NUREG/CR-1219

Analysis of the Three Mile Island Accident and Alternative Sequences

Prepared by R. O. Wooton, R. S. Denning, P. Cybulskis

Battelle, Columbus Laboratories

Prepared for
U. S. Nuclear Regulatory Commission

Available from

GPO Sales Program
Division of Technical Information and Document Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-1219

Analysis of the Three Mile Island Accident and Alternative Sequences

Manuscript Completed: December 1979
Date Published: January 1980

Prepared by
R. O. Wooton, R. S. Denning, P. Cybulskis

Battelle, Columbus Laboratories
505 King Avenue
Columbus, OH 43201

Prepared for
Three Mile Island Special Inquiry Group
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC Contract No. -04-76-293-08

TABLE OF CONTENTS

Page

ABSTRACT. v

1.0 INTRODUCTION AND SUMMARY. 1-1

2.0 CHRONOLOGY OF TMI ACCIDENT AND DESCRIPTION OF DATA. 2-1

3.0 THE MARCH COMPUTER CODE 3-1

3.1 Subroutine BOIL. 3-2

3.2 Subroutine PRIMP 3-8

3.3 Subroutine EXITQ 3-11

3.4 Subroutine PROPS 3-11

3.5 Subroutine ECC 3-12

3.6 Subroutine MACE. 3-12

4.0 ANALYSIS OF BASE CASE ACCIDENT SEQUENCE 4-1

4.1 MARCH Code Input 4-2

4.2 MARCH Results for 0 - 15 Hours 4-7

4.3 Sensitivity Analyses for the Time Period 0 - 4 Hours . . 4-23

5.0 MARCH ANALYSES OF ALTERNATIVE ACCIDENT SEQUENCES. 5-1

5.1 Case 1 5-2

5.2 Case 2 5-4

5.3 Case 3 5-4

5.4 Case 4 5-5

5.5 Case 5 5-8

5.6 Case 6 5-8

5.7 Case 7 5-11

5.8 Case 8 5-14

5.9 Case 9 5-14

5.10 Case 10. 5-18

TABLE OF CONTENTS
(Continued)

Page

6.0 MARCH ANALYSES OF CORE MELTDOWN SCENARIOS. 6-1

7.0 INTERPRETATION OF THERMAL AND HYDRAULIC CONDITIONS 7-1

7.1 General Description of Accident Conditions. 7-1

7.2 Reconciliation with Measured Data 7-3

8.0 ANALYSIS OF HYDROGEN BURNING DURING THE TMI-2 ACCIDENT . . . 8-1

8.1 Consideration of the Observations at TMI-2. 8-1

8.2 Effect of Containment Design. 8-4

9.0 REFERENCES 9-1

LIST OF TABLES

Table 2.1 Pressurizer Leakage Times. 2-5

Table 2.2 Makeup and Pump Seal Flow. 2-6

Table 2.3 ECC Injection From BWST. 2-7

Table 4.1 MARCH Input Data for TMI Accident. 4-3

Table 4.2 Calculated Core Temperature Maps 4-13

Table 4.3 Effect of Makeup Rate. 4-24

Table 4.4 Effect of Melting Assumptions. 4-25

Table 4.5 Effect of Rod-to-Steam Radiation Heat Transfer 4-27

Table 6.1 Early Core Meltdown. 6-5

Table 6.2 Delayed Core Meltdown. 6-5

Table 8.1 Typical Containment Design Parameters. 8-6

LIST OF FIGURES

Figure 2.1 TMI Data Plots. 2-3

Figure 3.1 MARCH Flow Diagram. 3-3

LIST OF FIGURES
(Continued)

Page

Figure 4.1 Comparison of TMI and Calculated Vessel Pressures, 0 - 15 Hrs. 4-8

Figure 4.2 Calculated Primary Coolant Leak Rate, 0-15 Hrs. . . . 4-10

Figure 4.3 Calculated Mixture Level, 0-15 Hrs. 4-11

Figure 4.4 MARCH Calculated Results for Base Case, 100-220 Min . 4-12

Figure 4.5 Comparison of TMI Hotleg Temperatures with Calculated Temperatures, 0-16 Hrs. 4-17

