UPDATE ON STANDARD PROBLEM, DATA BASE, AND UNCERTAINTIES[a]

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ABSTRACT

The Three Mile Island (TMI)-2 accident provides the only data for a full-scale, integrated water reactor facility during a severe accident. The data from this accident has been, and is being, collected, reviewed, and cataloged. The data will be placed in structured data bases. To date, three data bases have been completed: Plant Configuration, Sequence of Events, and Initial and Boundary Conditions. The data pertinent to the TMI-2 Standard Problem have been reviewed and categorized by type and level of uncertainty. As a result of the review process, it has been determined that further refinement of the estimated boundary conditions High Pressure Injection/makeup and letdown flow will be required. A demonstration calculation for the first 174 min of the accident has been completed using the RELAP/SCDAP integrated code. This calculation is in general agreement with the data for the first 100 min of the accident and in significantly less agreement beyond 100 min. Refinement of the model's nodalization is expected to correct some of the inconsistencies.

INTRODUCTION

A standard problem is a single problem upon which a specific class of computer codes may be tested. A standard problem may be based on either experimental or hypothetical conditions, depending on the nature of the codes and the desired test. To create a standard problem based on experimental data obtained from a nuclear reactor facility requires four basic types of data. These data types are the facility characteristics, experiment operating sequence, initial and boundary conditions for the experiment, and results of the experiment. The results may be provided to the analyst (open standard problem) or held back until completion of the problem (blind standard problem).

The Three Mile Island (TMI)-2 accident represents the only full-scale integrated facility data for a severe accident. Thus, a unique opportunity exists to benchmark the severe accident analysis computer codes. Secondly, an international consensus on code assessment and severe accident analysis may be achieved through participation of members of the Committee for the Safety of Nuclear Installations (CSNI). The purpose of this paper is to provide a synopsis of the ongoing efforts to provide the required benchmarking information in the form of a TMI-2 standard problem.

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ACCIDENT SEQUENCE

The TMI-2 accident can be divided into four phases. The division is based on major events during the accident. Phase 1, the small break loss of coolant accident (LOCA) phase, is bounded by the turbine trip at time zero and the final reactor coolant pump trip at 100 min. The turbine tripped on loss of main feedwater to the steam generators. Although the emergency feedwater pumps started in accordance with plant design, the block valves were closed preventing emergency (auxiliary) feedwater (AFW) flow to the steam generators. The steam generators started to boil dry and primary-to-secondary heat transfer decreased. With the loss of heat transfer, primary pressure increased and the pilot-operated relief valve (PORV) opened. Pressure continued to increase at a decreased rate and the reactor tripped at 8 s. Pressure decreased rapidly and the PORV should have closed. However, flow through the PORV continued unabated. By 1.5 min, the steam generators boiled dry. At 8 min, AFW flow was initiated by opening the block valves. By about 25 min, measurable steam generator water levels had been reestablished. The coolant inventory continued to decrease, and at 74 min, the B loop reactor coolant pumps were tripped by the operators to prevent operation beyond net positive suction head (NPSH) limits. At 100 min, the A loop pumps were tripped for the same reason and Phase 1 of the accident ended.

The reactor was now in pool boiling and Phase 2 of the accident had begun. Primary coolant inventory continued to decrease, the core uncovered, and the fuel rods started to heat up. At 139 min, the PORV block valve was closed. Fuel rods ballooned and ruptured and eventually rapid oxidation of the cladding occurred. By 174 min, some liquefaction had taken place and molten material flowed downward to the lower regions of the core. At 174 min, the 2B reactor coolant pump was restarted and Phase 2 of the accident ended.

Phase 3 of the accident covers the period of the 2B pump transient from 174 to 200 min. During this period, it is believed that approximately 28 m^3 (1000 ft^3) of coolant was pumped into the reactor vessel generating steam and hydrogen, and also providing some cooling to the damaged core.

Phase 4 includes the period from 200 to 300 min. During Phase 4, High Pressure Injection (HPI) was initiated, some of the molten core relocated to the lower plenum at 225 min, and a coolable geometry was achieved.

During the four accident phases, significantly different phenomena are taking place. It is then possible to simulate each phase independent of the other phases, if a sufficient set of phase initial and boundary conditions is provided. However, thermal hydraulics is significant to the overall system behavior throughout the four phases. In this regard, the accident thermal hydraulics must be reasonably well simulated if the core degradation simulation is to be reasonable.

