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TITLE: SYSTEM CALCULATIONS RELATED TO THE ACCIDENT AT THREE MILE ISLAND USING TRAC

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SYSTEM CALCULATIONS RELATED TO THE ACCIDENT AT THREE-MILE ISLAND USING TRAC*

by

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I. INTRODUCTION

The Three Mile Island nuclear plant (Unit 2) was modeled using the Transient Reactor Analysis Code (TRAC-P1A)[1] and a base case calculation, which simulated the initial part of the accident that occurred on March 28, 1979, was performed. In addition to the base case calculation, several parametric calculations were performed in which a single hypothetical change was made in the system conditions, such as assuming the high pressure injection (HPI) system operated as designed rather than as in the accident. The purposes of these calculations were to:

- Provide insight into the system thermal-hydraulic phenomena which occurred during the initial accident stages.
- Evaluate hypothetical alternative system/operator responses during the accident.
- Evaluate and assess the applicability of TRAC to non-LOCA accident scenarios.

Some of the important system parameter comparisons for the base case as well as some of the parametric case results will be presented in this paper. The parametric cases that will be discussed are as follows:

1. Auxiliary feedwater delay of 60 min. (compared to 8 min. delay in base case).
2. No auxiliary feedwater delay.
3. HPI system operating as designed (compared to throttled conditions in base case).

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

II. TRAC BASE CASE CALCULATION

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 (PORV) was modeled using a pipe module, allowing a direct calculation of the flow out of the PORV. 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 for the base case to generate the initial conditions prior to the accident. These conditions are in very good agreement with available TMI-2 data.[2]

Using these self-consistent initial conditions, the TRAC transient calculation for the base case 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 120 min of the accident sequence are well simulated by TRAC, particularly system pressure (Fig. 1), loop temperatures (Fig. 2), and pressurizer level (Fig. 3). During the period from 30 to 81 min coolant is continuously lost through the PORV 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. After 81 min, a more finely noded vessel was used to provide better resolution to track phase separation and calculate two-phase natural circulation. 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 forced coolant circulation through the loops. Since upward-moving steam velocities are very low, the steam becomes very superheated in the upper part of the core and, as a result, the cladding and fuel heat up rapidly (Fig.

4). 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, enhancing the core cooling rate (TRAC calculations for the base case were terminated at 3 h since the core modeling was no longer realistic).

The results of this TRAC base case calculation show good agreement with measured system parameters out to nearly 3 h and provide a foundation for making detailed comparisons against alternative system/operator responses during the accident sequence.

III. TRAC PARAMETRIC CALCULATIONS

These parametric calculations were performed to investigate hypothetical variations to the base case to determine the significance of system/operator actions on the course of the accident. It is not intended to judge system design or operator response as related to the TMI-2 accident, rather, its purpose is to serve as a basis for future discussion on reactor system design, instrumentation, and operation.

The parametric case with HPI operating as designed resulted in significant deviations from the base case. After the pressure dropped below the HPI setpoint and full flow was initiated, the HPI flow was sufficient to maintain the system pressure at a higher level than the base case. This resulted in a higher break flow than the base case, but, more importantly, maintained the coolant in a subcooled state, preventing a core temperature excursion. This calculation indicates that no core damage would have occurred as long as HPI flow was supplied.

The influence of delaying the auxiliary feedwater flow 60 min compared to no delay results in a higher system pressure and a higher PORV flow rate, but the long-term behavior of the system is about the same. However, due to the higher PORV flow the system water inventory is about 15% less than the no delay case, which would probably result in about a 10-15 min earlier time to core uncovery. For a 3 h transient, however, this amount of time is not very significant.

In summary, the parametric calculations indicate that loss-of-core cooling was most influenced by the throttling of HPI flows, given the accident initiating events and the PORV failing to close as designed.

In conclusion, these TRAC calculations have provided some insight into the system thermal-hydraulics of TMI-2 and hopefully have assessed the significance of some of the system/operator actions on the course of the accident. Finally, these calculations show that TRAC can be applied to non-LOCA accident scenarios with reasonable confidence.

REFERENCES

1. "TRAC-P1A, An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory report LA-7777-MS (May 1979).

