occurred.

Two primary coolant samples taken on March 28 and April 10, 1979 showed very little strontium, ruthenium, and tellurium.[4] A sample of the reactor sump was also taken on August 28, 1979. Analysis of the Sr content indicated that approximately 2% of the Sr inventory in the fuel had been released. In addition, about 0.02% of the core inventory of ^{129m}Te was found. Based on the low release fractions of strontium, tellurium, and ruthenium, it was concluded that "no significant quantity of the fuel reached the melting point of UO_2."[4] There is general agreement on this aspect of the accident.

4.4 Particle Size Distribution

Estimates of the fuel particle size were obtained by considering the leaching rate of refractory elements from the fuel during the period between March 29 and April 10, 1979. As shown in Table 5, the concentrations of strontium and barium in the coolant were very small on March 29, but had increased to an average of about 1% and 0.1%, respectively, by April 10. Research has shown that the leaching rates are comparable to those from glass.[39,40]

A definitive calculation of the particle size distribution in TMI-2 is not possible, however, several estimates have been made.[4,30,41] It was generally concluded that a large portion of the core was fragmented and that the size of the particles was probably on the order of a few millimeters rather than dustlike. Powers[30] stated that particles equivalent to a sphere having a radius of less than 0.3 mm would be leviated by the coolant flow and would have escaped the reactor coolant system to a much greater extent than the remaining particles of larger radii.

Figure 5 illustrates particle size distributions which have been observed in a variety of tests, both in-pile and out-of-pile. The RIA 1-1[42] and PCM-1[20] in-pile tests had oxidized cladding which was fragmented to varying degrees. Preliminary data from out-of-pile tests were provided by the Waste Forms Response Project, that is developing fundamental information on particle size distributions in irradiated and unirradiated commercial fuel subjected to high impact loading. This work, performed by EG&G Idaho, Inc., supports Sandia National Laboratories Transportation Technology Center's studies of spent commercial fuel shipping casks. TMI-2 may have a particle size distribution representative of the data shown in Figure 5.