DATA BASES

The efforts to date have focused on collecting, reviewing, and cataloging the data required for a standard problem, and performing a demonstration calculation for Phase 1 and 2 of the accident. The required information has been placed in three data bases, two of which are electronic. The electronic data bases have been developed in SAGE. The SAGE system was written in the MODULA-2 language as a personal computer-based relational data-base-management system. SAGE is an advantage where operations on diverse data types, such as tabular and time series data, is required. The Plant Configuration data base contains the geometric, component, and performance data for the TMI-2 facility. This data base is in a report format. The Sequence of Events (SOE) and Initial and Boundary Conditions (ICBC) data bases are electronic data bases. These data bases provide the event sequence during the accident, the initial conditions at the beginning of each accident phase, and the boundary conditions for the accident. These data bases have been provided as part of the standard problem.

The data provided must have integrity. Herein data integrity means that each measurement has been reviewed, and has an established uncertainty. The review process involves a thorough review of the measurement system, its operation, and calibration. Computational models, methods, and assumptions were also evaluated. From this review and analyses of the measurement systems, uncertainties in the data are determined. Based on the review process, the measurements were then categorized on the basis of data quality.

The sources of data for the accident are the reactimeter, alarm printer, utility printer, analog strip charts, multipoint recorders, plant manuals and procedures, other analyses and reports on the accident, and operator interviews. The reactimeter was an on-line digital data-acquisition system, and is the most reliable source of information about the accident. However, only a small number (24 of the more than 100 digital data channels) were recorded on this system at a sampling rate of 20 per minute. The utility printer provides the hourly logs specific digital measurement groupings requested by the operators at various times and other automatic data printouts. The alarm printer provides a printed record of plant alarms as they occurred. These data were lost from 73.3 to 159.5 min, because the printer was running behind and the operators flushed the buffer so they could get more current information. The alarm and utility printer are also considered to be reliable sources of information.

Analog strip charts and multipoint recorders are quite useful, but the uncertainties in these data sources are in general larger than the digital data recorded on the reactimeter, alarm printer or utility printer. The remaining sources of information provide a background for understanding the context in which the data were recorded and a basis to test the consistency of the data.

The formal review process started with an analyst reviewing the data for consistency with other available data sources and performing any required calculations. The analyst's work was then presented to, and reviewed by, an internal peer review committee (the Data Integrity Review Committee). Based on the analyst's recommendation and the committee review, the quality level of the data was established.

Qualified data are data that have valid magnitudes and whose uncertainties are sufficiently small that the absolute magnitudes can be relied on. The TMI-2 data that typically fit this category are the reactimeter data, except when specific measurements were out of range.

Trend data are those data for which the reliability of the absolute data magnitudes cannot be assured due to unacceptably high uncertainties or inability to determine the uncertainties. However, the relative magnitudes of trend data can be relied on. Lack of a recent calibration for a measurement is one reason for data to be categorized as trend data. Inability to accurately read the recorded data (e.g., poorly printed multipoint strip charts) is another. Enigmas in the data represent a common reason for categorizing data as trend.

Composite data are assembled using data from more than one measurement channel. The TMI-2 composite data may be composed from reactimeter, strip chart, and/or utility printer sources. As an example, the primary system pressure during the accident has been constructed from three measurement channels depending on the time during the accident. The reactimeter recorded the primary system pressure from a narrow-range transmitter (RC-3B-PT1-R). Up to about 2 min after the turbine trip, the reactimeter data were used. At this time, the pressure fell below the range for this instrument. Except for time periods when the pressure increased to within the range of the narrow-range transmitter, a wide-range transmitter (RC-3A-PT3) recorded on the utility printer (-15 to +15 min) and on a strip chart that had to be used for the best estimate of primary pressure. Composite data may contain both qualified and trend data, depending on the data source for any specific time period.

Computed parameters are the result of post-accident manipulation of a measurement or set of measurements. Computed parameters are based on computational models for which the output is clearly related to the data. The best estimate steam generator levels, for example, are computed parameters based on the level measurements. The startup, operating, and full-range steam generator level measurements were put on a common basis using a computational model. The results from the computational model provide a better level estimate than do the values from any one measurement. Computed parameters may be categorized as either qualified or trend depending on the quality of the data and the computational model.

Estimates are based on calculations that rely on assumptions about the plant operation and behavior, and may require considerable manipulation of data or use of plant-modeling techniques. Estimates generally have large uncertainties and unreliable magnitudes.

Figure 1 shows the A loop hot leg temperature. For temperatures above 600 K, the data is from a strip chart and the uncertainty is significantly larger than for the data from the reactimeter. Thus, the data above 600 K has been categorized as trend data and the combined measurement as composite data.

