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

Meyer

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.

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Enclosure -
As stated

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On Tuesday, April 3, 1979 a team consisting of

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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.

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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 [illegible]eratures 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 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--RCM) 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

*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.

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Enclosure 1

D. H. 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. |
| 0.90 | 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 tempreature 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 ~1025 psia 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 indiciation is restored. It stabilizes out at 375 inches at 15 minutes. |
| 16.0 | Quench tank pressure drops suddenly, indicating the rupture disk has blown (setpoint = 200 ± 25 psig). |
| 18.0 | [illegible] 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 feet 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 $10^5$ 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. |
| [illegible] | Site information notes that EMOV relief line was isolated initially. RB 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 43 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 480 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. |

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| Time, Minutes | Event |
|---|---|
| Thereafter | Condenser vacuum re-established.<br>SG A begins steaming to condenser.<br>RCS cooled to approximately 300°F, 1000 psi.<br>Letdown line ceased to permit flow and relief valve being used (estimated 14-16 gpm flow).<br>Some fuel incore thermocouples reading about 600°F.<br>RB pressure below 1 psi.<br>High radiation in reactor containment and auxiliary building. |

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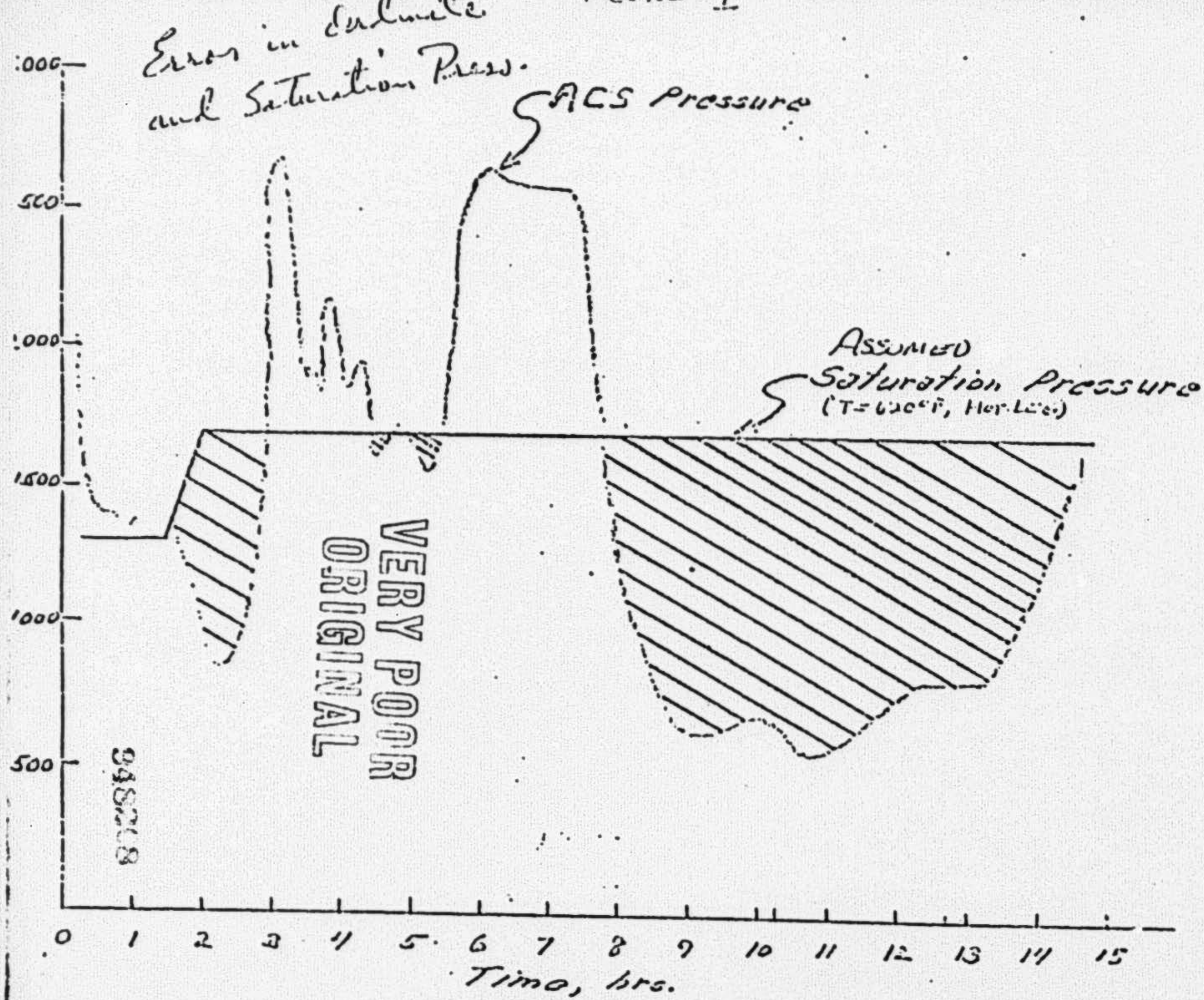
FIGURE 1