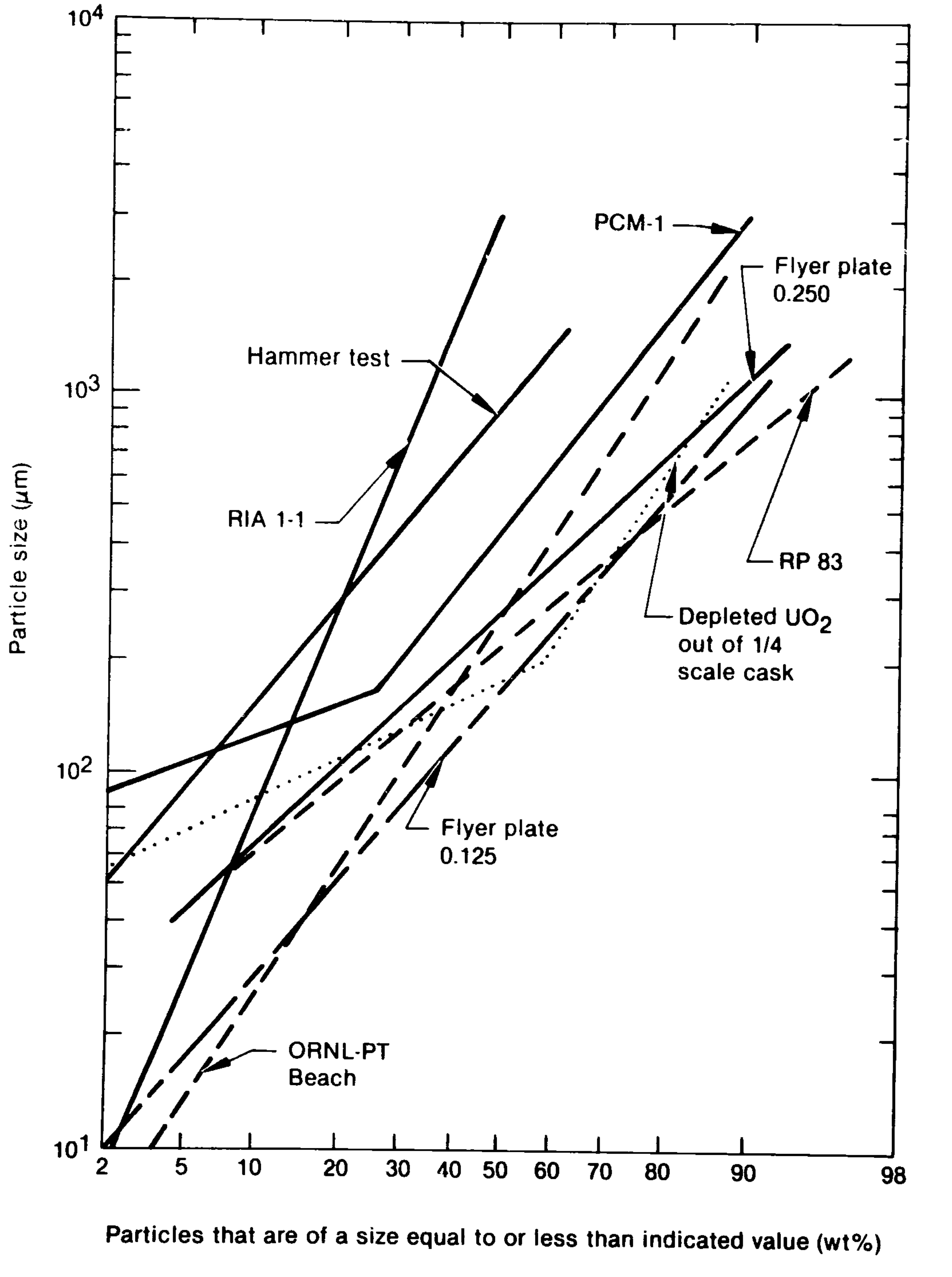

Figure 5. Particle size distributions from several types of tests.

5. CONCLUSIONS REGARDING TMI-2 CORE STATUS

Estimates of the core damage based on the interpretation of the thermal-hydraulic events, hydrogen generation, and fission product release have been reviewed. This section will summarize the limits of core damage and describe a "reference" core for the purpose of developing tooling and procedures for reactor disassembly, defueling, core inspection, and fuel sample acquisition. However, as a contingency, tooling should be available to encompass the minimum, reference, and maximum limits of core damage.

5.1 Damage Limits and Reference Core

Table 10 summarizes the damage limits estimated by various investigators and discussed previously. The estimates for each item in the table may not be self-consistent for the minimum and maximum estimates of damage since the estimates have been made by a variety of individuals. The "reference" core is also defined in the table and is self-consistent, lying between the minimum and maximum damage estimates.

Figures 6, 7, and 8 illustrate the minimum damage estimate, the reference core, and the maximum damage estimate, respectively. In constructing the figures, consideration was given to the following parameters: (a) the number of failed rods and the condition of the peripheral rods; (b) the estimated core blockage area; (c) the percent of cladding oxidized in the active core region; and (d) the embrittlement level in the core, based on the estimated minimum water-steam mixture level during the accident. Appendix D discusses the method of constructing the figures in greater detail. Three regions of cladding oxidation are shown as a function of the fractional height and radius of the active fuel region of the TMI-2 core. The height, Ho, and the equivalent radius, Ro, are equal to 3.66 m (12 ft) and 1.64 m (5.4 ft), respectively. A region of cladding immediately above the mixture level was assumed to be below the 17% embrittlement criterion and thus intact. Farther above the mixture level, a region of the cladding is expected to be embrittled, that is,

TABLE 10. SUMMARY OF DAMAGE ESTIMATES

	Minimum	Reference	Maximum
Failed Rods (%)	>90	~100	100
Fuel temperature (K)	Gross average in damaged region <2000	Peak ~2600	Peak ~2900
Cladding oxidized in active fuel region (%)	40	50	60
Liquified fuel	Locally possible	Present in several areas of central core	Present over most of core radius, perhaps extending downward to ~1 m above core bottom.
Molten fuel	None	None	Possible in a few localized areas of central core
Core slumping	Probable	Yes	Yes
Fuel rod fragmentation, debris bed formation	Yes	Yes	Yes
Peripheral rods	A few not breached, some embrittled	Few, if any not breached, most embrittled near top of core	All failed and embrittled, many with liquified fuel
Control rods and spacer grids	Molten	Melted	Melted
Instrument tubes	Most intact	Most in central region failed, peripheral tubes intact	All failed
Embrittlement level (m above bottom of core at centerline)	1.8	1.4	0.9
Upper plenum assemblies	No distortion, melting or fusing to other stainless steel components	Some distortion and local melting possible. May be fused to upper end fittings	Melting over the central lower region. Major slumping possible

