CONF

STATUS OF THE TMI-2 CORE:
A REVIEW OF DAMAGE ASSESSMENTS

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ABSTRACT

Assessments of the damage within the core of the Three Mile Island Unit 2 reactor, performed by reconstructing the transient thermal-hydraulic sequence of events, estimating the amount of hydrogen generation, and evaluating the amount of fission products released, are reviewed and summarized. Minimum and maximum bounds of damage to the core are identified.

INTRODUCTION

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 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 that have been made, identify the minimum and maximum bounds of damage, and establish a "reference" description for the current status of the core. This review of damage assessments, described fully in Reference 1, is the basis for the development of contingency tooling and procedures for inspection, sample acquisition, and defueling of the TMI-2 core.

THERMAL-HYDRAULIC EVENTS

Several investigators[2-8] 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, 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 min into the accident. Most, if not all of the damage to the core is believed to have occurred between 100 and 210 min, the period of core uncovery. The purpose of this section is to briefly summarize the results of the thermal-hydraulic

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studies as they relate to the behavior of the core materials during this period and to the core status at present.

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. Below the mixture level, the zircaloy remained relatively cool and retained its mechanical properties. At higher elevations, the zircaloy became 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.[3,5,6,9]

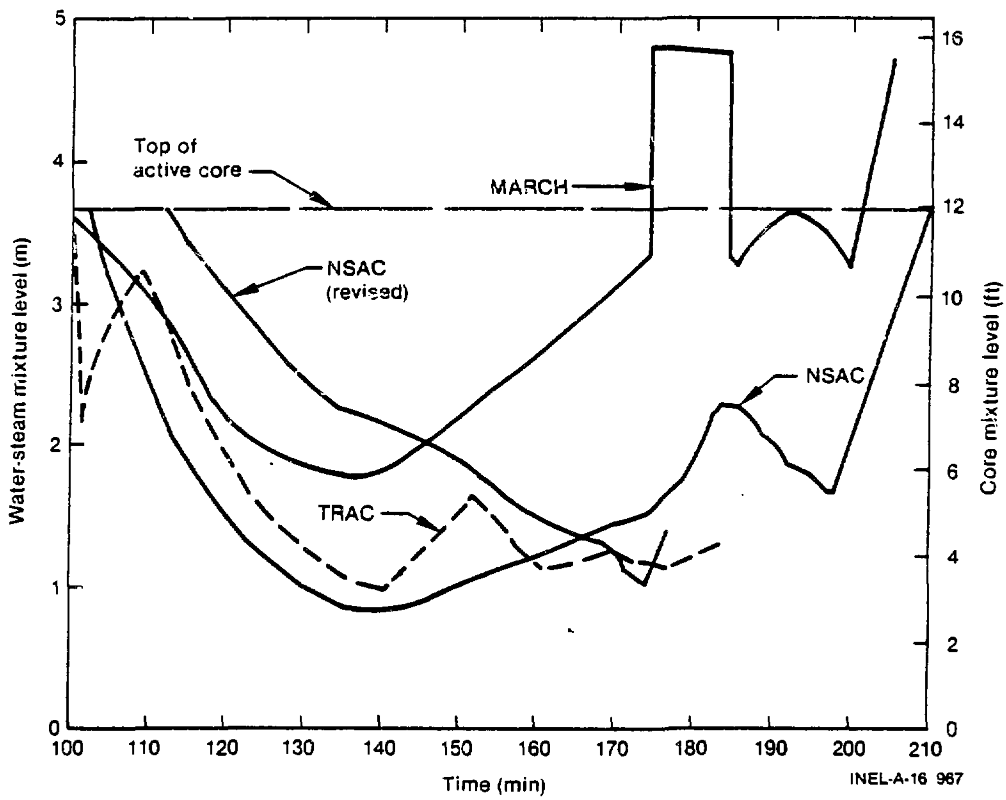

Figure 1. Water-steam and core mixture levels during uncovery from 100 to 210 min.

During the period of core uncovery from about 100 to 174 min, the fuel rods were heating up. As cladding temperatures reached a range of about 1030 K[2] to 1150 K[4], rupture of the rods began to occur. Since virtually all the rods reached temperatures of this magnitude, more 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 min after the start of the accident.[2] This coincides well with an estimated 3-min transport time of the fission products to the containment radiation monitors, which responded at 145 min.[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.[10] Approximately the upper half of the core was embrittled.

