LA-UR -79-2849

TITLE: ANALYSIS OF THREE-MILE-ISLAND ACCIDENT USING TRAC

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SUBMITTED TO: Seventh Water Reactor Safety Research Information Meeting, November 5-7, 1979.

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ANALYSIS OF THREE-MILE-ISLAND ACCIDENT USING TRAC*

by

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The Three Mile Island nuclear plant (Unit 2) was modeled using the TRAC code and a base case calculation, which simulated the first 3 h of the accident that occurred on March 28, 1979, was performed. The purpose of this calculation was to provide a better understanding of the system thermal-hydraulic response during the first 3 h and to evaluate how well TRAC could predict the overall accident scenario. As a result of this calculation, estimates were made regarding the extent of core damage and hydrogen production.[1,2] These estimates, along with some important system parameter comparisons will be presented in this paper. The loop fluid temperature comparisons are only shown to 2 h since the hot-leg temperature data was reading off scale after this time.

The TRAC model of the TMI-2 system for these calculations used 24 cells in the reactor vessel and 42 cells for the two system loops. The core fuel rods were modeled initially using three axial levels and two azimuthal regions per level, with average, high power, and low power fuel rods per region. This vessel noding was used to calculate the steady-state system conditions and the first 81 min of the transient. The pressurizer relief valve (EMOV) was modeled using a pipe module, allowing a direct calculation of the flow out of the EMOV. The once-through steam generators (OTSG) were modeled on both primary and secondary sides, but boundary conditions were used to model the balance of the secondary system. Based on the TMI-2 recorded power level, a TRAC steady-state calculation was performed to generate the initial conditions prior to the accident. These conditions are in very good agreement with available TMI-2 data.[1,2]

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*Work performed under the auspices of the United States Nuclear Regulatory Commission.

Using these self-consistent initial conditions, the TRAC transient calculation was initiated. Operator and system actions were simulated in TRAC using plant data, event chronologies, and in certain cases, assumptions necessary to give results which matched known system conditions. The first 30 min of the accident sequence are well simulated by TRAC, particularly system pressure, loop temperatures, and pressurizer level. During the period from 30 to 81 min coolant is continuously lost through the EMOV and the letdown system. Calculated core temperatures remain low, however, due to the good cooling provided by boiling in the core, which offsets the coolant losses and maintains a stable system pressure.

At 81 min, a more finely noded vessel model was used to provide more axial levels. This enhances the accuracy of predictions of the core thermal conditions and two-phase natural circulation through the system. Due to continual coolant loss, calculated core void fractions increase and primary coolant pump flow rates slowly decrease due to void formation in the coolant. The primary system pressure falls steadily after 91 min as increased auxiliary feedwater flow is introduced into the A loop OTSG. After the A loop pumps are tripped at 100 min, phase separation occurs throughout the system. This results in partial core uncovering and loss of coolant circulation through the loops. At 120 min, upper core temperatures begin to rise rapidly (0.25 K/s). At 138 min, the EMOV block valve is closed resulting in a gradual increase in core liquid inventory. At about 160 min, the water inventory in the core has boiled down again such that water is present in the lower plenum and partially in the lower core, resulting in a steep axial temperature gradient in the core. Since upward-moving steam velocities are very low (less than 0.1 m/s) the steam becomes very superheated in the upper part of the core and, as a result, the cladding and fuel heat up rapidly. When the cladding temperatures reach 1300 K, zirconium-steam reactions (exothermic) begin and the upper core temperatures begin rising at about 1.0 K/s. This temperature excursion was probably terminated in the accident when the HPI was returned to nonthrottled flow rates at 3 h and 20 min, enchancing the core cooling rate (TRAC calculations were terminated at 3 h since the core modeling was no longer realistic).

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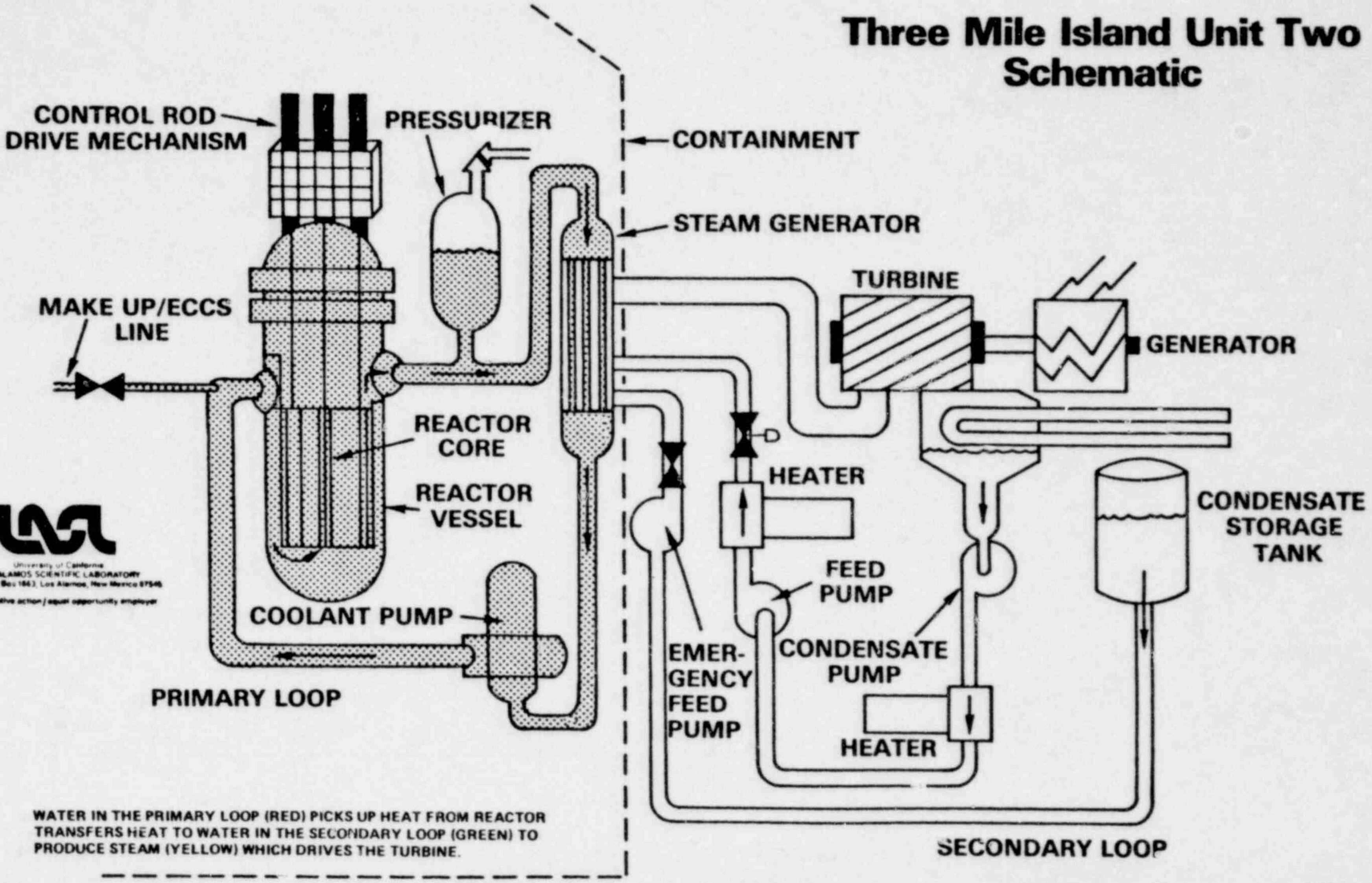
Overall, for the sequence of events and assumptions used, the TRAC results show good agreement with measured system parameters out to about 3 h. Using the calculated fuel rod temperature and system pressure histories, it was estimated that failure of nearly all the fuel rods in the upper 0.5 m of the core occurred at about 2 h and 30 min.[1] Also, cladding oxidation up to 3 h resulted in the production of approximately 40 kg of hydrogen.[1]

REFERENCES

1. "Preliminary Calculations Related to the Accident at Three Mile Island," Los Alamos Scientific Laboratory report LA-UR-79-2425, August 1979.

2. J. F. Jackson and M. G. Stevenson," Nuclear Reactor Safety Progress Report for the period July 1 - September 30, 1979," Los Alamos Scientific Laboratory report, to be published.

Three Mile Island Unit Two Schematic



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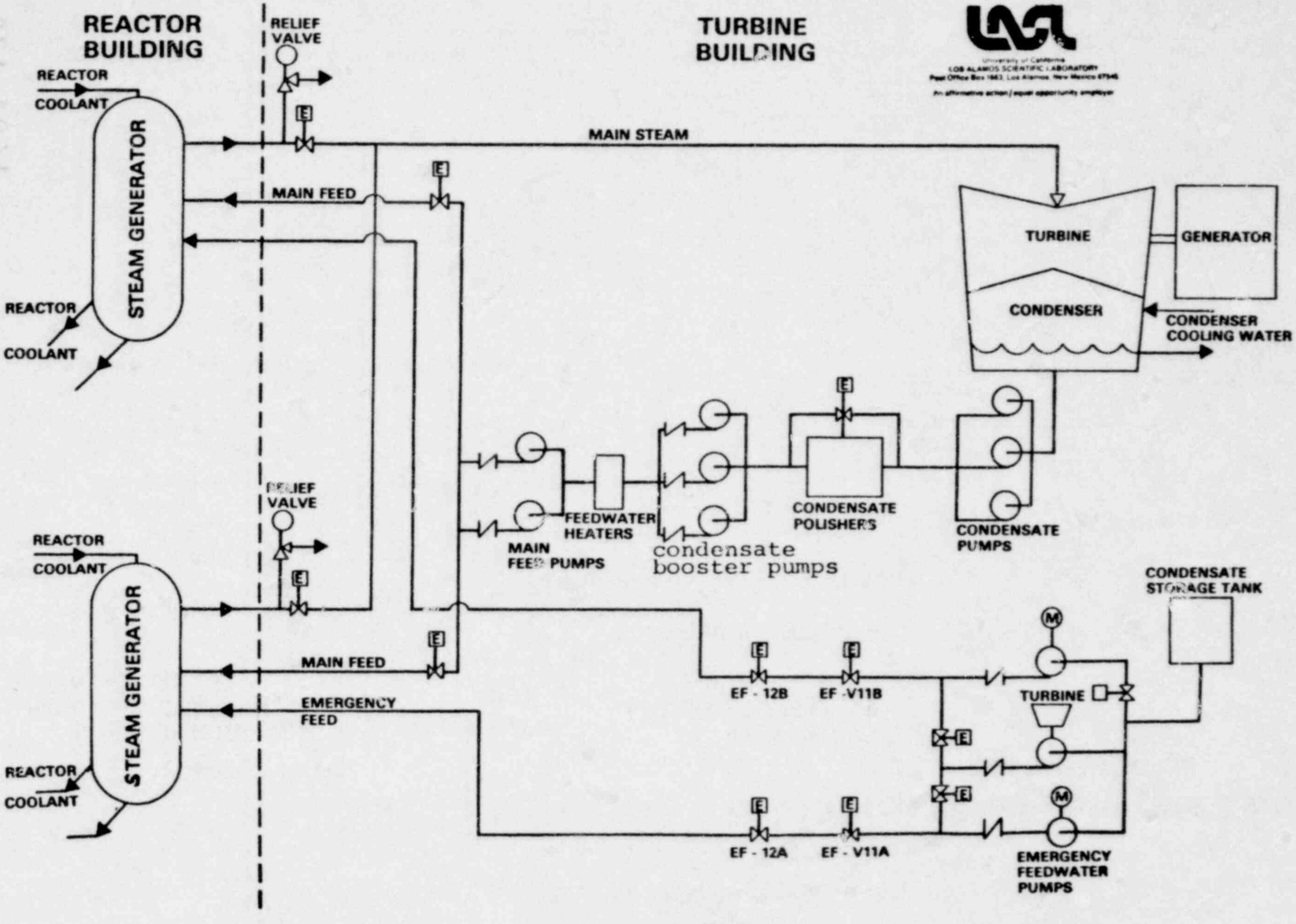
Steam, Feed and Condensate