Figure 4.6 Calculated Heat Loss of Steam Generators and the Total Water Leakage to the Containment, 0-15 Hrs. 4-19

Figure 4.7 Comparison of Decay Heat, Calculated Heat to Letdown Cooler, and Primary Coolant Pump Power, 0-15 Hrs. . . 4-20

Figure 4.8 Comparison of TMI and MARCH Calculated Saturation and Surge Line/Pressurizer Temperatures, 0-16 Hrs 4-21

Figure 4.9 Calculated Distribution of Hydrogen Produced in Initial Core Uncovery Between Containment and Vessel, 0-14 Hrs. 4-23

Figure 5.1 Calculated Results for Alternative Case 1 (Full HPI Flow), 0-100 Min. 5-3

Figure 5.2 Calculated Results for Alternative Case 4 (Emergency Feedwater Delayed to 1.0 Hr), 0-100 Min 5-6

Figure 5.3 Calculated Results for Alternative Case 6 (PORV Closure at 198 Min), 100-200 Min. 5-10

Figure 5.4 Calculated Results for Alternative Case 7 (No RCP's), 0-220 Min 5-15

Figure 5.5 Calculated Results for Alternative Case 8 (Loss of Power at 120 Min), 100-180 Min. 5-16

Figure 6.1 MARCH Results for Assumed Early Core Meltdown, 40 Percent Cladding Reaction, No Rapid Steam Production in Reactor Cavity. 6-6

Figure 6.2 MARCH Results for Assumed Early Core Meltdown, 100 Percent Cladding Reaciton, with Rapid Steam Generation in Reactor Cavity. 6-7

LIST OF FIGURES
(Continued)

Page

Figure 6.3 MARCH Results for Assumed Meltdown Accident Sequence Starting at 3 Days, 70 Percent Cladding Reaction, With Rapid Steam Generation in Reactor Cavity. 6-8

Figure 6.4 Bottom Pad Melt-Through Calculations with Subroutine INTER 6-11

ANALYSIS OF THE THREE MILE ISLAND ACCIDENT AND ALTERNATIVE SEQUENCES

by

R. O. Wooton, R. S. Denning, and P. Cybulskis

ABSTRACT

A number of analyses were performed with the MARCH computer code to assist the TMI Special Inquiry Group. The MARCH code predicts the thermal and hydraulic conditions in the reactor primary system and containment building in core meltdown accidents. The purpose of the analyses was to examine a number of variations in system operation in the TMI accident to evaluate their effect on the extent of core damage. The results indicate that:

1) The throttling of HPI had a major effect on core damage. If the system had been permitted to operate at high flow, the core would not have uncovered regardless of PORV position or the availability of emergency feedwater.
2) Closure of the block valve in the PORV line at 25 minutes into the accident would have permitted the operation of the reactor coolant pumps to continue and would have prevented core damage. An additional delay of one hour in closing the valve would have resulted in severe core damage and possibly core meltdown.
3) The delay in operation of the emergency feedwater system had little effect on the extent of core damage. However, a delay of one hour in the delivery of emergency feedwater would probably have resulted in more severe core damage and possibly core meltdown.

In this study, an interpretation of the thermal and hydraulic behavior during the first sixteen hours has been developed which is consistent with measured data from the plant. About 40 minutes after shutdown of the reactor coolant pumps, the water level fell to ~ 4 to 5 feet from the bottom

of the core. Although the operation of a reactor coolant pump at 2:54 was probably important in limiting the extent of core damage, the core was not recovered until operation of the HPI at 3:20. The top of the core was not uncovered again, although regions of the core remained vapor blanketed for days. For a number of hours following core recovery, flow through the hot legs was blocked by the presence of hydrogen and the hot leg temperatures remained in the range of 750 - 800°F, to which they had been heated during core uncovery.

Some analyses were performed with MARCH for sequences leading to complete core meltdown to examine the likelihood of different containment failure modes. Particular attention was given to the possibility of an atmospheric failure of containment resulting from a hydrogen explosion. The potential for containment failure was also evaluated for a variety of different containment designs for the same quantity of hydrogen as apparently was consumed in the burning event at TMI-2.