Table 1 provides a listing of measurements and their categories. Several key boundary conditions listed in this table are estimates. Auxiliary feedwater flow can be estimated from the qualified steam generator levels and secondary pressures. In the RELAP/SCDAP model, this has been accomplished using a control system to maintain steam generator level. There are two possible approaches to estimating HPI/makeup and letdown flows. The direct approach is to use the available information on the specific flow system and perform a direct estimate of the flow. The indirect approach is to infer the flow from the reactor coolant system conditions at various times during the accident.

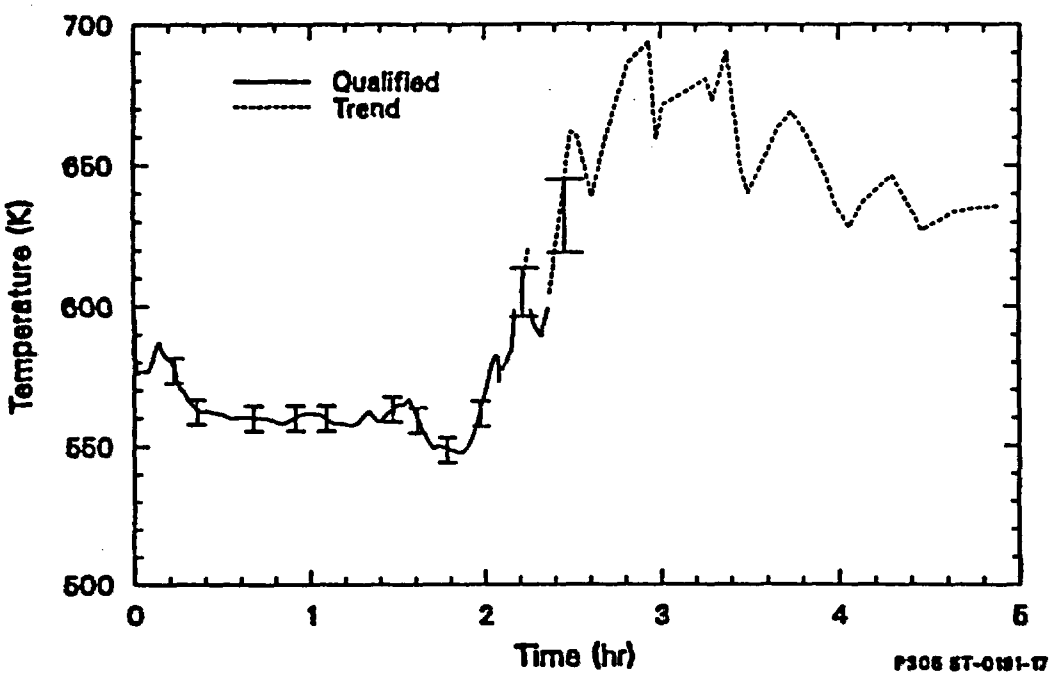

Figure 1. A loop hot leg temperature.

TABLE 1. DATA CATEGORY EXAMPLES

Measurement	Category
Primary pressure	Composite
Hot leg temperature	Composite
Cold leg temperature	
Loops 1A and 1B	Qualified
Loops 2A and 2B	Trend
Pressurizer level	Qualified
Steam generator level	Qualified
Secondary pressure	Qualified
HPI/makeup flow	Estimate
Letdown flow	Estimate
Initial core power	Computed parameter

The direct approach for makeup flow utilizes the hourly log data for makeup flow, data from the alarm printer to determine when the system was in the Emergency Safety Features actuation mode (HPI flow), makeup pump characteristics, and primary system pressure to estimate time-dependent makeup flow rates. The direct approach for letdown requires the use of the letdown cooler outlet temperature, reactor coolant cold leg temperature, and assumptions for the cooling water flows and temperatures. From the limited information and assumptions, a calculational model is utilized to estimate the flow rates.

The indirect approach to estimate the flows is based on a mass and energy balance of the primary and secondary systems using first principle models and the accident data. There are two significant marker points for this analysis. The first marker is a rapid increase in the source-range monitor between 110 and 120 min. This has generally been interpreted as core uncovery[1]. The second marker is hot leg superheat, which occurs at 112 min. These markers coupled with a detailed mass/energy balance will provide another estimate of the HPI/makeup and letdown flows. Combining the results of both methods should refine the best estimate flows and reduce the level of uncertainty. The required refinement of the best estimate flows and uncertainties will be completed by the end of January 1987.