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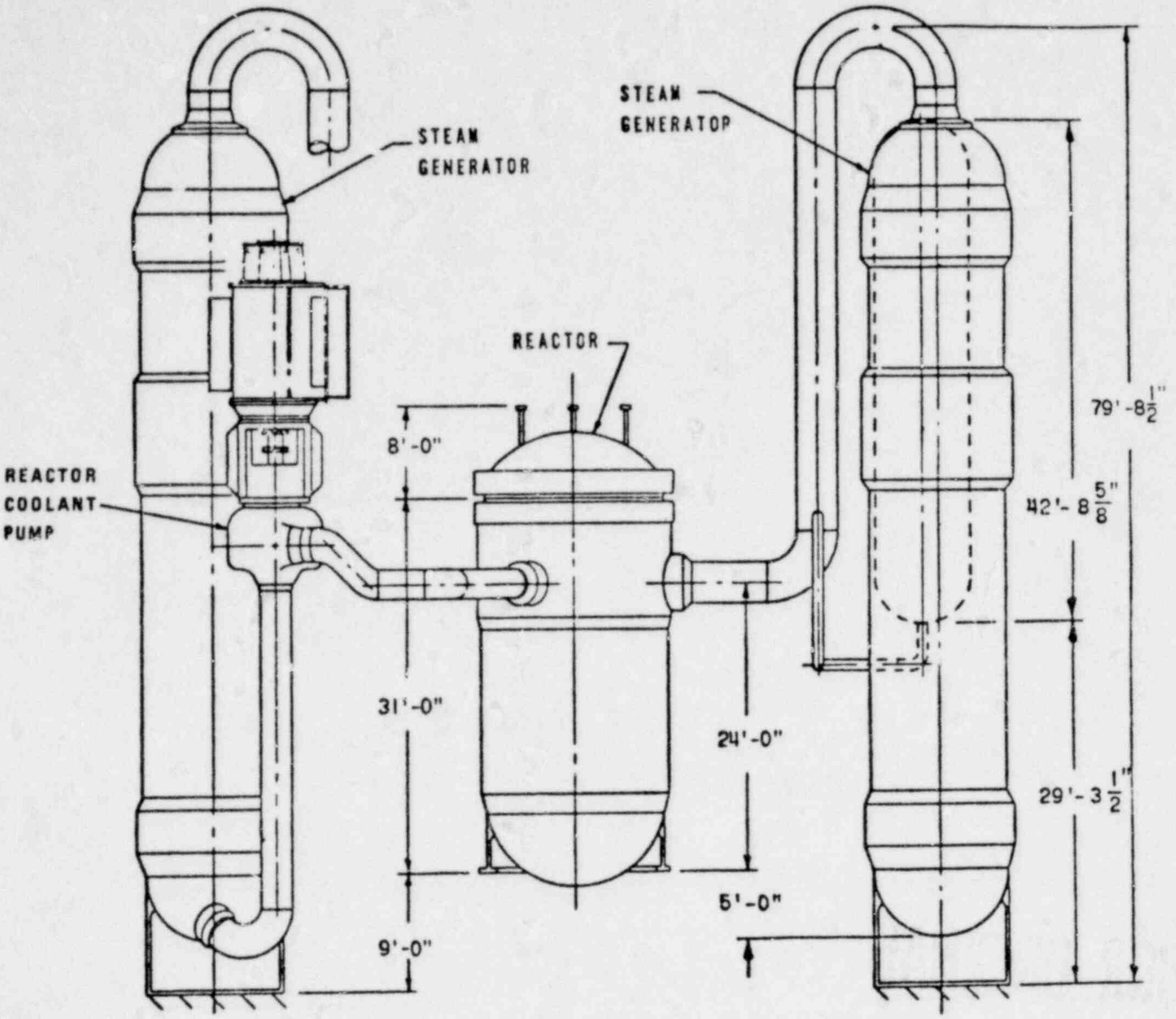

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REACTOR COOLANT SYSTEM ARRANGEMENT – ELEVATIC

THREE MILE ISLAND NUCLEAR STATION UNIT 2

FIGURE 5.1-5

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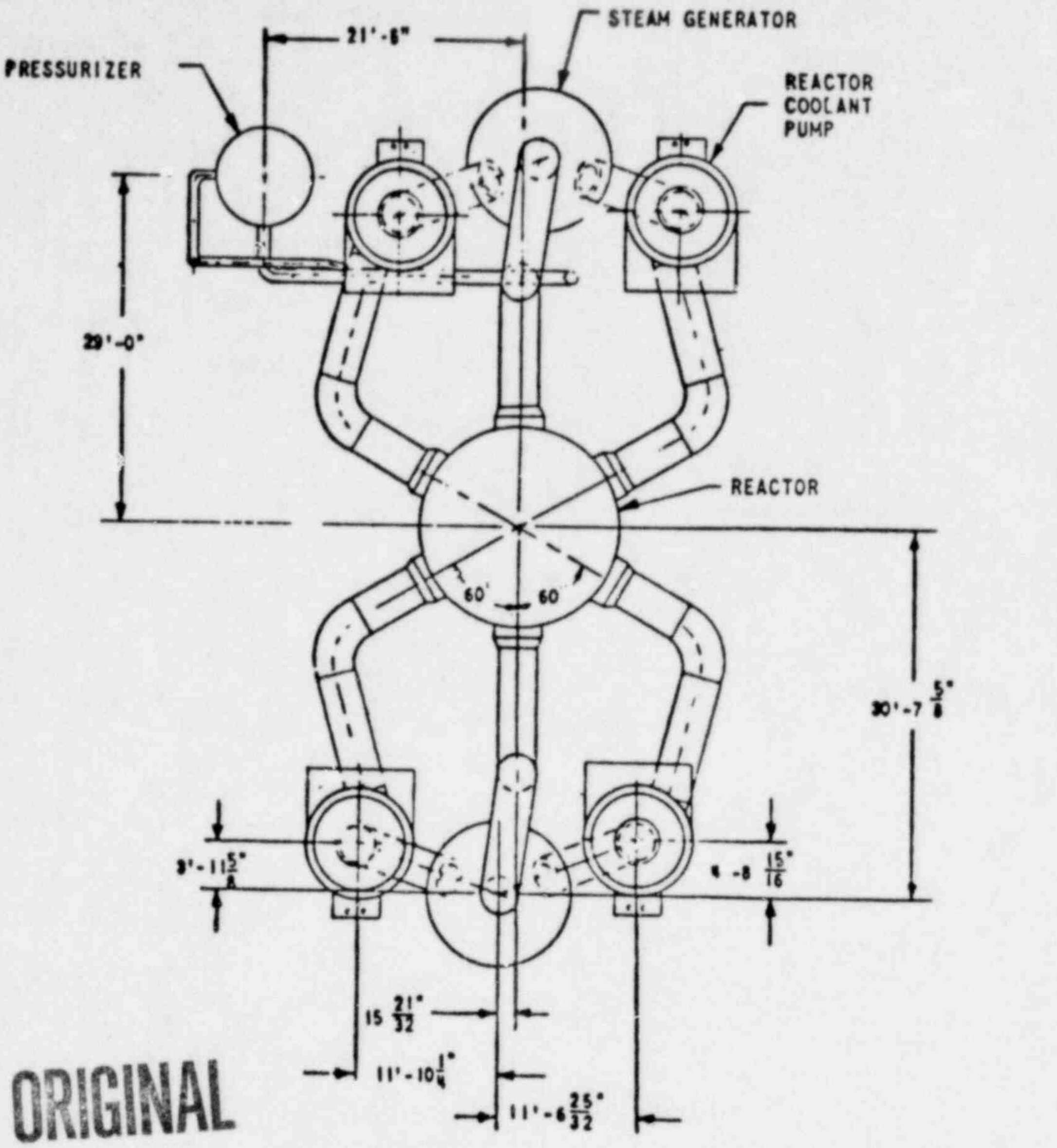
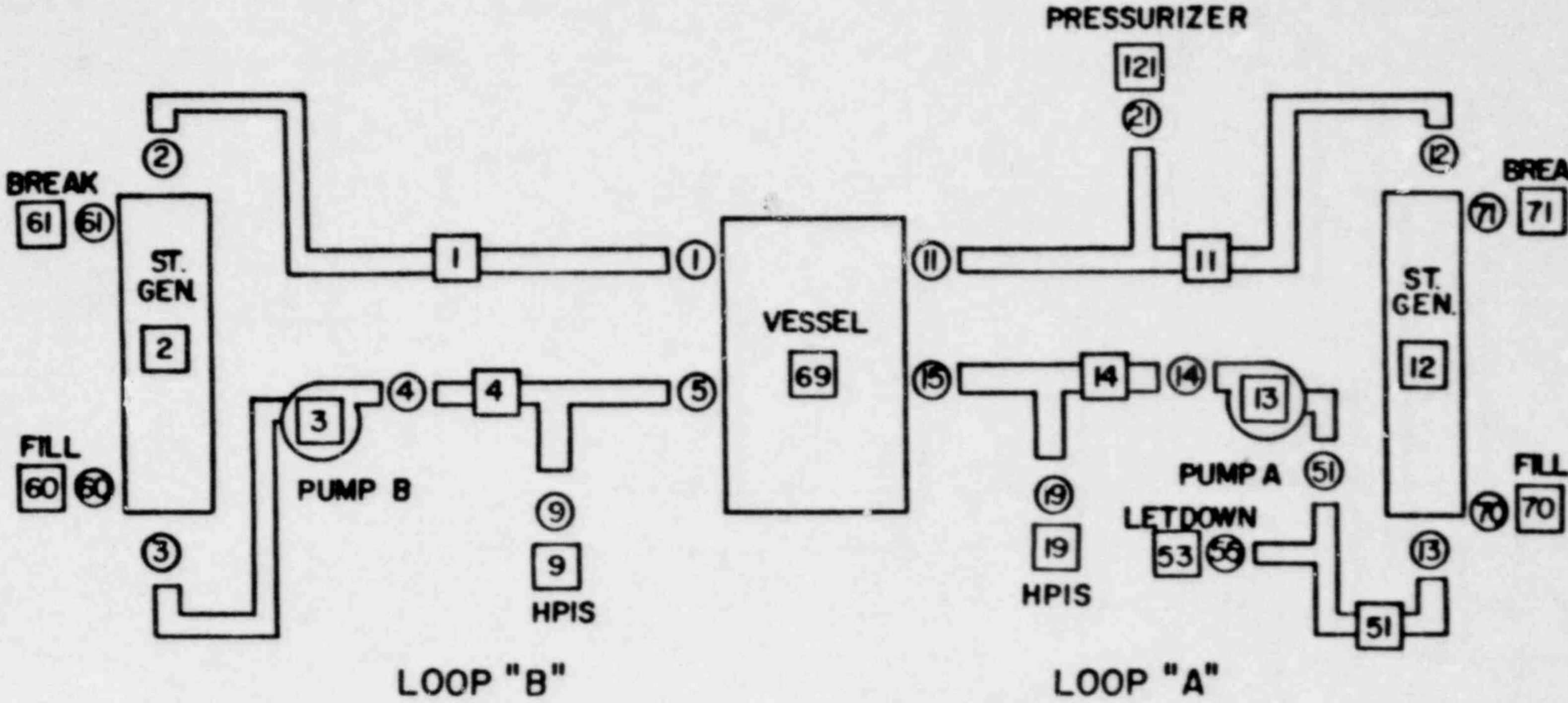