A range of fuel rod peak temperatures has been estimated. On the basis of TRAC calculations, peak temperatures were estimated to be 2400 K[8] to 2600 K[11] prior to resumption of high pressure injection flow. Fuel rod plenum temperatures were estimated to be about 1700 K.[11] The Nuclear Regulatory Commission's (NRC's) Special Inquiry Group and Westinghouse-Nuclear Energy Systems estimated slightly higher fuel peak temperatures of 2700 K[4] and 2900 K,[7] respectively. The President's Commission concluded that the fuel temperatures may have exceeded 2475 K throughout 30 to 40% of the core volume, and 2200 K throughout the upper 40 to 50% of the core.[12] These estimates are higher than the estimate by the Nuclear Safety Analysis Center (NSAC)[13] that the gross core average temperatures did not exceed 2000 K. However, a quantitative comparison cannot specifically be made, since the NSAC specified neither the size of the damaged region nor the peak temperatures.

At temperatures above 2175 K, it has been experimentally observed[14] 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.[12,13] The 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 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 nonfueled rods may only have been about 10 K less than that of the fueled rods.[15] However, for the period up to 174 min in the accident, the NRC Special Inquiry Group suggested that "percolation" may have occurred in the instrument and control rod guide tubes.[4]

Evidence suggests that the instrument tubes may not have survived. Figure 2 is a cross section of the instrument string comprised of seven self-powered neutron detectors (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. Prior to the accident, the swagelock fitting on one instrument string was removed. A movable in-core detector was inserted into the central hole of the instrument string which was located in the center of the N-8 assembly at about the midradius of the core, four rows from the center assembly. On April 8, 1979, the detector was inserted with great difficulty from its full-out position to an elevation of about 1 m above the bottom of the core.[16] During removal, the detector stuck about 1 m below the core and is currently immovable. Based on experience, it is believed that "the difficulty encountered during the insertion and withdrawal was caused by fine particles—grit-like particles—inside the movable detector guide tube,

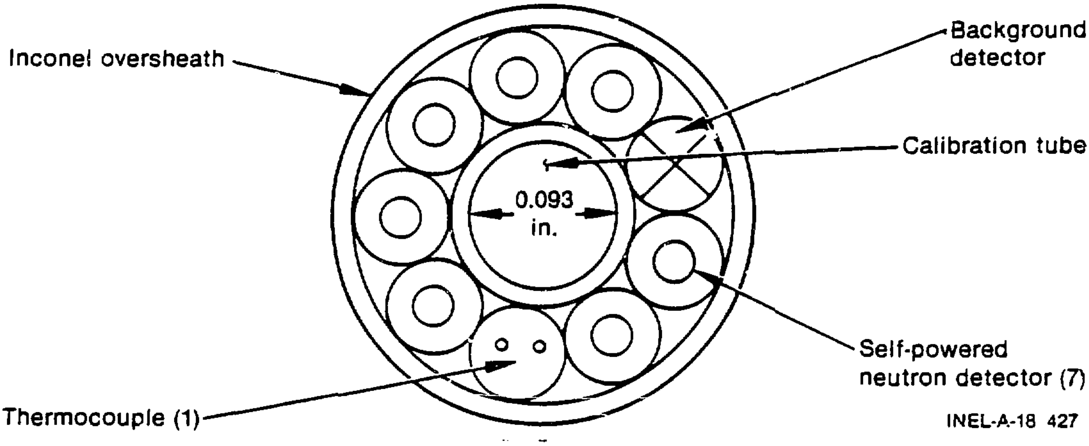

Figure 2. Cross-section of the in-core instrument string.

and that these particles entered the tube during or after the reactor accident."[16] The detector itself is failed, perhaps due to water entering the insulation through a rupture in the detector sheath. If, in fact, this instrument string at the midradius of the core has failed, then it is very possible that a large majority of the instrument strings in the central region of the core have failed. Although early reports indicated that boron crystals are present on the seal table where the in-core detectors terminate, potentially deposited from primary cooling water emanating from a failed instrument string, reports from later entries into the containment building do not confirm this.[17] However, radiation levels at the seal plate drain for several in-core detectors are high in comparison with floor drains in the nearby area, indicating failure of the first two pressure boundaries of the instrument string.[17]