DEMONSTRATION CALCULATION

A demonstration calculation for Phases 1 and 2 has been completed using the available boundary conditions at the time of the calculation. The purpose of this calculation is to show that a reasonable simulation of the TMI-2 accident is possible. The nodalization used for this calculation is shown in Figures 2 through 4. Figure 2 shows the overall system nodalization. The two cold legs in each loop was modeled as a single equivalent cold leg and pump. The core has been divided into three regions: center, mid, and peripheral assemblies, as shown in Figure 3. The core-heat structures shown in Figure 4 have been modeled with SCDAP fuel and control-rod components.

The results of the calculation are shown in Figures 5 through 9. Figures 5 and 6 show that the calculated pressure and pressurizer level are generally consistent with the measurements for the first 100 min of the accident. For the pressure calculation to be close to the data, the mass and steam generator heat-transfer computations must be approximately correct. The calculated pressure starts to deviate significantly from the measurement at about 130 min. At 140 min, the deviation becomes nearly constant to 170 min. The deviation is probably due to excessive steam generator heat transfer or excessive mass loss or both.

The hypothesis of excessive steam generator heat transfer may be supported first by the large volumes in the steam generators. The large volumes may result in the large boiling and condensation heat-transfer coefficients to be spread over an unrealistically large region of the steam generators. This hypothesis can be tested by renodalization of the steam generators. A second support for excessive steam generator heat transfer occurs at 94 min when the calculated drop in pressurizer level is much larger than was measured (see Figure 6). This corresponds to the time at which a rapid level increase in the A loop steam generator commenced. The

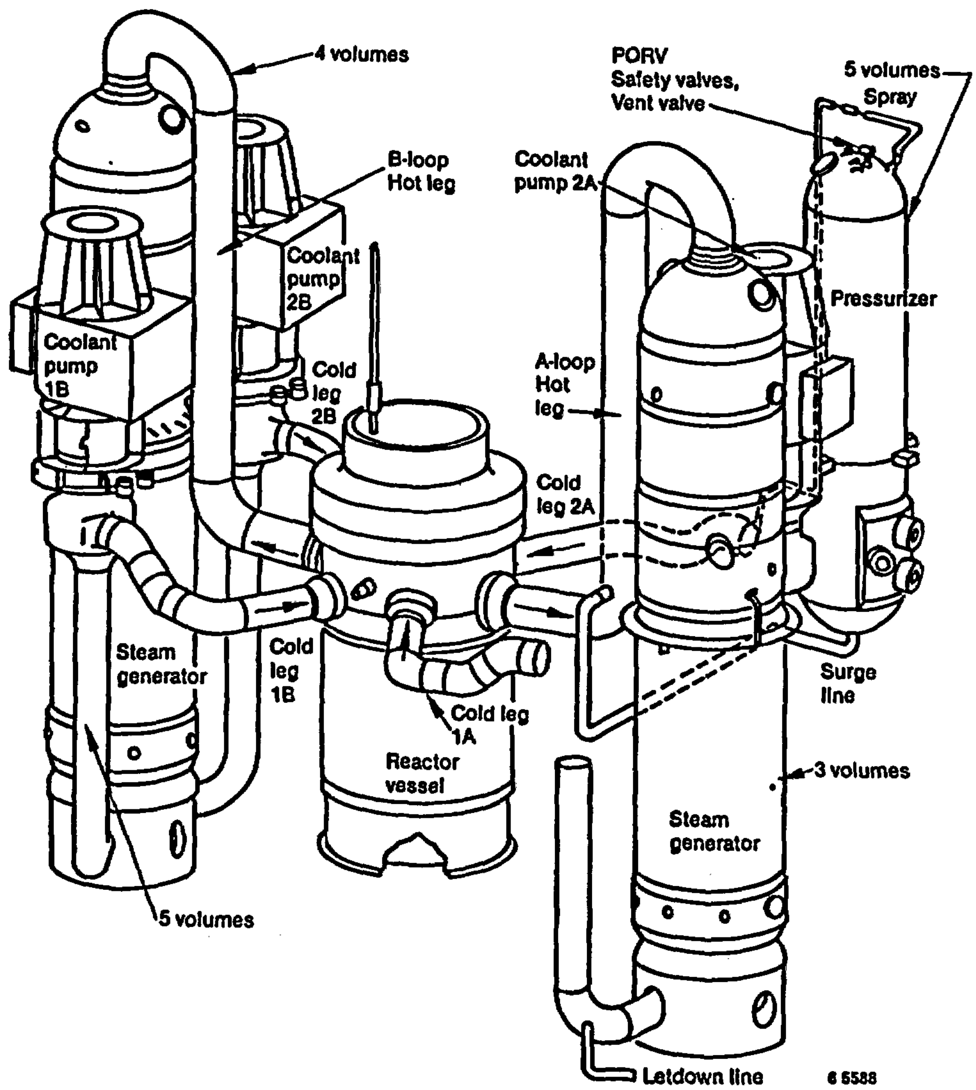

Figure 2. Basis of RELAP/SCDAP model.

