ACRS SM-0091

PDR 5/2/79

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 27, 1979

M. Bender
M. Carbon
J. Ebersole
W. Kerr
S. Lawroski

D. Moeller
M. Plessset
J. Ray
C. Siess

NRC MEMO ON CORE DAMAGE ASSESSMENT FOR TMI-2

Attached is a copy of an NRC memo on an assessment of TMI core damage. This memo is a follow-up to an April 6, 1979 NRC report that estimated fuel damage based on the earliest available information. Copies of the subject memo have been provided to ACRS members attending the May 8, 1979 Reactor Fuel meeting, under separate cover.

Paul Boehnert

Paul Boehnert
Reactor Engineer

Attachment: as stated

cc: ACRS Technical Staff

7905140154

106 147

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 26, 1979

P. Shewmon
H. Etherington
C. Mark
W. Mathis
D. Okrent

A. Bement
J. Crocker

NRC MEMO ON CORE DAMAGE ASSESSMENT FOR TMI-2

Attached is a revised copy of an NRC memo on TMI core damage that was attached to my April 25, 1979 notice of the May 8 Reactor Fuel Meeting. Some pages were omitted from the earlier copy.

Paul Boehnert

Paul Boehnert
Reactor Engineer

Attachment: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 13 1979

MEMORANDUM FOR: Roger J. Mattson, Director, Division of Systems Safety, NRR

FROM: R. O. Meyer, Reactor Fuel Section Leader, Core Performance Branch, Division of Systems Safety, NRR

SUBJECT: CORE DAMAGE ASSESSMENT FOR TMI-2

Attached is our assessment of the core damage at TMI-2 for use in the SER for natural circulation. It represents our independent evaluation of the facts available and of the industry/vendor/licensee analysis, which we have heard in several briefings.

An earlier estimate of fuel damage was made by Rubenstein et al, and a recent meeting was held at NRC with industry experts. Memoranda describing those evaluations are attached to this document.

Ralph O. Meyer

Ralph O. Meyer, Section Leader
Reactor Fuels
Core Performance Branch
Division of Systems Safety

Attachment:
As stated

cc: R. Tedesco
K. Kniel
P. Check
C. Berlinger
D. Crutchfield (FOIA File)
D. Houston
M. Tokar
V. Stello
B. Grimes
G. Knighton
J. Voglewede
D. Powers
PDR

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Pdr

File: 50-320 106 149

CORE DAMAGE

A. Introduction

For the usual analysis of hypothetical accidents, initial core conditions are assumed and consequences are calculated. This would involve complex thermal-hydraulic calculations and fuel behavior analyses. At Three Mile Island, however, some of the consequences are known (i.e., some information on fission product release, hydrogen generation, and instrument readings is available), so we will use "reverse engineering" as our principal method of backing out an assessment of core damage.

We start with the assumption that the core was uncovered and allowed to heat up for significant periods of time. Figure 1 shows the system pressure history for March 28, which includes three periods of significant uncovery. The periods of uncovery correspond approximately with the major periods when system pressure was below the saturation pressure. We will assume that the first core uncovery began shortly after 92 minutes into the accident at which time excore ion chambers show a response spike corresponding to the loss of water shielding. Although the two later periods of uncovery may have produced additional core damage, we will focus on the first period because decay heat was larger then and because that period produced the large radiation instrument reading (at 150 minutes) in the containment indicating major fuel damage.

Because the fuel damage to be discussed below is so extensive, we will

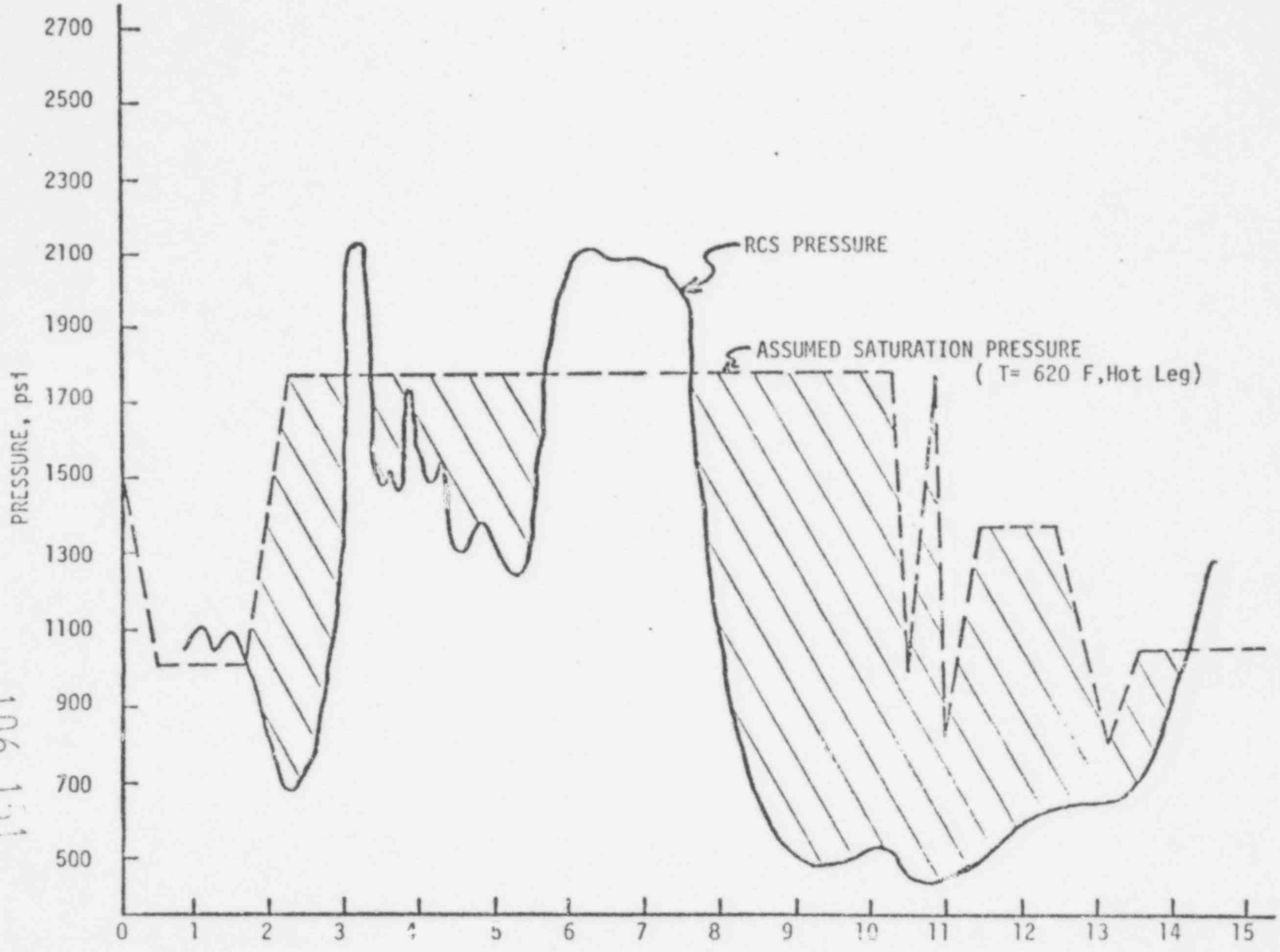

FIGURE 1.

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conclude without demonstration that virtually all of the fuel rods in the core failed in the sense of experiencing defects large enough to release gas. Furthermore, the rods probably failed by a LOCA-like ballooning-and-rupture mechanism. Because of the massive oxidation that followed, the mode of failure is probably immaterial.

As a point of reference, Table I lists melting temperatures of the various materials used in the fuel system.

B. Fuel Rods

Fission product and hydrogen measurements at TMI-2 give important clues about the condition of the fuel rods. We will deal with fission product releases first.

Air and water samples containing fission products have been analyzed. While we have analyzed both for indications of fuel conditions, we have concentrated on the Xe-133 concentration in the air sample. This isotope was selected for analysis for several reasons: (a) it is a noble gas and will not react, plate out or condense, (b) it has a relatively long half life (5.29 days) and a high production rate (6.8 atoms per 100 fissions) and therefore will be abundant thus reducing measurement errors, and (c) fission product release correlations are much better established for noble gases than for other fission products.

Bettis (BAPL) has evaluated the Xe-133 activity and concluded it is equivalent to 31% of the total core inventory. We have independently checked this calculation (but, of course, not the sample activity) and agree (31.5%).

TABLE 1. MELTING TEMPERATURES

MATERIAL	TEMPERATURE, ^{o}C	TEMPERATURE, ^{o}F
UO_2	2805	5080
ZIRC-4	1850	3362
ZrO_2	2715	4919
INCONEL 718	1260-1286	2300-2346
304 SS	1399-1421	2550-2590
Al_2O_3-B_4C	2030	3686
Ag-In-Cd	800	1472
UO_2-Gd_2O_3 *	2750	4982

* Two fuel assemblies contained gadolinia test rods.

Fission products including gases are normally retained by the UO_2 pellets. A normal pellet release to the fuel rod internal voidage is only 1 or 2% (even for a successfully terminated LOCA) so that a 30% release indicates additional release from fuel pellets not just a release of the gap activity.

Fuel pellet releases are strongly dependent on temperature, and Figure 2 shows a correlation of release versus temperature for Xe-133 (from a recent ANS-5.4 draft standard). The correlation, however, is for steady-state releases and we are dealing with a transient. Further errors are possible because of kinetics changes due to oxidation to U_4O_9 or U_3O_8. Nevertheless, it is a reasonable approximation and is consistent with recent short-time annealing experiments (private communication 4-10-79, R. A. Lorenz, ORNL) and earlier annealing work (G. W. Parker et al., ORNL-3981 - See attachment A).

Parker heated irradiated samples in a furnace for 5.5 hours. The samples had burnups ranging from trace to 4000 MWd/t (about the same as TMI-2). Parker measured releases of about 5% at 1600°C, 15% at 1800°C and 40% at 2000°C with an uncertainty of about a factor of 2 in release. These experimental releases for conditions roughly similar to TMI-2, but for different isotopes, are in fair agreement with Figure 2.

Using Figure 2 we could conclude that (a) the fuel was uniformly heated (uniform in axial and radial directions) to about 1750°C, or (b) 30% of the fuel melted while 70% remained below 1200°C, or (c) any intermediate condition existed. Because of the core uncovery sequence, Xe-133 FISSION GAS RELEASE

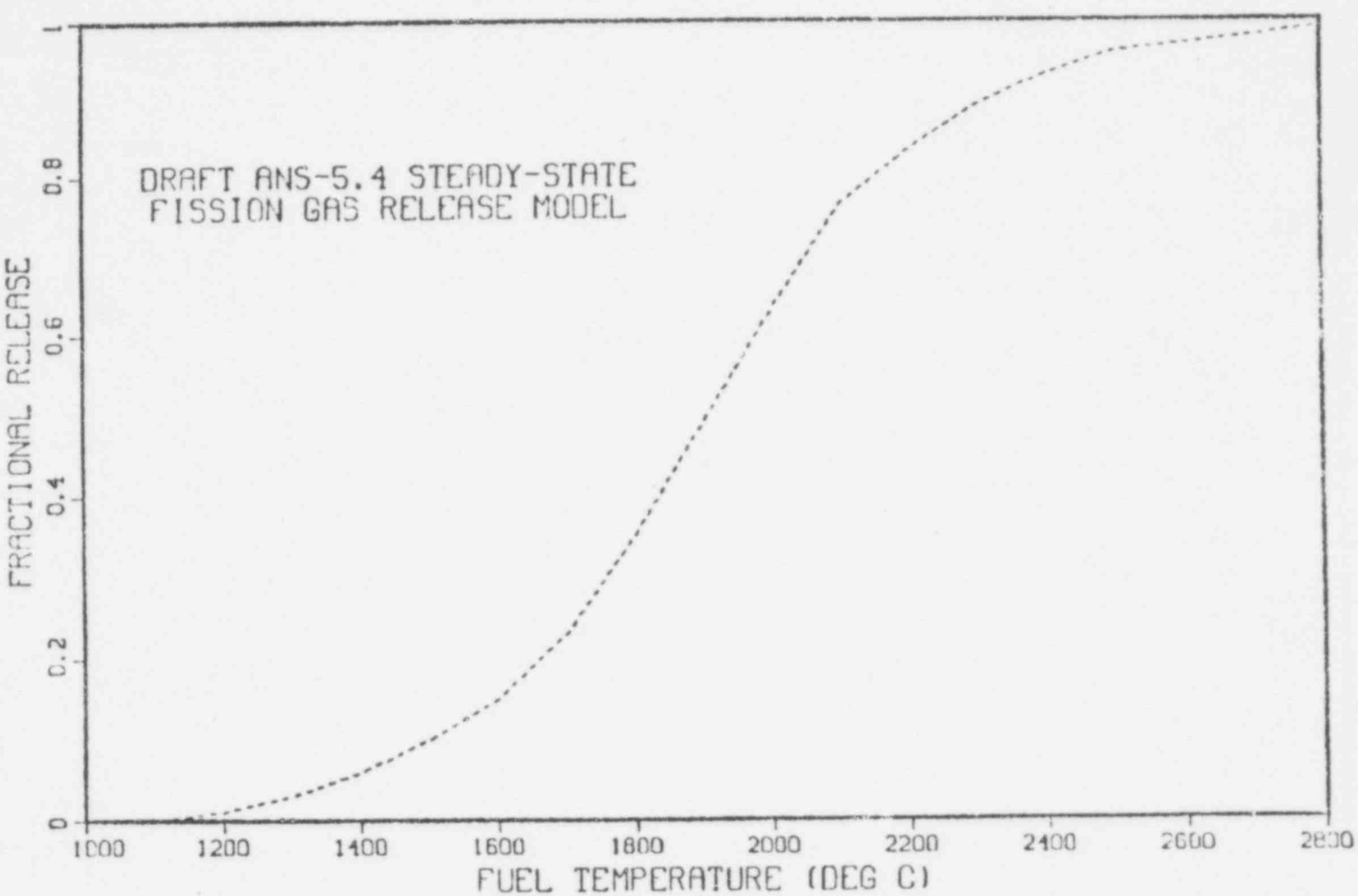

FIGURE 2.

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the fuel rods probably did not heat up uniformly in the axial direction. It is reasonable, however, to treat the fuel rods as isothermal in the radial direction because of the low heat flux. Figure 3 illustrates this point with a comparison of a full-power radial temperature profile and a decay-heat-power temperature profile.

There are physical limits on how hot the fuel can get during the periods of core uncovery because the fuel rods [illegible] a large heat capacity and a low heat generation rate. If one assumes zero heat removal (this would produce the most rapid heatup rate possible) during the first period of uncovery, the heatup rate is still fairly slow. Figure 4 shows the adiabatic temperature increase with time for the peak-power axial location, for the low-power ends of the rods, and for the average location. Since there must have been some heat removal thus further slowing down the temperature rise, pellet temperatures probably did not reach the melting point. Figure 5 shows the temperature changes with time for a surface heat transfer coefficient of 0.5 BTU/hr-ft^2-^{o}F, which is a very small value.

The results on temperature distribution are, therefore, not conclusive. It is unlikely that fuel temperatures were uniform and no lower than 1750^{o}C, and it is also unlikely that any fuel (UO_2) melting took place. The fuel, however, did get very hot compared with its normal operating temperatures.

Oxidation of Zircaloy by steam and the attendant decomposition of water provided the major source of hydrogen in the TMI-2 vessel and containment.