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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 44% 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 geam.

- 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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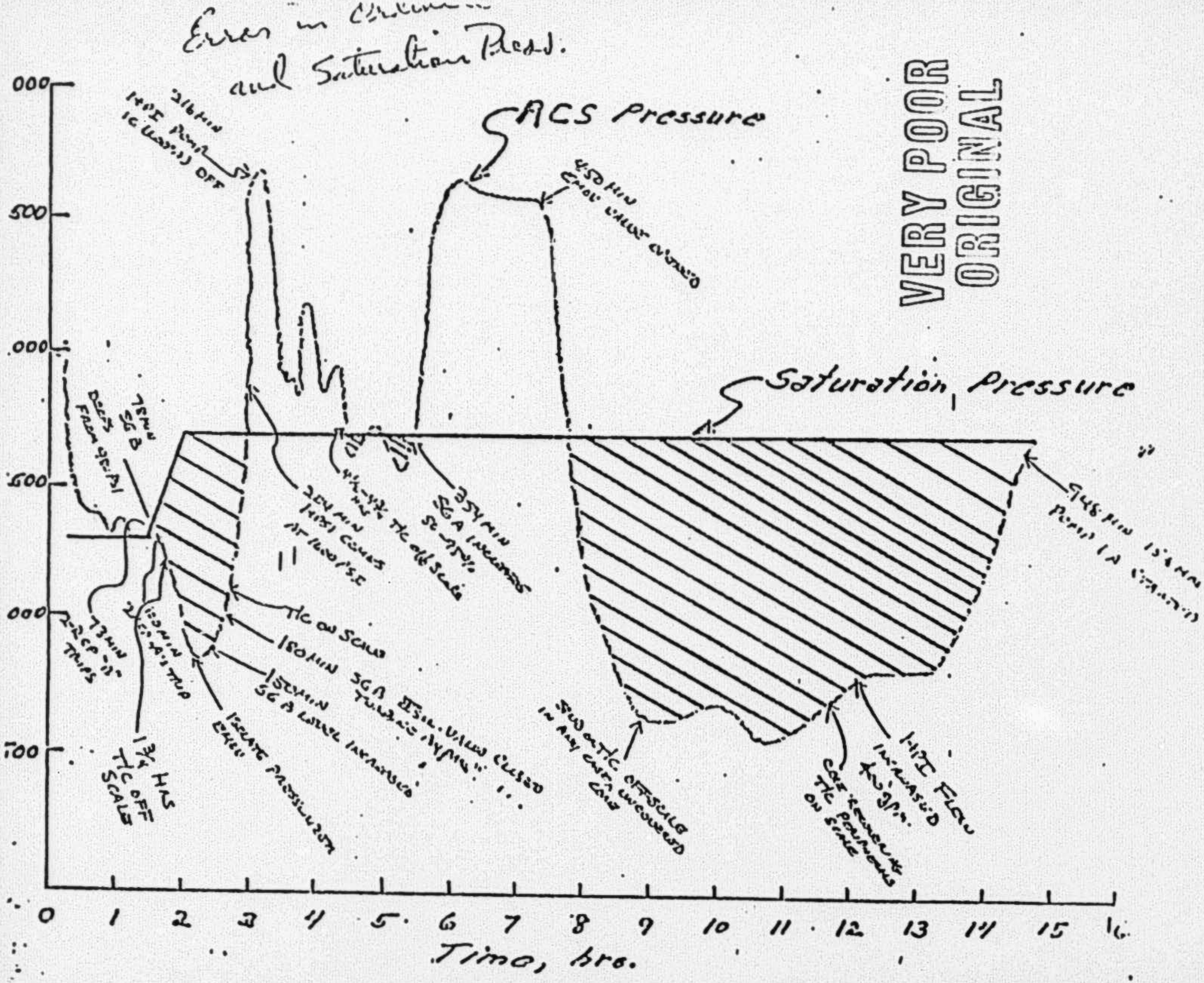

