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Docket File

MEMORANDUM FOR: Richard P. Denise, Acting Assistant Director
for Reactor Safety
Division of Systems Safety

FROM: Laurence Phillips, Acting Branch Chief
Analysis Branch
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SUBJECT: DOCUMENTATION OF TMI-2 BENCHMARK CALCULATIONS

Since the first few days after the TMI-2 accident, several calculations were performed in the Reactor Analysis Section of AB. The purpose of this memo is to document these calculations and to point out conclusions and questions which may be considered for further study by the task forces now assigned to the TMI-2 review.

(1) Core Uncovery Calculations

Based on studies of the plant process data which were plotted for the period immediately after the accident, it was concluded that core uncovery first occurred at 105 minutes after turbine trip. This was indicated by primary coolant temperature data and steam generator pressure data which indicated a loss of primary to secondary heat transfer, therefore inferring cessation of primary coolant circulation and condensation in the steam generator. Subcooling of primary coolant had ceased at approximately sixty minutes after the event and all reactor coolant circulation pumps were turned off by 100 minutes.

Calculations were performed to estimate the necessary primary coolant mass discharge rate to achieve top of the core uncovery at 105 minutes. The calculations were based on the RCS pressure history and the assumptions which follow:

(1) No HPI coolant after approximately eight minutes.
(2) Mass discharge rate varied as the square root of the pressure times density product,
(3) All liquid mass above the core exit elevation (including pressurizer) was discharged at 105 minutes.

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Results of these calculations are indicated in Figure 1 and show that a mass discharge rate of 395,000 lbm/hour (rated @2250 psi) would be required. This compares to a rated steam discharge rate of 118,000 lbm/hour @2250 psi through the power operated relief valve (PORV).

Since the required mass discharge could not be achieved by steam flow through the PORV alone, the effect of varying the discharge quality using various critical flow models was studied. These results are indicated in Figure 2.

It was concluded that very nearly 100 percent liquid discharge and little or no HPI flow for the 100 minutes preceding core uncovery would be required to explain the system mass loss by discharge through the PORV alone. Since the new HPI flow (in excess of letdown flow) is believed to have been 30 to 100 gpm or more during this period and since continuous discharge of very low quality coolant through the PORV for this entire interval is highly unlikely, a leakage source other than normal PORV discharge is inferred by these calculations. Additionally, later compilations of sequence of events data show that the Reactor Building Sump high level alarm (4.650 feet above the bottom of the sump) was received at about eleven minutes after turbine trip. Since the rupture diaphragm on the reactor coolant drain tank did not burst until fifteen minutes, PORV discharge does not explain the high building water level.

The core uncovery calculation was continued based on steam generation as a function of the decay heat rate and the portion of the core covered. The predicted core steaming rate during the core uncovery interval is given in Figure 3 based on (a) mass and energy conservation calculations and (b) mass and volume conservation calculations. The corresponding plots of core water level are given in Figure 4; an early B&W estimate of the core water level is also indicated on the plot. The plots indicate that the minimum core water level during this interval was no more than three feet and may have been below the bottom of the core.

Corresponding core heat-up calculations during the uncovery interval were performed using TOODEE. These calculations indicate that the clad melting point occurred in advance of total zirconium oxidation. More details of the core heat-up results will be documented separately.

(2) Once Through Steam Generator Heat Sink Capacity

Calculations were performed to estimate the effects of PORV setpoint and reactor trip response on the response of once through steam generators to the Loss of Feedwater Transient, assuming that no auxiliary feedwater is available. Design data for the Midland plant were used as the basis for the calculations. Three cases are tabulated in Table I. Case 1 assumes pre-TMI setpoints for PORV pressure relief and for reactor trip. Case 2 assumes a reduction in the overpressure reactor trip setpoint to 2300 psig, versus a 2450 psig higher setpoint for the PORV which may preclude its opening during the transient. Case 3 further reduces the time to reactor trip by providing a reactor trip on turbine trip.

An energy balance was performed for the first 110 seconds of the transient, which was the time required to boil the steam generator dry for Case 1. Steam generator heat transfer response and the minimum primary coolant temperature of 552F corresponding to secondary steam pressure control conditions were taken from B&W calculations of the transient. Decay heat rates are based on 1971 ANS data with a best estimate multiplier of 0.9. A feedwater coastdown of 10 seconds is assumed.