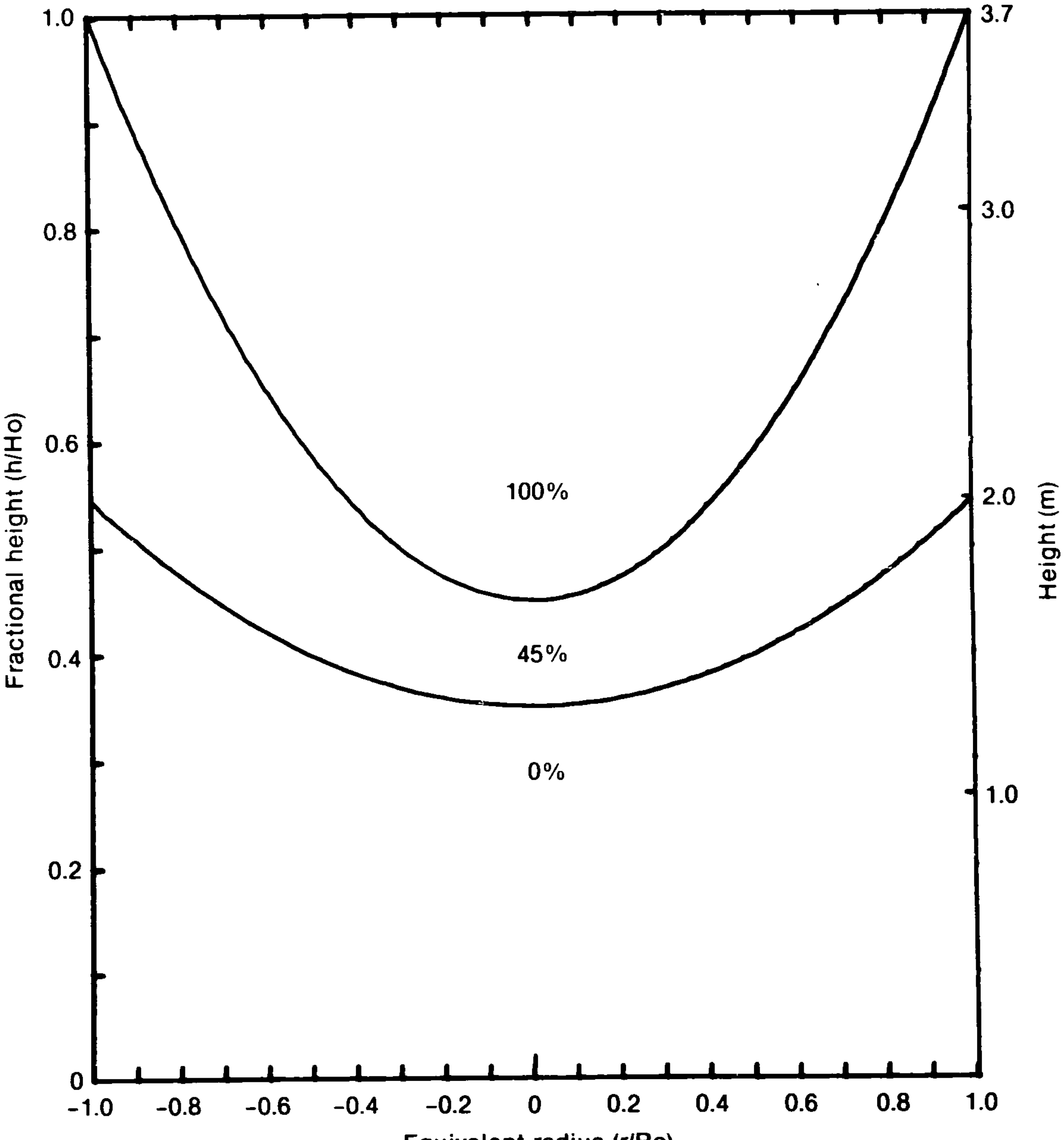

Figure 6. Regional average oxidation and estimated minimum damage configuration of the TMI-2 core.