1.0 INTRODUCTION AND SUMMARY

Analyses of the TMI-2 accident and a number of alternative accident sequences have been made to assist the TMI Special Inquiry Group of the U. S. Nuclear Regulatory Commission. Detailed calculations were performed with the Battelle-developed MARCH computer code. This report contains a description of the MARCH code. In addition to the results of the MARCH calculations, a semi-quantitative description of the hydraulic behavior of the primary system is presented. In this description we have attempted to reconcile our interpretation of the course of the accident with measured TMI data using hand calculations. These can be more easily understood and checked by the reader than the results of computer codes.

The major conclusions of our analysis of the TMI accident are:

- Between 150 and 200 minutes after accident initiation, temperatures in the upper region of the core were achieved which would have led to cladding oxidation, severe damage to the integrity of fuel, and quite possibly to fuel melting.
- The predictions by the MARCH code of the extent of core damage are very sensitive to the coolant makeup rate to the vessel during the period of core uncovery. Variations of 25 gpm from the base case would have resulted in very limited damage to the core for a higher makeup rate or to conditions for a lower makeup rate which could have resulted in core meltdown.
- Because of the sensitivity to uncertainties in boundary conditions and modeling assumptions, it is not possible to predict the extent of core damage or fuel melting in the accident. A final understanding of the severity of damage to the fuel must await fuel examination.
- The only period of core uncovery occurred between 1.7 and 3.5 hours although some covered regions of the core apparently remained vapor blanketed for days.

- Between 3.5 and 10.5 hours there was sufficient coolant flow through the core and out the pilot operated relief valve to carry away all of the decay heat with mixed mean outlet conditions that were subcooled. The flow-through of this coolant was the major mechanism for decay heat removal in this time period.
- Stagnant gases were trapped in the upper portions of the hot legs, steam generators, and reactor vessel after 3.5 hours. The observed primary system pressures after 3.5 hours can be explained by simple compression and expansion of these gases due to changes in the primary system water levels as makeup and coolant leakage rates were varied.
- Hydrogen was vented from the primary system when the surge line connection to the hot leg was uncovered in the 3.2 - 3.8 hours, 8.3 - 9.0 hours, and 10.8 - 11 hours periods.

Ten alternative accident sequences were examined. For some of the alternative sequences the thermal-hydraulic conditions were sufficiently different from the actual TMI conditions that the relatively simple MARCH models (in comparison to the RELAP and TRAC codes) were judged not to be definitive. Bounding calculations were performed for those cases. The alternative cases and the results of our analyses are:

- Case 1. <u>HPI flow not throttled</u>.
  The primary system remains full and a normal cooldown is achieved.
- Case 2. <u>Emergency feedwater delayed one hour and HPI flow not throttled</u>.
  The primary system remains full and a normal cooldown is achieved.
- Case 3. <u>Emergency feedwater is delivered as designed</u>.
  The core damage is the same and begins at about the same time as in the base case (actual TMI-2 accident).
- Case 4. <u>Emergency feedwater is delayed one hour.</u>
  MARCH predicts that core damage begins about 80 minutes earlier than in the base case. The thermal-hydraulic conditions are significantly altered from the base case.

- Case 5. <u>PORV leakage is stopped at 25 minutes.</u>
  The transient is stopped after 15 percent of the primary coolant is lost. The core is not uncovered, no core damage occurs, and a normal cooldown is achieved.
- Case 6. <u>PORV leakage continues until 3.3 hours.</u>
  Our best estimate is that additional core damage occurs relative to the base case. Whether or not core meltdown would result cannot be determined. There is a possibility for this case that a partial dump of the core flood tanks occurs. If the core flood tanks do discharge, MARCH predicts much less core damage than in the base case.
- Case 7. <u>The primary coolant pumps are immediately tripped.</u>
  Our best estimate is that the core damage is about the same as in the base case. A second possibility exists for this case that a steam bubble can eventually be maintained in the pressurizer. MARCH models are not adequate to evaluate this possibility. However, if a steam bubble can be established, MARCH predicts core uncovery will not occur prior to closure of the PORV at 2.3 hours, and no core damage would result.
- Case 8. <u>Loss of all AC electric power.</u>
  Loss of power at 2 hours was assumed. Complete core meltdown is predicted by 2.9 hours assuming that power is not restored.
- Case 9. <u>PORV remains closed after 2.3 hours and HPI not initiated at 3.3 hours.</u>
  Additional core uncovery and heatup occurs after 3.3 hours. The core is about half molten by 5 hours, and eventual meltdown would be expected.
- Case 10. <u>HPI is not initiated at 3.3 hours.</u>
  MARCH predicts similar results for alternative Cases 9 and 10.