For Case 1, the primary heat sources including stored energy released when the primary system drops from 582F average temperature to 552F provide sufficient energy to boil dry the 109,000 pounds of mass in the steam generator and to produce 592F superheated steam at an assumed saturation pressure of 1020 psia. However, it is important to note that the steam generator dry out process has reduced the energy level of the system so that 538 seconds of decay heat would be required to return the primary system to initial average temperature of 582F. Therefore, initiation of auxiliary feedwater could be delayed up to nine minutes without rise in the primary system energy level above the initial level.

Case 2 results in about one second earlier trip time and corresponding reduction in the full power seconds generated after turbine trip. Three percent of the steam generator coolant inventory remains available at 110 seconds, and approximately 585 seconds of decay heat are required to restore the system to its initial energy level at 582F.

Case 3 results in instantaneous trip with less than one second at full power. The primary coolant system is reduced to the assumed minimum temperature of 552F (secondary saturation temperature is 547F) with 29 percent of the secondary coolant inventory remaining. Approximately 17 minutes of decay heat are required to boil the balance of steam generator coolant and restore the system to its initial energy level of 582F.

It can be concluded that the early trip time is equivalent to increasing the steam generator coolant inventory by: .29 x 109,000 = 31,610 pounds. Typical steam generator coolant inventory and boil-off data were computed for loss of ac/dc Power Task Action Plan A-30 as follows:

Plant	Type	Power (Mw)	S.G. Mass (lb.)	Time to Boil Dry*
Midland	B&W	2552	92000	17 minutes
St. Lucie	CE	2570	258000	61 minutes
Zion	W	3238	357146	70 minutes

*Assumed 1971 ANS decay heat with nc multiplier.

It is clear that an early reactor trip setting is of little significance to the post-accident response of CE and W steam generators with large coolant inventory, even if auxiliary feedwater is not available. For B&W steam generators with low water inventory (probably less than the design value), an early trip would significantly delay the occurrence of steam generator dry out and provide more time for initiation of auxiliary feedwater.

It is also noteworthy that the steam generator coolant inventory depletion can be delayed by higher secondary pressure relief setpoints (CE vs. B&W). However, higher secondary pressure also limits the primary cooling at a higher temperature level, so that less time is required to reheat the primary system if the heat sink is lost. Therefore, there appears to be no ultimate advantage to higher secondary pressure relief setpoints.

(3) Evaluation of TMI-2 Benchmark Calculations

It was noted that both B&W and INEL benchmark calculations produced overcooling of the primary system during the first two minutes of the transient. An energy balance based on plant process data was performed by hand calculations for the intervals (0-110) sec., (110-300) sec., (300-360) sec., and (360-540) sec., in order to better understand the indicated transient and reasons for the error in computer models.

Table II is a tabulation of the bases and results for the Reactor Coolant System (RCS) energy balance calculations.

The steam generator mass, including feedwater added during a linear ten second coastdown, was depleted during the (0-110) second interval. Mass inventory of 55,970 pounds was computed from the measured level of 160 inches based on a shell side flow area of 44.4 ft.^{2} in each steam generator. This compares to a B&W reported mass of 97,000 pounds which is believed to have been used in their benchmark calculations. The difference of 41,030 pounds is believed due to the difference between actual heat transfer performance and design heat transfer performance. Better performance results in lower liquid level in a once through steam generator. The additional mass would correspond to an added heat sink of approximately 30,000 Mw-sec and would lead to an over prediction of the Reactor Coolant System cooldown during steam generator dry out. In fact, the LOFW analysis would look like Case 1 of Table I with the RCS cooled to 552F compared to the measured temperature of 577F. The sensitivity of safety analyses to assumed mass inventory of the steam generator should be considered in future review of plants having once through steam generators.

The energy balance during the first 110 seconds resulted in excess energy of 1495 Mw-sec. Decay heat energy is believed to be at least as great as the estimate and possibly 10 percent more. Uncertainty in the number of full power seconds or in the steam generator heat sink could account for

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the deficit. For the purposes of the tabulation, the amount of steam that could be generated by the excess energy of 13.6 Mwt each second is assumed to flash and discharge through an unidentified leak (e.g., a steam generator tube).

One peculiar aspect of the TMI-2 data is the mismatch in reactor coolant cold leg temperatures at the start of the transient. Loop A was at 568F while Loop B was at 557F. The control system would normally be expected to maintain a much closer match between these temperatures; e.g., ± 2F, and the reason for this condition should be investigated.