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

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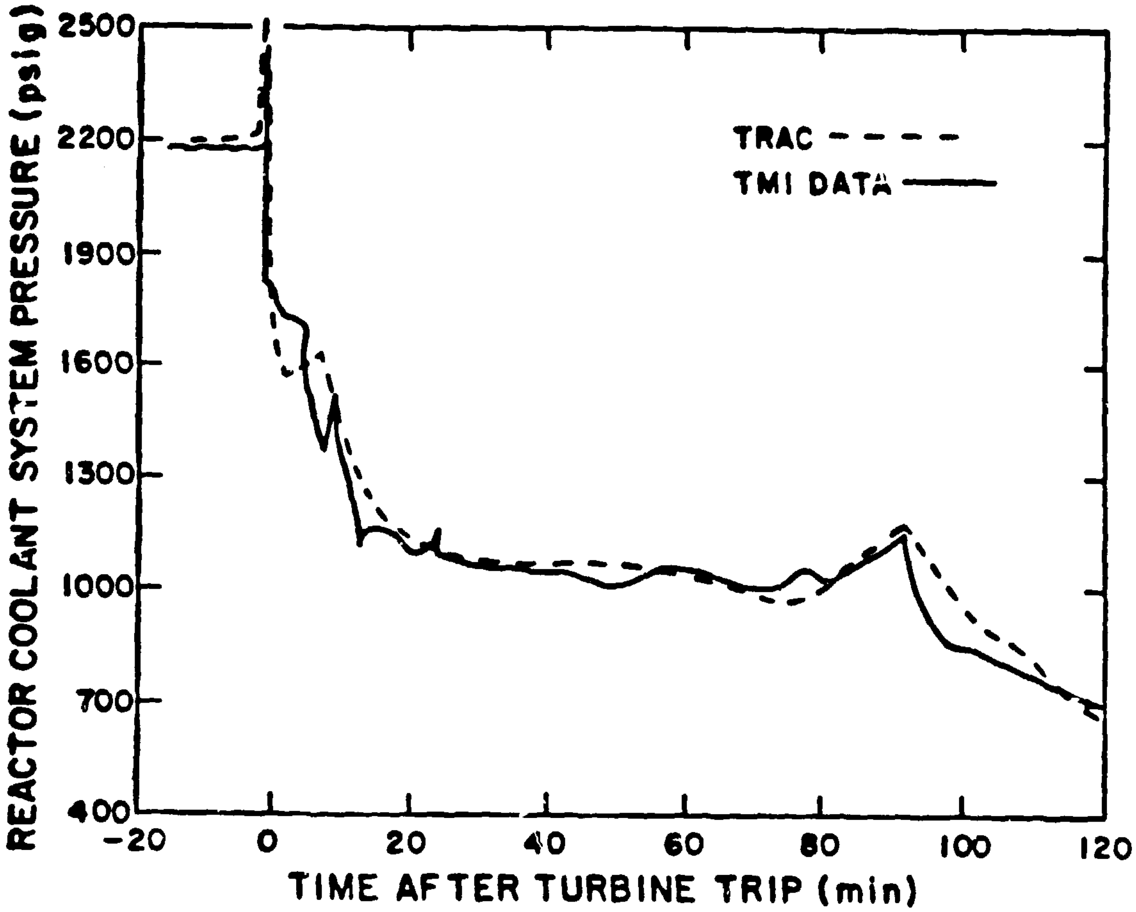

Fig. 1.

System pressure comparisons out to 120 minutes.