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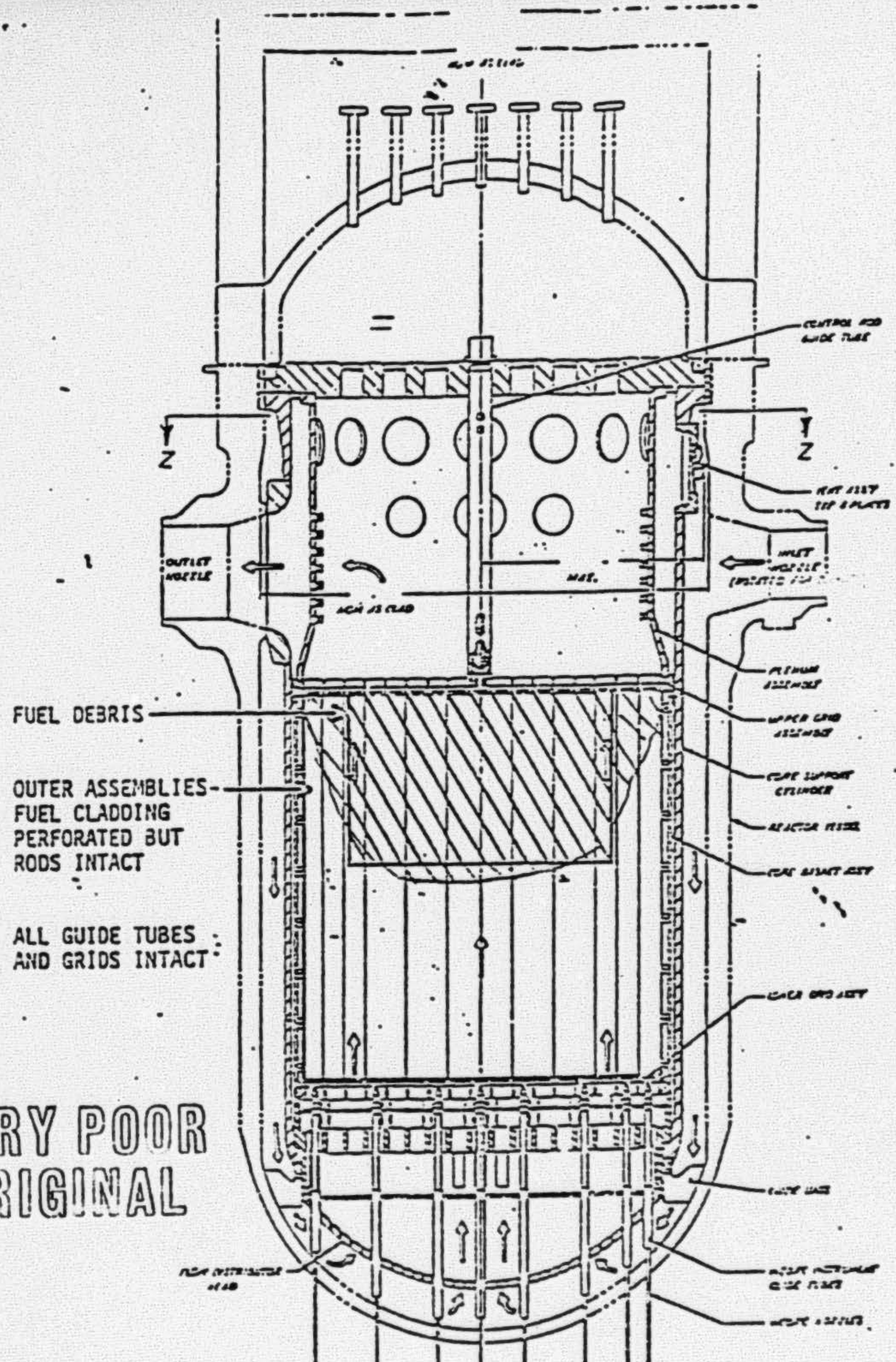


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