The (110-300) second time interval should be ideal for an energy balance. During this interval, the HPI was known to be operating at full capacity and was the only heat sink other than normal heat loss to the containment. The RCS average temperature was nearly constant during this interval. The only heat sources were decay heat (estimated uncertainty of -0 + 10%) and the reactor coolant pumps. This heat balance shows a large deficit of 5219 Mw-sec or 27.5 Mwt per second. The only plausible explanation of the deficit is flashing of the primary coolant. Since at least 10F subcooling is indicated during this interval, flashing could only occur at a leak location. Leakage through the stuck open PORV would be supplied by flashing in the pressurizer, which was analyzed separately from the RCS heat balance. Leakage flow rate for this energy deficit would be 47.8 pounds per second of steam, compared to 26.9 pounds per second for the deficit indicated during the first 110 seconds.

The balance was continued for the interval of (300-360) seconds when the RCS temperature rises to 582F and reaches saturation temperature. An energy deficit of 12 Mwt per second (equivalent steam leakage of 19.8 pounds per second) is indicated for this interval. However, there was greater uncertainty in the HPI coolant injection rate and in the energy supplied to the reactor coolant system during the reheating. A calculation performed for the six to nine minute interval with RCS at saturation and rising in temperature showed only a 3.6 Mwt/sec deficit, indicative of very little flashing during that period. The latter result is surprising and possibly indicative of lower quality leakage from the saturated system. However, the calculations may be in error after six minutes since the pressurizer cannot be properly separated from the RCS heat balance after saturation is reached.

Table III is an energy balance of the pressurizer to evaluate the calculated leakage and the calculated level based on the system pressure history during the first six minutes of the transient. Time intervals were chosen to match Table II except that no balance was made after six minutes when saturation temperatures was reached. An equilibrium pressurizer model was assumed with flashing energy supplied by all of the hot fluid in the pressurizer at the beginning of a calculation interval. RCS water was not included in the balance.

Steam relieving rates through the PORV were normalized to a discharge pressure of 2250 psi for comparison of discharge capacity during the three time intervals and for comparison to the rated relief capacity of 118,125 lbm/hr. Excellent agreement with rated capacity was obtained, particularly during the (110-300) sec. period when the calculational uncertainty is at a minimum. This tends to confirm that pressurizer leakage was via the stuck open PORV and an additional RCS leak is needed to explain the mass balance.

Also of interest is the computed liquid level versus the measured liquid level. The calculations indicate that the indicated level was too high by 35 inches at 110 seconds, 151 inches at 300 seconds, and 137 inches at 360 seconds. These calculations are believed to be reasonably accurate and indicative of substantial error in the liquid level reading during the first several minutes of the transient. The high indicated levels suggest flashing in the reference leg during the depressurization.

In summary, the following conclusions were reached from the benchmark calculations.

(a) The B&W computer model for CADDS benchmarking of TMI appears to have several deficiencies; e.g., too much water inventory in the steam generator; 3% heat demand to simulate HPI cooling effect from two to five minutes (this is too much); etc., which could be pursued to obtain a more acceptable benchmark.

(b) The INEL model had several problems with improper handling of auxiliary feedwater being the largest error contributor. They are now aware of these problems and are making appropriate corrections.

(c) All calculations seem to point to leakage in addition to that through the stuck open PORV.

(d) A reactor trip on turbine trip has the same effect as additional inventory in the steam generator and appears to be of no value for plants having steam generators which reduce the primary temperature to near the secondary saturation temperature without boiling dry.

(e) Safety analyses sensitive to steam generator coolant inventory should be reviewed carefully to assure that conservative water levels are used.

(f) There remain sufficient questions about the TMI-2 response to warrant additional benchmarking analyses.

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Enclosures:
Tables

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TABLE I

PRIMARY COOLANT ENERGY RELEASE TO BOIL DRY
THE STEAM GENERATORS

Reference: Midland Data	CASE 1 (Pre-TMI)	CASE 2* (Post-TMI)	CASE 3** (Post-TMI)
PORV Set Pressure (psig)	2340	2450	2450
Time to Reactor Trip (sec)	8.93	8.0	0
Time to Rod Movement (sec)	9.63	8.7	0.9
Feedwater Flow Rate (lbm/sec)	3500	3500	3500
Feedwater Added During Coastdown (lbm)	17000	17000	17000
Steam Generator Secondary Water (lbm)	92000	92000	92000
Total Coolant Mass Boiled (lbm @ 1035psig)	109000	109000	109000
Avg. Reactor Coolant System Temp. @ 0 sec. (°F)	582	582	582
Avg. Reactor Coolant System Temp. @ 110 sec. (°F)	552	552***	552***
Avg. Secondary Steam Pressure (psia)	1020	1020	1020
Estimated PORV Flow (avg. lbm/sec)	-	-	-
Power Level (Mwt)	2552	2552	2552
PRIMARY HEAT SINKS (Mw-sec)			
(a) Steam Generator	74157	74157	74157
(b) PORV	NEGLECTED		
PRIMARY HEAT SOURCES (Mw-sec) to 110 sec. after Turbine Trip			
(a) Full Power (2552 Mwt) Energy Input	24575	22202	2297
(b) Fuel Stored Energy (1350F→550F)	12913	12913	12913
(c) System Stored Energy (582F→552F)	25392	25392	25392
(d) Decay Heat to 110 sec.	9297	9297	10057
(e) Reactor Coolant Pumps (18 Mwt) to 110 sec.	1980	1980	1980
TOTAL ENERGY SUPPLY TO S.G. @ 110 sec.	74157	71784	52639
FRACTION OF STEAM GENERATOR COOLANT NOT BOILED	0	.03	.29
TOTAL DECAY HEAT FOR ADIABATIC HEATING OF THE SYSTEM TO 582F ENERGY LEVEL	34689	37062	56967
TIME (sec.) required to generate decay heat	538	585	1006