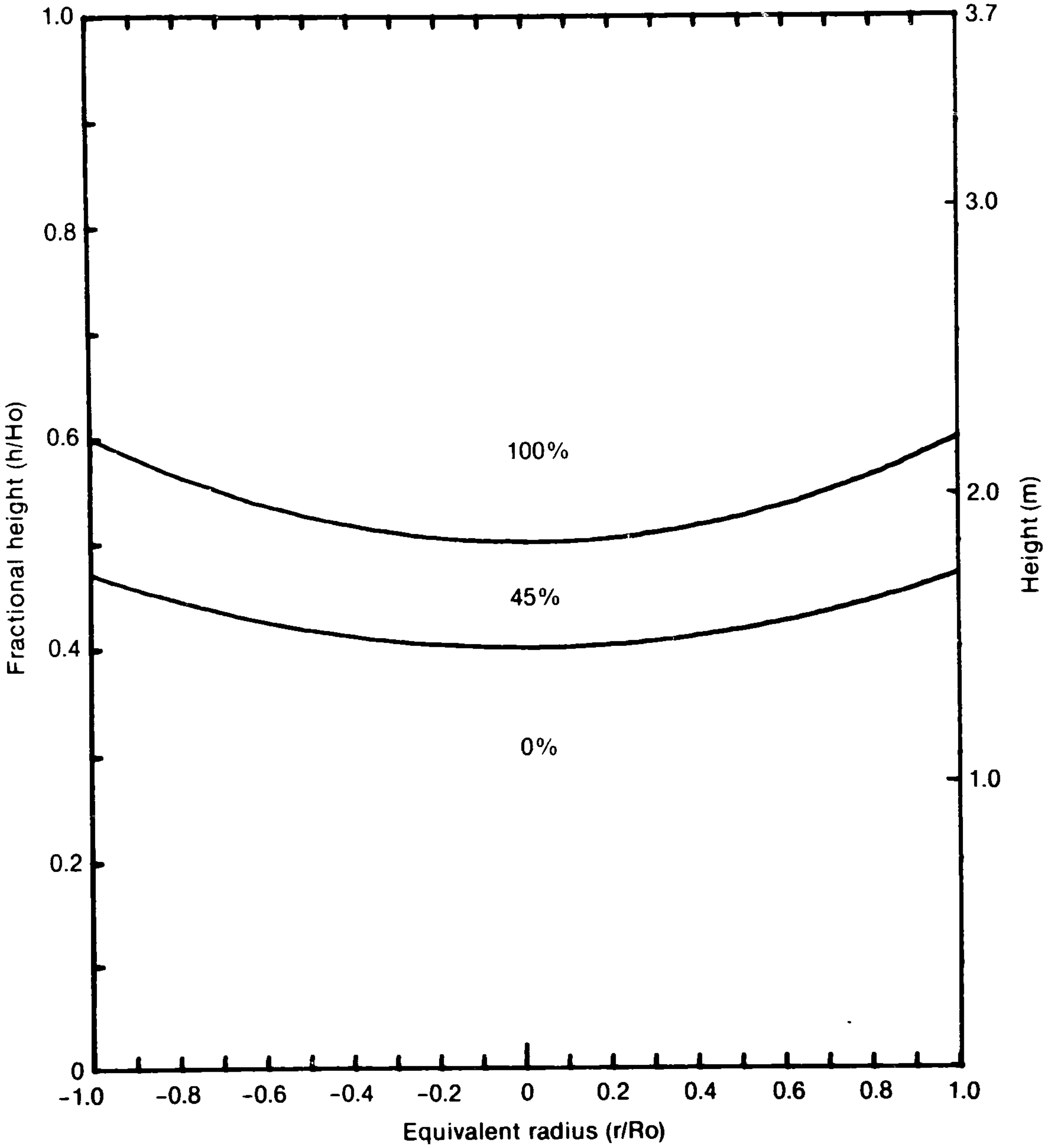

Figure 7. Regional average oxidation and reference configuration of the TMI-2 core.