REACTOR COOLANT SYSTEM ARRANGEMENT – PLAN

THREE MILE ISLAND NUCLEAR STATION UNIT 2

FIGURE 5.1-4



COMPONENT NO.

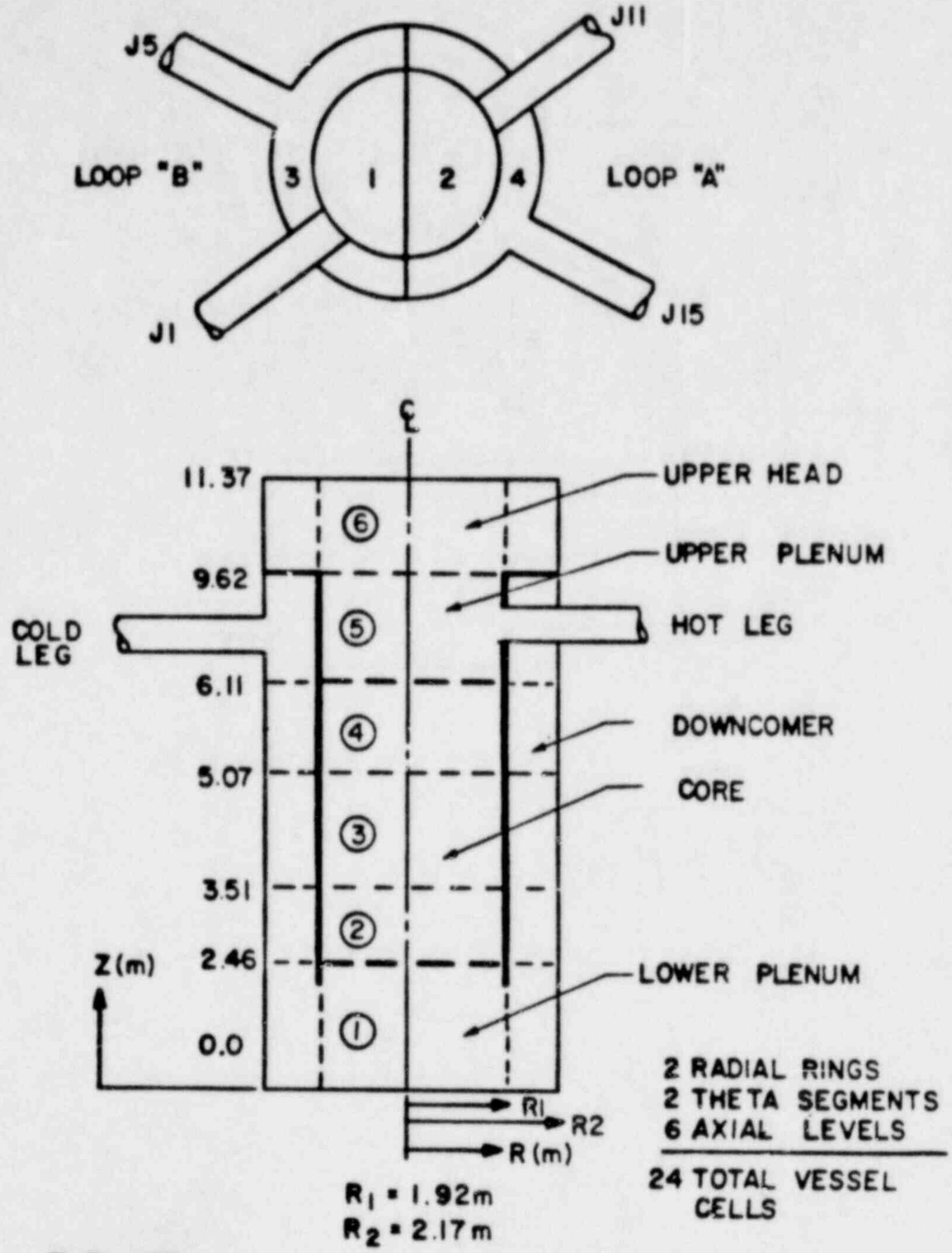
JUNCTION NO.

18 COMPONENTS

19 JUNCTIONS

24 VESSEL CELLS
42 I-D CELLS
66 TOTAL CELLS

TRAC noding schematic for TMI-2.



Vessel noding used for first 81 minutes.

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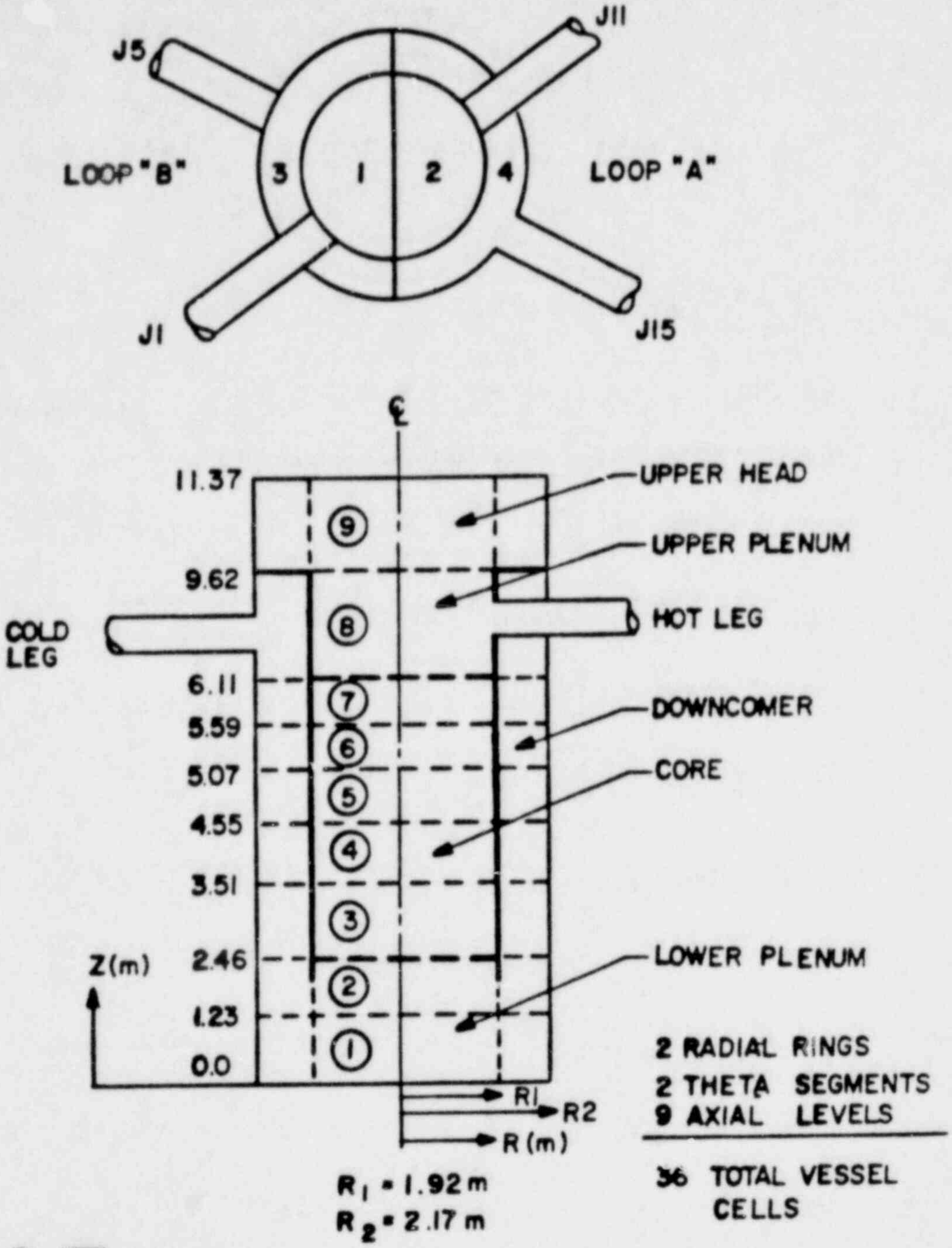

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Vessel noding used after 81 minutes.