In addition to the alternative cases above, MARCH calculations were also performed for two accident sequences deliberately designed to produce complete core meltdown. One sequence was intended to investigate an early meltdown and the other to produce a meltdown starting with the TMI conditions at 3 days. The purpose of these calculations was to investigate the course and timing of core meltdown and to evaluate possible mechanisms that could threaten the integrity of the containment. Since the containment coolers were operational, the greatest threat to containment integrity was felt to be from the rapid combustion of the hydrogen generated from metal-water reactions. MARCH analyses indicate the uncertainty bounds on hydrogen production in core meltdown range between the equivalent of 40 and 100 percent reaction of the core zircaloy. If the hydrogen concentration in containment corresponding to 100 percent cladding reaction were to accumulate well beyond the flammability limit, containment failure could result upon ignition. The most likely time for this to occur would be when the pressure vessel fails and the molten core falls into the reactor cavity. Whether, indeed, hydrogen would accumulate to critical levels without undergoing prior combustion and then explode with sufficient energy to fail containment, cannot be determined without further research.

Finally, analyses were performed to evaluate the impact that the hydrogen burning event that occurred in the TMI-2 containment would have, if it were to occur in other types of containment design. In general, the pressure suppression containment designs with lower design pressures are much more vulnerable to hydrogen explosion than large dry containments.

2.0 CHRONOLOGY OF TMI ACCIDENT AND DESCRIPTION OF DATA

The accident at TMI-2 began at 4:00 in the morning of March 28, 1979. The initiating event was a loss of feedwater to the steam generator secondary. The resulting degradation of heat transfer from the primary system caused an increase in pressure and shutdown of the reactor. The pilot operated relief valve (PORV) on the pressurizer opened at the setpoint of 2250 psia in response to the increase in primary system pressure, and water began to leak from the reactor. The coolant leakage through the open PORV continued until about 142 min* later when a block valve was closed. During this time, the water inventory in the primary system decreased from an initial value of over 500,000 lb to a value somewhat below 200,000 lb**. The coolant leakage rate during most of this period was about 2800 lb/min** while the net makeup rate from ECC injection was an order of magnitude less. The system pressure fell below 1300 psia at 15 min and remained at approximately 1100 psia until 101 min, when the last two primary coolant pumps were shut down in response to indications of pump cavitation. The first sign of core uncovery began at about 110 min when thermocouples in the hotlegs indicated the steam boiling out of the core was superheated. The hotleg temperatures remained in the 700-800 F range for nearly 10 hours. The system pressure decreased to a minimum of about 650 psia at about the time the coolant leakage was stopped at 142 min. Also, during this period between 110 and 142 min, thermocouples above the core and the self powered neutron detectors (SPND) began to indicate temperatures in the 1000 F range, source range core power level monitors began to read high in response to increased neutron flux from the uncovering core, and high radiation levels were observed in coolant samples and in the containment building as the result of fission product release from the over-heated core.

When the coolant leakage was stopped at about 142 min, the system pressure began to increase. At about the time reactor coolant pump 2B was

* In the following discussion, the event times refer to accident time, that is, time after 4:00 a.m. The uncertainty in most of the event times is generally less than a few minutes.

** Calculated value. Not directly obtainable from the data.

temporarily turned on at 174 min, the system pressure increased rapidly to over 2000 psi. ECC injection was significantly increased at about 200 min. (MARCH calculations indicate the core remained covered after 3.5 hrs). Over the next 14 hours, the primary system pressure varied between about 2200 and 550 psi in response to changes in the ECC injection rate and the opening and closing of the block valve in the line of the stuck relief valve. Containment building temperatures and pressures generally responded as expected to whether the relief valve line was open or closed. At about 10 hours, the containment pressure briefly increased by 28 psi, indicating a containment hydrogen burn. During most of the first 16 hours of the accident, the pressurizer water level indicated a full pressurizer. Under normal conditions, this would be an indication that the primary system was water-filled. Between 1.8 and 3.6 hours, and for about 60 min at 11 hours and 30 min at 13 hours, the pressurizer water level indicator fell below 90 percent full.