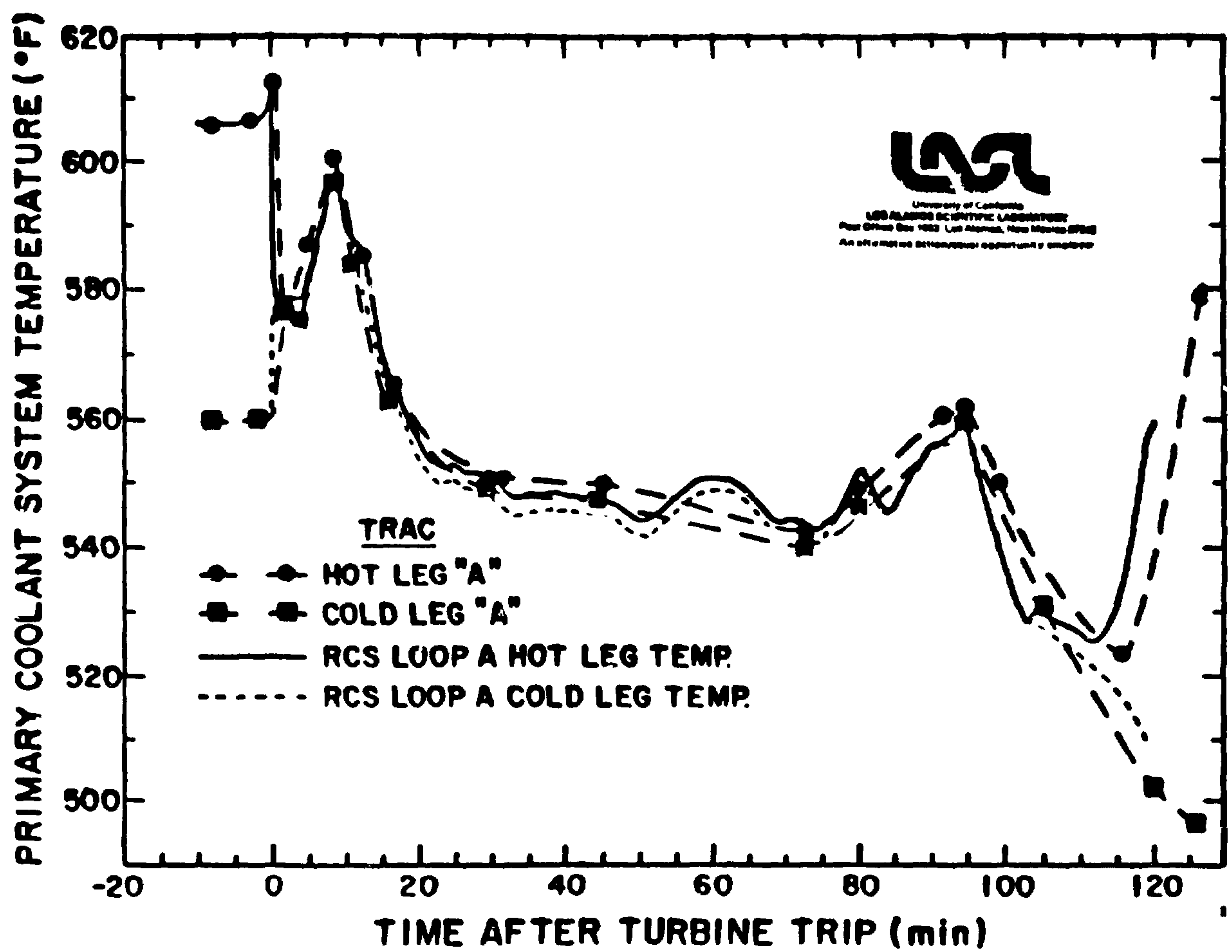

Fig. 2.

A LOop fluid temperature comparisons out to 120 minutes.