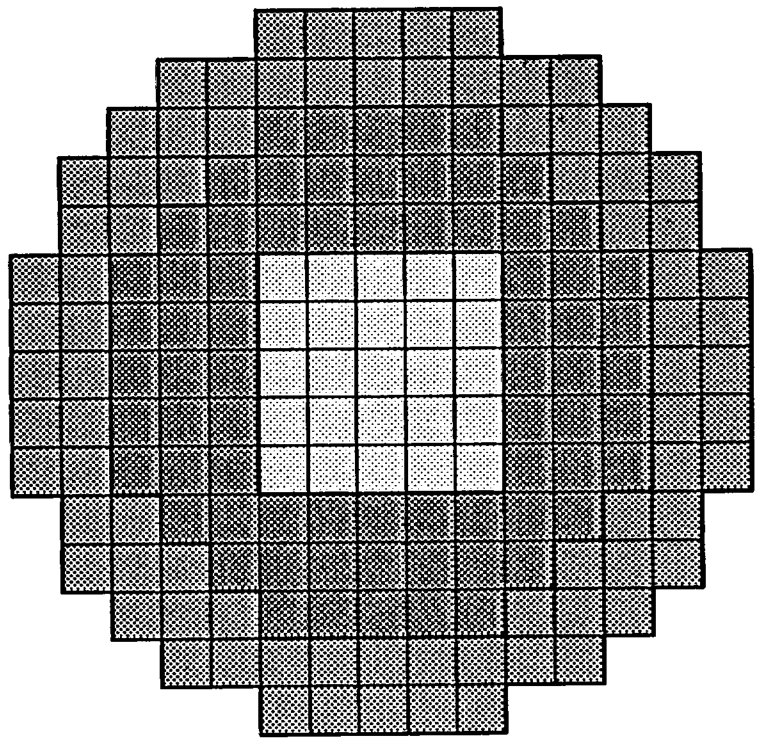

Figure 3. Plan view of reactor core.

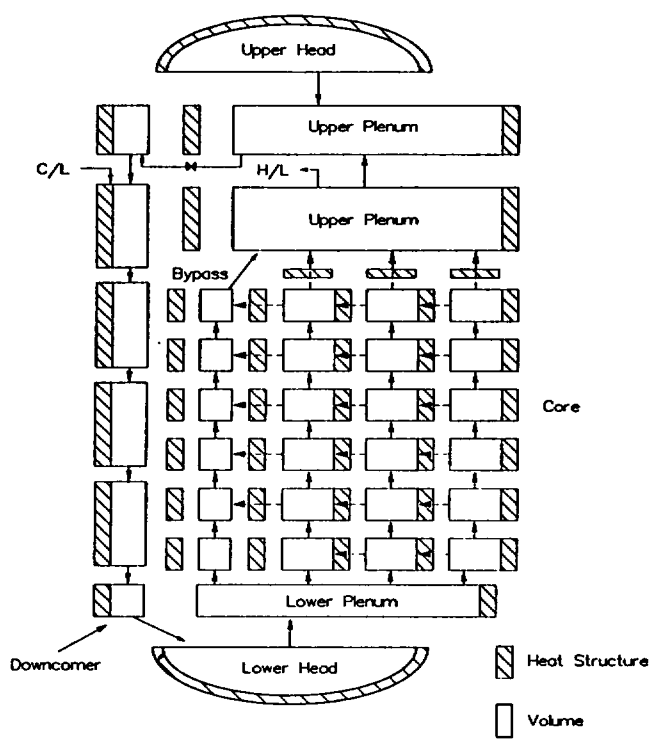

Figure 4. Reactor vessel, elevation.