The above is a very brief chronology of the TMI accident based upon the available instrumentation readings from this time period. The data do not provide sufficient boundary conditions to accurately describe the thermal and hydraulic conditions within the primary system at all times. They do, however, provide check points against which to test the validity of models of the system behavior. The interpretation of some of the data is still subject to debate.

The TMI data have been reported and analyzed by the utility, the Office of Inspection and Enforcement of the NRC[1], the Nuclear Safety Analysis Center[2], and the NRC Special Inquiry Group. These groups have made somewhat different interpretations of some of the data. Primarily, the differences result from the different dates the studies were performed and the resulting refinements of the interpretations. The authors of the present report have made no attempt to make an independent evaluation of the raw data.

Figure 2.1 shows plots of the TMI data. These data have been reported previously and are repeated here for reference.

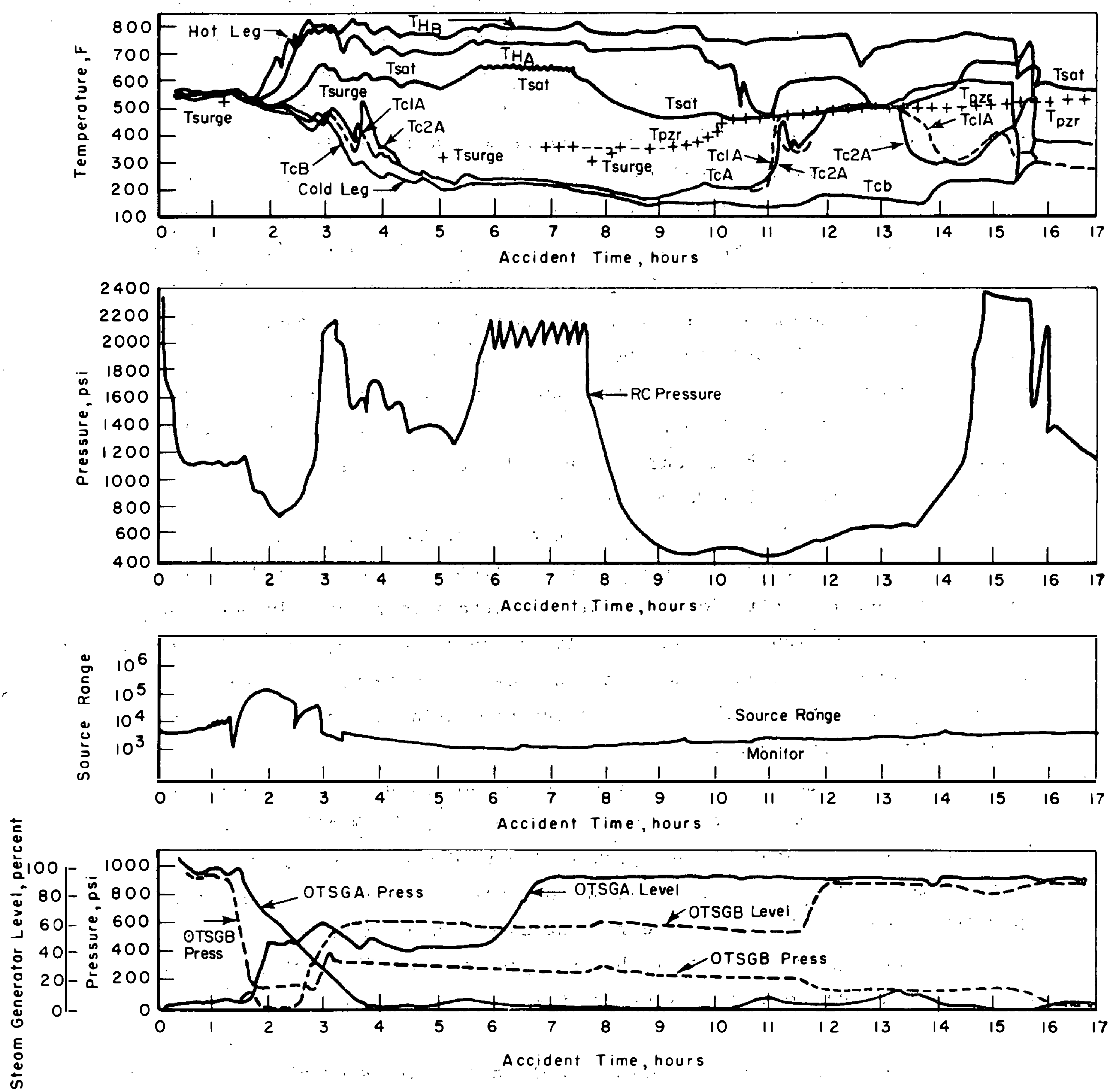

Figure 2.1 TMI Data Plots

Table 2.1 lists the opening and closing times of the PORV block valve. When the various references indicate disagreement, a range of times is listed. The most important of the differences, occurs in the 192-221 min time interval. The MARCH calculations of the system pressure in this time interval are sensitive to the relative timing of the opening and closing of the PORV and the initiation of ECC injection. Note that in the 459 to 550 min time interval, both the PORV block valve and the pressurizer vent lines are open. The vent line is a "non-safety" item, and its rated capacity is not known (to us). Based on the known diameter (1 inch) of the line, having both flow paths open could increase the effective leakage area to about 160 percent of the PORV area alone.

Table 2.2 lists the makeup flow rates for a portion of the accident period. The data are reproduced from NUREG-0600[1]. The average makeup rate is about 80 gpm. Under normal operating conditions, the letdown flow is balanced by the makeup flow so that there is little net water addition to the primary through the makeup/letdown flow. The return flow is cooled by the letdown cooler, however, so that there is a net energy loss. Over the short term (few hours) the makeup and letdown flows may not be balanced. Accurate values of makeup flow rate and letdown flow rate are not known as a function of time during the TMI accident.