The axial power shaping rods, control rods, burnable poison rods, spacer grids, and guide tubes were also damaged. Since the melting points of the Ag-In-Cd alloy and the 304 stainless steel are about 1075 K and 1675 K, 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.[15] Both materials contributed to formation of the debris bed and fusing of portions of the rubble. Since the zircaloy cladding of the burnable poison rods oxidized over the same region as the cladding on the fuel rods, 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.[15] Since the melting point of Inconel 718 is about 1550 K, 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 at 185 min into the transient, a time when fuel peak temperatures were calculated by TRAC to be 1800 K.[5,11] Calculated fuel rod plenum temperatures were 1200 K.[11] The fuel rod plenum and fuel peak temperatures were extrapolated to about 1900 K and 2600 K,[11] respectively, prior to resumption of high pressure injection flow at about 200 min. 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 and 1800 K. Temperatures in this range would imply, first, that brazements of Beryllium-Nickel 5, which 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. Second, stainless steel components, such as the fuel assembly upper end fittings that have a melting range of 1670 to 1695 K, would have melted or fused at their contact points with the plenum. Control rod spider failure and leadscrew distortion would be likely.

At 174 min, with the coolant mixture level at about 1.5 m above the bottom of the core, one primary coolant pump in the B loop was turned on for 19 min. 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. 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 and 226 min 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 were reached at locations 25 to 75 cm above the bottom of the fuel rods.

Although it is educated speculation, additional core damage apparently occurred during this time. Given the existence of oxidized and embrittled cladding prior to about 225 min, it is possible that "unstable thermalhydraulic 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. On the basis of available instrumentation, no apparent change in the condition of the core occurred after about 226 min.

Independent assessments of the core flow resistance following the accident were made by Babcock & Wilcox (B&W) and Battelle Pacific Northwest Laboratories (PNL).[18] The assessments indicated that a large portion of the core was blocked. B&W made two estimates by comparing reactor coolant system flowmeter readings, with one pump operating, before and after the accident and by 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. Although these determinations are complicated by the potential damage to many of the core exit thermocouples and an estimated decay heat source due to fission product release, 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 $\sim$90% had occurred. Temperatures in the peripheral assemblies indicated that minimum blockage had occurred in these areas. However, since more than 20% of the core flow area is made up of the peripheral assemblies, some blockage at the edge of the core is expected.

HYDROGEN GENERATION

A number of mechanisms in light water reactors may result in the generation of hydrogen. For the TMI-2 accident, the generation of hydrogen by radiolysis and oxidation of UO_2 fuel is expected to have been small when compared with the volume of hydrogen that was produced by oxidation of the zirconium in the cladding by the steam in the reactor vessel.[4] Assessment of the amount of hydrogen generated yields an estimate of the amount of zircaloy oxidized, and hence embrittled, in the reactor core.

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 1. The Electric Power Research Institute (EPRI)[3] 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%[19] to 10%[3] 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.[3] However, if oxygen were depleted by oxidation of core materials other than zircaloy, less oxidation of the zircaloy would result.

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 1 was calculated using the 0.19-MPa pressure pulse in the containment building[3] and the containment atmosphere composition. The amount of free hydrogen in the primary system was calculated on the basis of estimates of bubble size by Metropolitan Edison,[22] and the temperature, pressure, and free volume of the containment building. Hydrogen in solution in the primary system was estimated on the basis of the primary coolant temperature and the hydrogen overpressure.