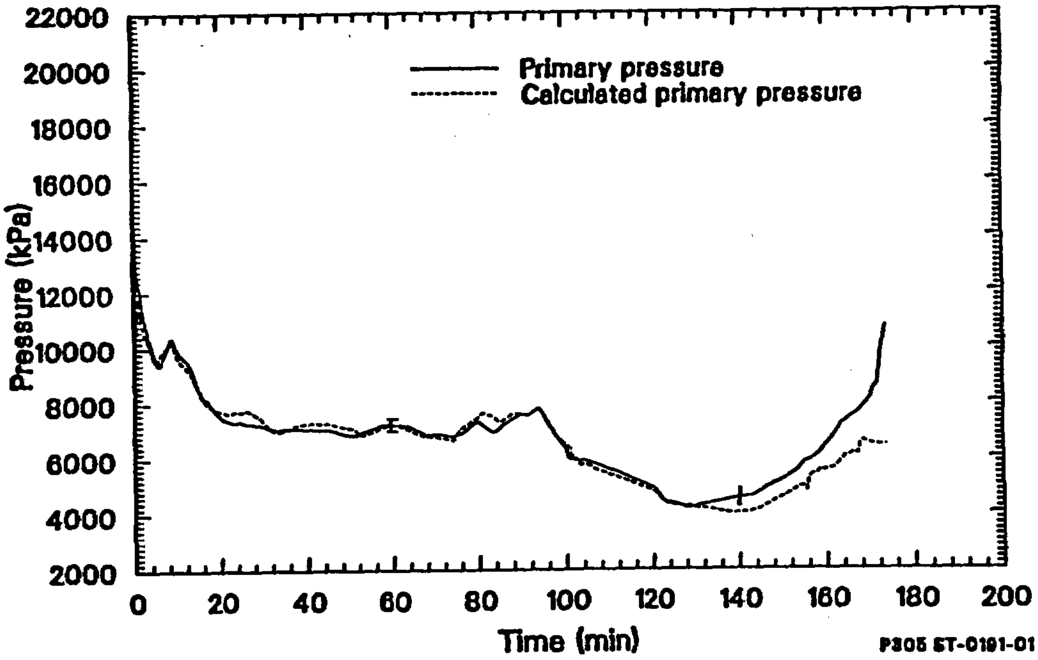

Figure 5. Reactor coolant system pressure.

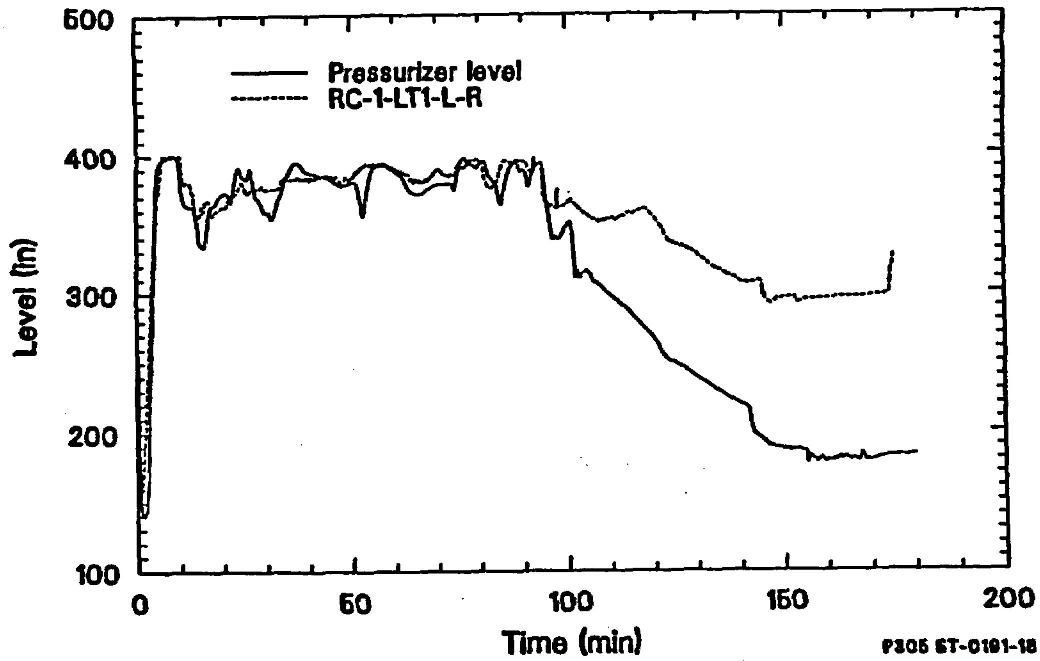

Figure 6. Pressurizer level.

calculated primary side condensation is probably somewhat larger than actually took place. This would, in turn, cause the calculated pressurizer outflow to the primary loop to be greater than actual. This again is probably correctable by finer nodalization in the steam generator.