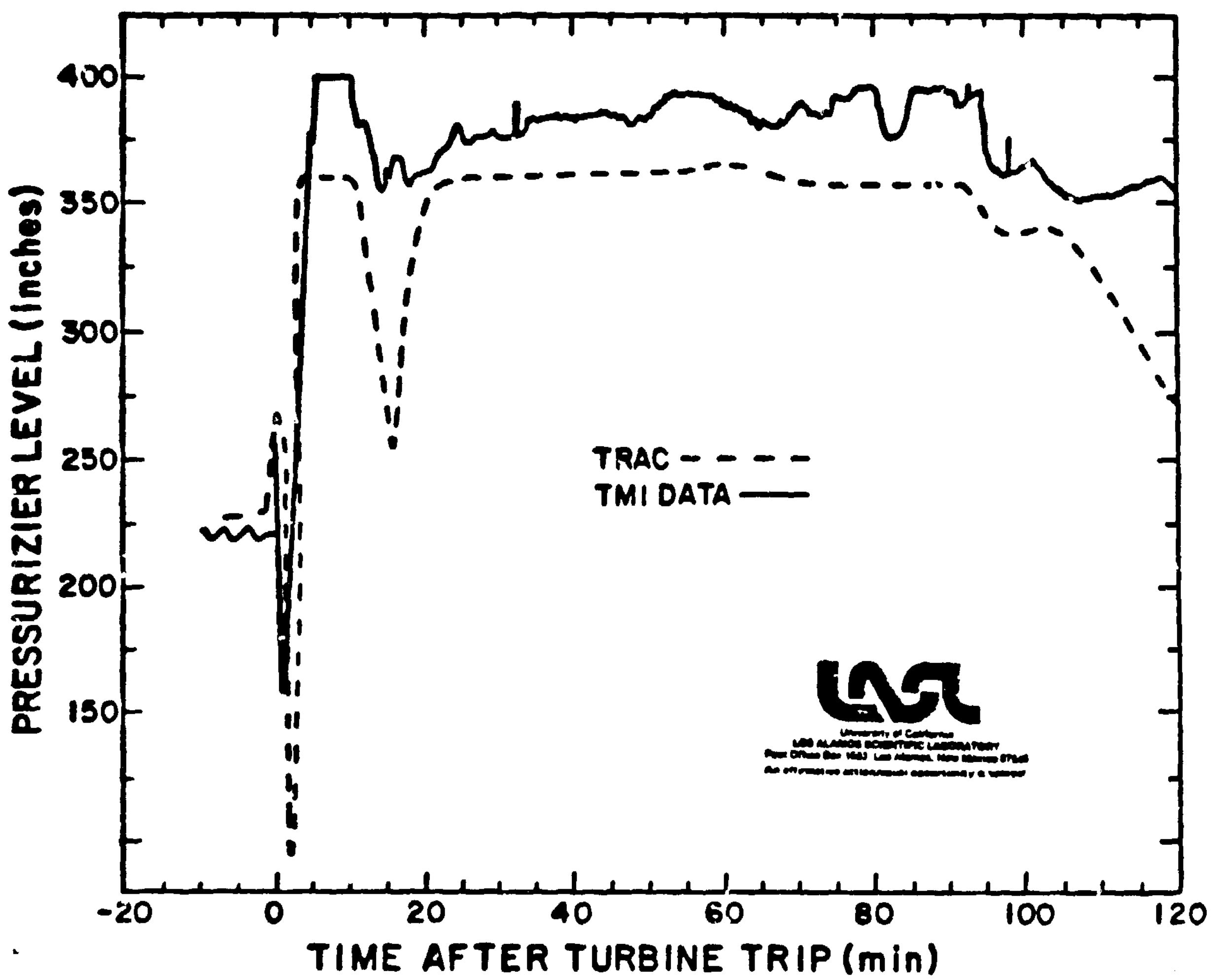

Fig. 3.

Pressurizer water level comparisons out to 120 minutes.

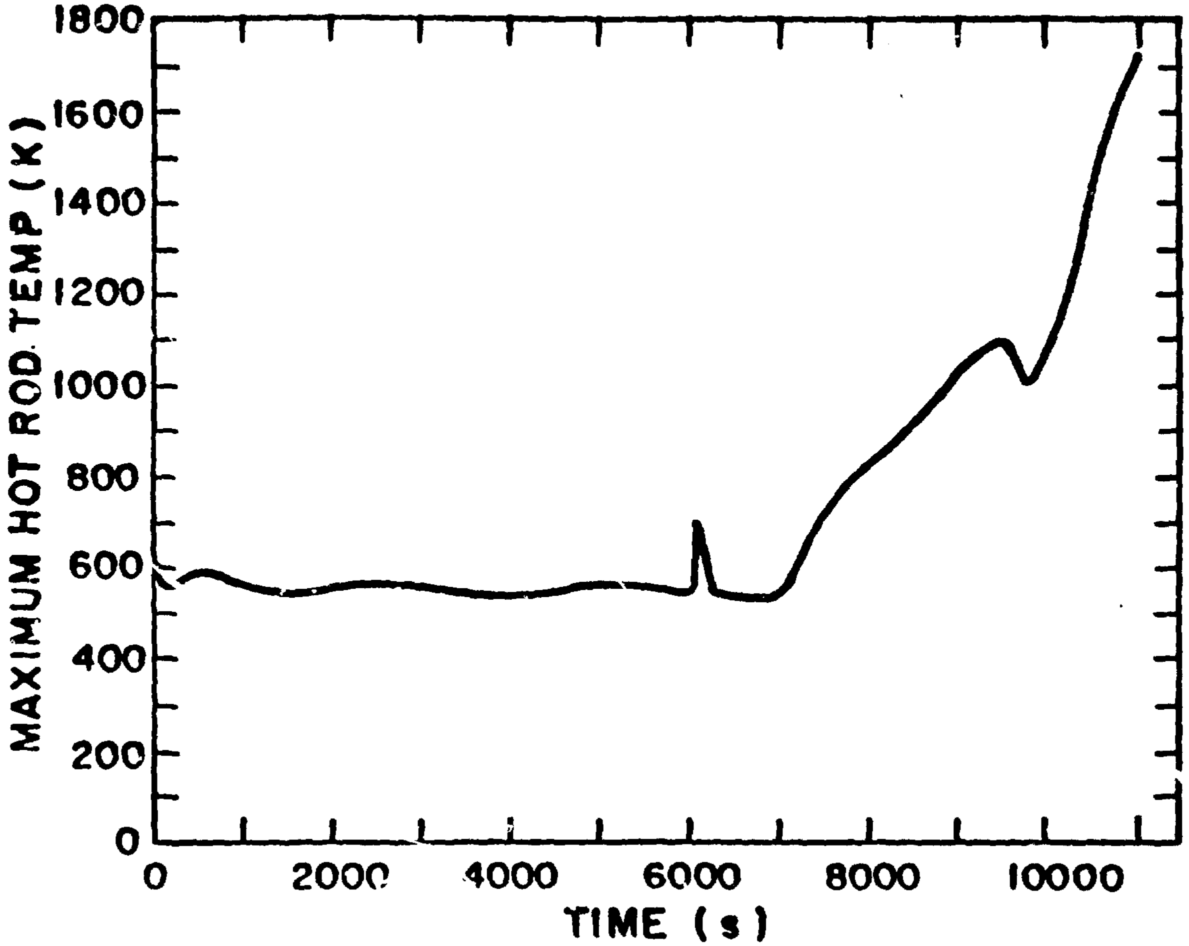

Fig. 4.

Maximum hot-rod cladding temperature.