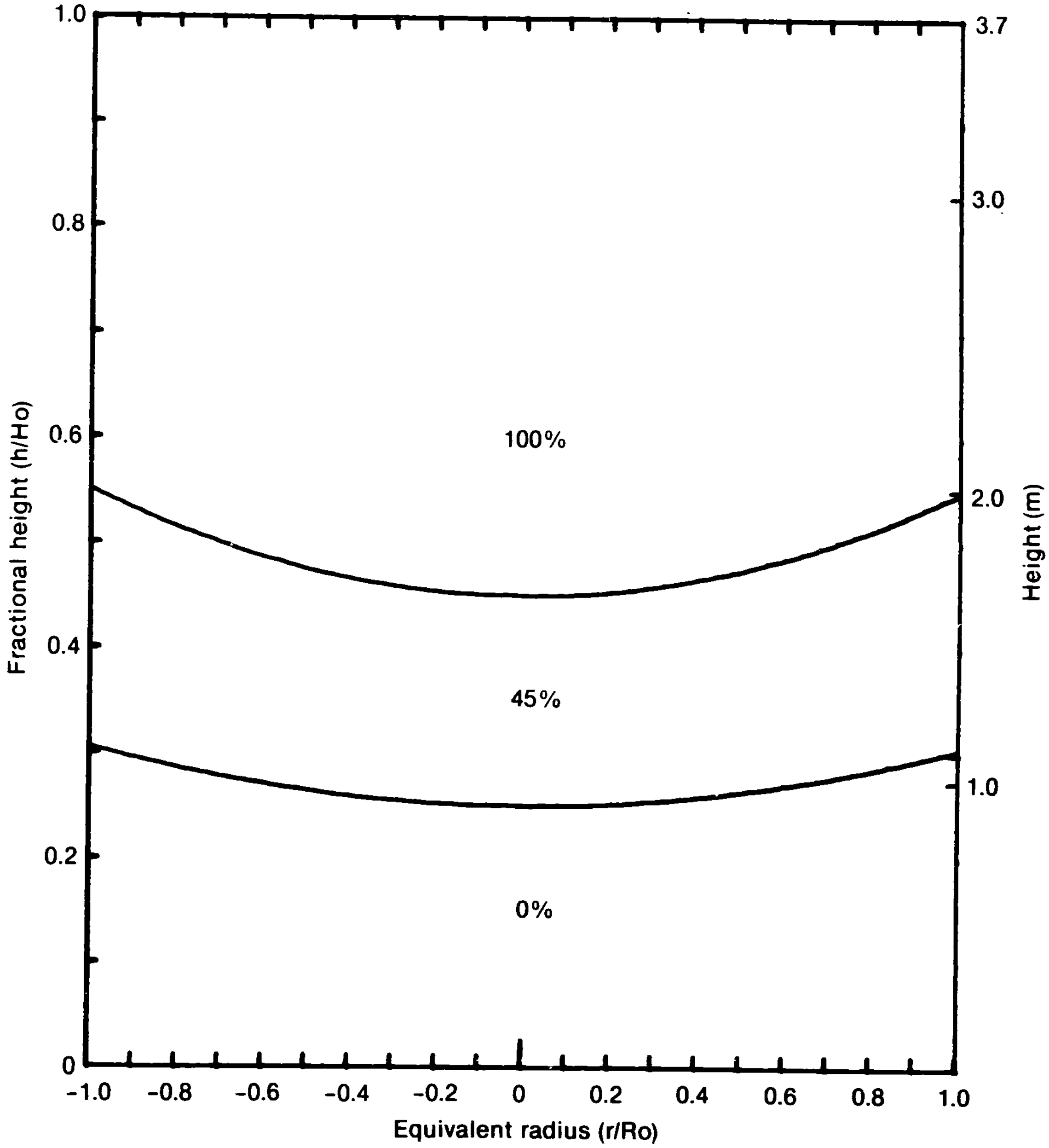

Figure 8. Regional average oxidation and estimated maximum damage configuration of the TMI-2 core.

greater than the 17% embrittlement criterion, but not fully oxidized. For this region, an average oxidation of 45% of the cladding was assumed. From this region to the top of the active core, 100% oxidation was assumed.

The minimum damage estimate illustrated in Figure 6 was based on 40% cladding oxidation, and partially embrittled peripheral rods. Although the minimum coolant mixture level was estimated to be 1.8 m (5.9 ft), this criterion cannot be met without either reducing the amount of cladding oxidation or allowing the damage to be more uniformly distributed across the core. Based on the core flow area blockage measurements discussed in Section 2.2 and the isotopic ratios of uranium and plutonium found in the reactor building sump mentioned in Section 4.1, the core damage is expected to be more uniform.

Figure 7 illustrates the reference core. Embrittlement of the peripheral rods, 50% cladding oxidation, and minimum embrittlement level of 1.4 m (4.6 ft) was assumed. The maximum damage estimate is illustrated by Figure 8 in which fairly uniform damage, 60% oxidation, and a minimum embrittlement level of 1.1 m (3.6 ft) was considered.

As evidenced by the highly qualitative discussion in Section 2.2 regarding estimated plenum temperatures, few definitive comments can be made regarding the condition of the plenum. If the calculated temperatures on which the damage estimates were based are actually lower, the upper plenum may remain fully intact. However, melted control rod guide tube brazements, and partially molten or fused stainless steel components would characterize an estimate of maximum damage to the upper plenum structures. For instance, fuel assembly upper end fittings could be fused to the upper core support plate and control rod spiders could be fused to their male coupling pins. It is also likely that some components may rest on top of the core debris.