Table 1, Section A, shows that, except for the calculated inventory based on the April 1-2 samples, estimates of the total amount of hydrogen produced are fairly consistent, averaging 510 kg and ranging between 450 and 582 kg. 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 1 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 knowledge that two moles of hydrogen are produced for each mole of zirconium consumed. The percent of zircaloy oxidation shown in Section C was obtained by dividing the amount oxidized by the inventory, 23 922 kg.[21]

Since Sections B and C of Table 1 are not significantly different, it is concluded that about 50% of the zircaloy in the core (11 961 kg) is oxidized. About 10% of the zircaloy inventory (2288 kg) is in the plenum region of the

TABLE 1. HYDROGEN AND CLADDING INVENTORIES AND CLADDING OXIDATION

	Preburn[a]	March 31[a]	March 31[b]	March 31[c]	March 31[d]	April 1-2[a]	June 1[a]	August 2[a]	Average[e]
A. Hydrogen Inventory (kg)									
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[f]	256[g] to 469[f]	267[g] to 395 ± 30[f]	270	181[f]	490[f]	526[f]	381
4. Remaining in containment	308	65	--	72 ± 4	80	81	23	24	--
5. Remaining in reactor coolant system (6 + 7)	167	111	--	115	100[g]	<42	<1[h]	<1[h]	--
6. In solution	31	33	--	33	26[i]	33	<1	<1	--
7. In bubble	136	78	--	82 ± 20	74[i]	<9[j]	0	0	--
B. Cladding Inventory (kg)	24 040	24 040	--	22 585	24 040	24 040	24 040	24 040	--
1. Cladding Oxidized (kg)	10 818	12 261	--	9 937 to 14 282	~12 020	6 731	11 780	12 501	11 943
(%)	45	51	--	44 to 63	~50	28	49	52	50.6
C. Cladding Oxidized[k] (kg)	11 053	12 752	--	10 565 to 13 543	10 472	7 074	11 938	12 799	11 875
(%)	46	53	--	44 to 57	44	30	50	54	49.6

a. Reference 3.

b. Reference 6.

c. Reference 20.

d. Reference 4. This reference judged these values to be "most likely" after reviewing calculations from several sources, including Reference 20.

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

f. Based on oxygen depletion.

g. Based on hydrogen burn.

h. Hydrogen recombiners had been operating.

i. At 16 h.

j. Bubble virtually gone by this time.

k. Results from this work based on 23 922 kg of zircaloy in the core (see Reference 21).

fuel rods.[21] 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 by 21 634 kg. 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,[11] 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.

FISSION PRODUCT RELEASE

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 the Los Alamos Scientific Laboratory (LASL)[23] using CINDER and EPRI-CINDER, and by B&W[19] 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 contained in Reference 20. In addition, a gas sample was obtained from the containment building atmosphere on March 31, 1979; the results from BAPL's analysis are in Reference 24. 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. 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.

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 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 were compared with the measured ratios in the reactor sump.[4] The measured ratios compared 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[4] 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.

FUEL TEMPERATURES

Estimates of the fuel temperatures can be made on the basis of the types of isotopes released and on their release fractions. Isotopic release fractions are summarized in Table 2. All of the data shown represent a nearly complete

TABLE 2. SUMMARY OF FISSION PRODUCT RELEASE FRACTIONS

	Release Fraction (%)				
	March 28, 1979	August 28, 1979			
Isotope	Reference 19	Reference 25	Reference 4	Reference 20	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 20).

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 20).

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 on the basis of 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.

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 noble gases, halogens, and alkali metals was released to the coolant.[4] A small amount of tellurium and a minute amount of the less volatile isotopes were released. The average values shown in Table 2 would tend to support the higher end of this range.

Investigators drew a variety of conclusions regarding fuel temperatures from these data. On the basis of analyses of the water sample taken on March 29, 1979, BAPL[22] concluded that (a) most of the volatile fission products were released to the coolant (b) 2 to 12% of the fuel reached temperatures of 1900 to 2500 K. From the air sample on March 31, 1979, BAPL[22] concluded that (a) the cladding of about 90% of the 36 816 fuel rods ruptured and (b) about 30% of the fuel exceeded 2200 K.

Considering only the ^{133}Xe release fraction of 57%,[26] the Technical Staff of the President's Commission speculated on the fuel temperatures.[12,20] Lorenz[27] 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. During the thermal transient between 101 and 210 min 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[27] stated that a fuel temperature of 2675 to 2775 K would be required. On the basis of these considerations, the staff concluded that 50%[20] to 66%[12] of the core exceeded temperatures of 2475 K, and that more than 90% of the fuel rods ruptured.[20] 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.[28] 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.[4,19,25]

FUEL MELTING

The occurrence of fuel melting was judged from two primary coolant samples taken on March 28 and April 10, 1979, which showed very little strontium, ruthenium, and tellurium.[4] A sample of the reactor sump was also taken on August 28, 1979. Analysis 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. On the basis of 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.