*Overpressure trip @ 2300 psig; S.G., would not be dry @ 110 sec.

**Reactor trip on turbine trip; S.G. would have substantial inventory @ 110 sec.

***Primary coolant temperature cannot be lowered below 552F due to limiting secondary saturation temperature of 547F which limits further heat transfer.

TABLE II

TMI ENERGY BALANCE BENCHMARK

Basis: Plant Data Describing the Accident

	(0-110) sec	(110-300) sec	(300-360) sec	(360-540) sec
PORV Set Pressure (psig)	2,255	OPEN	OPEN	OPEN
Time to Reactor Trip (sec)	9	--	--	--
Time to Rod Insertion (sec)	10	--	--	--
Feedwater flow Rate (lbm/sec)	3,180	NONE	NONE	NONE
Feedwater Temperature (°F)	463	--	--	--
Feedwater Added During Coast-down (lbm)	15,897	--	--	--
Steam Generator Coolant Inventory	55,970	NONE	NONE	NONE
Total Coolant Available (lbm)	71,870	NONE	NONE	NONE
Steam Superheat Temperature (°F)	592	--	--	--
Avg. Secondary Steam Pressure (psia)	1,035	950	865	825
Avg. Reactor Coolant Temp. @ 0 sec (°F)	582	582	582	582
Final Reactor Coolant Temp. Avg. (°F)	577	578	582	595
Avg. Primary Steam Press. (psia)	1,900	1,560	1,375	1,425
*Estimated Leak Flow (Avg. lbm/sec)	26.9	47.8	19.8	6.1
HPI Coolant Added (Gallons)	243	2,136	200	600
Power Level (Mwt)	2,688	--	--	--
PRIMARY HEAT SOURCE (Mw-sec)				
(a) Full Power (2688 Mwt) Energy Input	26,880	0	0	0
(b) Fuel Stored Energy (1350°F - T Coolant)	12,590	0	-81	-212
(c) System Stored Energy	4,250	-850	-2,800	-9,100
(d) Decay Heat	9,792	13,290	3,604	9,970
(e) Reactor Coolant Pumps (18 Mwt)	1,980	3,420	1,080	3,240
TOTAL ENERGY SUPPLY	55,492	15,860	1,804	3,898
Fraction of S.G. Coolant not boiled	0	--		
TOTAL DECAY HEAT FOR ADIABATIC HEATING OF THE SYSTEM TO 582°F ENERGY LEVEL	14,042	--	--	--
Time (sec) Required to Generate Decay Heat	175 sec	--	--	--
PRIMARY HEAT SINKS (Mw-sec)				
(a) Steam Generator	52,610	0	0	0
(b) HPI Coolant Heating	1,167	10,261	961	2,882
(c) System Heat Losses	220	380	120	360
(d) Deficit	1,495	5,219	723	656
**ENERGY Deficit per sec (Mwt)	13.59	27.47	12.05	3.64

*For 100% quality leakage from the RCS (excluding the pressurizer) to compensate the energy deficit