Table 2.3 lists the injection rate of water from the borated water storage tank. This injection represents addition of new water to the system. The injection rates shown are average rates over the indicated periods. The actual rate at a particular time may differ significantly from the average since the high pressure injection system was activated and later throttled by the operators during some time periods.

We have not seen sufficient information to conclusively determine the importance of heat transfer to the steam generator during the accident. Data that are available include the secondary side pressure and water levels in the steam generators. The mode of heat rejection of the steam generators, whether they are dumping heat to the condenser or boiling off through the steam dump valves, are also known. Information on the flow rates and temperatures of the feedwater to the secondary are generally lacking. The initiation of emergency feedwater at 8 min is well established. NSAC investigators[2]

TABLE 2-1. PRESSURIZER LEAKAGE TIMES

Time, min	State of Valve
0 - (138, 142)*	PORV** full open
(138, 142) - (192, 193)	no leakage
(192, 193) - (195, 210)	PORV full open
(195, 210) - (198, 221)	no leakage
221 - 318	PORV full open***
318 - 348	no leakage
348 - 459	Block valve open 40% of time
459 - 550	PORV and pressurizer vent valve open
550 - 600	no leakage
600 - 666	PORV full open
666 - 756	no leakage
756 - 765	PORV full open
765 - 771	no leakage
771 - 780	PORV full open
780 - end	no leakage

* Indicates range of times from various sources.

** The block valve is also open. The leakage rate is determined by the PORV capacity.

*** Recent analysis by the TMI Special Inquiry Group indicates the PORV was closed 64 percent of the time between 221 and 276 min.

TABLE 2-2. MAKEUP AND PUMP SEAL FLOW

Time, Hrs.	Makeup valve MU-V17 flow, gpm	Pump Seal Flow, gpm
0	16	32
1	118	32
2	97	32
3	125	32
4	8	32
5	58	32
6	52	32
7	49	32
8	49	32
9	97	32
10	113	32
11	95	32
12	116	32
13	128	32
14	43	32
15	72	32
16	95	32

TABLE 2-3. ECC INJECTION FROM BWST

Time, min	Flow Rate, gpm
0.3 - 4.7	672
4.7 - 200	25 (165 injection, 140 letdown)
200 - 207	1000
207 - 236	360
236 - 240	0
240 - 244	1000
210 - 415	640 (average)
415 - 555	360
555 - 800	150
800 -	470

have concluded there was emergency feedwater to the A-steam generator between 90 and 124 min. For much of the accident, the role of the steam generators as a heat sink for the primary must be inferred from the overall primary-secondary system response.