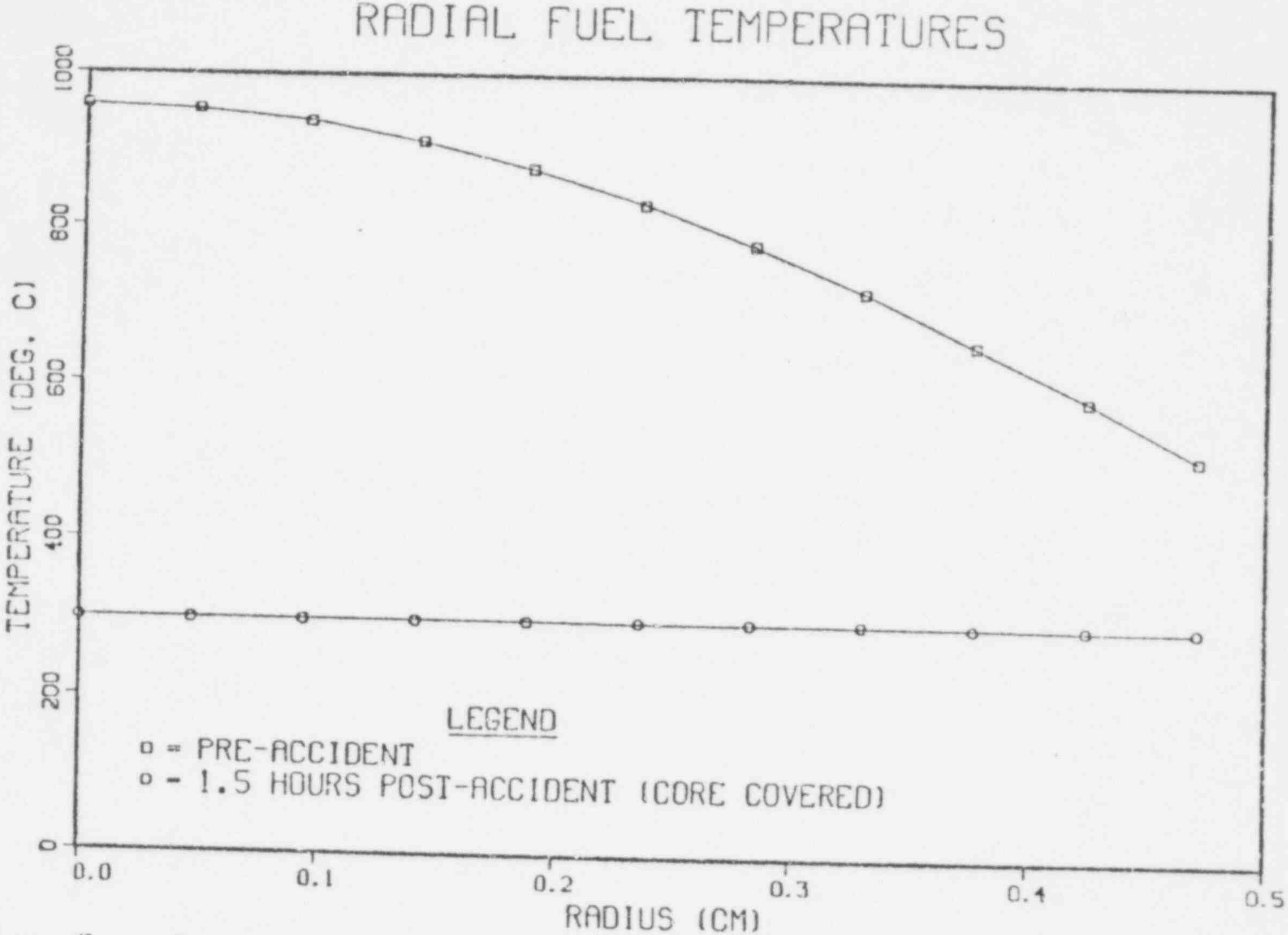

FIGURE 3.

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ADIABATIC HEATUP

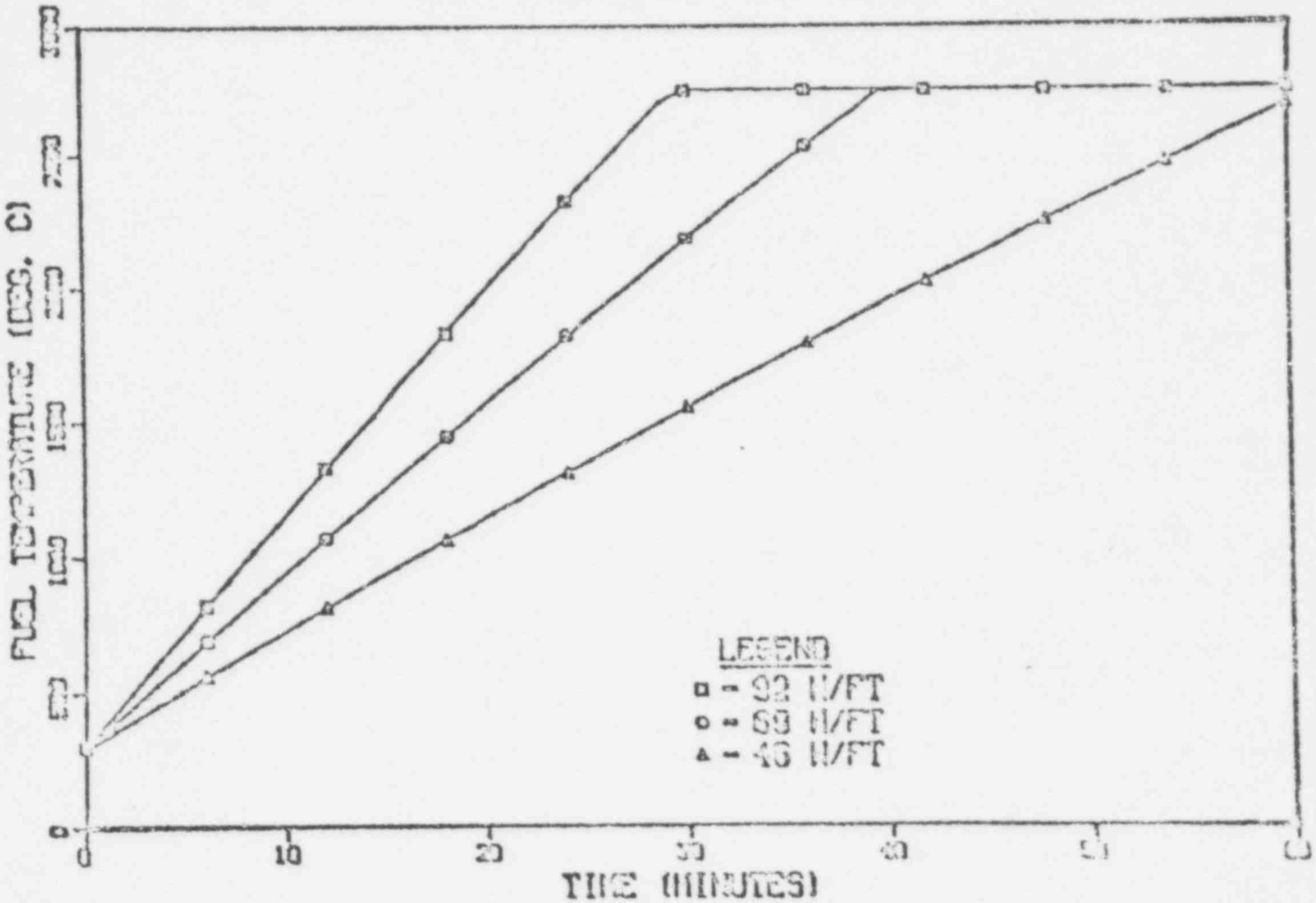

FIGURE 4.

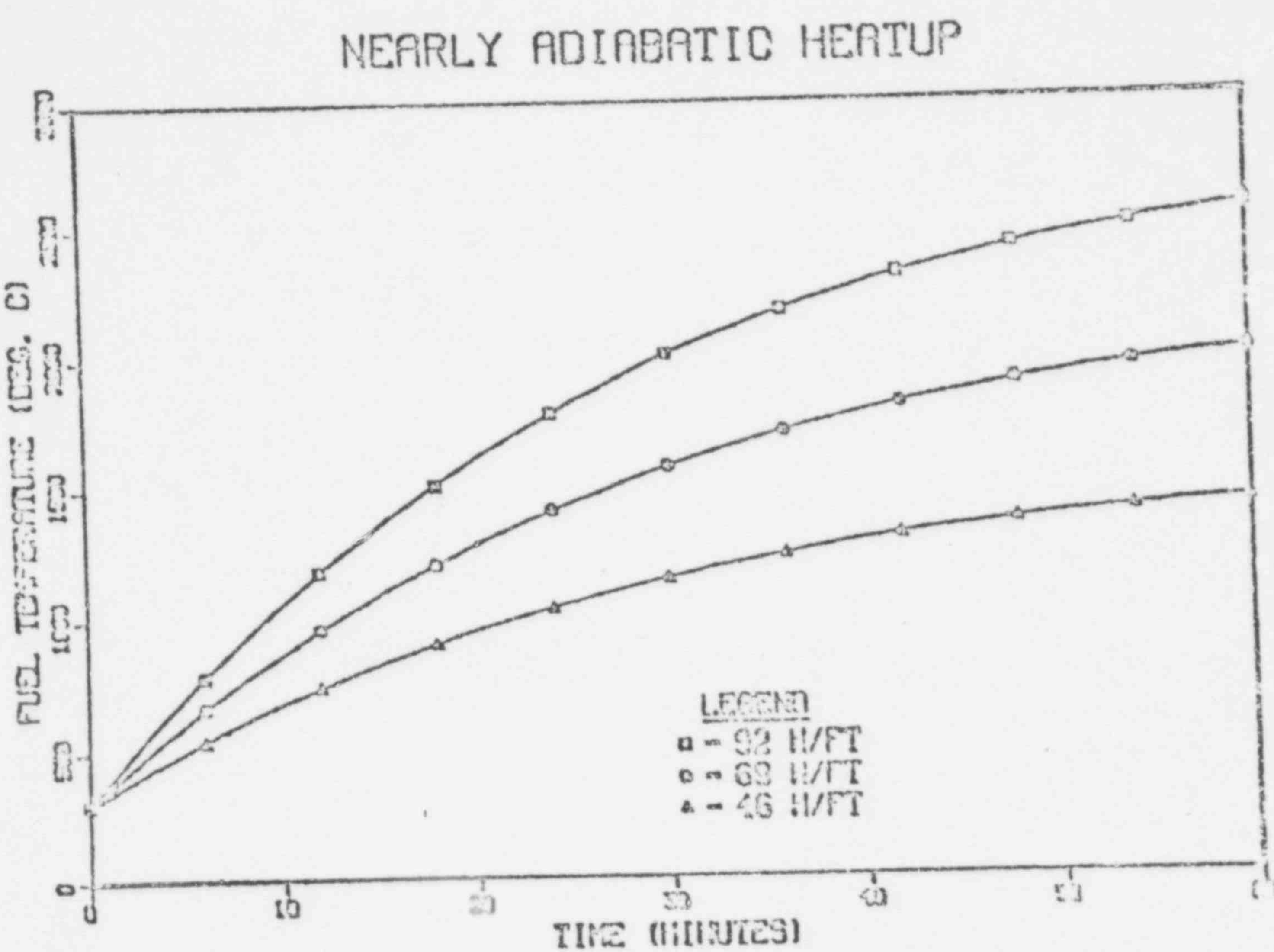

FIGURE 5.

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The Containment Systems Branch has estimated the amount of hydrogen present in the plant (Attachment B) after the periods of core uncovery that caused fuel damage. They included amounts (a) consumed by the hydrogen explosion (226 lb mole), (b) remaining in containment after the explosion (80 lb mole), and (c) in the primary system bubble (76 lb mole), which was corrected for radiolysis.

Comparing the above amounts with the total amount of hydrogen that could have been produced if all of the Zircaloy in the fueled region reacted with water, we get 41%. As with the temperatures, an ambiguity exists. This could mean that (a) about 40% of the cladding wall thickness is uniformly oxidized throughout the core, or (b) 40% of the fueled region of the core has fully oxidized cladding, or (c) any intermediate condition exists.

Figure 6 shows the time required for total wall thickness oxidation as a function of temperature (Cathcart-Pawel correlation). It is clear from Figure 6 that complete oxidation is possible in cladding segments that reached temperatures of around 2000^{o}C during the period of core uncovery. It is also clear from Figure 6 that all of the cladding did not experience sustained temperatures of around 1750^{o}C else it would all have oxidized. This is further evidence that fuel temperatures were not uniform throughout the core, and that temperatures locally were very high.

Based on early estimates by the Analysis Branch of core uncovery, we will assume simplified uncovery histories shown in Figures 7 and 8 for the

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TOTAL OXIDATION TIME

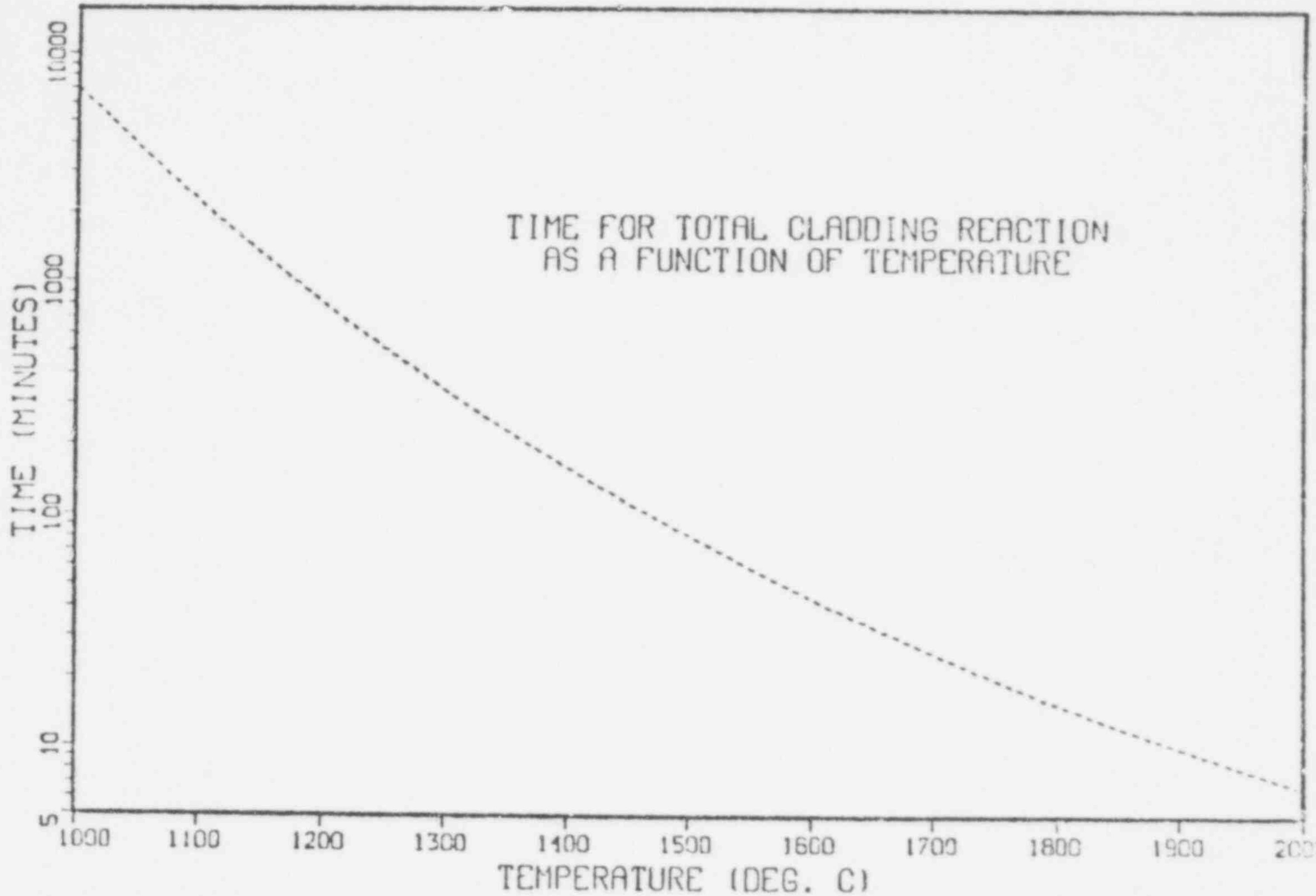

FIGURE 6.