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TABLE I

THREE MILE ISLAND - UNIT 2

TRAC INPUT PARAMETERS

	PARAMETER	VALUE
1.	INITIAL POWER (97% OF RATED)	$2.711\,78 \times 10^9$ W
2.	RELATIVE AXIAL POWER SHAPE (3 LEVELS - BOTTOM TO TOP)	0.64, 1.0, 0.76
3.	RELATIVE RADIAL POWER SHAPE	1.0
4.	CORE AVERAGE LINEAR POWER	$2.014\,4 \times 10^4$ W/m
5.	PEAK ROD LINEAR POWER	$2.444\,2 \times 10^4$ W/m
6.	HIGH POWER ROD LINEAR POWER	$3.589\,2 \times 10^4$ W/m
7.	LOW POWER ROD LINEAR POWER	$1.197\,5 \times 10^4$ W/m
8.	PRESSURIZER PRESSURE	$1.477\,21 \times 10^7$ Pa

TABLE II

THREE MILE ISLAND - UNIT 2

CALCULATED INITIAL CONDITIONS AT STEADY STATE

	PARAMETER	TRAC	CRAFT-2
1.	AVERAGE HOT-LEG TEMPERATURE AT VESSEL OUTLET (K)	592.3	593.0
2.	AVERAGE COLD-LEG TEMPERATURE AT VESSEL INLET (K)	564.1	564.5
3.	TOTAL PRIMARY SYSTEM FLOW RATE (2 LOOPS) (KG/S)	17 027.0	17 375.5
4.	AVERAGE HOT-LEG PRESSURE AT VESSEL OUTLET (PA)	1.475×10^7	1.472×10^7
5.	AVERAGE COLD-LEG PRESSURE AT VESSEL INLET (PA)	1.504×10^7	1.534×10^7
6.	PUMP ΔP (PA)	7.87×10^5	7.87×10^5
7.	STEAM GENERATOR SECONDARY SIDE FLOW RATE (EACH) (KG/S)	700.0	
8.	AVERAGE STEAM GENERATOR SECONDARY SIDE PRESSURE (PA)	65.5×10^6	
9.	CLADDING SURFACE TEMPERATURES AT CORE LEVEL 2: (K)		
	A. AVERAGE ROD	605.0	
	B. HIGH POWER ROD	614.1	
	C. LOW POWER ROD	595.0	
10.	TOTAL PRIMARY SYSTEM WATER MASS (KG)	2.774×10^5	2.765×10^5

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TABLE III

THREE MILE ISLAND - UNIT 2

BOUNDARY CONDITIONS

1. REACTOR POWER VS TIME

2. PUMP SPEED VS TIME:

A.	PUMP LOOP B:	$0 \le T \le 4\,380.0$	125.7 RAD/S
		$T > 4\,330.0$	0.0 RAD/S
B.	PUMP LOOP A:	$0 \le T \le 6\,000.0$	125.7 RAD/S
		$T > 6\,000.0$	0.0 RAD/S

3. HPI FLOW VS TIME

4. PRESSURIZER RELIEF VALVE BACK-PRESSURE VS TIME

5. STEAM GENERATOR STEAM LINE BACK-PRESSURE VS TIME

6. STEAM GENERATOR FEEDWATER FLOW VS TIME

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TABLE IV
THREE MILE ISLAND - UNIT 2
SEQUENCE OF EVENTS
USED FOR BASE CASE CALCULATION

TIME(S)	EVENT
0.0	LOSS OF FEEDWATER FLOW
10.5	TRIP REACTOR POWER
13.0	START MAKE-UP PUMP 1A FULL FLOW 27.5 KG/S
120.0	START MAKE-UP PUMP 1C FULL FLOW 27.5 KG/S
194.0	THROTTLE PUMPS 1A AND 1C TO 6.1 KG/S EACH
278.0	TRIP PUMP 1C - CONTINUE PUMP 1A AT 6.1 KG/S
300.0	INITIATE LETDOWN FLOW OF 8.6 KG/S
418.0	REDUCE LETDOWN FLOW TO 4.5 KG/S
480.0	START AUXILIARY FEEDWATER FLOW OF 31.3 KG/S (EACH OTSG)
624.0	TRIP PUMP 1A - CONTINUE LETDOWN FLOW
700.0	START PUMP 1A (MAKEUP + HPI = 1.85 KG/S)
3824.0	TURN OFF LETDOWN
4380.0	TRIP PRIMARY PUMPS - LOOP B
4860.0	TURN OFF HPI AND AUXILIARY FEEDWATER FLOW
5460.0	INITIATE HPI - 4.4 KG/S INITIATE LETDOWN - 4.4 KG/S INITIATE AUXILIARY FEEDWATER FLOW TO OTSG "A" - 31.3 KG/S

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CONTINUED TABLE IV

TIME(S)	EVENT
6000.0	TRIP PRIMARY PUMPS - LOOP A
6060.0	REDUCE HPI - 2.2 KG/S
	INCREASE LETDOWN - 15.0 KG/S
7170.0	TURN OFF HPI AND DECREASE LETDOWN - 4.5 KG/S
8280.0	SHUT PRESSURIZER BLOCK VALVE AND TURN OFF LETDOWN

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TABLE V

ASSUMPTIONS FOR TMI BASE CASE

1. DECAY POWER OBTAINED FROM "NUCLEAR LEGISLATIVE ADVISORY SERVICE," ISSUE 17, APRIL 13, 1979.

2. FEEDWATER FLOW VS TIME RAMPED TO ZERO OVER A 90 s TIME INTERVAL AT BEGINNING OF TRANSIENT. (90 s WAS USED IN ORDER TO ACCOUNT FOR THE STORED WATER MASS IN THE OTSG DOWNCOMER.)

3. MAKE-UP PUMP FULL FLOW CAPACITY OF 27.5 KG/s (EACH).

4. THROTTLED FLOW RATE FOR MAKE-UP PUMPS OF 6.1 KG/s.

5. LETDOWN FLOW IS ASSUMED TO BE EQUAL TO MAKE-UP FLOW FOR $T < 13$ s AND FOR $T > 8\,280.0$.

6. LETDOWN FLOW GREATER THAN MAKE-UP + HPI FOR $600 \leq T \leq 8\,280$ s.

7. AUXILIARY FEEDWATER FLOW IS 31.3 KG/s FOR EACH OTSG (LATER REDUCED TO MATCH SECONDARY SIDE WATER LEVEL).

8. PRESSURIZER RELIEF VALVE NODING DETERMINED BY USING RATED SATURATED STEAM FLOW CONDITIONS OF 15.0 KG/s.

9. FROM T = 101 MINUTES UNTIL 120 MINUTES, 15 KG/s LETDOWN FLOW WAS USED TO MATCH PRIMARY SYSTEM PRESSURE.

10. PRESSURIZER HEATERS AND SPRAYERS WERE NOT MODELED.

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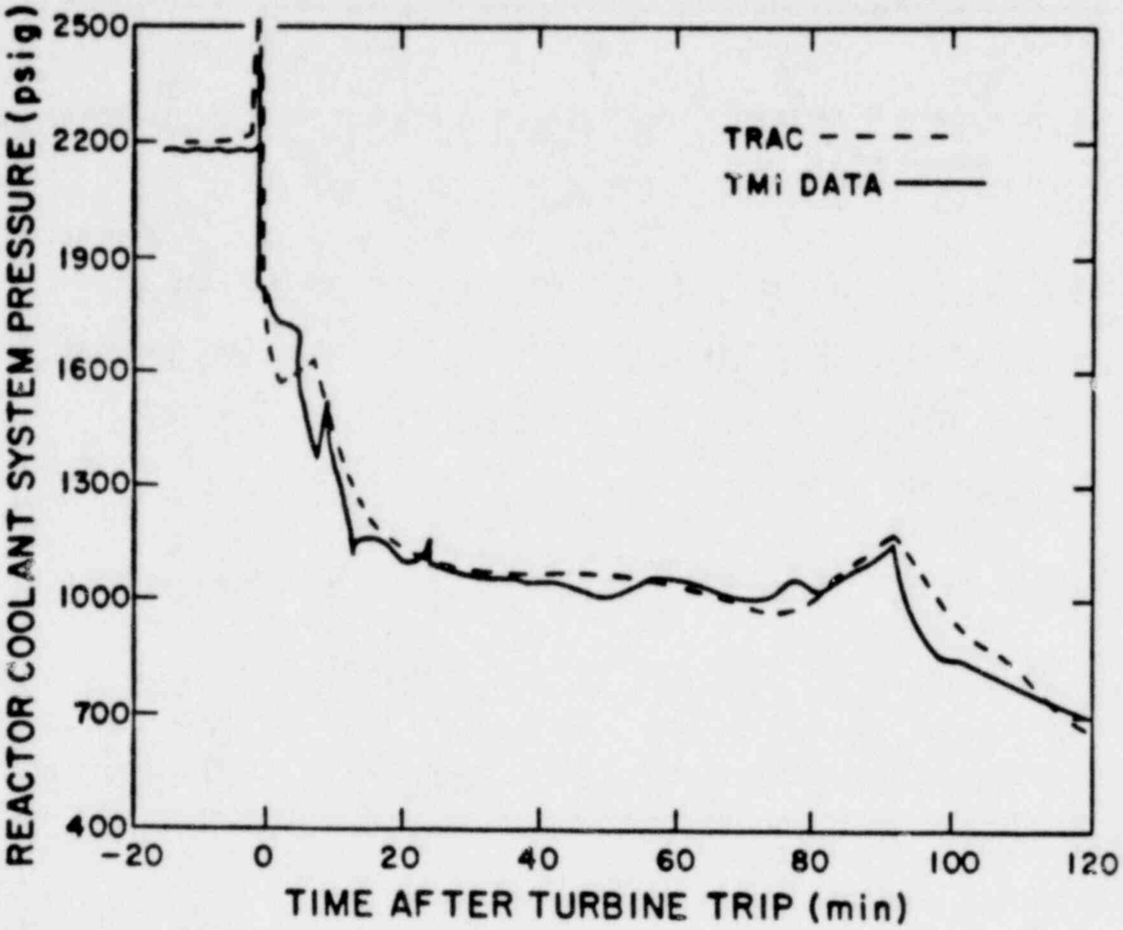

System pressure comparisons out to 120 minutes.