**Note that a lower average energy discharge indicates that lower quality coolant is being discharge.

TABLE III

TMI-II PRESSURE VOLUME/ENERGY BALANCE

	(0-110) sec	(110-300) sec	(300-360) sec
PZR Pressure (psig)	2150 - 1745	1745 - 1400	1400 - 1355
Coolant Properties: $V_g (ft^3/lbm)$	.1685 - .2275	.2275 - .2974	.2974 - .3094
$V_f (ft^3/lbm)$	.0265 - .0245	.0245 - .0232	.0232 - .0230
h_g(BTU/lbm)	1123.3 - 1153.3	1153.3 - 1172.6	1172.6 - 1175.0
h_f(BTU/lbm)	689.1 - 641.1	641.1 - 600.7	600.7 - 594.7
Electric Heater Input (Mw-Sec)	180.18	311.22	98.28
Pressurizer Volume per ft. Level Change (ft^3/ft)	38.48	38.48	38.48
Reactor Coolant Average Temperature (°F)	582 - 577	577 - 578	578 - 582
Fraction of Hot PZR Fluid Flashed	.0937	.0706	.0103
LIQUID VOLUME (V_f), Ft^3	795.35 - 604.4	604.4 - 811.2	811.2 - 932.4
DELTA LIQUID VOLUME (. V_f)			
RCS Shrinkage, Ft^3	-99.4	0	+99.4
PZR Shrinkage, Ft^3	-60.0	-32.1	-7.0
Liquid Flashed, Ft^3	-68.9	-40.4	-5.3
Liquid Boiled by Heaters Ft^3	-8.6	-12.6	-3.7
Liquid Added to RCS by HPI Ft^3	+45.9	+291.9	+37.8
STEAM VOLUME (V_g) Ft^3	704.65 - 895.6	895.6 - 688.8	688.8 - 567.6
STEAM MASS (W_g)			
Steam at Beginning of Interval lbm	4181.9	3936.7	2316.2
Added by Flashing, lbm	2812.6	1741.7	231.4
Added by Heating, lbm	350.	544.4	161.8
Steam at end of Interval, lbm	3936.7	2316.2	1834.5
Balance lost thru PORV, lbm	3407.8	3906.6	874.9
Steam Relieving Rate (lbm/Sec)	31.0	20.6	14.6

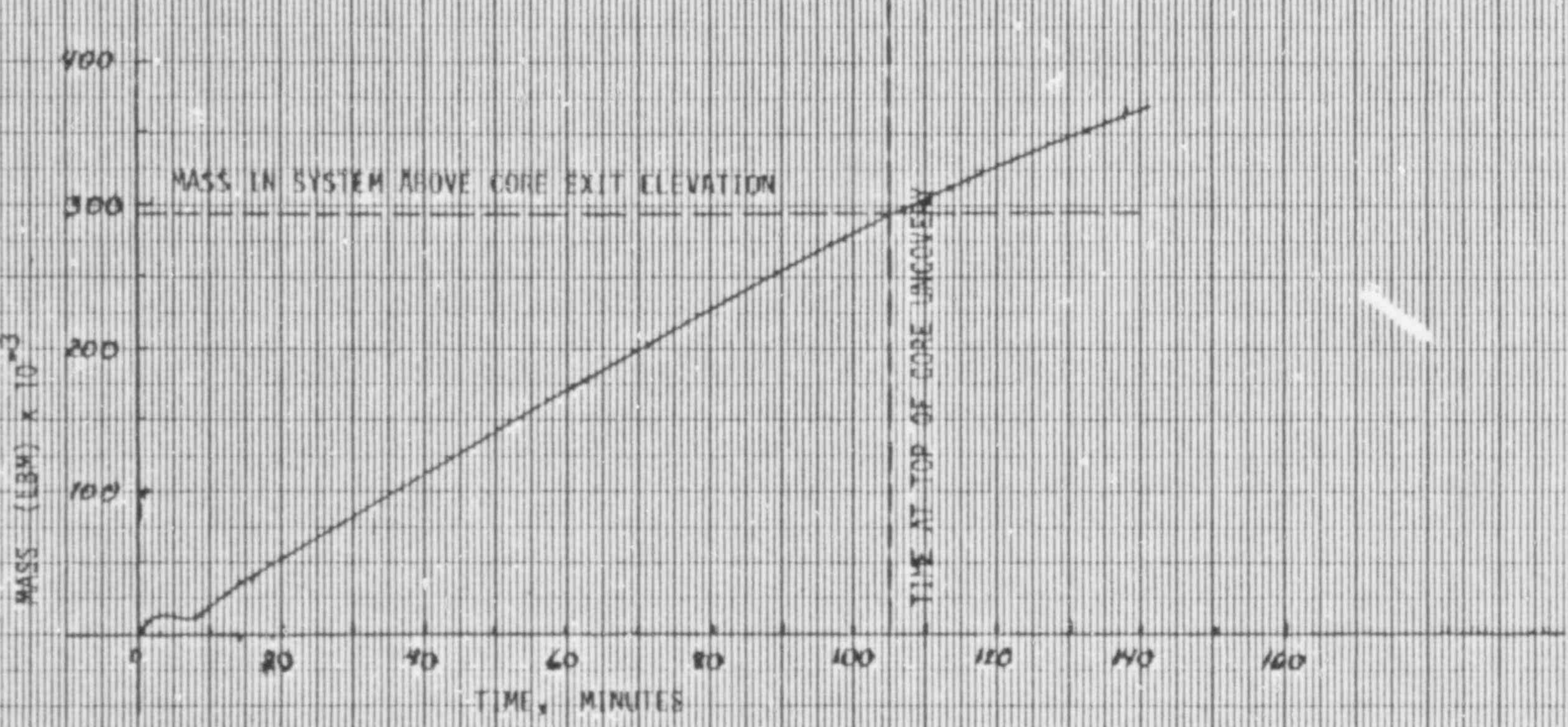
TABLE III (Cont'd)