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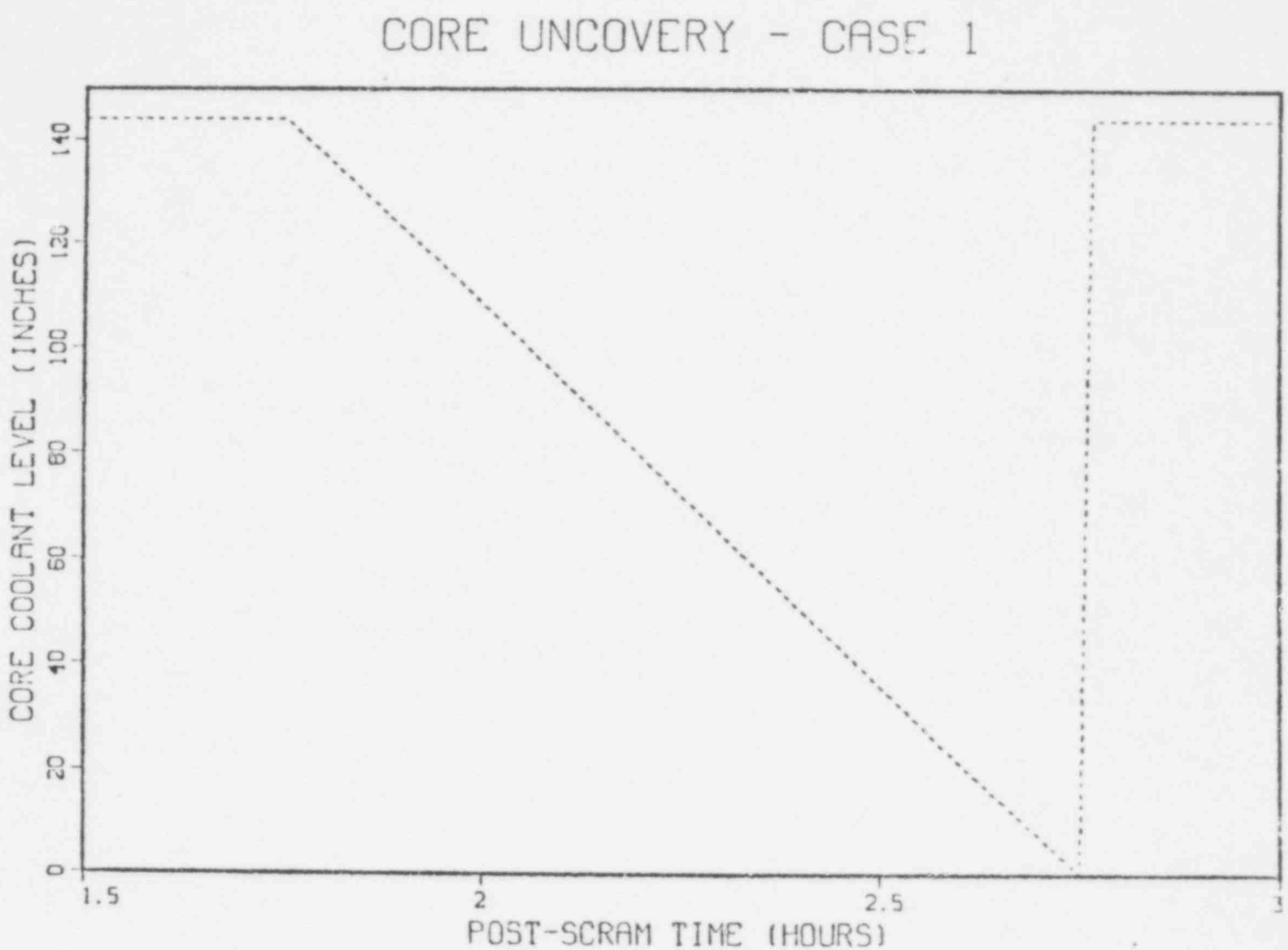

FIGURE 7.

106 162

CORE UNCOVERY - CASE 2

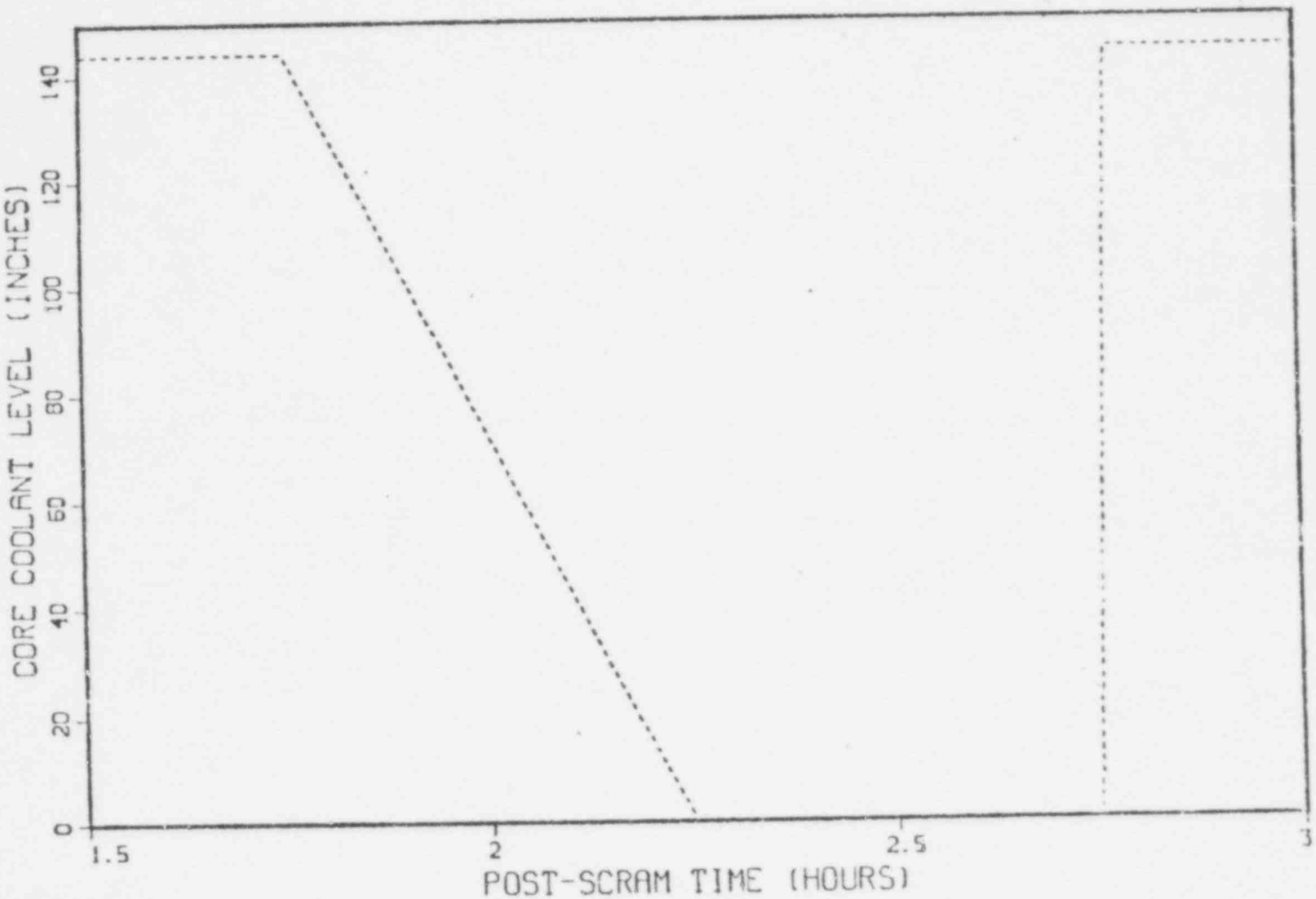

FIGURE 3.

106 163

following calculation. Fuel that is covered will be considered to be cold (i.e., no cladding oxidation). Fuel that is uncovered will be allowed to heat up; fuel that heats up will be given a heat transfer coefficient that is adjusted such that the total integrated oxidation is 40%. These calculations give the oxidation distributions shown in Figures 9 and 10, and these distributions are insensitive to many of the assumptions that were made. Figures 9 and 10 thus are more probable distributions than 100% oxidation over 40% of the core or 40% oxidation over 100% of the core.

Figure 11 is a recent best-estimate embrittlement correlation (Kassner et al., ANL) that shows high-temperature fragmentation of quenched tubes at about 30% oxidation. Using this correlation, Figures 9 and 10 indicate that a fragmented region of about 5 ft. in height exists near the top of the core. It may well be right at the top of the core as a result of simplifications in our analysis. In any event, at least 4 to 6 ft. of intact (but partially oxidized) fuel rods remain standing at the bottom of the core.

Figure 12 shows fragmented Zircaloy cladding after oxidation in a simulated-LOCA test. Kassner (ANL-78-25 and ANL-78-49 reports that at high temperatures (> 1250^{o}C) many fragments are produced whereas at lower temperature the rod may simply break into two pieces. Inasmuch as TMI-2 temperatures were higher than 1250^{o}C and oxidation was severe, small fragments of the size shown in Figure 12 should be expected along with larger tube-like pieces.

CORE OXIDATION - CASE 1

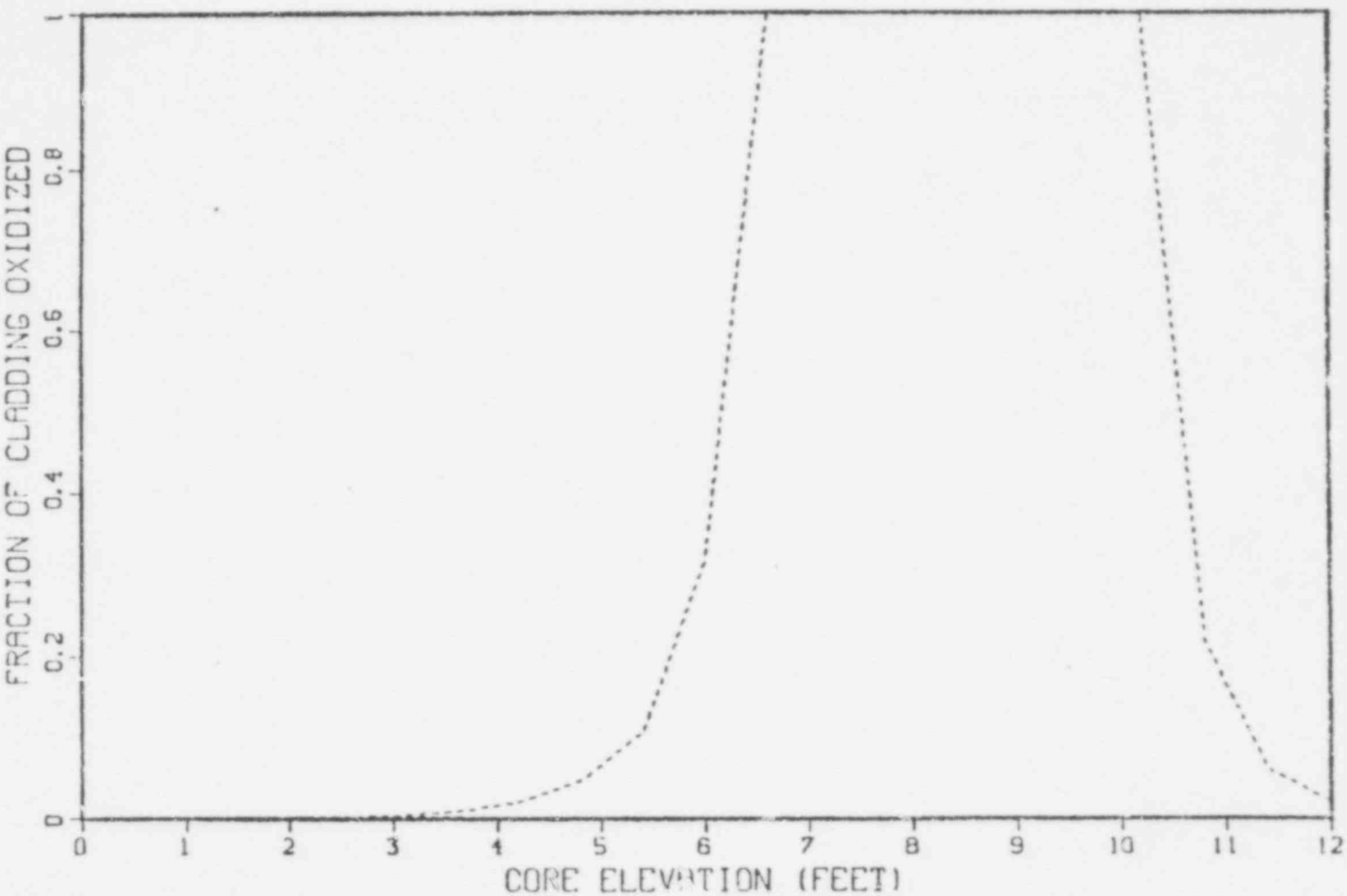

FIGURE 9.

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CORE OXIDATION - CASE 2

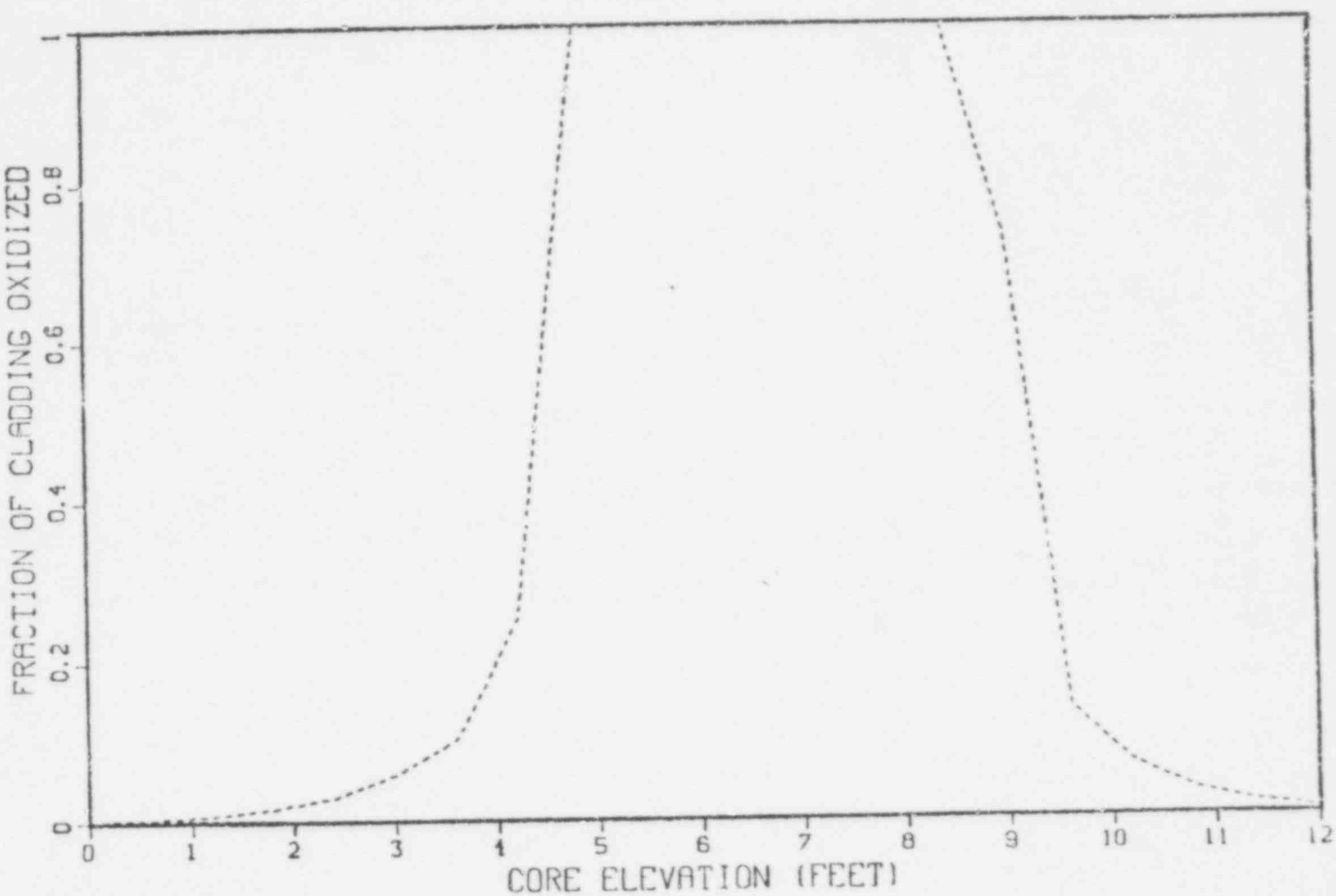

FIGURE 10.