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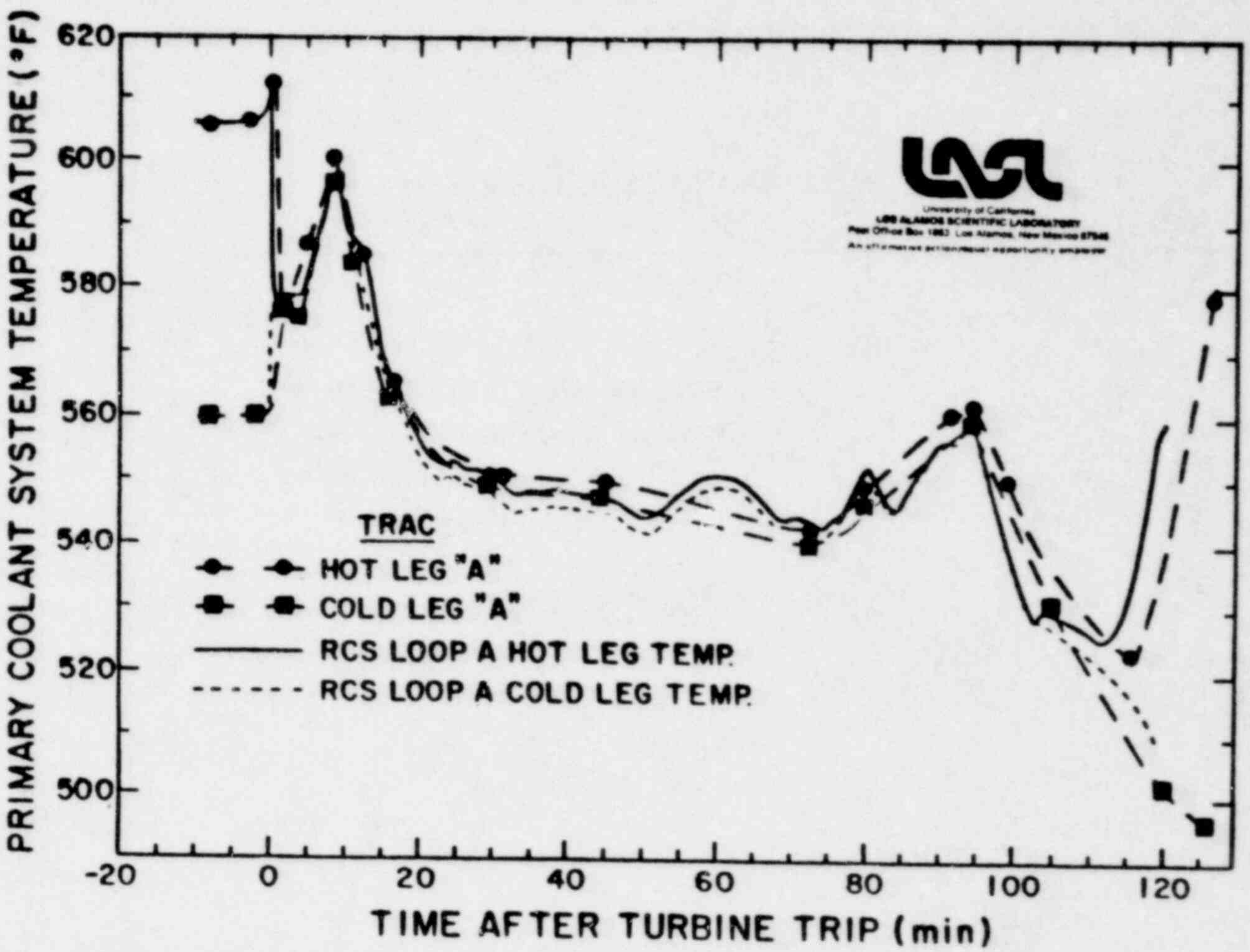

A LOop fluid temperature comparisons out to 120 minutes.

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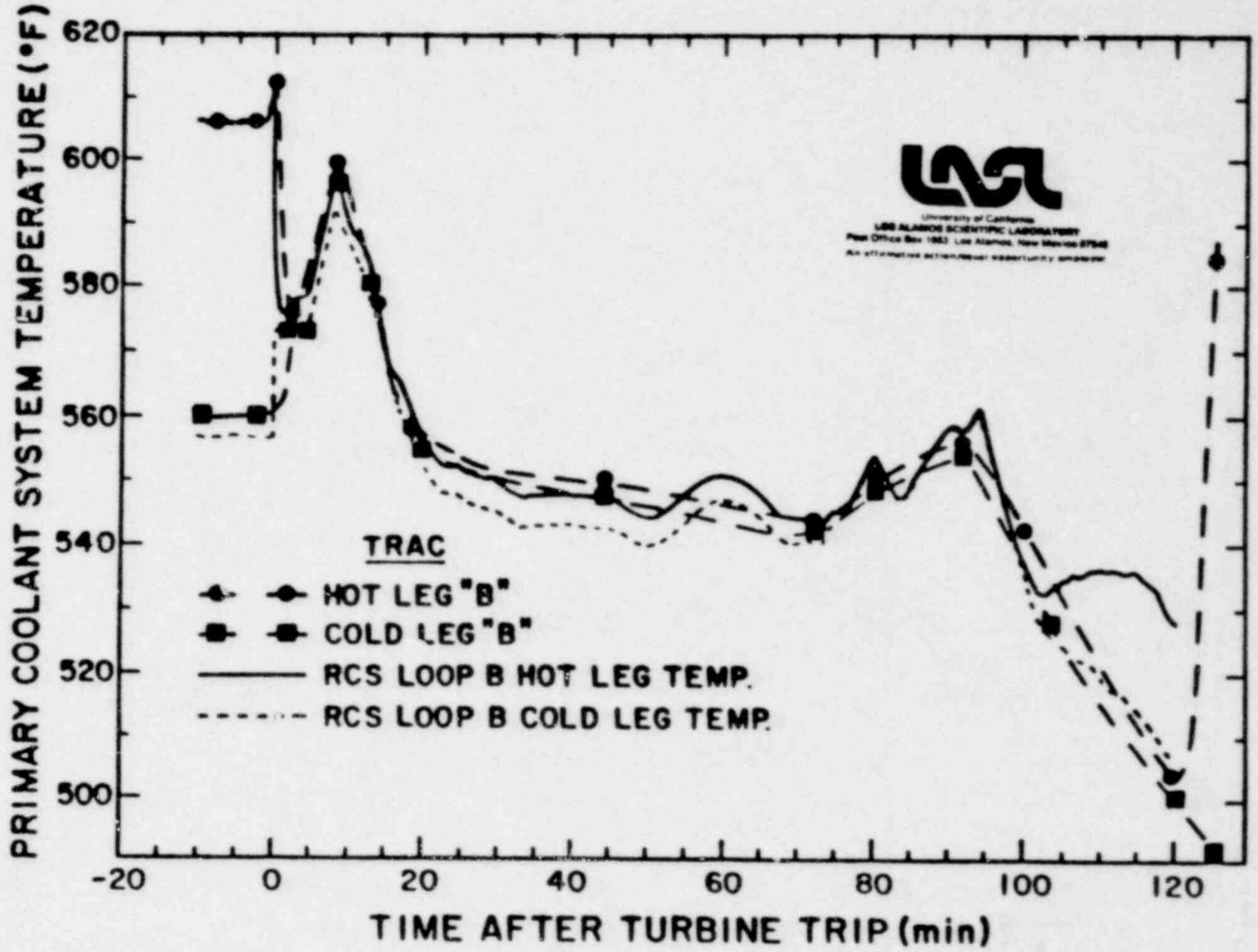

B Loop fluid temperature comparisons out to 120 minutes.