PARTICLE SIZE DISTRIBUTION

A definitive calculation of the particle size distribution in TMI-2 is not possible; however, several estimates have been made.[4] Consideration was given to the leaching rate of refractory elements from the fuel during the period between March 29 and April 10, 1979. 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.[29] 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 dust-like. Powers[4] stated that particles equivalent to a sphere having a radius of less than 0.3 mm would be levitated by the coolant flow and would have escaped the reactor coolant system to a much greater extent than the remaining particles of larger radii.

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. Table 3 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.

Figure 3 illustrates the reference core. In constructing the figure, consideration was given to the following parameters whose values are summarized in Table 3: (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 estimated minimum water/steam mixture level during the accident. 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, H_o, and the equivalent radius, R_o, are equal to 3.66 m and 1.64 m, 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, 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.

On the basis of 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 extends downward to between 0.9 and 1.8 m 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 range from moderately intact (not fully embrittled) to partially liquified and oxidized. Liquified fuel formed, fusing core components and debris in several areas. As evidenced by the highly qualitative discussion regarding estimated plenum temperatures, few definitive comments can be made regarding the condition of the plenum. If

TABLE 3. 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.

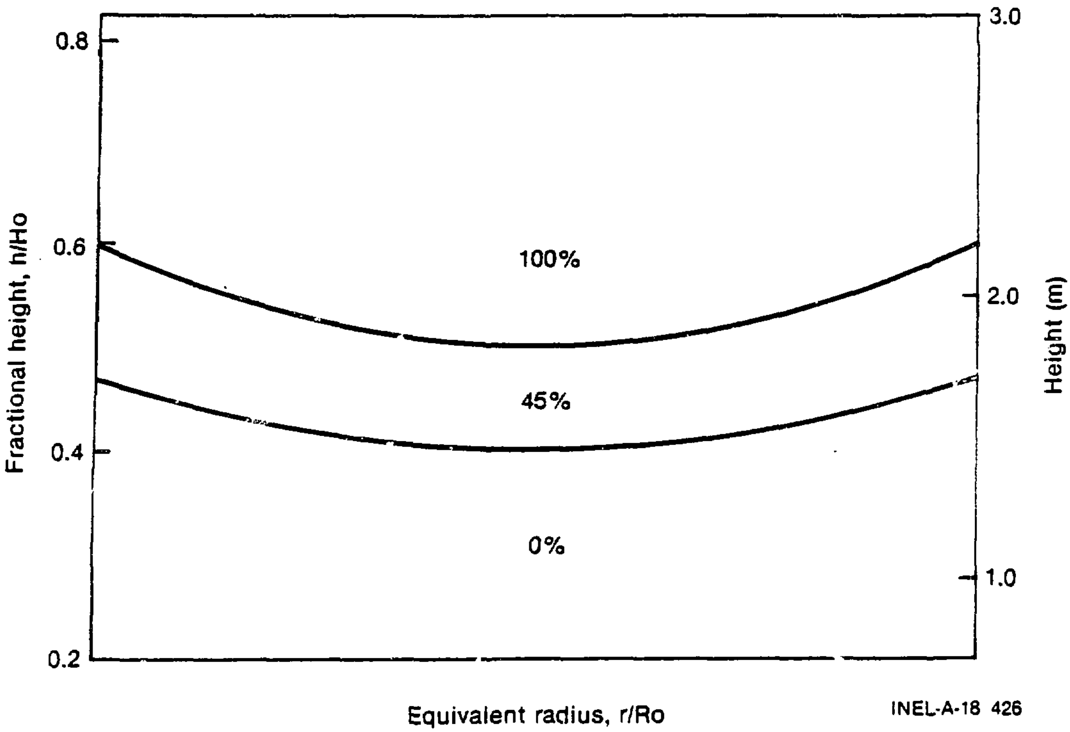

Figure 3. Regional average oxidation and reference configuration of the TMI-2 core.

the calculated temperatures upon 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 couplings. It is also likely that some components may rest on top of the core debris.

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