106 166

FAILURE MAP FOR ZIRCALOY-4 CLADDING BY THERMAL SHOCK OR NORMAL HANDLING

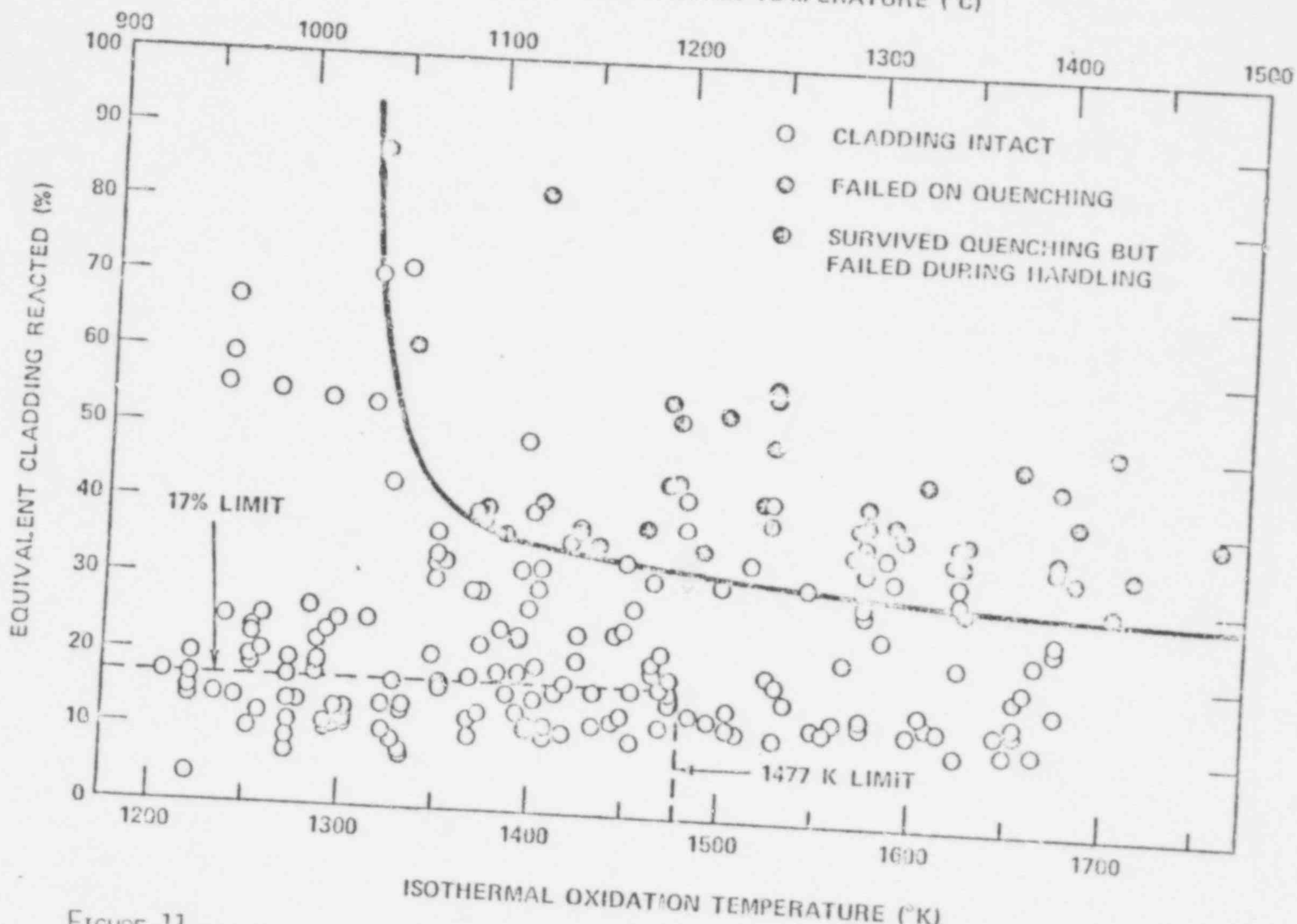

FIGURE 11.

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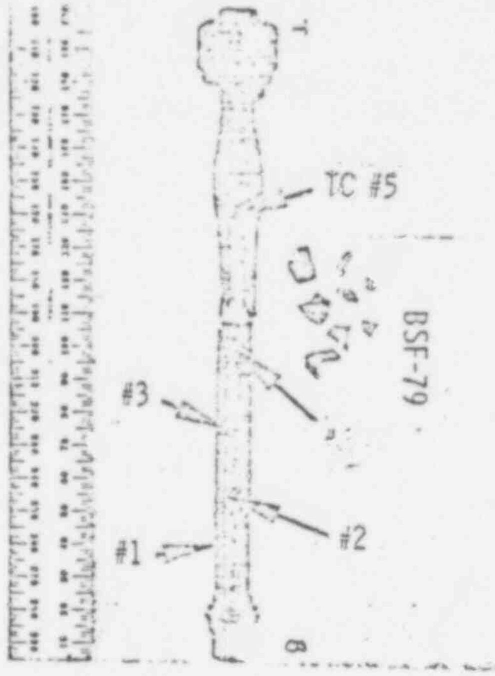


Fig. III.12

Zircaloy-4 Cladding after Thermal-shock Failure Showing Location of Thermocouples That Produced the Temperature-vs-Time Curves in Fig. III.10. ANL Neg. No. 306-78-223.

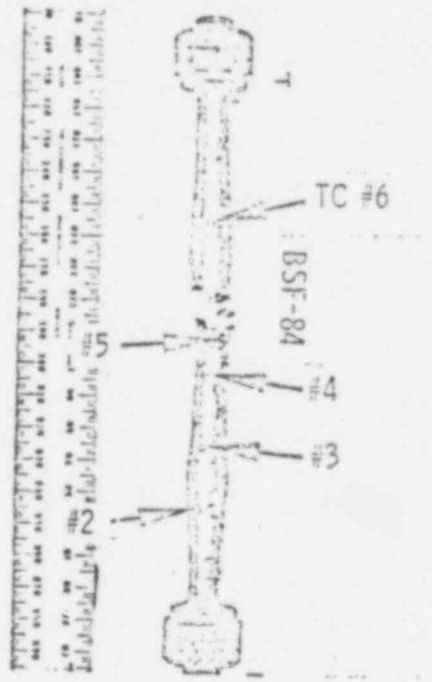


Fig. III.13

Zircaloy-4 Cladding after Thermal-shock Failure Showing Location of Thermocouples That Produced the Temperature-vs-Time Curves in Fig. III.11. ANL Neg. No. 306-78-224.

FIGURE 12.

106 168

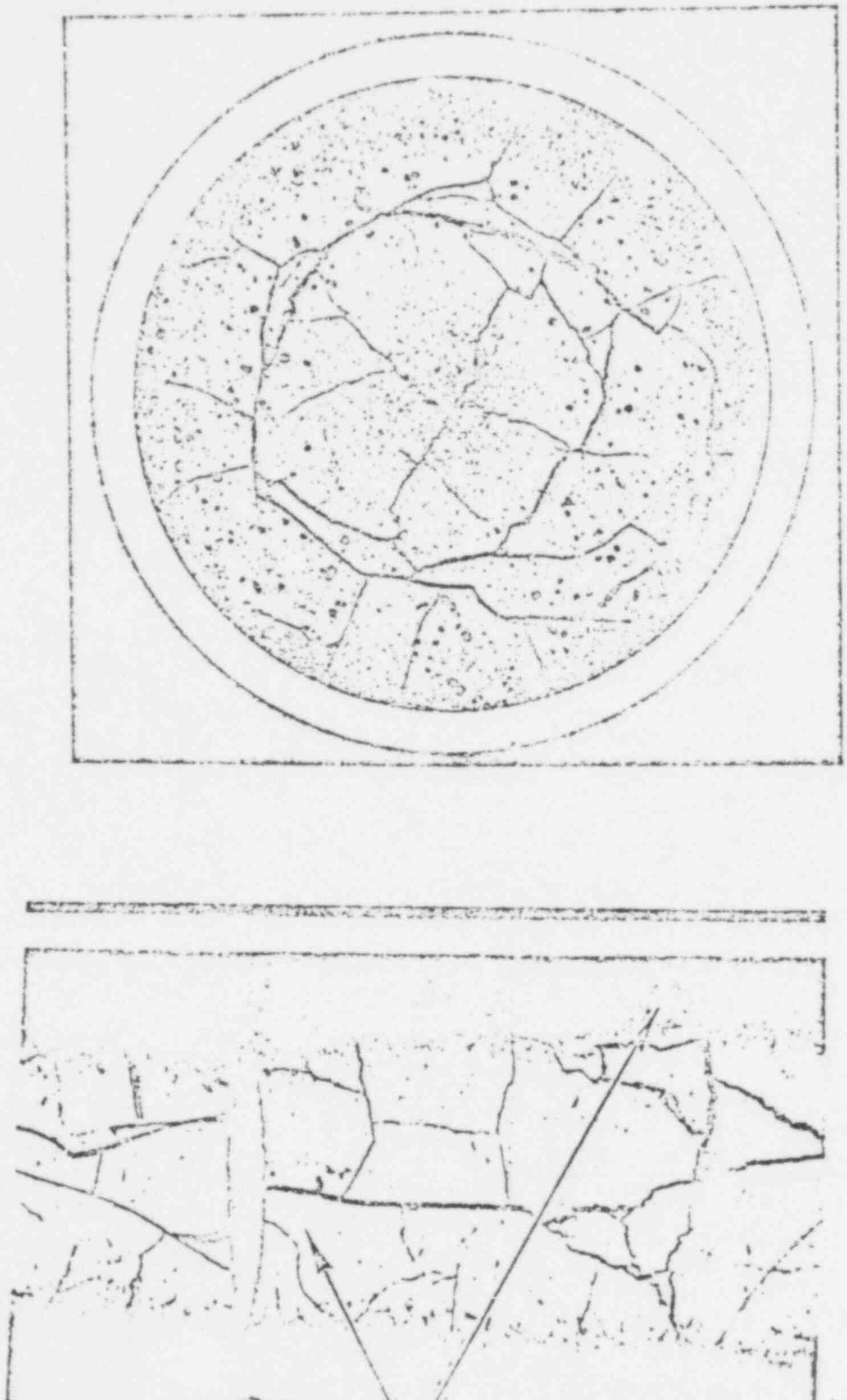
Fuel pellets normally crack during operation and crack healing can occur at power. Figure 13 is a typical example of a cracked pellet. Quenching during core flooding may also promote fragmenting of the pellets. Severely fragmented regions are commonly seen in fuel pellets as a result of extreme temperature conditions in test reactors. Powdered regions in fuel pellets have also been seen in some PBF tests, but these tests are characterized by very high powers (> 20 kw/ft) and very steep temperature gradients unlike the low-power uniform (radial) temperature TMI-2 fuel. Therefore we would expect the TMI-2 fuel to be in millimeter-size granules and larger pieces including whole pellets.

C. Unfueled Components (Control rods, guide tubes, etc.)

Figures 14 through 17 show the control rods, the burnable poison rods, the power shaping rods, and the central instrument tube. All of these rods and the instrument tube are inserted into Zircaloy guide tubes in the fuel assembly. The materials of which these components are made are indicated on the figures.

An important clue about the condition of unfueled components is provided from instrument readings. The fact that all 52 thermocouples worked throughout the accident and most continue to give credible information suggests that a central tubular structural member survivied. It is tempting to conclude that all Zircaloy guide tubes also survived, but this may not be the case since the thermocouple is well protected by multiple barriers.

106 169



106 170

FIGURE 13.

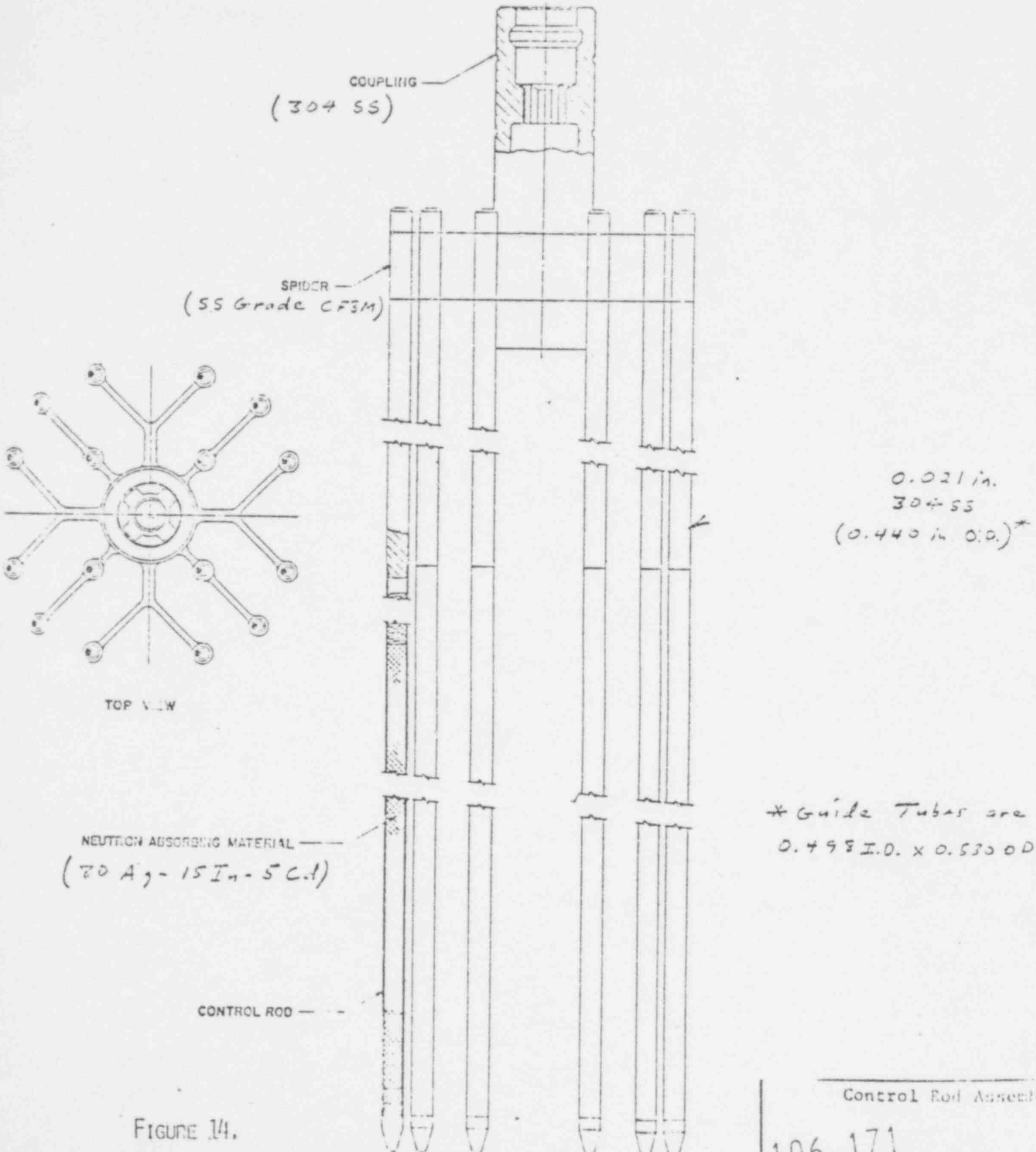


FIGURE 14.