5.2 Tooling Development for Defueling

There are several primary core damage parameters that will affect the development of tooling for reactor disassembly and defueling. They are: a) the condition of the upper plenum assembly; (b) the extent of cladding embrittlement, (c) the presence of previously liquified or molten materials; and (d) the condition of the instrument and guide tubes. This section concentrates on these four parameters.

As a contingency, tooling and procedures capable of handling the estimated maximum amount of damage to the upper plenum assembly components should be developed. In the event that normal control rod disconnect procedures are unsuccessful at all positions, tooling and techniques must be developed to remove the remaining control rod drive mechanisms or to sever the leadscrews either above or below the reactor vessel head. The vessel head should be removed without significant problems. Techniques for identifying and separating fused fuel assembly upper end fittings from the upper core support plate should be devised. As a last resort, major cutting may be required.

As a result of the significant cladding oxidation and fuel rod fragmentation which occurred in the TMI-2 core, investigators have suggested that a debris bed is being supported by portions of rods which are embrittled to varying degrees. The upper portion of these rod stubs may be brittle enough to fracture while the debris bed above them is being disturbed or removed, perhaps by the use of a vacuum technique. Once the bed is removed, attempts to handle them during core defueling are expected to fragment the rods further.

Based on the assessments reviewed in Section 2, the presence of previously liquified or molten material cannot be discounted. Because of this, contingency tooling should be developed. The minimum damage estimates state that localized areas of liquified fuel are possible. In the worst case, liquified fuel is expected to be present over most of the core radius, perhaps extending downward to approximately 1 m (approximately 3 ft) above the bottom of the core. In essentially all of the damage assessments, the UO_2 fuel is not expected to have become molten during the course of the accident. Nonfuel core components such as unoxidized zircaloy cladding, spacer grids, stainless steel cladding, and the control rod poison material are expected to have melted over large regions of the core. The once-liquid material may span a height c˘ 2 m (6.5 ft). Cementation of the debris into larger masses by these materials will probably necessitate the use of saws or other cutting tools to separate fuel assemblies or reduce the size of debris masses before removal and packaging.

Since damage to the instrument and guide tubes also cannot be discounted, contingency tooling should be developed for removing fuel assemblies having severely damaged tubes. For those assemblies that have intact, structurally sound tubes and upper end fittings, normal or slightly modified procedures could be used for fuel assembly removal. For assemblies having failed tubes, whose upper portions are embrittled, the tubes may still be used for withdrawing the assembly once the debris is removed. A tool, perhaps an expanding mandrel, could be inserted into the tube down to a location below the embrittled region, however, the presence of cemented debris may preclude insertion of such a tool. Alternatively, a can with grappling fingers at the bottom could lift the assembly from the bottom.

Table 11 summarizes the types of materials expected to be present in the TMI-2 core.

5.3 Conclusions

The minimum and maximum bounds of damage have been identified and a reference description of the TMI-2 core has been defined. These damage limits have been established by reviewing core damage assessments based on reconstruction of the thermal-hydraulic events, determinations of the

TABLE 11. SUMMARY OF POTENTIAL RANGE OF CORE CONDITIONS TO BE USED IN PLANNING OF CONTINGENCY TOOLING

Material	Operation	Quantitative Assessment
Upper plenum components	Disassembly, removal	Potential for substantial melting or fusing to fuel assembly upper end fittings in central region
Loose core debris	Removal	32 000 to 58 000 kg[a] (70 000 to 128 000 lb)
Fused core debris	Separation, removal	Length of fused region, 0 to 2 m (0 to 6.5 ft)
Embrittled cladding	Removal	13 000 to 32 000 kg[a] (29 000 to 70 000 lb)
Structurally intact rods	Removal	32 000 to 51 000 kg (70 000 to 113 000 lb) length, 0 to 3.89 m (0 to 12.8 ft)

a. Total weight of potential debris and embrittled cladding is 64 000 to 83 000 kg (140 000 to 184 000 lb).

hydrogen generated, and fission product release data. The information contained in Figures 6, 7, and 8 and in Tables 10 and 11 should be considered in planning contingency tooling and procedures for reactor disassembly, defueling, core inspection, and fuel sample acquisition.

