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TMI-1 file

DATE: July 3, 1979

TO: Ed Throm, NRC-DSS

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SUBJECT: ANALYSIS OF THE THREE MILE ISLAND ACCIDENT

An analysis of the Three Mile Island (TMI) accident has been partially completed. The calculation of the first 2250.0 seconds of the transient is complete and it is planned to continue the calculations past the time when the reactor coolant pumps were tripped off and the heatup phase of the accident started.

The study presented here was designated Three Mile Island Run 106. This analysis reflects several important improvements over an analysis presented earlier to the NRC, designated Three Mile Island Run 9. These include a more detailed RELAP4 model of the reactor and changes in the initial conditions and events.

The results from this calculation are promising. The analysis will be continued to find out if the core heatup phase of the TMI accident can be predicted. Results of a calculation to 2.5 hrs of transient should be done by July 6, 1979.

Initial Conditions - Three Mile Island Run 106

Reactor Power	- 2568.0 Mwt
System Pressure	- 2156.0 psia (hot leg pressure)
Core Flow	- 37562.6 lbm/sec
Hot Leg Flows	- 19153.3 lbm/sec
Pressurizer Level	- 30.0 ft
Steam generator levels	- 25.96 ft
Steam generator mass	34900 lbm

A set of trip controls was input that simulated the sequence of events which occurred during the accident. These controls are:

Event	Setpoint
S.G. feedwater flow off	0.0 seconds (12. second ramp)
Pressurizer valve opens	2270.0 psia in a loop hot leg
Reactor scram	2370.0 psia in a loop hot leg
HPIS start (2 pumps)	124.0 seconds
HPIS off	278.0 seconds
Makeup flow start	278.0 seconds (140 gpm)
Let down flow start	300.0 seconds (140 gpm)
S.G. auxiliary feed on	480.0 seconds
Trip pump B-loop	4440.0 seconds
Aux feed to A-loop S.G. off	5100.0 seconds

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Event	Setpoint
Aux feed to B-loop S.G. off	5220.0 seconds
Aux feed to A-loop S.G. on	5650.0 seconds
Trip pump A-Loop	6000.0 seconds
Aux feed to A-loop S.G. off	7440.0 seconds
Shut EMOV	8280.0 seconds

RELAP4 Model

The RELAP4 model used in the analysis is shown in Figure 1. The model was originally developed to check out RELAP4/MOD5[1]. It was then modified for analysis of the TMI transient. A description of most of the components of the model is contained in Reference (1).

The HPIS is modeled with Junctions 31, 38, 39, 40, 41 and 42. Junctions 31, 38 and 39 model the normal HPIS flow while Junctions 40, 41 and 42 model the flow with one pump only.

There are uncertainties in the steam generator conditions during the early part of the transient. The turbine bypass flow (Junctions 26 and 29) which was given as 15% of normal flow, the steam generator relief valve flows (Junctions 27 and 30) and the steam generator auxiliary feed flow (Junctions 26 and 29) are uncertain. Steam generator conditions are critical to the analysis because the energy removed from the reactor system is dependent on these conditions. To eliminate some of the problems in calculating steam generator conditions, volumes 23 and 24 were added. These are steam filled time-dependent volumes with pressures set to match the measured steam generator pressures during the accident. Insertion of these volumes will hold the pressures in steam generator volumes 20 and 21 at the measured pressures. The mean volume flow and volume quality in 20 and 21 will still be functions of the input turbine bypass flow, relief valve flows, and auxiliary feed flow.

The RELAP4 bubble rise model was selected for volumes 1, 2, 3, 4, 6, 14, 34, 36, 37 and 38. This was done to allow a calculation of core dry out during the heatup phase of the accident. The computer calculation time for the analysis was very long when bubble rise was selected for the entire transient. Therefore, the bubble rise calculation was not started until after the reactor coolant pumps were tripped off.

The pressurizer was modeled as a vertical stack of six volumes with vertical slip selected at the junctions connecting the volumes. This method has been proven to be the most accurate and efficient method of modeling swelling phenomena such as occurs in the pressurizer during this transient.