Control Rod Assembly

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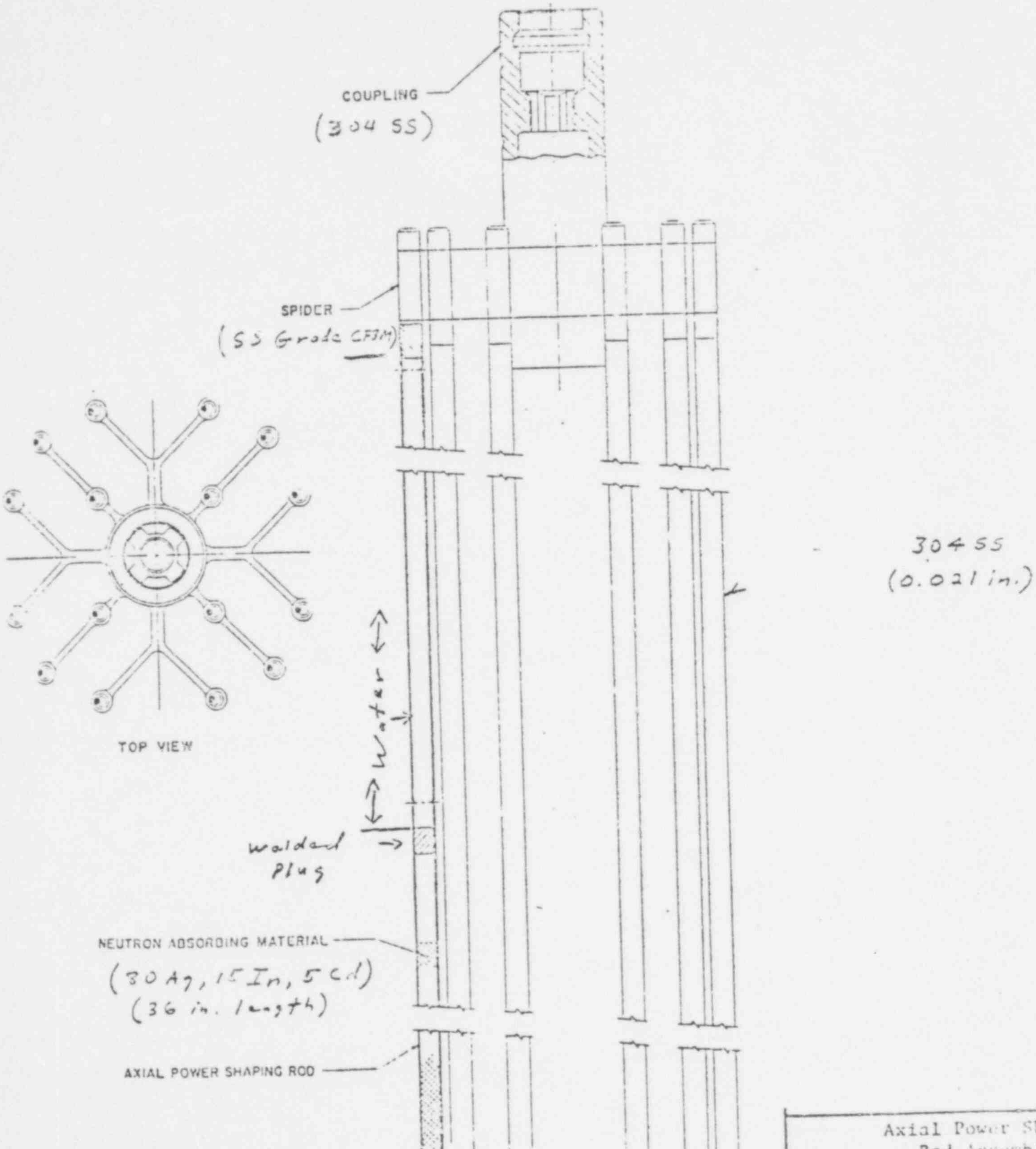

FIGURE 15.

Axial Power Shaping Rod Assembly

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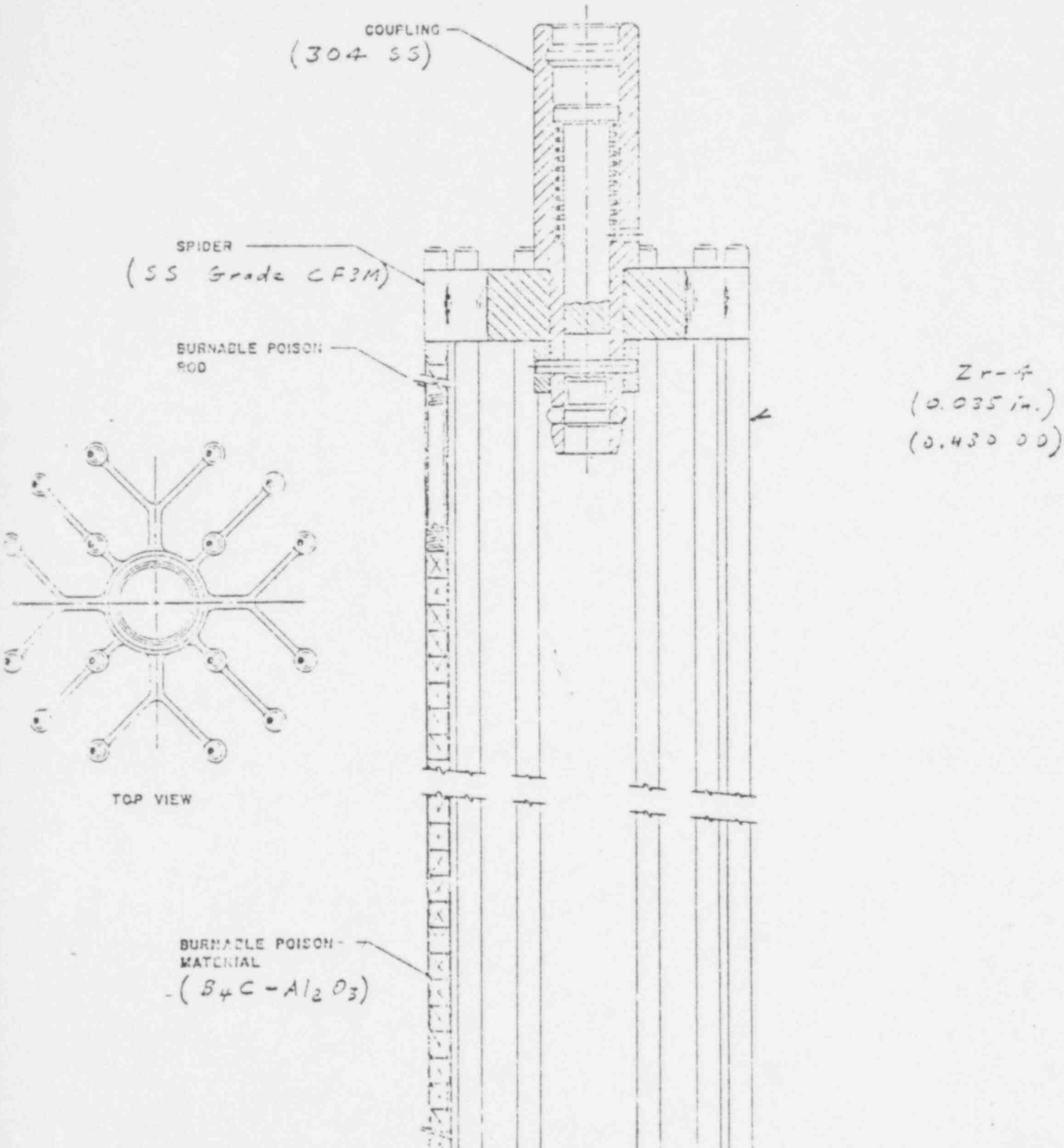

FIGURE 16.

Burnable Poison Rod Assembly

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Fixed SPND assembly cross section

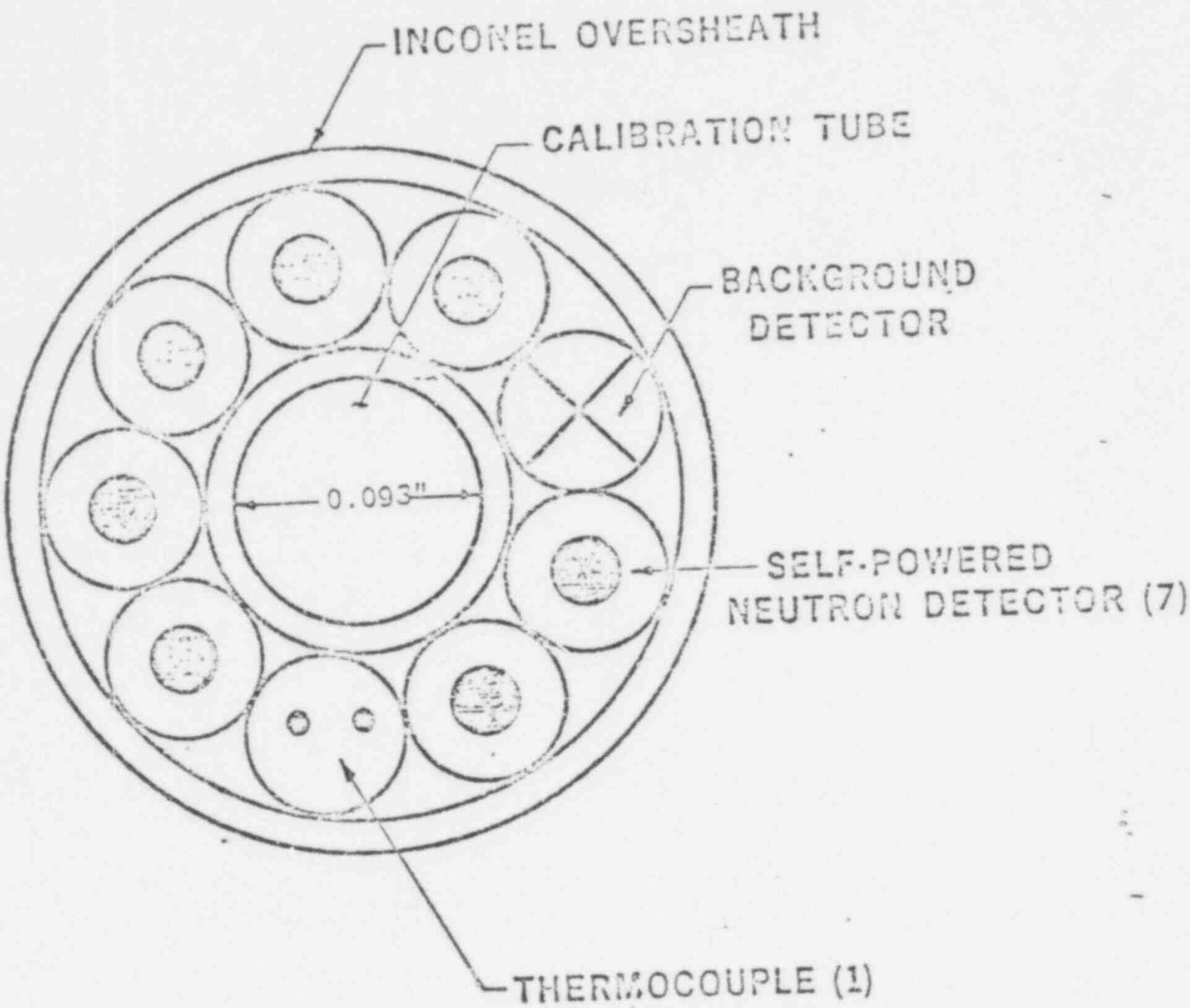

FIGURE 17.

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INEL has made calculations of guide tube temperatures by parametrically varying heat transfer conditions (see Attachment C). Their results show that guide tube temperatures lag the fuel rod temperatures by only about 20^oF. Babcock & Wilcox has performed similar calculations and concluded that there is a much larger spread in temperatures. We believe the INEL calculations are more nearly correct and that temperatures of unfueled components were close to fuel rod temperatures. Since fuel rod temperatures are believed to have exceeded 1750^oC in the hot region of the core, then in that region (a) Ag-In-Cd and its stainless steel cladding would have melted, (b) Inconel spacer grids would have melted, (c) Zircaloy guide tubes would have oxidized, and (d) Zircaloy cladding of the burnable poison rods would have oxidized.

In the cooler parts of the core below about the 4 to 6 ft. elevation, we would expect all unfueled components to be intact, although perhaps damaged, just as the fuel rods are expected to be intact. Control rod segments could have only fallen about 3 inches if severed by melting in the hot region, and the Ag-In-Cd absorber should be in place because it is an insoluble metal. Although the burnable poison rods would also be expected to be in place, their poison is probably lost; boron is known to leach out of B_4C-Al_2O_3 pellets when exposed to water in a radiation environment.

D. SUMMARY

Many or all fuel rods may have ballooned and ruptured, but this mode of initial defecting is probably irrelevant in light of later more extensive damage.

In the hot upper central region of the core, fuel temperatures probably exceeded 1750°C releasing large quantities of fission products; about 30% of the total core inventory of noble gases was released.

About 40% of the Zircaloy cladding reacted with water. This region of severe oxidation was localized above the 4 to 6ft elevation and may not have included peripheral bundles. The severely oxidized fuel probably fragmented into pieces ranging from millimeter size to whole sections of rods.

The temperature of unfueled components lagged the temperature of fuel rods by only about 20°F so that they also experienced temperatures above about 1700°C. Consequently, in the hot region of the core Zircaloy components should have oxidized, and components with Inconel, stainless steel, and Ag-In-Cd should have melted. Because of many layers of protection, the thermocouple tubes have survived even in the damaged core region, although the outer sheath of the instrument tube may be badly damaged.

Nearly all of the broken and oxidized fuel debris should remain trapped in the upper core region because the upper end fittings have a grillage that would act as a screen. Furthermore, the compaction of fuel debris

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is limited because it is fabricated with a packing fraction of about 46% and the theoretical maximum packing fraction (for a bed of spherical particles) is only about 63%. It is very likely that fuel debris are also trapped in some mixing cups (See Figure 18) contributing to non-uniform thermocouple readings.

An earlier estimate of fuel damage in TMI-2 was made at NRC by Rubenstein, Meyer, Tokar, and Johnston. That estimate is in general agreement with the present estimate although our current evaluation is more refined. A memorandum summarizing the earlier estimate is attached as Attachment D.

E. Recommendations

Reactor fuel is rugged, and it is unlikely that limits for natural circulation conditions will be related to fuel behavior. The general criterion with regard to the fuel should be that additional Zircaloy oxidation and fission gas release should be avoided.

Significant oxidation rates do not occur until 900 or 1000^{o}C (See Figure 6). Significant fission gas releases do not occur until even higher temperatures (See Figure 2). These temperatures should be avoided in the (relatively) undamaged regions of the TMI-2 core, but these temperatures are so high that other limits will probably prevail.