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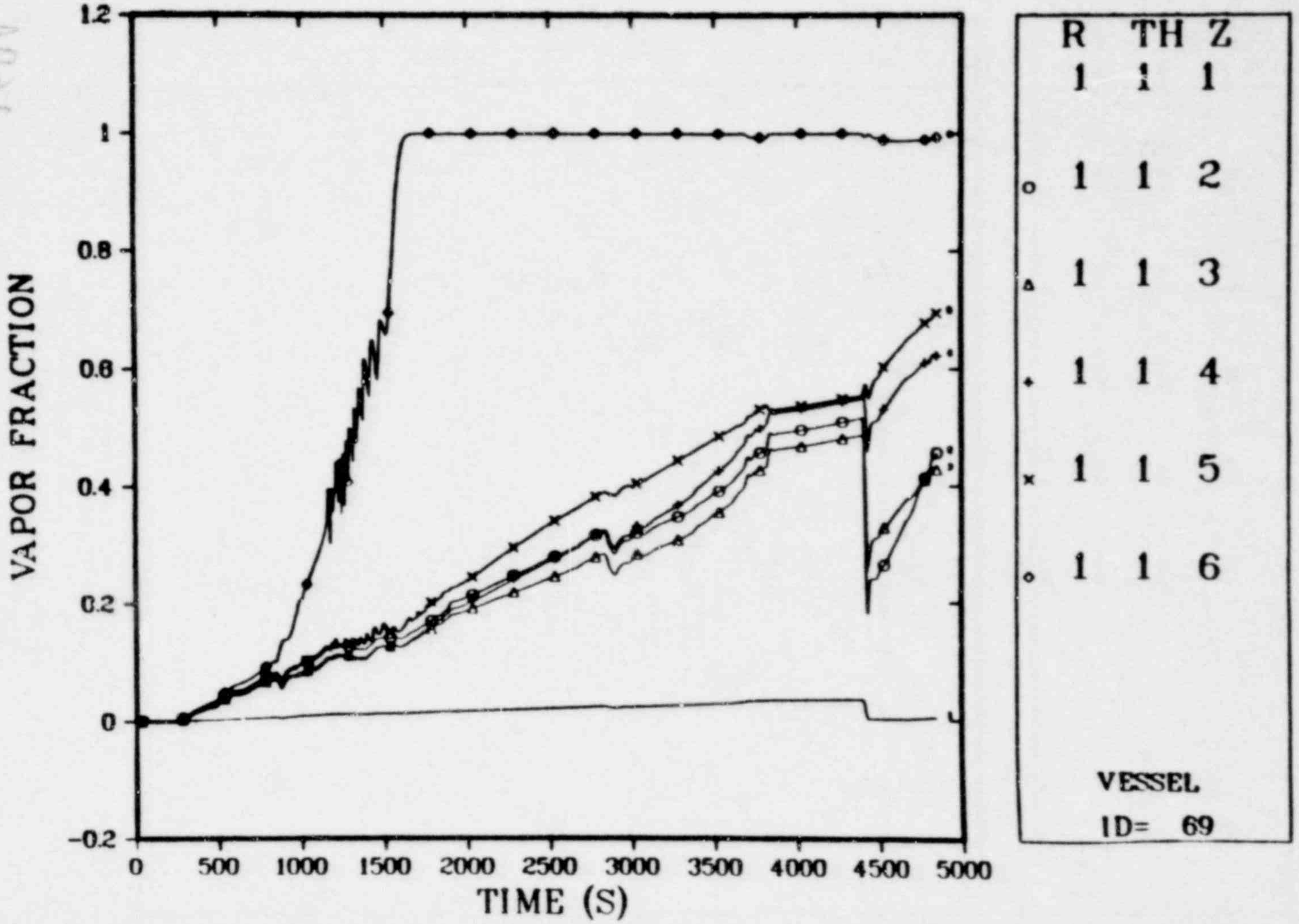
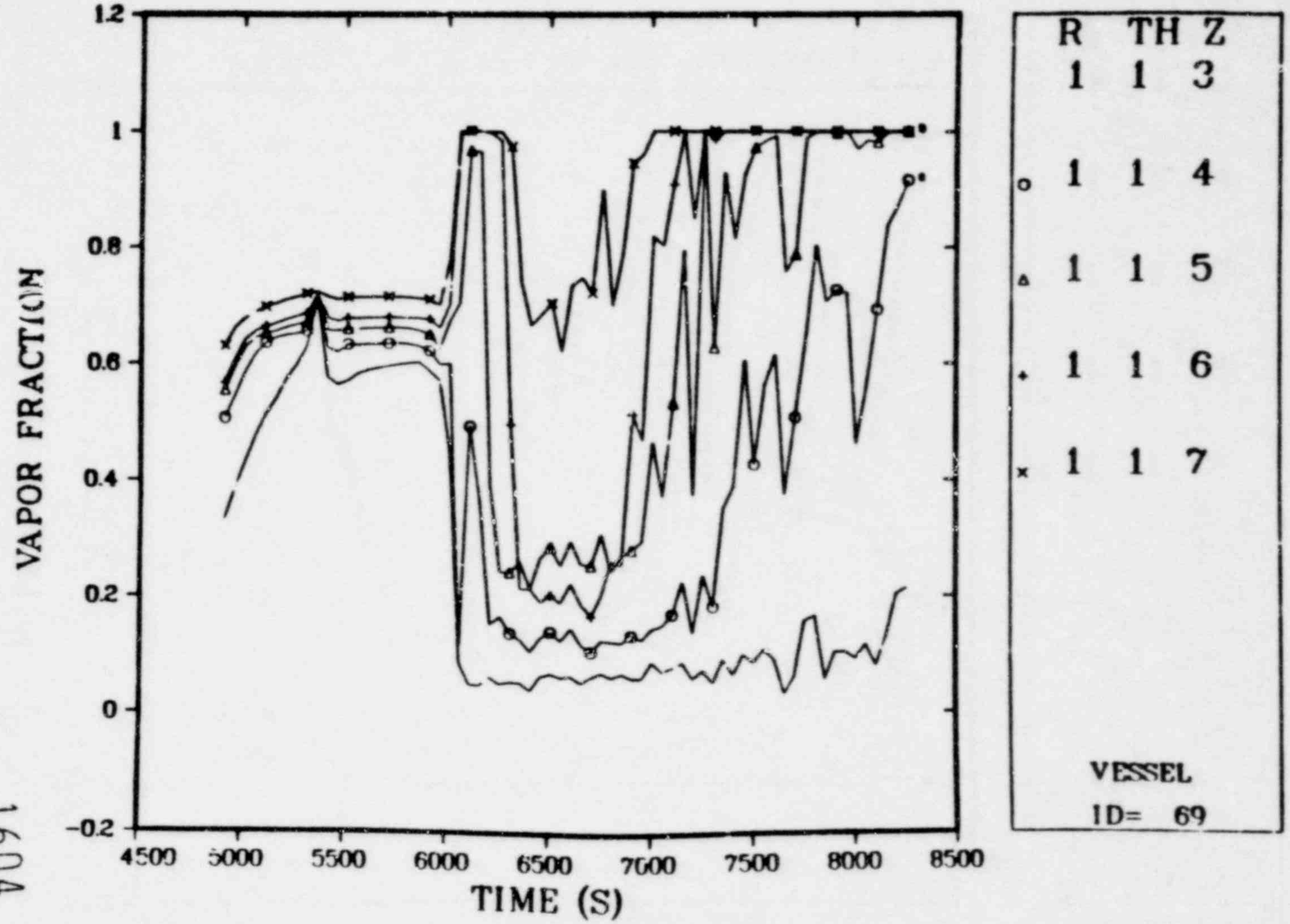


Vessel void fraction profile out to 81 minutes.



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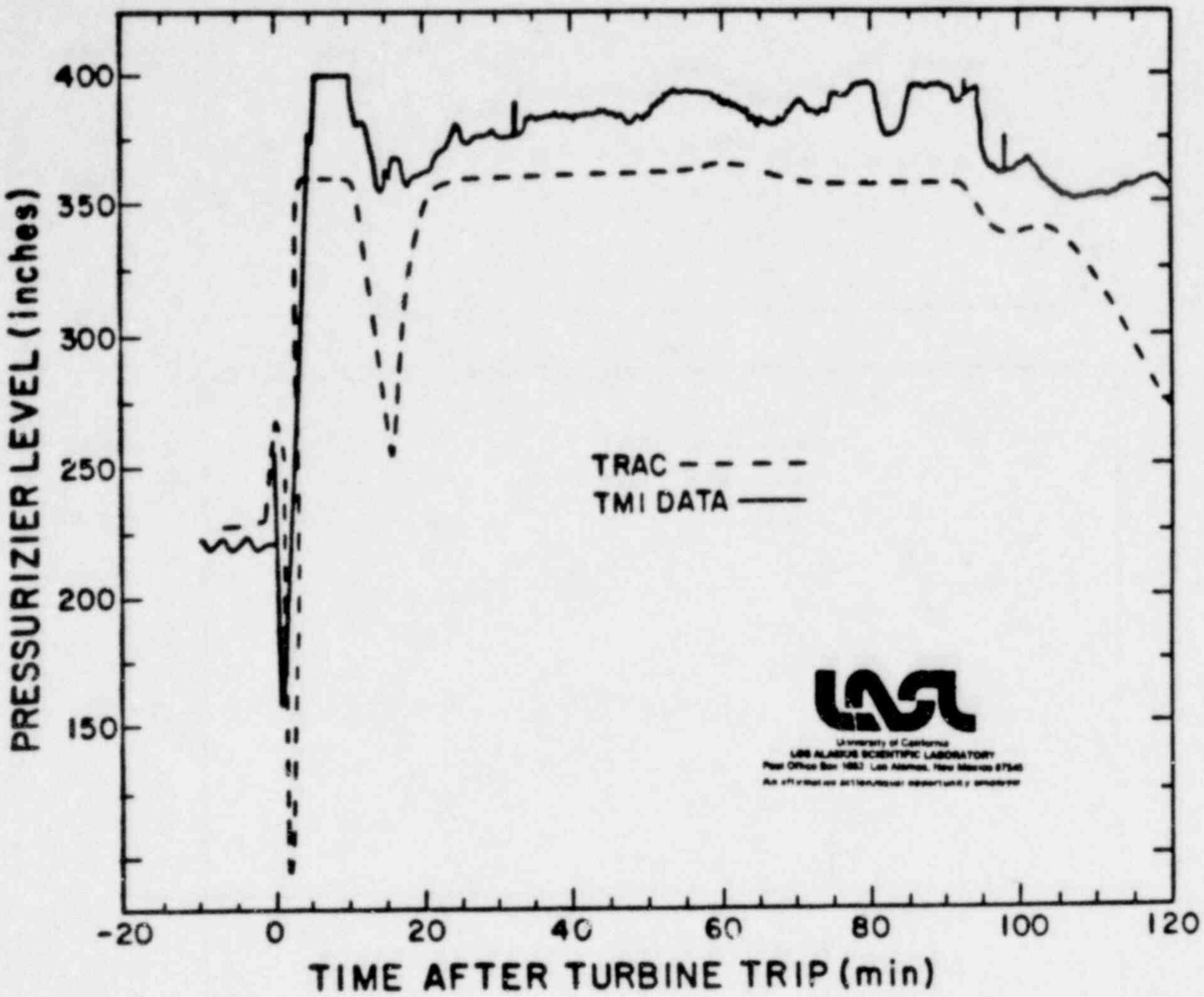
Core void fraction axial profile after 81 minutes.



Pressurizer water level comparisons out to 120 minutes.