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TABLE A-1. FUEL ASSEMBLY COMPONENTS, MATERIALS AND DIMENSIONS[A-1]

Item	Material	Dimensions, in.
Fuel Rod (208 rods/assy)		
Fuel	UO_2 dish-end, sintered pellets (92.5% TD), 2.57 wt% ^{235}U (core average, first cycle)	0.370 diameter
Cladding	Zircaloy-4	0.430 OD x 0.377 ID x 153.125 long
Fuel rod pitch	--	0.568
Active fuel length	--	144
Nom. fuel-cladding gap (BOL)[a]	--	0.007
Ceramic spacer	ZrO_2	0.366 OD
Fuel Assembly		
Number Pitch	--	177 8.587
Overall length	--	165.625
Control rod guide tubes (16)	Zircaloy-4	0.530 OD x 0.498 ID
Instrument tube (1)	Zircaloy-4	0.493 OD x 0.441 ID
End fittings (2)	304 SS (castings)	--
Spacer grids (8)	Inconel-718	--
Spacer sleeves (7)	Zircaloy-4	0.554 OD x 0.502 ID

a. Beginning of life.

TABLE A-2. CONTROL ROD ASSEMBLY DATA[A-1]

Item	Data
Number of full-length CRAs	61
Number of control rods per assembly	16
Outside diameter of control rod, mm (in.)	11.18 (0.440)
Cladding thickness, mm (in.)	0.533 (0.021)
Cladding material	304 SS, cold-worked
End plug material	304 SS, annealed
Spider material	SS grade CF-3M
Poison material	80% Ag, 15% In, 5% Cd
Female coupling material	304 SS, annealed
Length of poison section, m (in.)	3.40 (134)
Stroke of control rod, m (in.)	3.53 (139)

TABLE A-3. AXIAL POWER SHAPING ROD ASSEMBLY DATA[A-1]

Item	Data
Number of APSRAs	8
Number of APSR/assy	16
Outside diameter of APSR, mm (in.)	11.18 (0.440)
Cladding thickness, mm (in.)	0.533 (0.021)
Cladding material	304 SS, cold-worked
Plug material	304 SS, annealed
Poison material	80% Ag, 15% In, 5% Cd
Spider material	SS, grade CF-3M
Female coupling material	304 SS, annealed
Length of poison section, m (in.)	0.91 (36)
Stroke of APSR, m (in.)	3.53 (139)

TABLE A-4. BURNABLE POISON ROD ASSEMBLY DATA[A-1]

Item	Data
Number BPRAs	68
Number of burnable poison rods per assembly	16
Outside diameter of burnable poison rod, mm (in.)	10.92 (0.430)
Cladding thickness, mm (in.)	0.889 (0.035)
Cladding material	Zircaloy-4, cold-worked
End cap material	Zircaloy-4, annealed
Poison material	Al_2O_3-B_4C
Length of poison section, m (in.)	3.20 (126)
Spider material	SS, grade CF-3M
Coupling mechanism material	304SS, annealed and 17-4PH, condition H1100

TABLE A-5. ORIFICE ROD ASSEMBLY DATA[A-1]

Item	Data
Number of orifice rod assemblies	97
Number of orifice rods per assembly	16
Outside diameter of orifice rod, mm (in.)	12.19 (0.480)
Orifice rod material	304 SS, annealed
Spider material	SS, grade CF-3M
Coupling mechanism material	304 SS, annealed, and 17-4 PH, condition H1100

TABLE A-6. CORE DESIGN DATA[A-1]

A.	Reactor	
	1. Design heat output (MWt)	2772
	2. Vessel coolant inlet temperature, K (°F)	565 (557)
	3. Vessel coolant outlet temperature K (°F)	592.8 (607.7)
	4. Core coolant outlet temperature K (°F)	594.4 (610.6)
	5. Core operating pressure, MPag (psig)	15.1 (2185)
B.	Core and Fuel Assemblies	
	1. Total fuel assemblies in core	177
	2. Fuel rods per fuel assembly	208
	3. Control rod guide tubes per assembly	16
	4. In-core instr. positions per fuel assembly	1
C.	Fuel Assembly Volume Fractions	
	Fuel	0.303
	Moderator	0.580
	Zircaloy	0.102
	Stainless steel	0.003
	Void	0.012
		1.000
D.	Total UO_2 Beginning of Life	
	UO_2, first core (metric tons)	93.1
E.	Core Dimensions	
	Equivalent diameter, m (in.)	3.27 (128.9)
	Active height, m (in.)	3.66 (144.0)

TABLE A-7. MELTING POINTS OF CORE MATERIALS[A-2]