By now the adiabatic heatup rate is low (See Figure 19) and ample time will be provided to detect fission gas or hydrogen releases. Therefore, on-line methods of such detection, if feasible, should give adequate

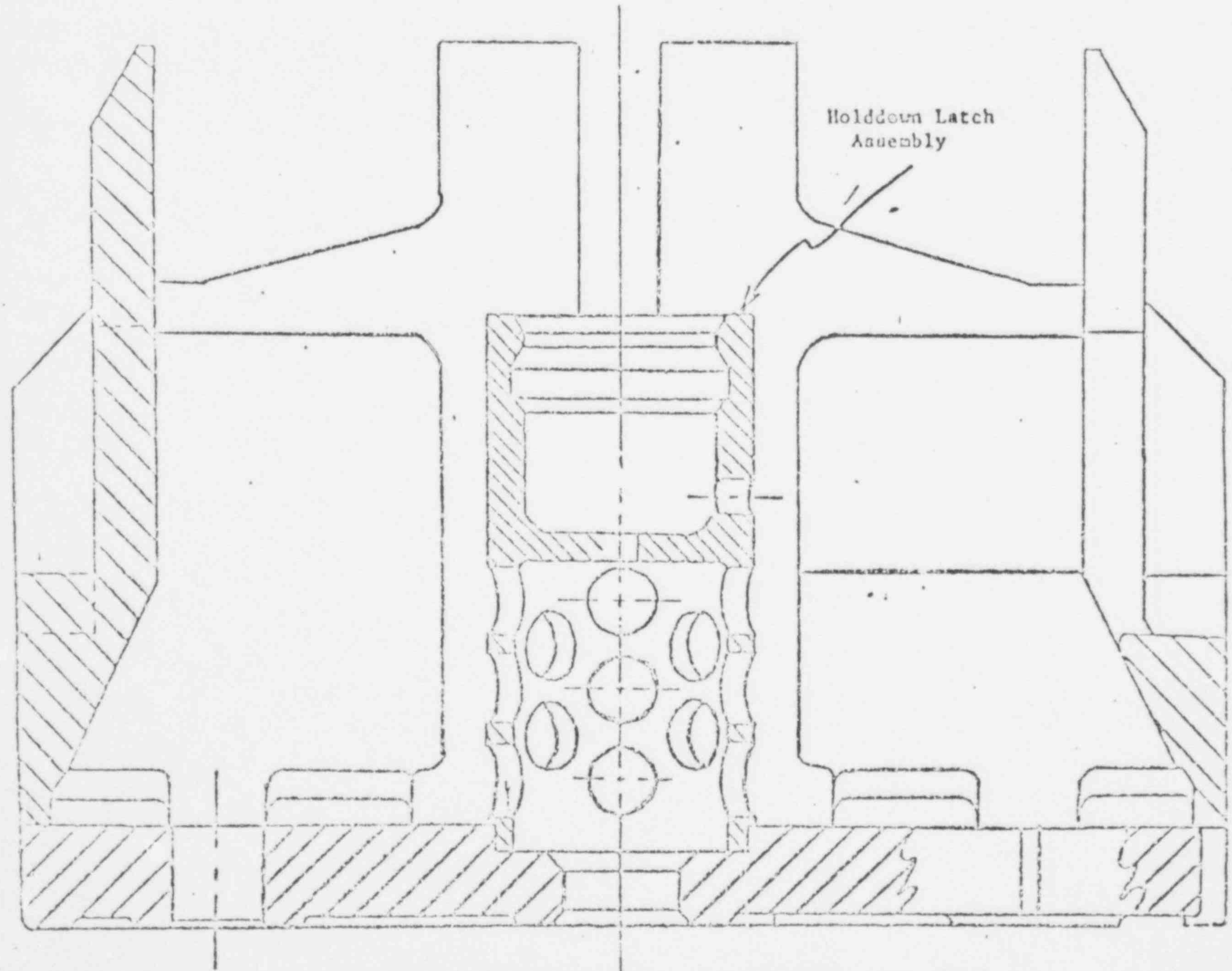

FIGURE 18. UPPER END FITTING AND HOLDDOWN LATCH ASSEMBLY

106-178

SUBJECT

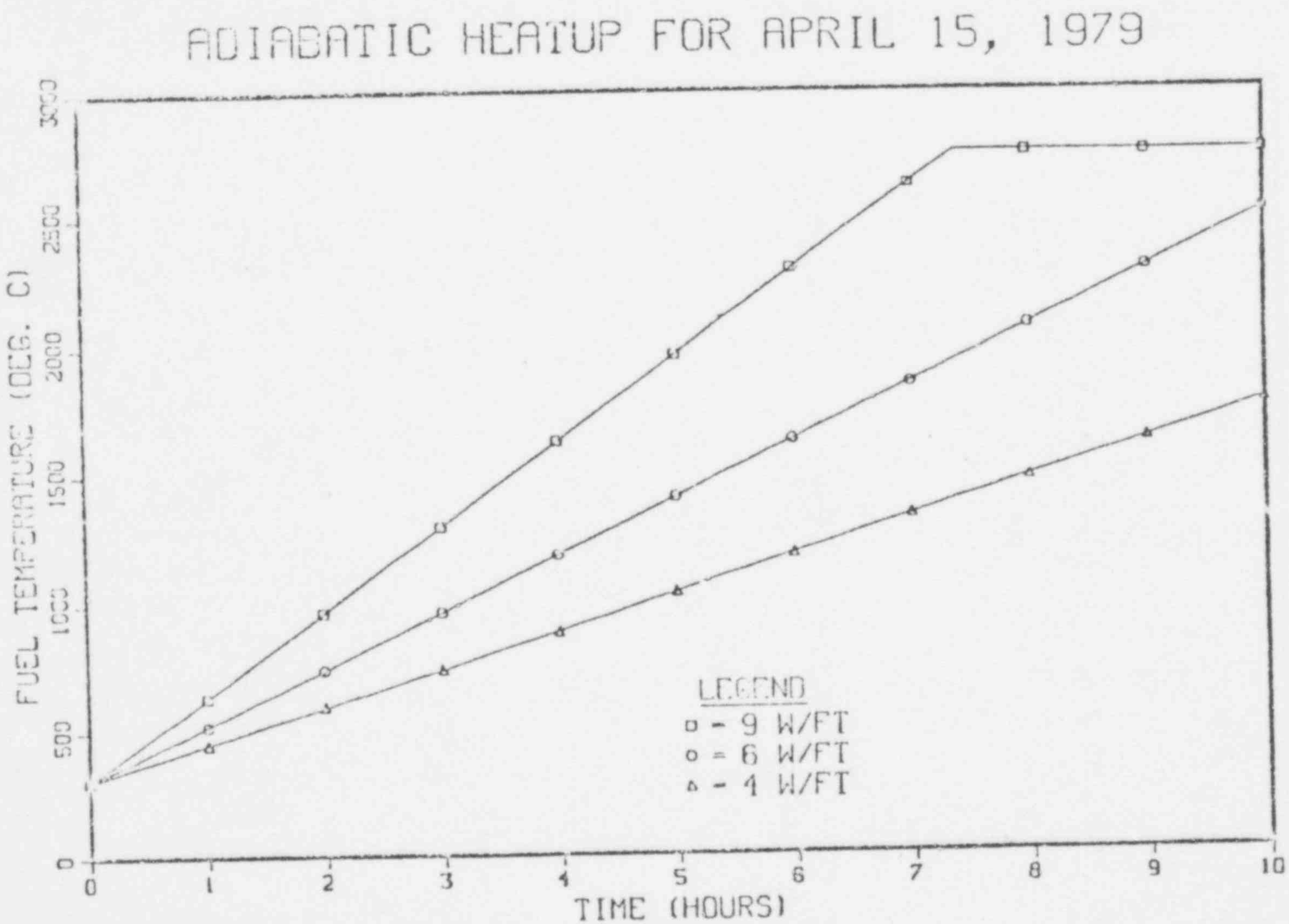

Figure 19.

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warning of fuel damaging conditions.

A discussion of instrument responses relevant to fuel behavior was held with a group of fuel experts from across the industry. A summary of those discussions was prepared by W. V. Johnston and is attached as Attachment E. One consensus of that group was that in-core thermocouple readings should be recorded continuously. A recommendation for such data recording was made and is attached as Attachment F.

106 180

ATTACHMENT A.

R.O. Meyer
P-1114

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ZNR UUUUU
R 112045Z APR 79
FM R A LORENZ OAKRIDGE NAT LAB OAK RIDGE TENN
TO R O MEYER NRC PHONE 0 7003 NRC WASHINGTON DC 20555
BT
BT
UNCLASSIFIED/N O N N D
DATA FROM 53 POST IRRADIATION ANNEALING EXPERIMENTS CONDUCTED BY
R. W. PARKER ET AL. WERE ANALYZED IN ORDER TO OBTAIN ESTIMATES OF
XENON..IODINE AND CESIUM RELEASED FROM UO2. THESE TESTS WERE
CONDUCTED WITH UO2 IRRADIATED TO BURNUPS OF TRACE TO 4000 MWD MT
HEATED AFTER IRRADIATION IN FLOWING INERT ATMOSPHERES FOR
5.5 HR. THE TESTS ARE SUMMARIZED ON P. 80 OF REPORT ORNL 3971.
THE FOLLOWING NUMBERS ARE THE PERCENTAGE RELEASES OF XENON
CORRESPONDING TO MINIMUM PROBABLE..MOST PROBABLE..AND MAXIMUM
PROBABLE..RELEASE.
AT 1600 DEG C THE ESTIMATED RELEASE PERCENTAGES ARE 2.2..5.3..
AND 11.5. AT 1800 DEG THEY ARE 7..15..AND 30. AT 2000 DEG C
THEY ARE 19..37..AND 61. RELEASE PERCENTAGES FOR IODINE AND
CESIUM AVERAGED APPROXIMATELY TWICE THE ABOVE VALUES.
BT
#0184

7904270083

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PDR
File 50-320

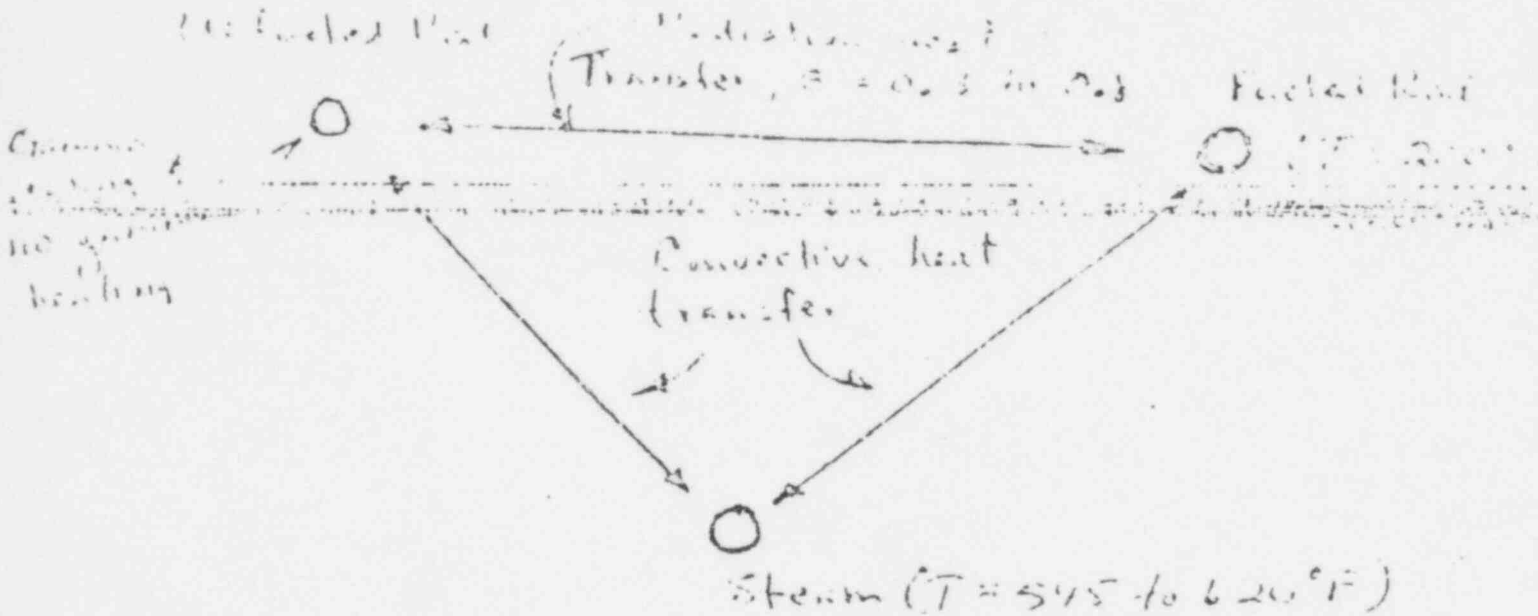
106 181

Attachment B

See memorandum from W. Butler, Chief of the Containment Systems Branch, to R. Tedesco, Assistant Director for Reactor Safety, and entitled, "Three Mile Island, Unit 2: Analysis and Evaluation of Selected Containment Related Issues," to be issued on or about April 16, 1979.

106 182

ATTACHMENT C.



Results

Heat transfer coeff, h = 1.76 to 2.58 English units

Temp. unfueled rod lags fueled rod by 15°F
(essentially same for all cases run)

Assumptions
1. Steam ... 0.0011 ... /ft.

2. ... heating ...

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3. ... fuel rods and ... unfueled rod

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
April 6, 1979

Meyer

ATTACHMENT D.

MEMORANDUM FOR: E. G. Case, Deputy Director
Office of Nuclear Reactor Regulation

FROM: TMI Fuel Team

SUBJECT: ESTIMATE OF FUEL DAMAGE IN THREE MILE ISLAND (TMI)

Enclosed is a brief report describing the preliminary conclusions of the team formed to analyze the probable damage to the fuel system at TMI.

L.S. Rubenstein
L. S. Rubenstein, PSS/NRR

Ralph Meyer
R. O. Meyer, DSS/NRR

M. Tokar
M. Tokar, DSS/NRR

W. V. Johnston, RSR/RES
W.V. Johnston

Enclosure -
As stated

7904270090

PDR
File 50-320

106 184

On Tuesday, April 3, 1979 a team consisting of

L. S. Rubenstein, PSS/NRR
R. O. Meyer, DSS/NRR
M. Tokar, DSS/NRR
W. V. Johnston, RSR/RES

was formed to survey the fuel groups analyzing the damage to the fuel system of TMI and draw some preliminary conclusions from their deliberations regarding that damage.

The following individuals and organizations were contacted on April 3 and 4, 1979:

E. L. Zebroski (EPRI) -
(representing the Metropolitan Edison Group)
J. Taylor/J. Tulenko, B&W
R. Denning, BCL
D. McCloskey, Sandia
J. Scott, LASL

In addition to the information obtained from conversations with these organizations and the NRC staff, the team obtained a "sequence of events" from B&W (Enclosure 1) a group of curves describing the pressure, temperature changes at TMI-2 during the first 15 hours from D. Eisenhut, and a BAPL radiochemical analysis of the primary coolant taken at 1600 hours March 29, 1979 and decay corrected to 0700 hours March 30, 1979.

The primary information used in our analysis of fuel system damage was obtained from the B&W Company, the Metropolitan Edison Industry Group, and from calculations of the NRC staff (Reactor Fuel Section, CPB; Fuel Behavior Branch, RES).

System Effects

Using the chronology of events obtained from B&W and the control room strip chart tracing of system pressure for the first 15 hours of operation, we were able to determine that there were three periods in which the primary system pressure was below a saturation pressure corresponding to a temperature of 620°F. The system changes which caused these periods are described in the sequence of events provided by B&W enclosed with this report. The details of what occurred to cause the pressure changes in the primary system are not discussed here as these are considered in other staff reports (see e.g., IE Bulletin 79-05A, Nuclear Incident at Three Mile Island) and will be evaluated by others.