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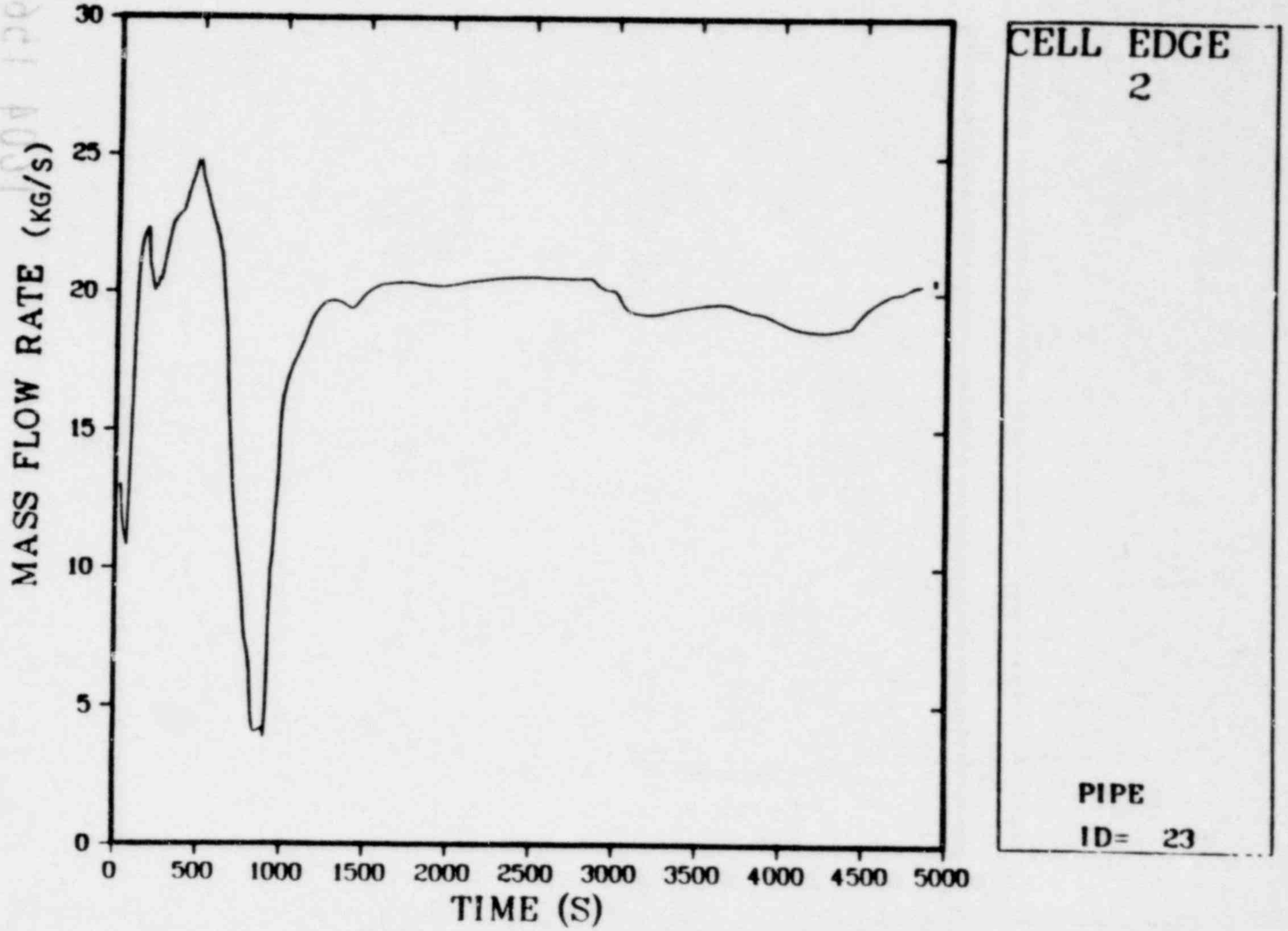

Pressurizer relief valve flow rate for first 81 minutes.

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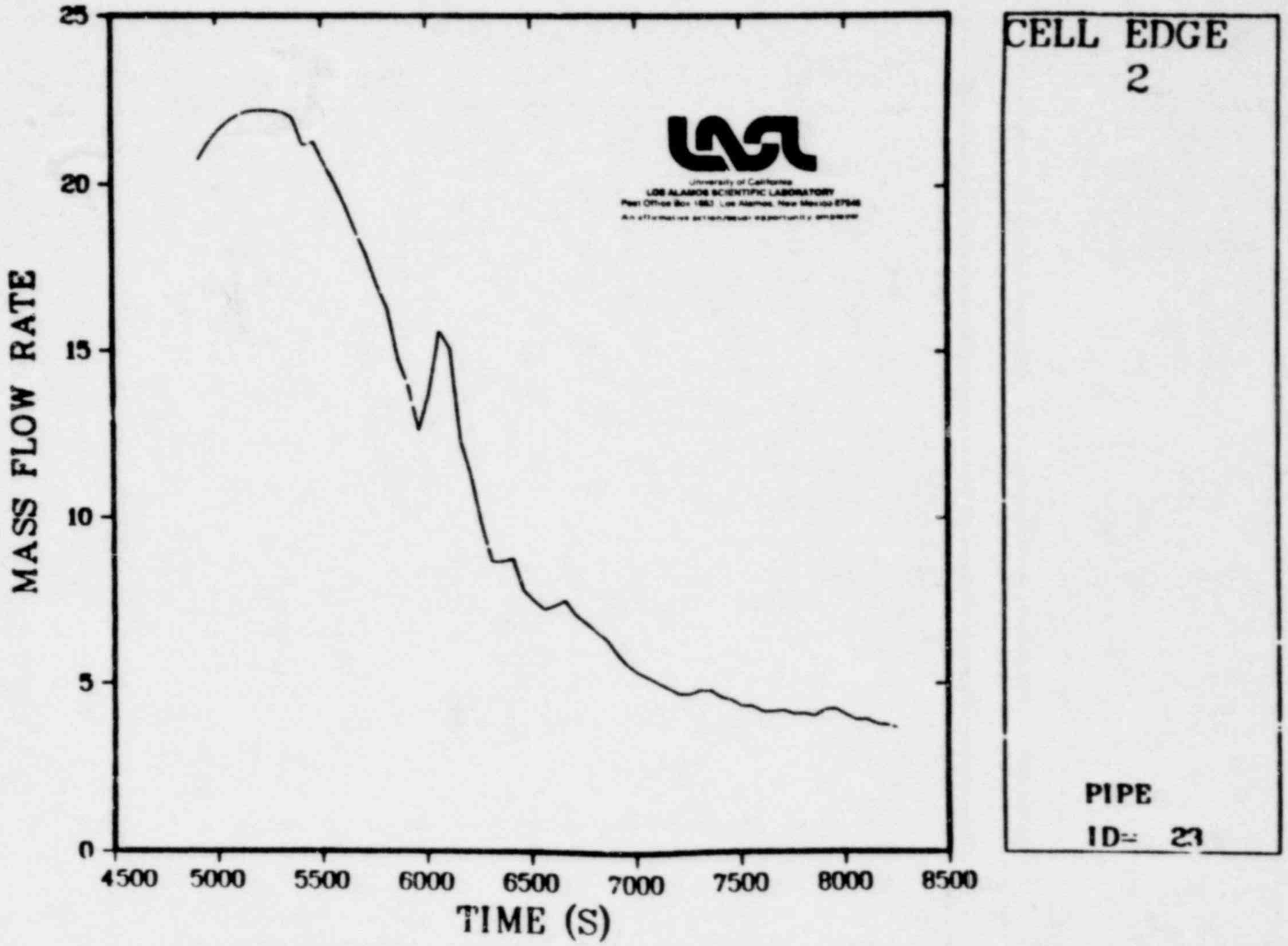

Pressurizer relief valve flow rate after 81 minutes.

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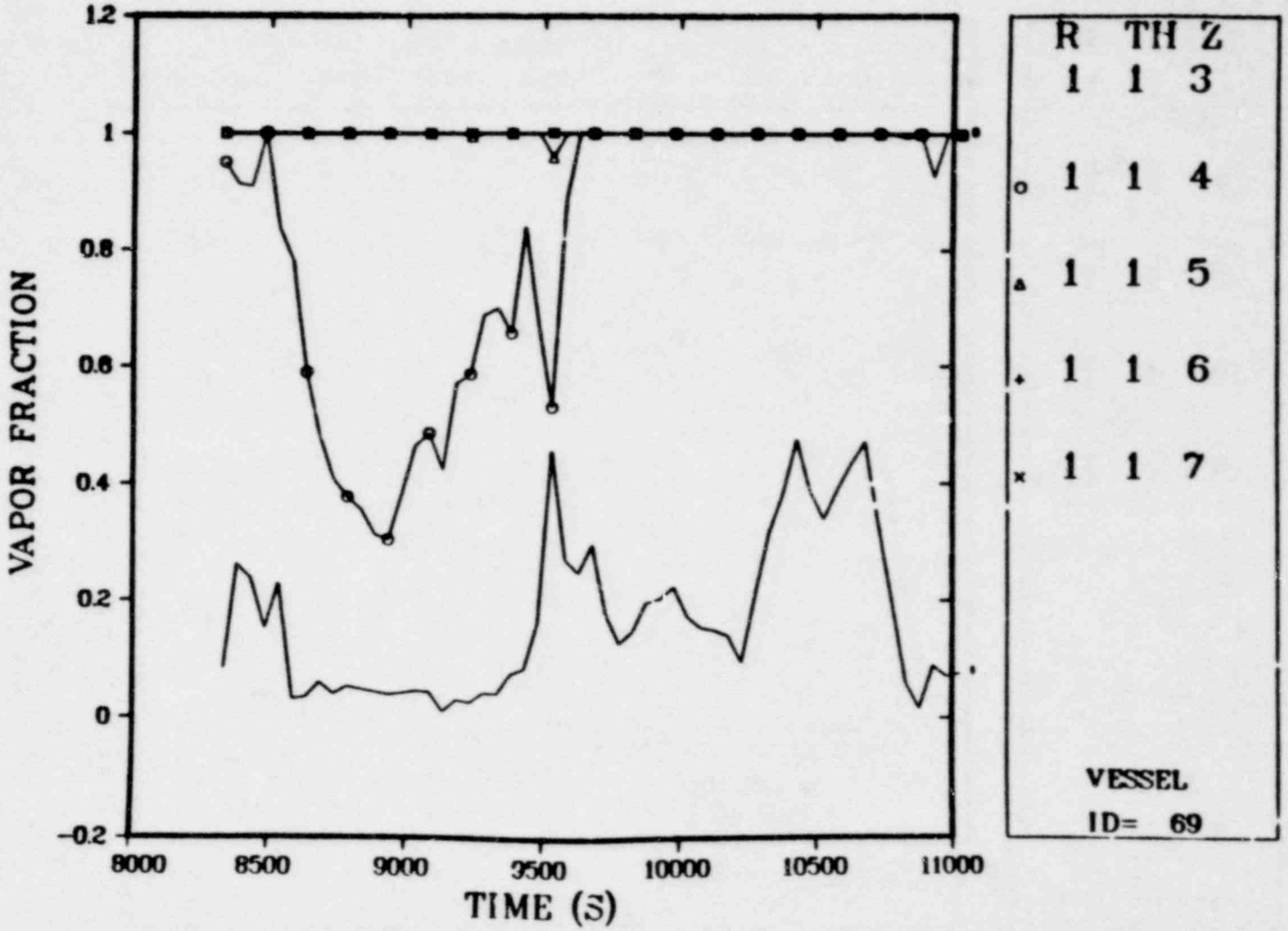

Core void fraction axial profile after 81 minutes.