Material	Melting Temperature (K)	Melting Temperature (^oF)
UO_2	3078	5080
Zircaloy-4 (Fuel Rod Cladding and Guide Tubes)	2123	3362
ZrO_2 (By-Product Metal-Water Reaction)	2988	4919
Inconel 718 (Spacer Grid) 1559	1533-1559	2300-2346
Ag-In-Cd (Control Rod Poison)	1060	1472
304 SS (Cladding of Control Rods and Axial Power Shape Rods)	1672-1694	2550-2590
SS, grade CF-3M	1698	2600
Al_2O_3-B_4C (Axial Power Shape Rods)	2303	3686
UO_2-Gd_2O_3 (2 fuel assemblies contained gadolinia test rods)	3023	4982

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APPENDIX B
ZIRCALOY OXIDATION

Oxidation of zirconium in steam proceeds according to the exothermic reaction

$$2H_2O + Zr \rightarrow ZrO_2 + 2H_2\uparrow$$

such that two moles of hydrogen are produced for every mole of zirconium consumed to become zirconium dioxide. Based on the calculated mass of hydrogen produced, the amount of zirconium oxidized can be calculated as follows:

$$Zr\ (kg) = \frac{H\ (kg)}{2\ kgH/mole\ H_2} \times \frac{1\ mole\ Zr}{2\ moles\ H_2} \times \frac{91.22\ kg\ Zr}{1\ mole\ Zr}$$

$$Zr\ (kg) = 22.805 \times H\ (kg)$$

Since zircaloy (Zry) is 98% zirconium,[B-1] the amount of zirconium oxidized for each kilogram of hydrogen produced is

$$Zry\ (kg) = Zr\ (kg)/0.98 = 23.27\ H\ (kg).$$

This equation was used to obtain the values in Table 3 of cladding oxidized. Dividing the amount of zircaloy oxidized by the amount of zircaloy in the core, 23 920 kg,[B-2] from Appendix C, yields the fraction of zircaloy oxidized.

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APPENDIX C

ZIRCALOY INVENTORY AND DISTRIBUTION IN TMI-2

From Reference C-1, the preaccident zircaloy inventory in the core of 177 assemblies is distributed as follows:

Component	Mass kg	Mass lbm
Fuel rod cladding	20 411	45 000
Upper and lower end plugs	635	1 400
Spacer discs (inside fuel rod near top)	318	700
Assembly guide tubes	1 678	3 700
Instrument tubes	318	700
Burnable poison cladding	562	1 240
TOTAL	23 922 kg	52 740 lb

The fuel rod cladding, assembly guide tubes, and instrument tubes are 3.89 m (153.125 in.) long; the fueled length of the rods is 3.66 m (144 in.). Thus, the amount of zircaloy in the active region of the core is the ratio of the lengths multiplied by the weight of those three components and added to the weight of the burnable poison cladding. The end plugs and spacer discs are considered to be out of the core.

Performing the calculation yields a value of 21 634 kg (47 703 lbm) for the weight of zircaloy in the active region of the core. This is 90.4% of the total amount of zircaloy in the core.

REFERENCE

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APPENDIX D

CONSTRUCTION OF CORE DAMAGE LIMIT ILLUSTRATIONS

APPENDIX D

CONSTRUCTION OF CORE DAMAGE LIMIT ILLUSTRATIONS

This appendix briefly outlines the manner in which Figures 6, 7, and 8, illustrating the minimum, reference, and maximum damage estimates, were constructed.

Four assumptions were common to all the damage estimates. First, a general parabolic shape was chosen for the interface between different oxidation regions of the core. Second, three types of regions were assumed to exist: (a) cladding in the upper elevations was assumed to be 100% oxidized; (b) cladding in the central elevations was expected to be embitted (greater than 17% oxidized), but less than 100% oxidized due to uncovery during only part of the accident, so 45% was chosen; and (c) cladding in the lower regions was not uncovered, remaining below the 17% embrittlement criterion, so 0% was chosen. The volume of each region was obtained by integrating the parabolic equation as a solid of revolution about the core centerline. Multiplying the regional volumes by the fractional amount of oxidation within each region yielded the total amount of zircaloy oxidation in the active core region.

Third, it was assumed that the minimum water-steam mixture level controlled the elevation of the zircaloy embrittlement, namely, the location of the boundary between the bottom and central regions of oxidation. Lastly, the estimate of 90% blockage tended to indicate that even the peripheral fuel assemblies were damaged to some extent.

In obtaining Figure 6, the minimum damage estimate, it was specifically assumed that some of the peripheral rods were intact. Oxidation of 100% was chosen to extend across the whole core diameter, but only to a very small depth at the edges of the core. In the remaining estimates, fairly uniform damage was assumed.