Examination of Figure 1 shows that the first period in which the system pressure was substantially below saturation pressure occurred approximately 1.75 to 3 hours after start of the transient. The second period, which was relatively short in duration, occurred in the 4.5 to 5.5 hour time frame and resulted in a small decrease in primary system pressure below saturation pressure. The final period of decreased primary system pressure extended from approximately 8-14 hours after start of the transient. It was during these 3 periods that the core was exposed to extensive amounts of steam cooling and experienced fuel damage. The group was able to infer from examination of these pressure histories, reports of fuel channel temperature changes with time obtained from the incore thermocouples, the behavior of the incore rhodium self-powered neutron detectors (SPND's), and 3'-long Intermediate Range Ex-Core Detectors, and the containment radiation monitors some details of when the fuel pins lost their integrity, the depth of the core which was exposed to steam cooling, the probable time periods of that exposure, and the amount of damage to the fuel.

As previously stated, the evidence for the level of uncovering was obtained from a B&W analysis of the incore SPND's. It can be shown that :

> Above about 700°F, incore SPND's (Rh) act as thermionic elements and generate currents which are correlatable to temperature. Thus, if a discontinuity is observed in current measurement, a transition in temperature may be inferred. It was assumed that this discontinuity represents an elevation at which voiding of the coolant has occurred.

Similarly, the excore Intermediate Range Detectors may be used to provide an indication of voiding.

The information obtained from these detectors was consistant with the results from the Industry Group calculation that, in approximately one hour without introduction of makeup water, the core could boil down to full uncovering.

Fuel System Conditions During Period of 1st Uncovering

During the first period of major uncovering of the core (at least 5 feet of the core was uncovered for about an hour, and perhaps all of the core may have been uncovered for about one-half hour), the uncovered portion reached temperatures high enough to fail fuel rod cladding. At this point, fission products were released into the primary coolant as evidenced by the subsequent alarming of the containment activity monitors. Based on the measured coolant activity and the amount of hydrogen release from reaction of the Zircaloy cladding with water[1], all of the fuel rods probably defected and released fission products.

Fuel temperatures were estimated from calculations based on the fission product analysis of the sample of primary coolant, and also from heat transfer considerations. Based on back-calculations[2] that accounted for temperatures and temperature-dependent release rates that would be required to produce the measured level of activity, fuel temperatures of 1400 to greater than 1600°C were obtained. Estimates by ORNL based on their experiments indicated that the Cs and I releases measured would have required fuel temperatures of at least 1300°C for an hour. The heat transfer calculations indicated, on the other hand, that the fuel temperature may have been only about 1100°C[3]. In either case since the melting point of UO_2 is 2840°C, fuel melting was unlikely. These temperature differences[2] can be rationalized by considering that a small portion of the core may have been at the higher temperatures. There is also a possibility of some eutectic formation between UO_2 and ZrO_2 at temperatures above approximately 1800°C, but no significance was attached to the occurrance of such a eutectic. Later analysis by members of ANS-5.4 fission gas working group (including one of US--ROM) indicates fuel pellet temperatures as high as 2000°C based on Xe^{133} data and the assumption that half of the core remained cool. While noble gas activities lend themselves to smaller analytical uncertainties than iodine or cesium activities, the uncertainty in the core fraction that is responsible for the release still renders this result inconclusive.

Hydrogen balance calculations indicate that from 15 to 30%[3] of the total Zircaloy inventory has been oxidized[4]. Some of the oxidation, however, undoubtedly occurred during the latter uncoverings. The extent of the oxidation probably varies as a function of height in the core, with the greatest amount of oxidation having occurred in the uncovered (upper) portions of the fuel rods. Later calculations accounting for hydrogen in the bubble, in the containment, lost in the hydrogen explosion, and gained by radiolysis suggests that almost 40% of the Zircaloy in the fuel region may have been oxidized.

1 CPB Staff Calculation
Industry Group Calculation
B&W Calculation

2 CPB Staff Calculation
3 B&W, Industry Group and NRC Staffs
4 Industry Group & NRC Staffs

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As the primary coolant level was restored during the latter portion of the time period of the first uncovering, thermal and mechanical shock loadings of the oxidized and embrittled cladding are believed to have occurred and to have resulted in cladding fragmentation.

At the end of the period of first uncovering, virtually all of the fuel rods had defected and released fission products. Although temperatures had been high enough for a long enough time to have caused severe cladding oxidation, continued operation of incore instruments strongly indicates that fuel assembly structural members such as guide tubes remained intact. Control rod materials are believed to have remained in place, as indicated by the absence of silver in the primary coolant.

Fuel System Conditions and Effects During Period of 2nd Major Uncovering

At about 4 1/2 hours into the event, the core level again decreased to expose the upper 5 feet of the fuel assemblies. The duration of this additional uncovering was shorter than the first, the system pressure was higher, and the overall temperature effects were less severe, as evidenced by the fact that the thermocouples in the outer periphery of the core remained on-scale. Because of the reduced severity of the core conditions during the second uncovering, as compared with the first uncovering, less damage is believed to have occurred to the fuel system.

Fuel System Conditions During Period of 3rd Uncovering

At about nine hours into the event, the core coolant level again decreased, possiblity down to 7 to 7 1/2 ft. from the top of the active fuel level.* The core remained uncovered at this level for about one to three hours, after which the coolant level was again raised and covered the core. The low system pressure (~450 psi minimum), the rather lengthly period of uncovering, and the additional length of fuel surface uncovered, undoubtedly resulted in additional fuel system damage due to Zircaloy oxidation and embrittlement (followed again by more fragmentation due to thermal shock during the recovering of coolant level), although the amount of additional damage is presently unquantifiable.

Fuel System Damage Summary

The picture of the core that has emerged is that the core configuration currently consists of a basket-like shape of relatively intact assemblies that surround a central region of severely oxidized, and probably fragmented, fuel rods in the upper central part. The fuel

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*Based on information received via telecommunication from B&W (April 3) rods are less damaged in the lower central part of the core. Although the fuel rods in the upper central region may be completely fragmented, the guide tubes, grids, and end plates are believed to be intact thus providing a skeletal structure which supports the remaining portions of the damaged assemblies. Partial flow blockage caused by accumulation of fuel debris is thought to be responsible for continuing elevated thermocouple readings. The assymetry of the incore thermocouple readings suggests that a region of the core is more heavily damaged than the average.

Enclosure 1

4/5/79
D. W. LaBelle

PRELIMINARY SEQUENCE
OF EVENTS
(TMI-2, 3/28/79 INCIDENT)

The following sequence of events for the TMI-2 incident of 3/28/79 has been formulated by B&W engineers using available plant data. This chronology has been constructed from numerous sources and has not been totally confirmed. It may not be precise in either event occurrence or sequence.

Time, Minutes	Event
Prior to turbine trip	The initiating events could have come from numerous postulated causes. For purposes of this sequence, they are relatively unimportant. The prime effect is that it led to a loss of main feedwater (MFW) booster pumps.
0	Main feedwater pumps are tripped. Almost simultaneously, the turbine trip occurs.
0.10	Pressurizer pressure increases to the EMOV setpoint of 2270 psig.
0.15	Secondary side pressure peaks at 1070 psig and is limited by steam relief valves.
0.20	RC pressure trip setpoint reached (2355 psig at hot leg tap) and system pressure peaks at about this value. Indications from pump discharge pressure are that auxiliary feedwater pumps (one turbine driven, two electric) are running at this point; however, no level change occurs in steam generators.
0.25	Pressurizer level peaks at 255 inches (indicated) and starts to decrease with system contraction.
0.30	Quench tank pressure is increasing.
0.50	Pressurizer level is at a minimum of 158 inches and starts to increase. Hot leg temperature is at a minimum of 577°F and starts to increase slowly.
1.0	OTSG level indication on the startup range is 10 inches. OTSG pressure holds at about 1025 psig.
2.0	OTSG pressure starts a steady decrease. HPI flow is initiated by ESFAS on low RC pressure (HPI setpoint = 1600 psig).
3.0	The quench tank's increasing pressure levels off at 120 psig. Relief valve setpoint is 150 psig.
4.75	The hot and cold leg temperatures start increasing at a more rapid rate. Analytical simulation indicates that this occurs when the HPI is turned off. Site information notes that operator terminates HPI fully at 5.1 minutes.

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Time, Minutes	Event
5.0	Pressurizer level indicates a slowing and then continues to increase as the hot leg temperature if increasing.
6.0	Pressurizer level indicates a full pressurizer and the quench tank pressure increases beyond the relief valve setpoint of 150 psig.
	RC pressure reaches a minimum of 1350 psig with a hot leg temperature of 584°F. This indicates hot leg is in saturation condition.
8.0	Auxiliary feedwater flow is initiated to both OTSG's. This is indicated by immediate OTSG repressurization to ~1020 psig and OTSG level change.
9.0	RC pressure peaks out at 1500 psig and starts to decrease. Hot leg temperature peaks out at 597°F.
11.0	Pressurizer level indication is restored. It stabilizes out at 375 inches at 13 minutes.
15.0	Quench tank pressure drops suddenly, indicating the rupture disk has blown (setpoint = 200 ± 25 psig).
18.0	The decreasing RC pressure stabilizes at 1115 psig.
22.0	The RCS temperature stabilizes at a hot leg of 553°F and a cold leg of 548°F. The temperature decrease from start of auxiliary feedwater to this stabilization represents a 200°F/hr cooldown. Reactor building pressure is 1.4 psig and increasing. Two foot level is restored in both OTSG's.
50.0	The startup level indication shows OTSG B level increasing and OTSG A level decreasing. Pressure increases in both OTSG's.
60.0	During the 22-60 minute period, the system parameters have stabilized in the saturation condition of a pressure of ~1015 psig, temperature of ~550°F. RC flow indication is decreasing from 60 (initial) to 50 x 10^6 lb/hr. The reactor building pressure is 2.2 psig and increasing.
73.0	Two RC pumps are tripped (in Loop B). Reactor coolant flow rate decreases in Loop B.
78.0	OTSG B pressure drops from 950 psig to 140 psig in 18 minutes.
90.0	T_{hot} follows T_{sat}. ΔT across the core equals about 5°F.
100.0	Both remaining RC pumps are tripped.
114.0-120.0	T_{hot} and T_{cold} diverge rapidly. T_{hot} > 620°F in less than 15 minutes.
132.0	Site information notes that EMOV relief line was isolated initially. RC pressure starts decreasing more rapidly.

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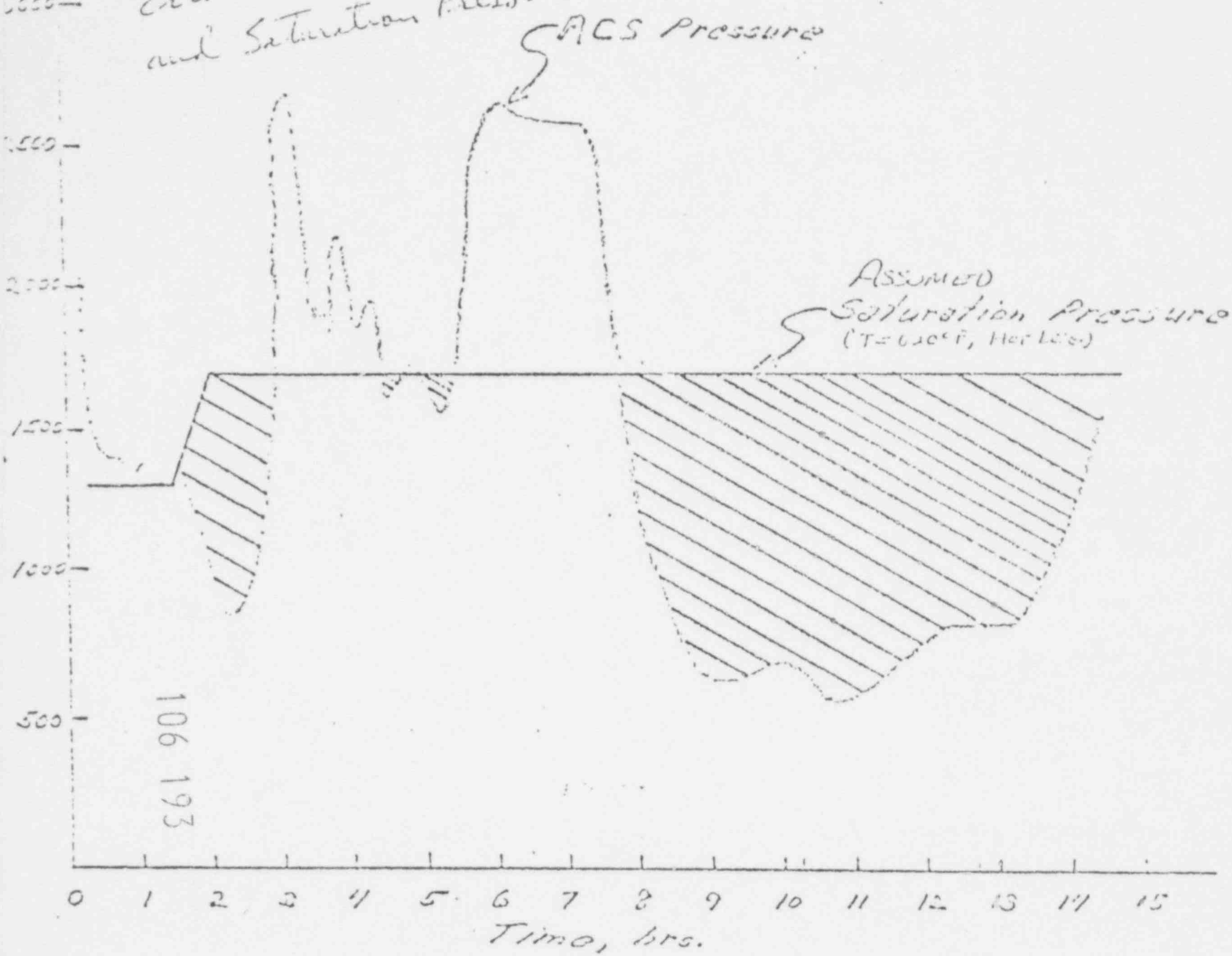
Time, Minutes	Event
135.0	RCS has depressurized to 670 psig and RCS hot leg temperature is at maximum scale of 620°F. At 620°F, system would have superheating at upper elevations as long as pressure was below saturation pressure of 1772 psig.
	RCS shows rapid re-pressurization.
150.0	OTSG B level ramped up from 5% to 65% in 48 minutes.
180.0	OTSG B main steam isolation valves and turbine bypass valves are closed. RCS pressure peaks at 2120 psig.
180.0-204.0	Regulation by EMOV block valve reduces RCS pressure.
204.0	HPI comes on (1600 psig signal).
216.0m	HPI pump 1c to Loop A turned off. RC pressure decreases stepwise. RB pressure increases stepwise.
290.0m (4.83 hr)	RB pressure hits 4 psig. Building fan cooler comes on.
318.0m (5.3 hr)	RCS pressure increases rapidly from 1250 to 2120 psig in 35 minutes. The EMOV block valve is closed, one HPI (1A) is on.
354.0 (5.9 hr)	OTSG A level is ramped up from 50% to 95% on operating range in 1 hour and to 100% in 1.5 hour. OTSG A pressure starts to decrease toward zero.
450.0 (7.5 hr)	The EMOV block valve is opened. RCS pressure starts to decrease (2050 psig to 400 psig in 1 hr, 45 min).
519.0 (8.65 hr)	RC system pressure reaches 600 psig, core flood tank setpoint.
588.0 (9.8 hr)	RB pressure spike to 28 psig occurs.
630.0 (10.5 hr)	T_{hot} Loop A reappears on scale, decreases to 525°F in 1/2 hr.
678.0 (11.3 hr)	T_{cold} Loop A increases in about 5 minutes from 190°F to 400°F.
750.0 (12.5 hr)	HPI flow increased to 400 gpm. T_{hot} in Loop A decreases.
810.0 (13.5 hr)	T_{cold} Loop A decreases.
948.0 (15.8 hr)	Pump 1A is started.