Equivalent Relief Rate (lbm/Hour) @ 2250 psi		134150	114502	92279
PZR LIQUID LEVEL:	Measured (in)	220-195	195-376	376-400
	Calculated (in)	220-160.4	160.4-224.9	224.9-262.7

FIGURE 1

TMI-2

INTEGRAL MASS FLOW OUT OF SYSTEM NECESSARY
TO UNCOVER CORE AT T = 105 MINUTES



NECESSARY RELIEVING RATE =
395,000 LBM/HR @ 2250 PSI

$W_{out} = (P/V)^{1/2}$

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K-E 10 X 10 TO THE CENTIMETER 18 X 25 CM
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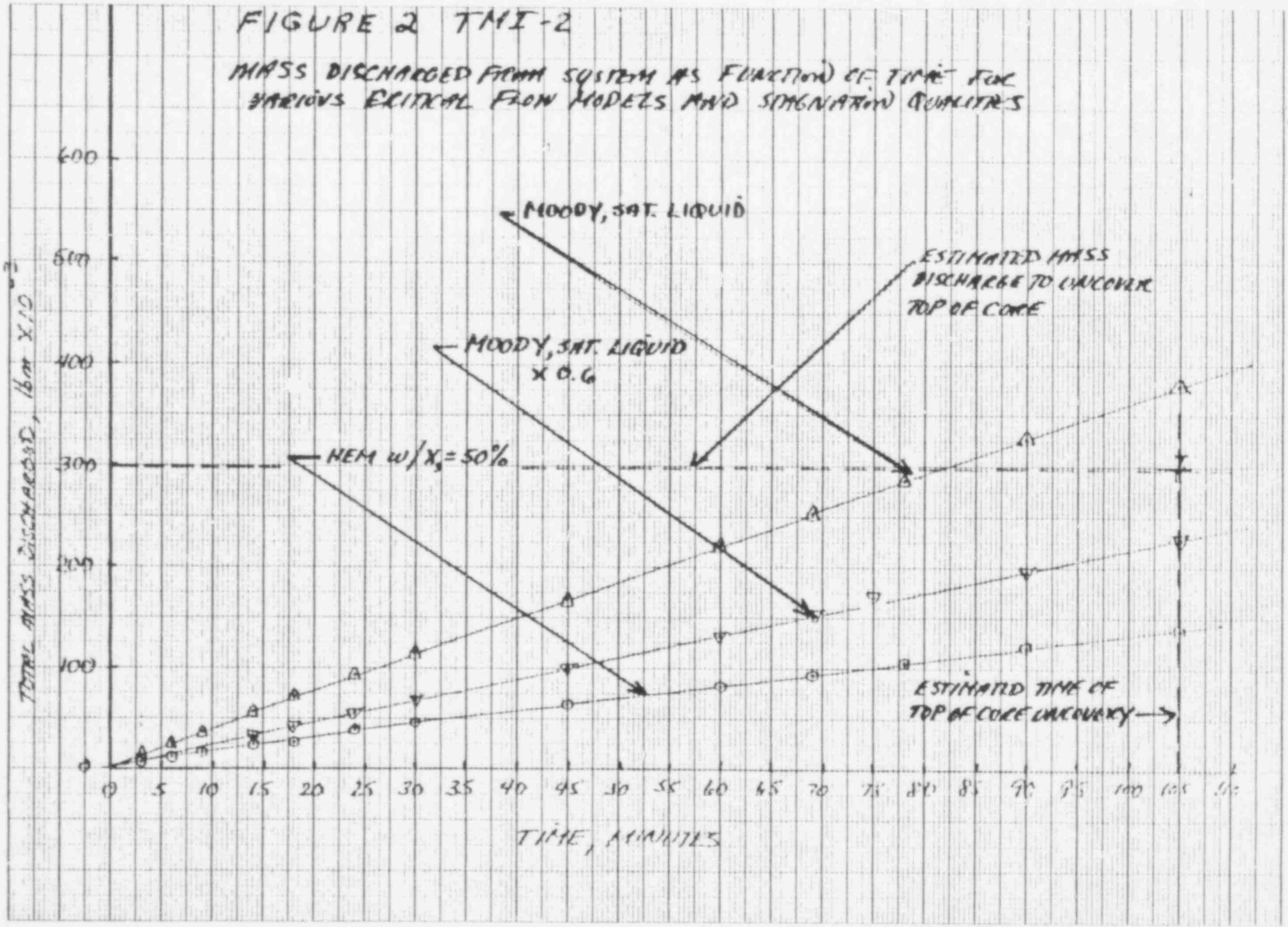
FIGURE 2 TMI-2