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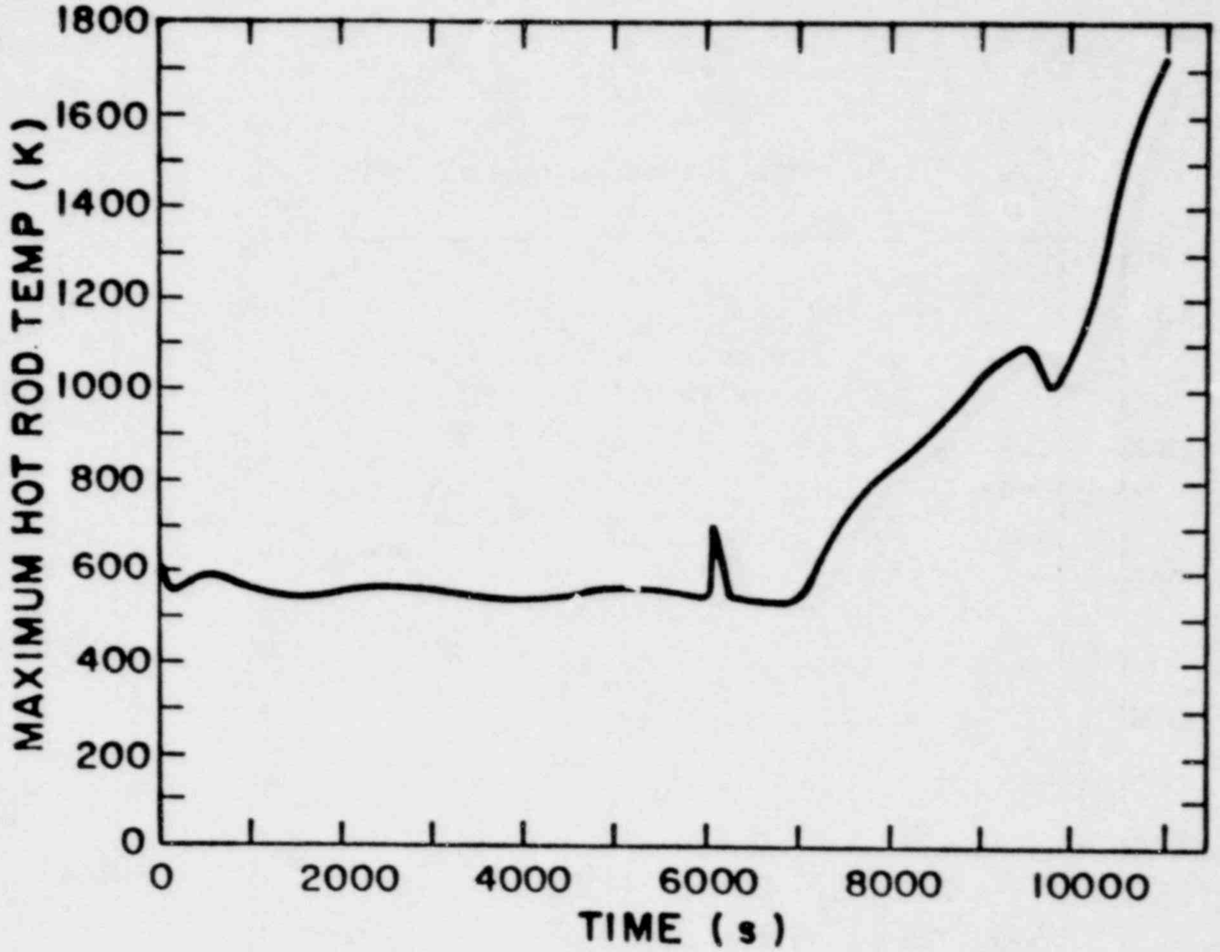

Maximum hot-rod cladding temperature.

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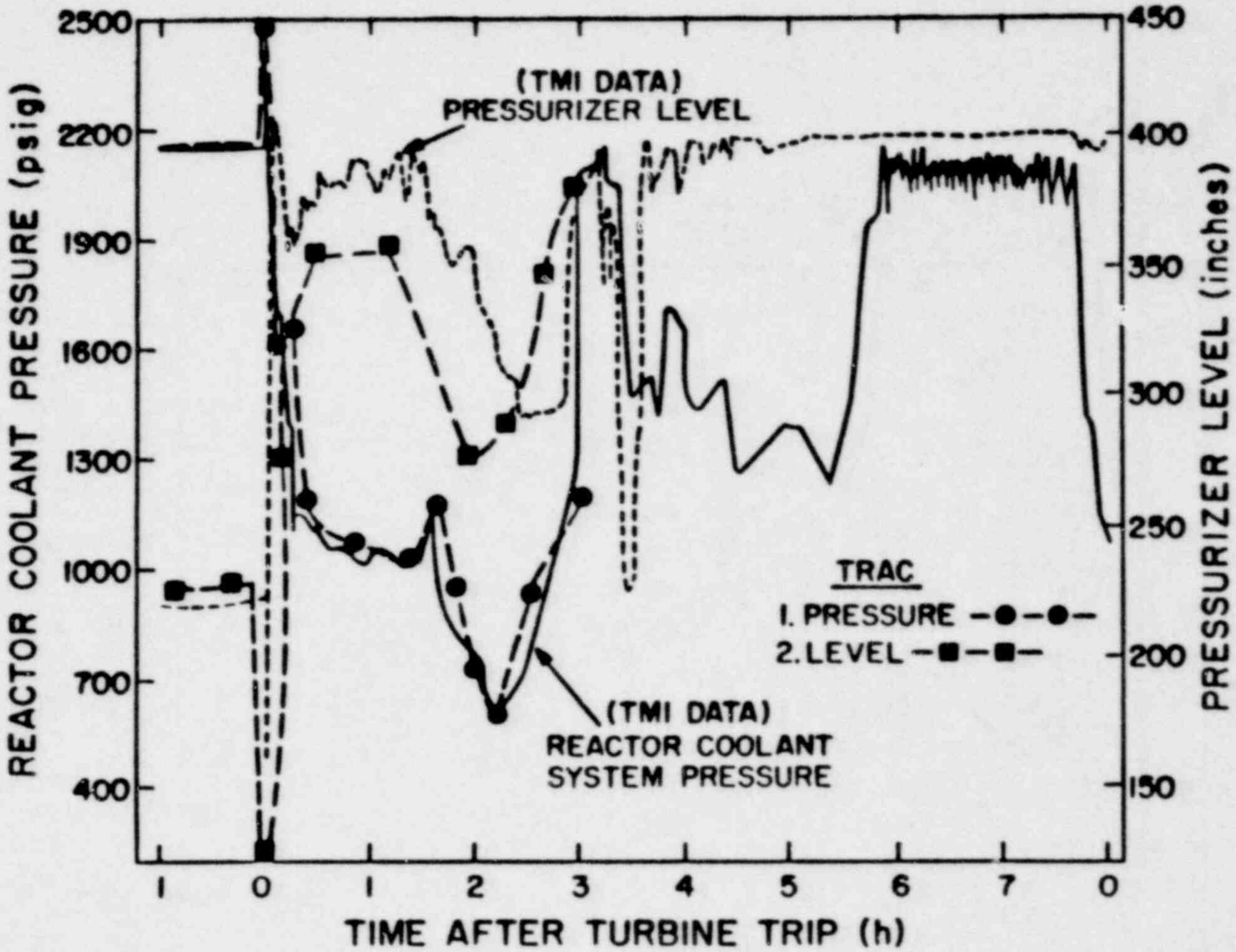

TRAC comparisons with TMI data out to 3 hours.

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UNCERTAINTIES IN BASE CASE CALCULATION

- MAKE-UP, HPI, AND LETDOWN FLOWRATES THROUGHOUT TRANSIENT.
- PRESSURIZER RELIEF VALVE FLOWRATE.
- AUXILIARY FEEDWATER FLOWRATES.
- IMPORTANCE OF PRESSURIZER SPRAYERS AND HEATERS DURING TRANSIENT.
- PUMP HEAT SOURCES.
- CORE GEOMETRY AFTER 2 HOURS.
- EFFECT OF NON-CONDENSIBLES (HYDROGEN).

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SUMMARY

- OVERALL SYSTEM COMPARISONS ARE REASONABLE OUT TO 3 HOURS INTO THE TRANSIENT FOR THE ASSUMPTIONS USED.

- INITIAL CORE UNCOVERY OCCURS AT ABOUT 101 MINUTES DUE TO TRIPPING OF LOOP A PUMPS.

- FAILURE OF NEARLY ALL THE FUEL RODS IS PREDICTED TO HAVE OCCURRED IN THE UPPER 0.5 M OF THE CORE AT ABOUT 2 HOURS AND 30 MINUTES. THIS IS CONSISTENT WITH RADIATION MONITORS AT TMI-2.

- CLADDING OXIDATION (ZIRCONIUM - STEAM REACTION) UP TO 3 HOURS RESULTED IN THE PRODUCTION OF APPROXIMATELY 40 KG OF HYDROGEN (ABOUT 27 M^3).

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