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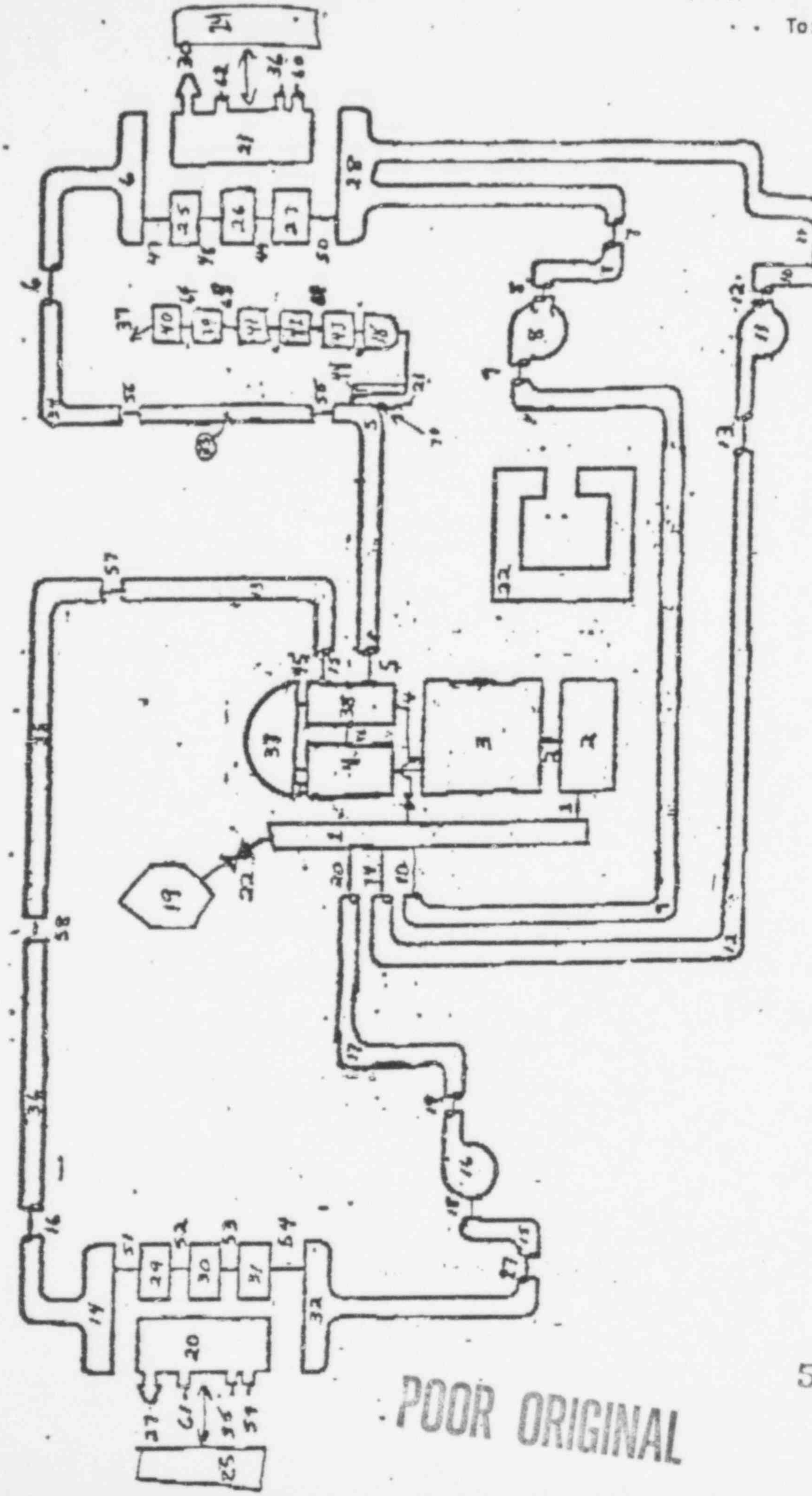

Figure 1. RELAP4/MOD7 MODEL of TMI-2

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Preliminary Results

Figure 2 shows the B-loop hot leg temperature plotted versus the measured data. There is a sharp drop in temperature after reactor scram at 10 seconds. When the steam generators dry out at 60.0 seconds the temperature increases. The RELAP4 calculation matches the trends shown in the data at these times.

After steam generator dry out the temperature increases faster in the code calculation than in the measured data. Apparently, the HPIS flow prevents a sharp temperature rise but this trend did not appear in this calculation. When HPIS flow is turned off at 278.0 seconds the measured temperatures increase at the same rate as the calculation. The temperatures peak at 480.0 seconds after the steam generator auxiliary flow starts. The rate of temperature decrease after 480 seconds is smaller in the calculation than the data indicates. This is believed to be due to heat transfer calculations not reflecting the steam generator tube rewetting.