3.0 THE MARCH COMPUTER CODE

The MARCH code calculates the thermal-hydraulic behavior of a water-cooled reactor during a meltdown accident. MARCH is an acronym for Meltdown Accident Response CHaracteristics. MARCH is being developed by Battelle-Columbus for the Probabilistic Analysis Staff of the NRC. The development of the MARCH code is an extension of the meltdown analysis work performed by Battelle-Columbus for the Reactor Safety Study. The original BOIL[(3)] code, which models core heatup and meltdown, was developed as part of that effort. BOIL is incorporated as a subroutine of MARCH. Many of the other analyses in the Reactor Safety Study, including the calculation of reactor vessel failure, concrete decomposition, and containment building pressurization, were independently performed hand calculations. MARCH provides a continuous, coupled analysis of the thermal and hydraulic behavior of the containment building and reactor from the time of accident initiation. Primary system response is modeled for both large and small pipe break loss-of-coolant accidents and for transients. An effort is currently underway to test, document, and make MARCH available for external distribution. An unverified version of MARCH is being prepared for preliminary release by the end of 1979. Containment pressures for large pipe break accidents and primary system pressures and blowdown rates for small cold-leg pipe breaks appear to be in reasonable agreement with vendor analyses reported in safety analysis reports. No direct comparisons have been made previously with the CONTEMPT and BEACON (containment) codes or the RELAP and TRAC (primary system thermal-hydraulics) codes. Of course, some of the MARCH models are not easily tested since few other codes attempt to model core melting.

During the blowdown phase of a pipe break accident, MARCH performs the function of a coupled RELAP/CONTEMPT code. However, MARCH does not generally approach the level of sophistication or detail of these codes. Since many of the physical processes of core meltdown are not well understood, detailed modeling of system thermal-hydraulic behavior has not been justified. As a result the code is fast running and inexpensive. This is convenient since the processes of core meltdown frequently must be analyzed for accident times of hours or days. MARCH computational times can be orders of magnitude less than those of RELAP. For example, the MARCH calculation of the first 15 hours of the TMI accident required about 400 sec computational time on a CDC 6500. The calculations of alternative accident sequences discussed in Section 5.0 generally required 100-200 sec computational time.

Figure 3.1 illustrates the overall structure of the MARCH code. For the present TMI study, subroutines BOIL and PRIMP are of primary interest. BOIL performs the primary system heat transfer and mass and energy balances, and PRIMP calculates the primary system pressure and leakage rates. These subroutines will be discussed in more detail below. Subroutines HEAD, HOTDROP, and INTER model phenomena occurring following complete core meltdown and will not be discussed in detail. HEAD calculates bottom head failure due to the combined effects of melting and overstressing. HOTDROP predicts the boiloff of water (if present) from the reactor cavity below the reactor vessel following head failure. The decomposition and penetration of the concrete floor of the containment building is calculated by INTER. (The INTER[4] code was developed by Sandia Laboratories.) MACE calculates the containment building thermal-hydraulics. The analysis of the alternative accident scenarios primarily involves thermal-hydraulic conditions in the primary system. The MACE routine will be briefly discussed, however, since the containment building temperature and pressure response are of interest for the core meltdown scenarios in Section 6.

3.1 Subroutine BOIL

The BOIL subroutine calculates the core heatup and meltdown and performs energy and mass balances for the primary system. Most of the BOIL models are the same as those reported in the Reactor Safety Study. However, a number of changes have been made. A review of BOIL follows.