The hypothesis of excessive mass loss is supported by reactor vessel coolant inventory. Figure 7 shows that the calculated coolant inventory reaches the bottom of the core at about 140 min and remains approximately constant. Based on the observed end-state core condition, the water level probably did not go very far below 30 in. above the bottom of the fuel[2].

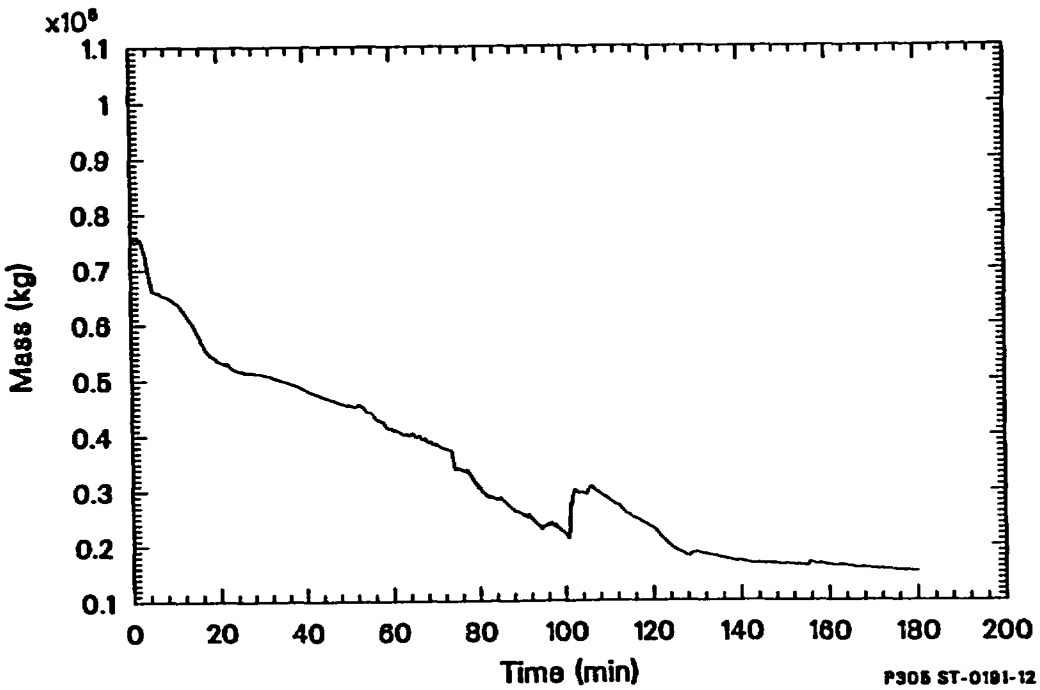

Figure 7. Coolant inventory in the reactor vessel.

Figures 8 and 9 depict the time dependence of the calculated fuel rod temperatures. Figure 8 shows the SCDAP calculated temperatures in the center, mid, and peripheral regions immediately below the core center plane. The differences in temperatures are due to the differences in decay-heat production for the three regions. Figure 9 shows the heatup in the core center region as a function of axial position. As would be expected the heatup commences at the top of the core and progresses downward. Relocation of molten material is predicted to occur at 160 min. This is shown by the rapid increase in temperature in the two lower core nodes as molten material passes through these nodes.