Figure 3 shows the RELAP4 code calculation of A-loop hot leg pressure and measured data. The measured data shows that the pressure in the TMI reactor decreased during the first 350.0 seconds of the transient. A pressure increase then occurred between 350.0 to 500.0 seconds. The rise was caused by the liquid mixture in the pressurizer rising to the top. This decreased the volumetric flow rate through the pressurizer relief valve. At 500.0 seconds the system started to depressurize again due to cooling caused by the steam generator auxiliary feed flow.

As seen in Figure 3, the calculation does not match trends shown in the data during the first 500.0 seconds of the transient. This problem is believed to be caused by an inaccurate prediction of pressurizer behavior during the early part of the accident. However, after 500.0 seconds the calculation matched the measured depressurization rate.

Figure 4 shows the RELAP4 calculation of the reactor core power overlayed with the power transferred through both steam generators. This graph illustrates the importance of correctly calculating steam generator behavior during a small break transient because there is a mismatch between power in and power out throughout the accident. In this analysis, the results indicate that the power out exceeds power in after 500.0 seconds which resulted in the temperature decrease seen in Figure 2.

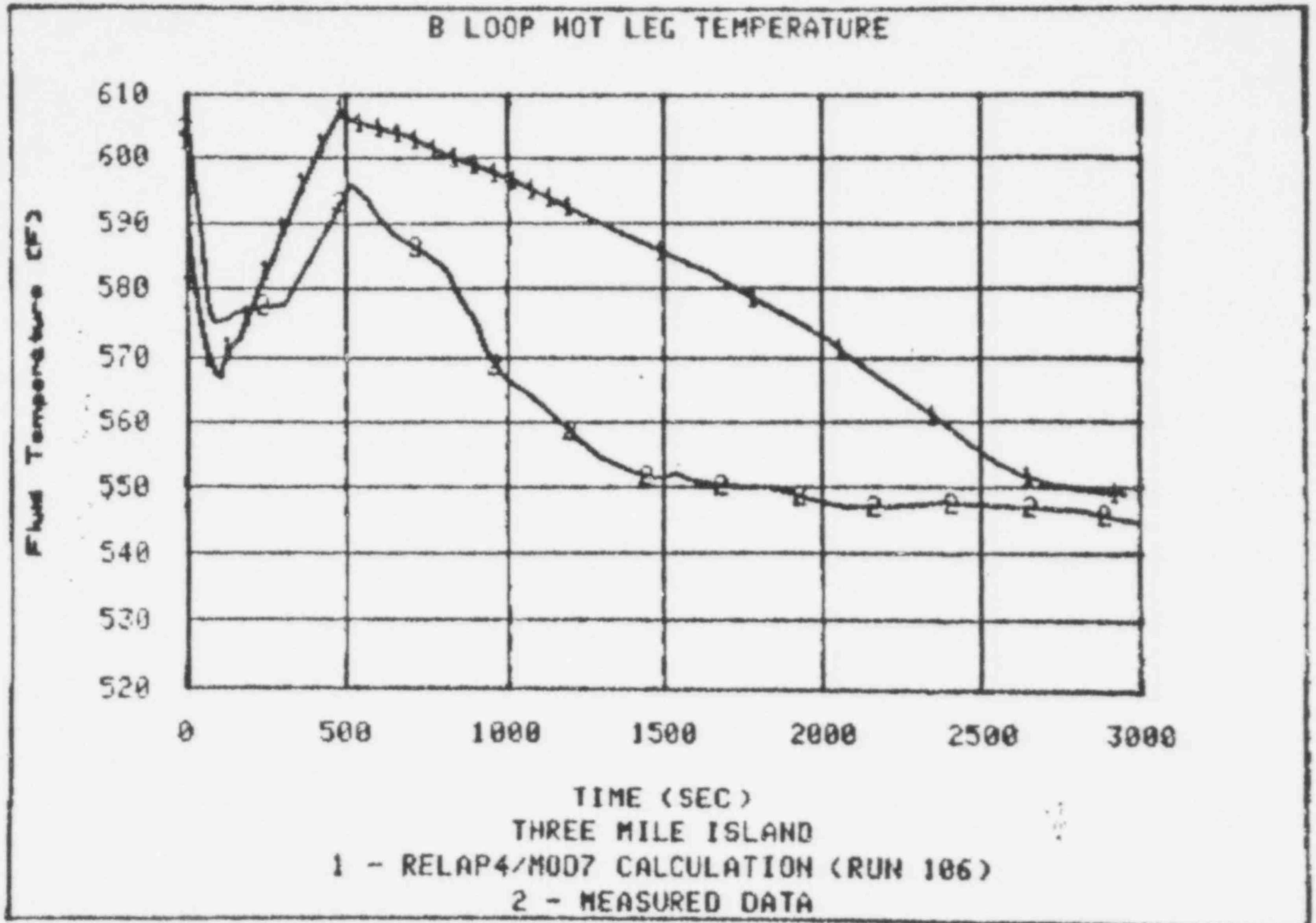

Figure 2

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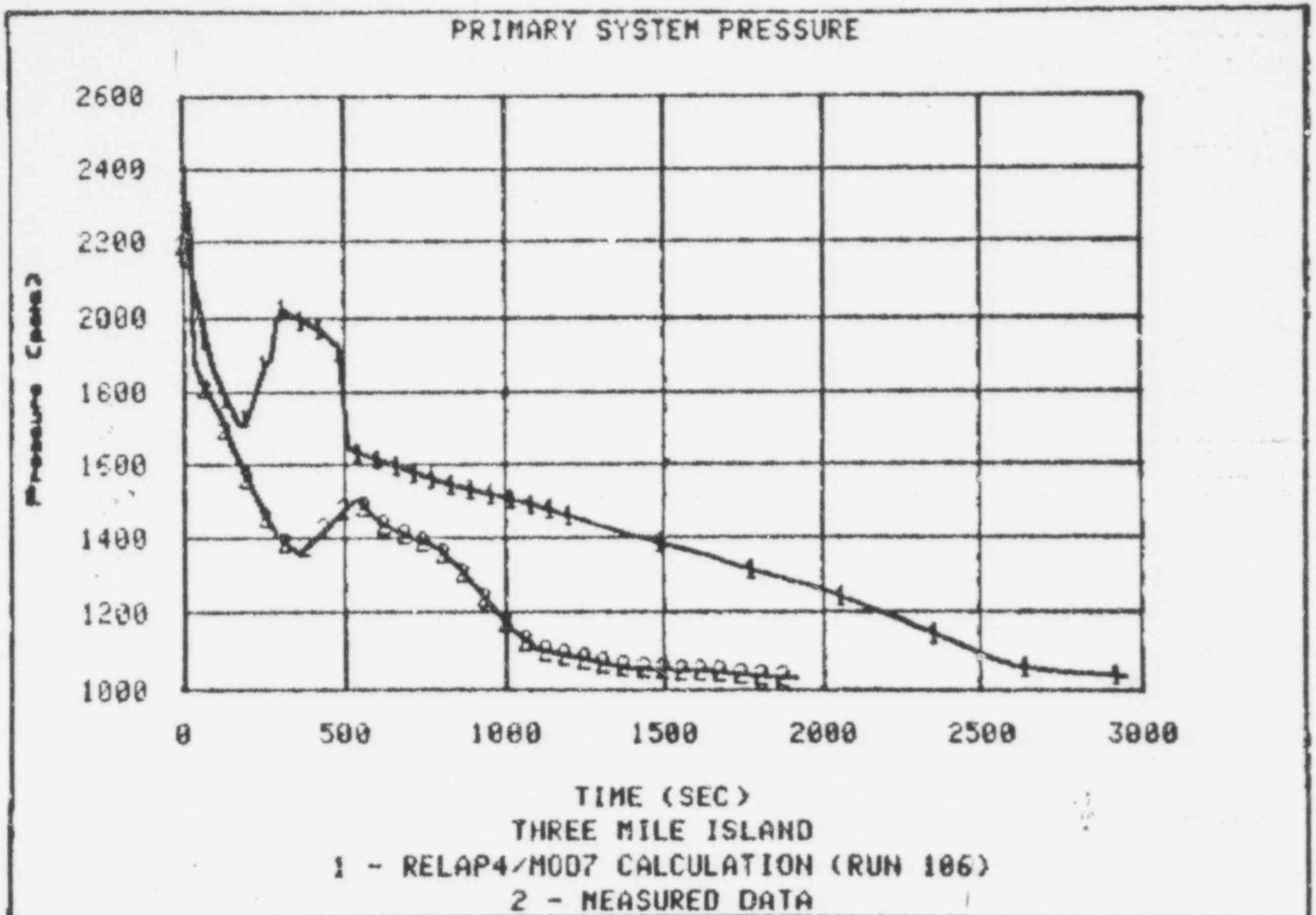