FIGURE 1

Error in Estimate and Saturation Press.



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Prepared for Eisenhut's April 4
Commission Briefing

Core Coolant Conditions

- At 2 hours after turbine trip the core had become partly uncovered and remained uncovered for about one hour.

- During this period activity alarms came on indicating significant fuel failure.

- Core was recovered when high pressure injection pump came on.

- Two additional periods of extensive core uncovering followed at about 5 and again at 9 through 12 hours after turbine trip.

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Number of Fuel Rods with Defects

- Based on measured coolant activity, all of the fuel rods probably released fission products.

- Amount of hydrogen released from oxidation of cladding (metal/water reactions) also indicates all fuel rods are damaged.

Maximum Fuel Temperatures

- Calculations based on Fission product analysis indicate fuel temperatures of 1400 to 1600°C.

- Heat transfer calculations indicate temperature of about 1100°C.

- The melting point of UO_2 fuel is 2840°C so that core meltdown was not approached.

- The absence of Sr and Ba activity in the coolant confirm the avoidance of fuel melting.

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Extent of Fuel Damage

- Hydrogen balance calculations indicate from 15 to 40% of the Zircaloy cladding has been oxidized.
- Continued operation of incore instruments indicates that fuel assembly structural members remain intact.
- Absence of silver in coolant suggests that control rod materials remain in place.
- Continued low thermocouple readings at periphery suggest that peripheral fuel assemblies retained much of their original geom.
- The picture that emerges is that the upper central part of the core is severly oxidized, probably fragmented, and largely confined to the core region (based on loose parts monitoring data).
- Partial flow blockage caused by accumulation of fuel debris has probably occurred and is responsible for elevated thermocouple readings.

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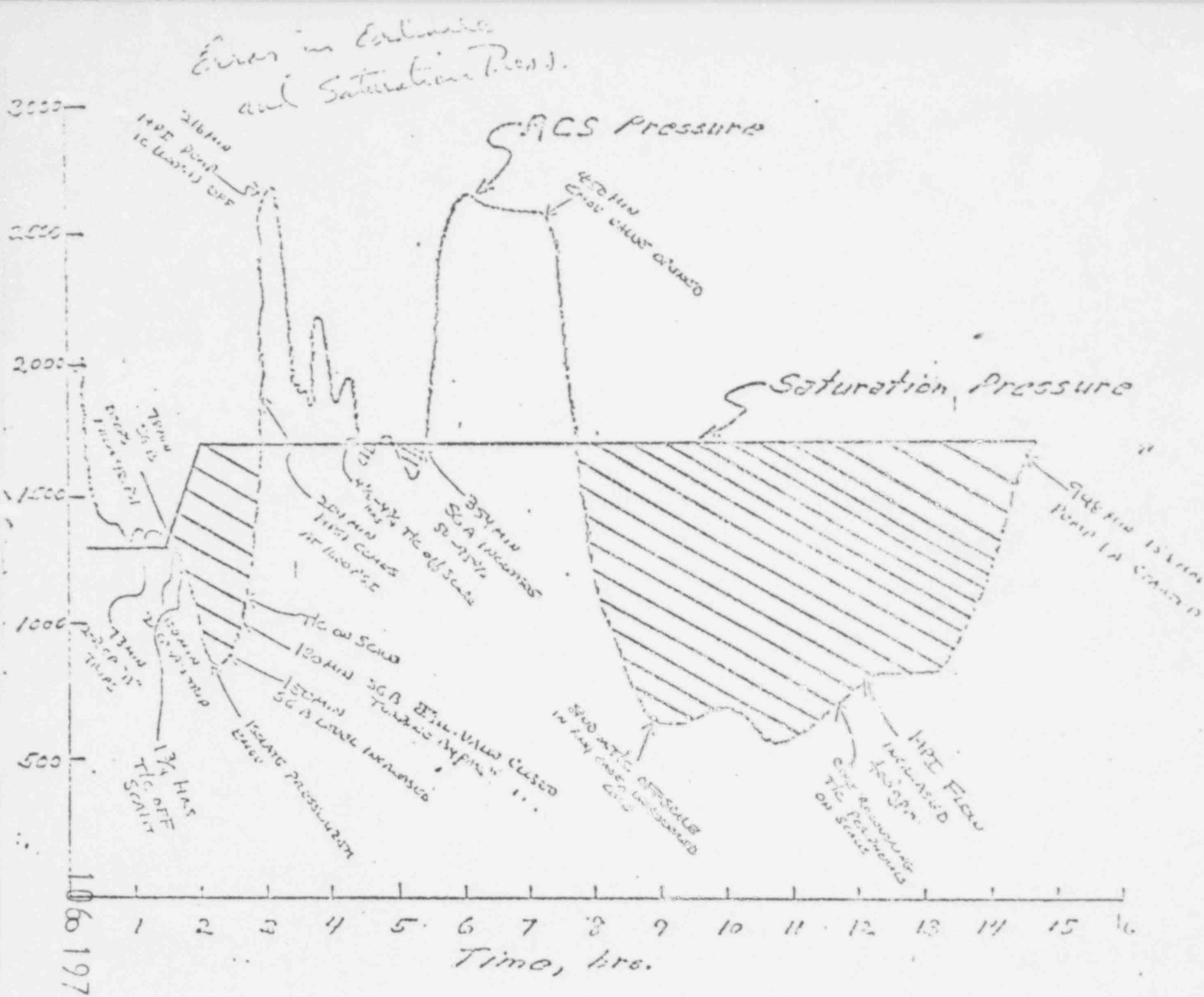

RCS Pressure
Saturation Pressure
3000
2500
2000
1000
500
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
Time, hrs.
T/C ON SCALE
1¾ HRS
T/C OFF
SCALE
HPI FLOW
INCREASED

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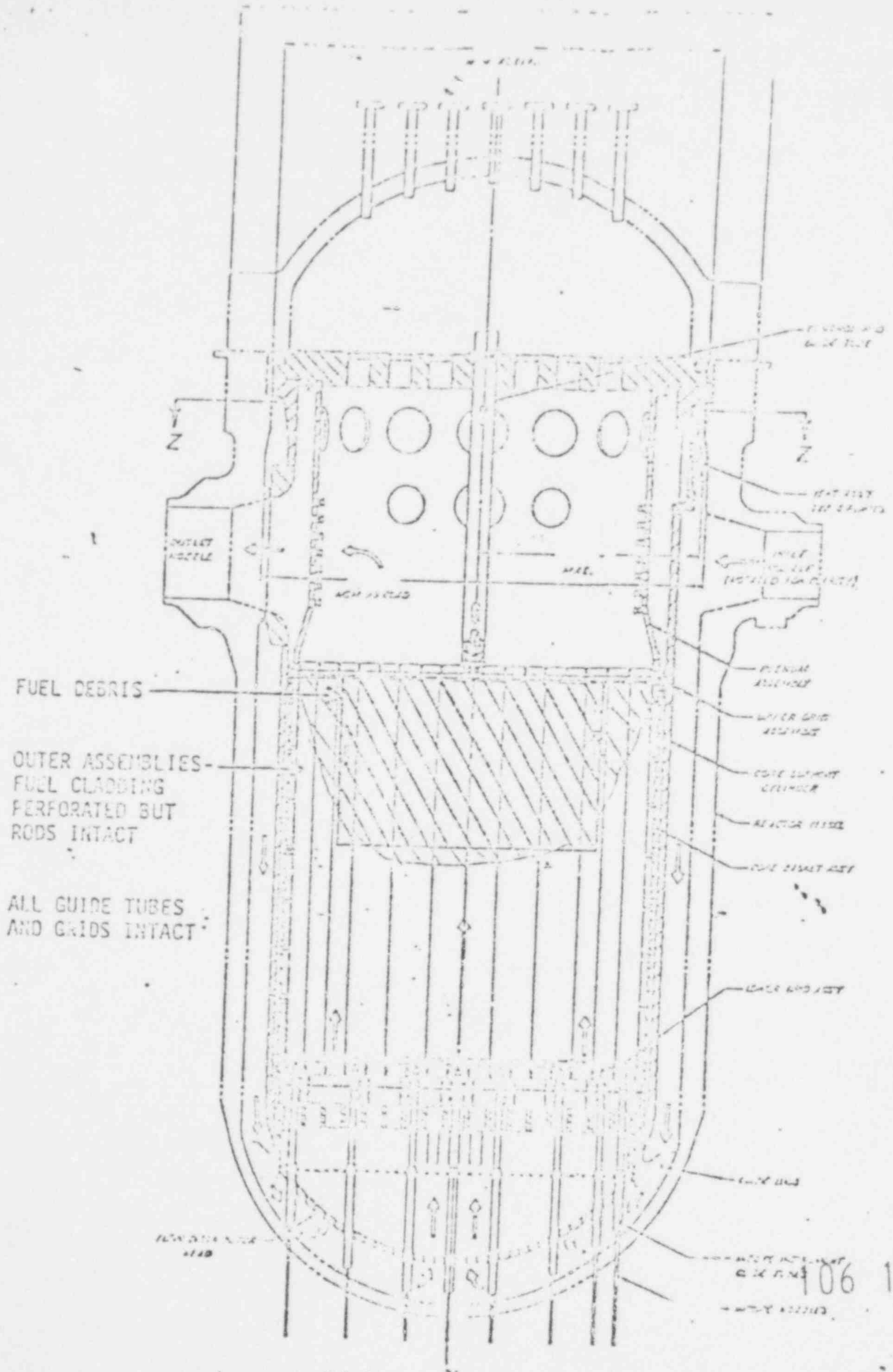

REACTOR VESSEL & INTERNALS

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ATTACHMENT E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 13 1979

MEMORANDUM FOR: D. Ross, Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation

FROM: W. V. Johnston, Chief
Fuel Behavior Research Branch
Division of Reactor Safety Research
Office of Nuclear Regulatory Research

SUBJECT: FUEL EXPERTS MEETING ON CONDITION OF THE TMI CORE

A meeting of nuclear fuel experts was held on April 12 to update the estimates of the damage to the TMI core and to consider its effect on the desirability of moving to natural convection cooling of the core. The Experts Group consisted of the following persons: J. S. Tulenko, B&W; R. DeMars, B&W; T. Kassner, ANL; R. A. Proebstle, GE; K. A. Jordan, W; R. Duncan, CE; T. Fernandez, EPRI; T. Buhl, NRC; R. Meyer, NRC; W. Johnston, NRC, Chairman. Additional attendees included L. Rubenstein, NRC; C. Berlinger, NRC; M. Tokar, NRC; R. Majors, ACRS Staff and T. Mott, TEC.

Summary

The group concluded that although the core is badly damaged, essentially all of the fuel has remained in the core and that the overall packing density of the settled portion is not expected to exceed 70%. Therefore, shutting off the RC pumps should not seriously threaten further damage to the reactor. It was further concluded that the thermocouples (Tc's) located in the upper end fittings are the most important indicator of core condition during transition to natural convection cooling. If feasible, the addition of a γ spectrometer to monitor the activity of the loop coolant for new fission products released during the transition to natural convection cooling will provide an independent alert to possible difficulties.

Two questions were considered: Is a pump trip likely to lead to an unsafe condition? and What signals will indicate undesirable conditions in the core?

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File 50-320

Summary of Core Status

Summaries of amount of damage to the core based upon measurements or calculations of fission gas release, hydrogen produced by zircaloy oxidation, coolant analysis, coolant boil-off rates, incore and excore instrumentation were presented by J. Tulenko and R. Meyer. A relatively large pressure drop across the core is inferred by TH calculations. If the pressure drop is real, blockage must also exist in the peripheral assemblies (perhaps by ballooning). The shift in location and magnitude of the high reading core Tc's following the pump trip on April 6 was believed to indicate either a change in the core flow path through more heavily damaged sections of the core to a redistribution of debris surrounding some of the thermocouple leads. An alternate explanation for the change in Tc temperature distribution patterns was presented by T. Mott of TSC. He suggested that the Tc temperature difference may be due to non uniform flow distributions caused by operation of a single pump rather than non uniformity within the damaged region of the core. Due to this non uniform flow distribution portions of the core may already be experiencing similar cooling to that expected during natural convection. Mott estimates smaller core pressure drop and suggests the B&W estimates may include substantial external pressure drops.

The group visualizes the core as consisting of a heavily damaged region resembling an inverted bell extending across nearly the full width of the top of the core and reaching down about five - six feet into the core at the center and a less damaged remainder of the core. In the heavily damaged region, 100% oxidation of the zircaloy and less of a regular geometry is expected. The guide tubes and poison rods are damaged similarly to the cladding. Spacer grids should be located at or near their original locations. The important coolability conclusions are that although some settling may have taken place, the overall packing density of the settled portion is not expected to be greater than 70% and that 85% to 90% of the fuel and cladding from this region is believed to have remained in the "core" region including the upper end fitting. The remainder of the core is less damaged although considerably oxidized. The original flow geometry is probably retained although the rods may be twisted or warped and broken in a few places and the spacer grids may have collected some loose debris.

The above conditions should not preclude satisfactory achievement of natural convection flows.

What should be monitored to determine undesirable changes in the core during the transition to natural convection cooling? The temperature distribution of the core exit thermocouples are the most important

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condition monitoring signals. The group believes that all exit Tc's should be continuously tracked and recorded. BCM suggested the following criteria for remaining in natural convection cooling: No more than Tc's above 800°F and at least 10 Tc's below Tsat. There were some reservations among the group about allowing so many Tc's to read above Tsat (as many as 30) and radiolysis was an expressed concern. There was a lot of discussion but no consensus on how many interior Tc's should be permitted to exceed Tsat. Tc's in peripheral assemblies should not exceed Tsat.

The following table summarizes the available instrumentation and its possible application to monitoring core condition.

Detector	Event-Core Overheating Criteria	Basis
1. Exit Tc's	Limit no. in film boiling Limit no. above 800°F	Not to exceed previous core damage reverse procedure.
2. RTD Hot leg Cold leg	Maintain positive ΔT across core.	No flow reversal permitted.
3. Ion chambers γ and N	Void formation If +, record for future interpretation, watch Tc's.	Ambiguous signal since some local superheat may be permitted.
4. Noise detection	If + indicates bubbles in core or loop, check Tc's, SG.	Same as above.
5. System Pressure	If increasing system effects Branch should review this.	Not direct indication of core condition, but for gas bubble formation detection.
6. Pressurizer Level	Same as above.	Same as above.

Additional Detection - Feasibility needs to be established.

γ spectroscopy of coolant via sampling line	Increasing activity of Xe, I_2	Overheated core alert for major error in procedure.
H_2O analysis on line monitor	Boron, O_2, H_2	Core criticality and chemistry control radiolysis and H_2 content control.

William V. Johnston

W. V. Johnston, Chief
Fuel Behavior Research Branch
Division of Reactor Safety Research [illegible]

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April 13, 1979

Attachment F.

NRC ADVISORY -- THERMOCOUPLE READINGS

We recommend that all incore thermocouples be read in a continuous manner with provisions for rapid retrieval and permanent storage. It is clear that a continuous reading of thermocouples will be needed during the transition to natural circulation. It is also clear that temperature trends, which were not recorded during the pump changeover on April 6, would have given additional clues to core behavior in natural circulation. Since future flow transients cannot be ruled out and since the transition to natural circulation could occur involuntarily, we recommend that the continuous recording of all thermocouple readings be initiated as soon as possible.

Ralph O. Meyer

Ralph O. Meyer, Leader
Reactor Fuels Section
Core Performance Branch

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