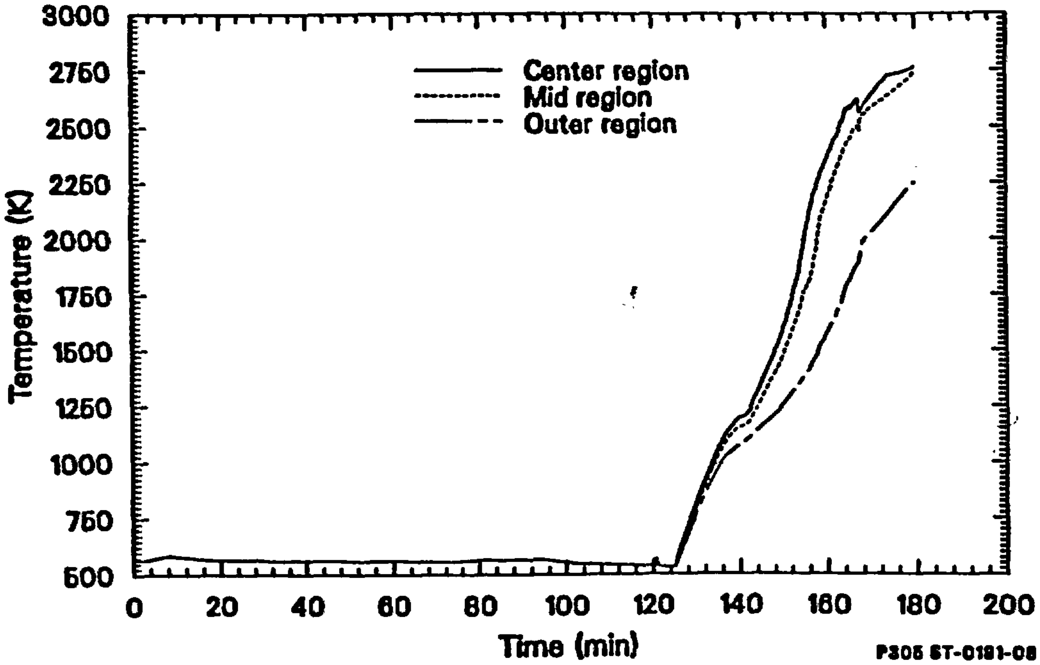

Figure 8. Calculated fuel rod temperatures for the three radial regions.

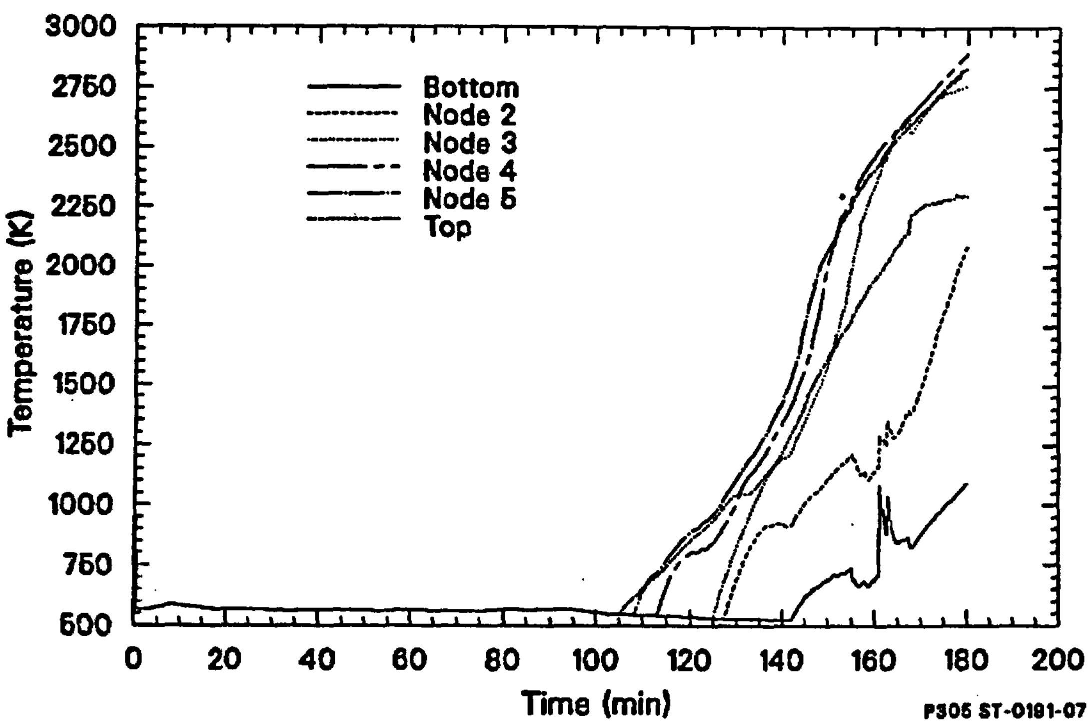

Figure 9. Fuel rod temperatures for the center region.

CONCLUSION

The above discussion is a brief overview and status of the TMI-2 standard problem. Although refinement of the estimates for several of the key boundary conditions is required, it is possible to perform the Phase 1 and 2 calculations as evidenced by the demonstration calculation. By the end of January 1987, the boundary conditions for HPI/makeup and letdown flows (with uncertainties) will be established, and the demonstration calculation will be redone to test the boundary conditions and refine the system nodalization.

REFERENCES

1. H. Warren et al, Interpretation of TMI-2 Instrument Data, NSAC-28, May 1982.

2. E. L. Tolman, "Accident Scenario Update", Proceedings of the 14th Water Reactor Safety Information Meeting, October 1986.