Figure 3

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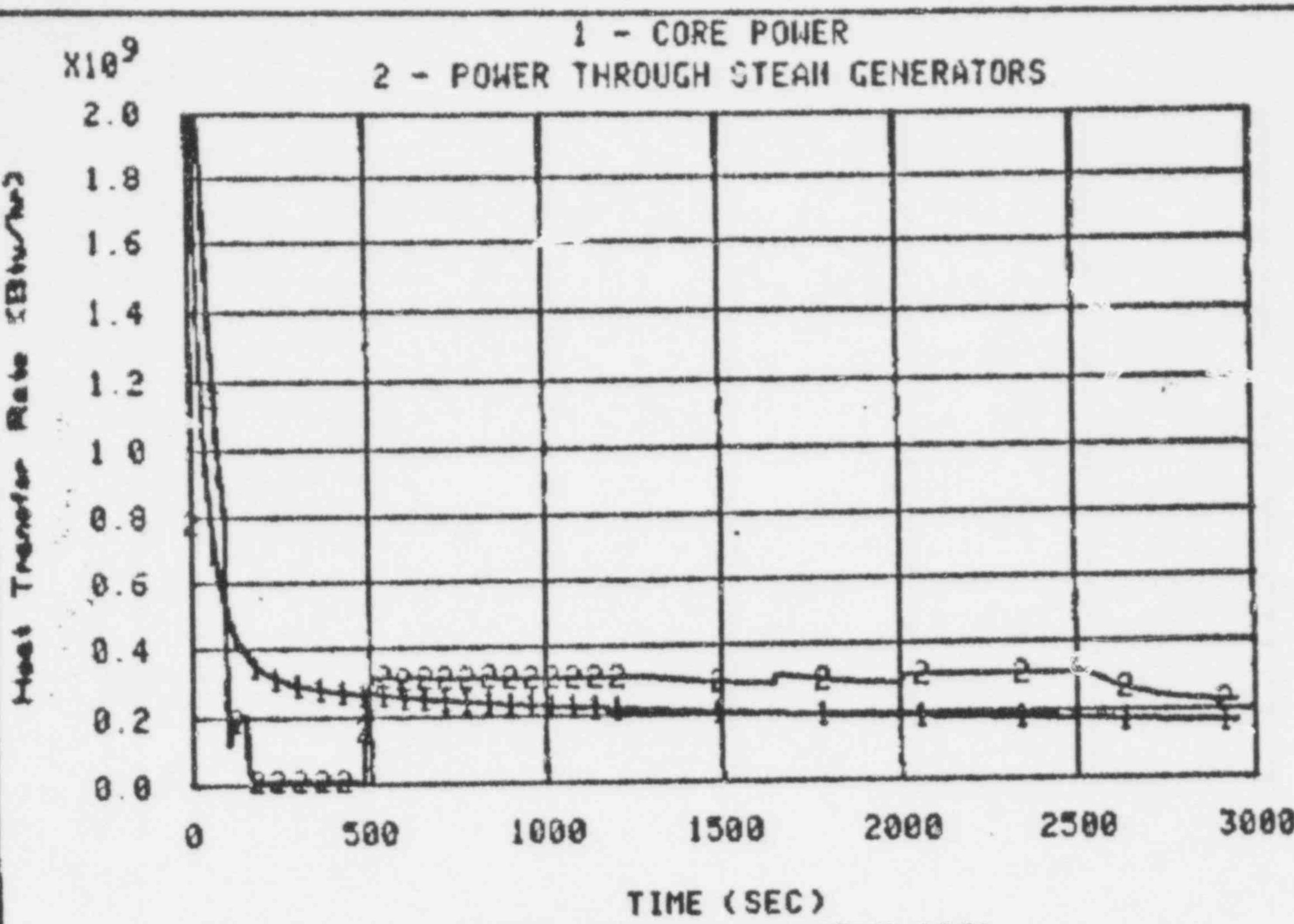

THREE MILE ISLAND (RUN 106)
TIME DEPENDENT STEAM GENERATORS
SEVEN NODE PRESSURIZER

Figure 4

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