MASS DISCHARGED FROM SYSTEM AS FUNCTION OF TIME FOR VARIOUS CRITICAL FLOW MODELS AND STAGNATION QUALITIES



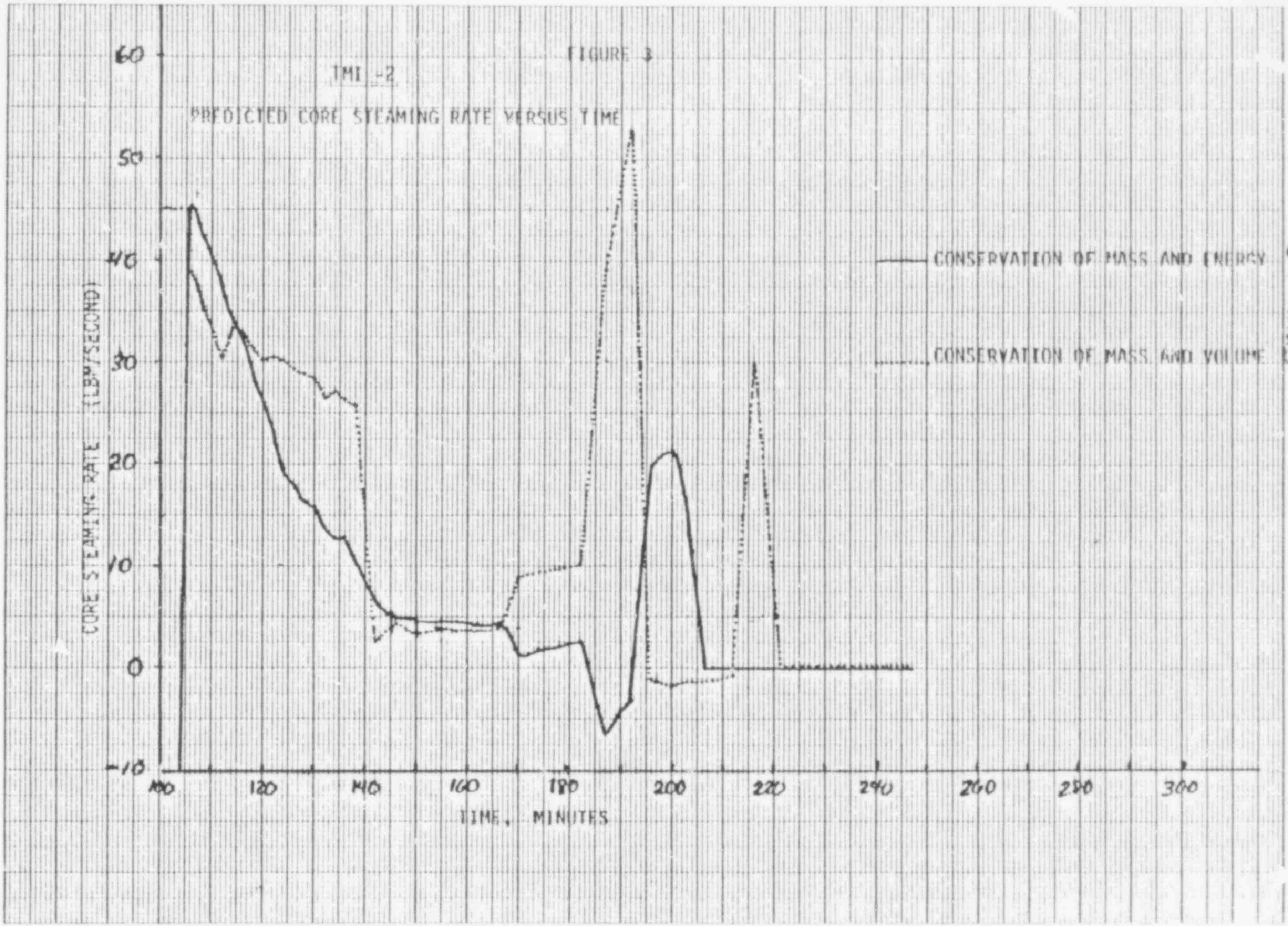


FIGURE 3

TMI -2

PREDICTED CORE STEAMING RATE VERSUS TIME

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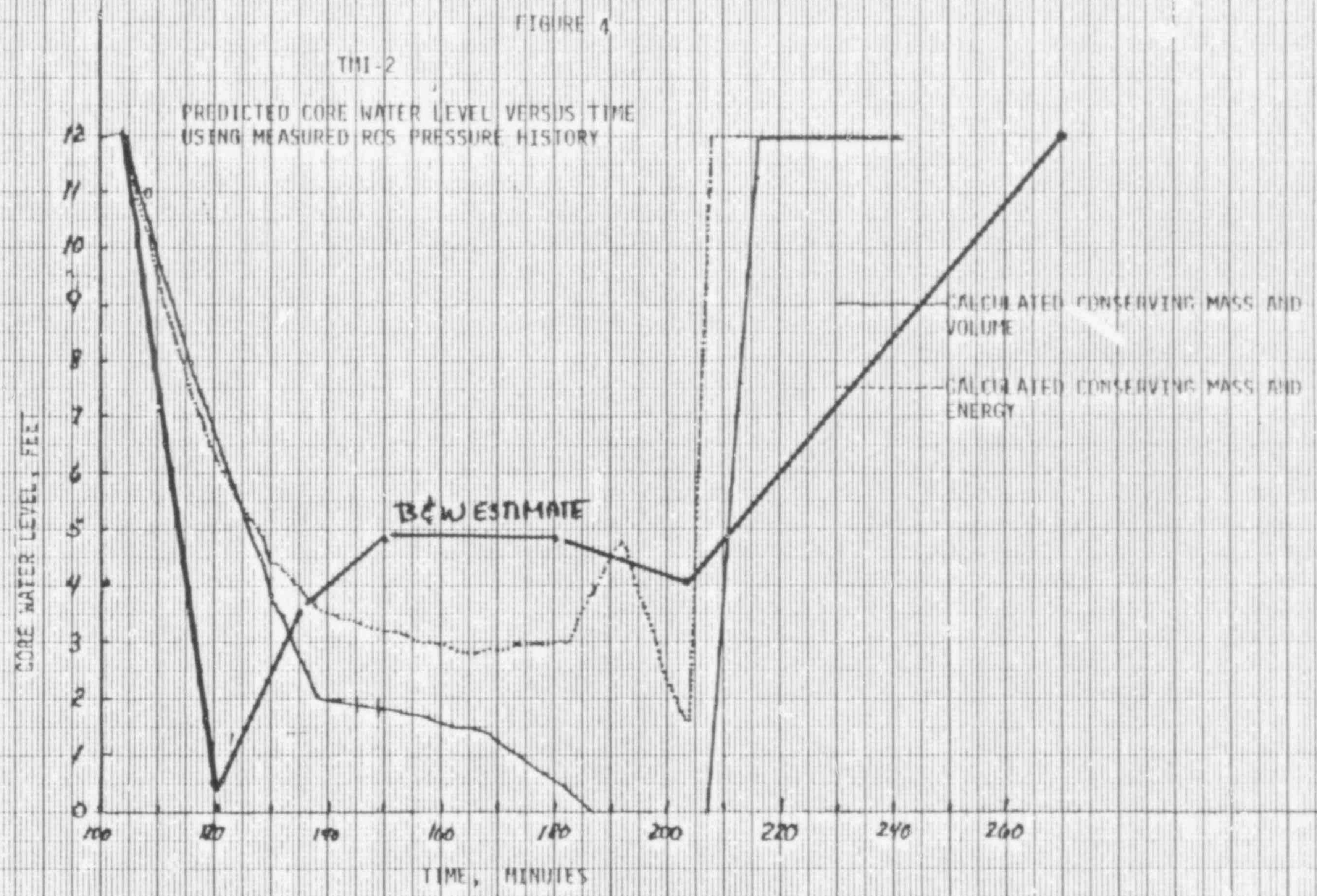
FIGURE 4

TMI-2

PREDICTED CORE WATER LEVEL VERSUS TIME USING MEASURED RCS PRESSURE HISTORY



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