The primary system is modeled as a single volume which is subdivided into a water space and a steam space. The water space incorporates a level swell model. Core nodes which are covered by the swollen mixture level are generally assumed to be well-cooled. Energy transferred to the water space from the core includes the decay heat of all mixture covered nodes in addition to heat coming from changes in the stored energy of the fuel nodes. These stored energy terms result from quenching of newly slumped

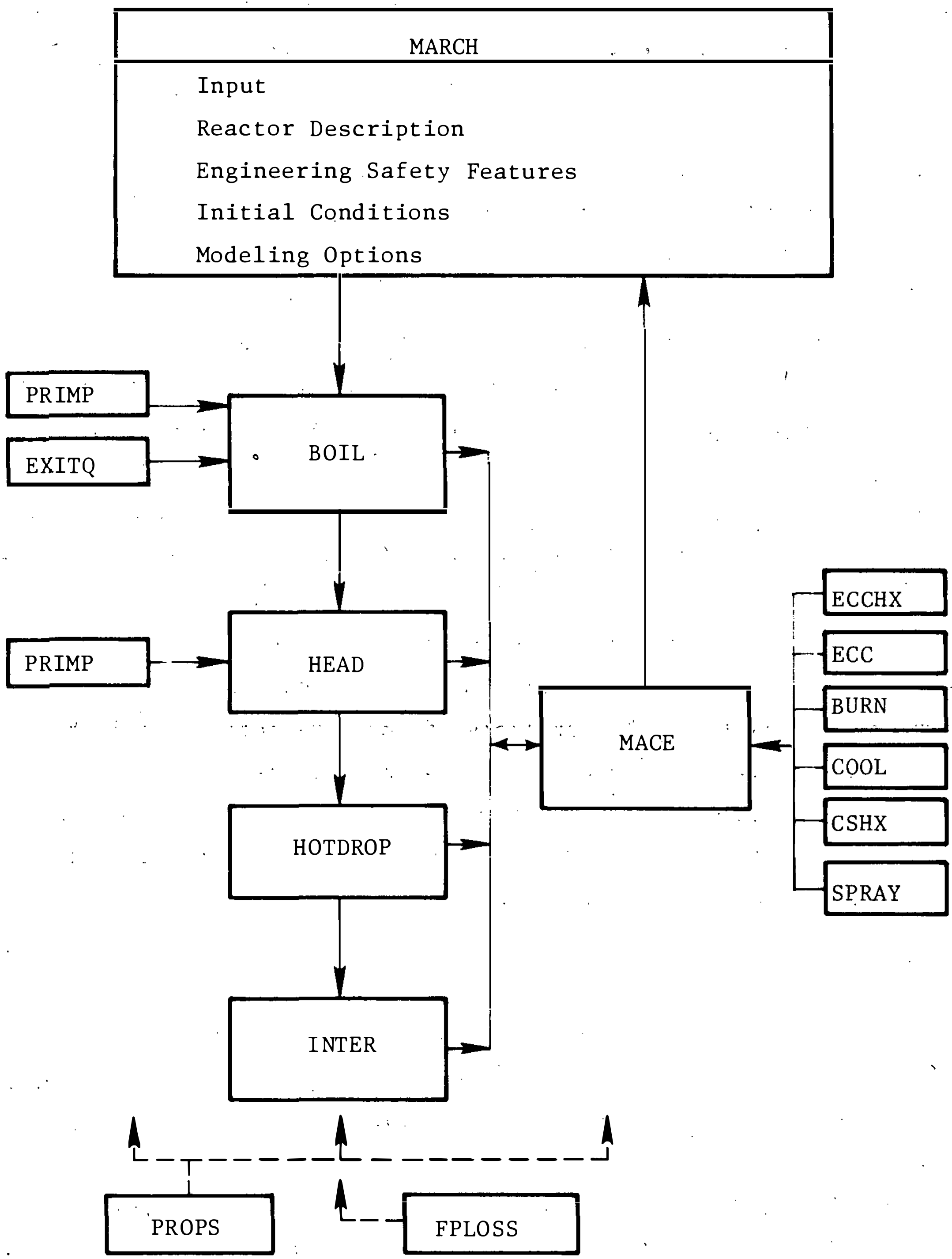

Figure 3.1 MARCH Flow Diagram

(molten) core nodes which fall into the water or from quenching previously dry core nodes which become covered by a rising mixture level due to ECC injection or increasing level swell.

Heat may be transferred from the primary system water space to the steam generator (for a PWR) or by passing primary water through a letdown/makeup cooler. Heat removal by the letdown cooler is modeled as

$$QMUP = WMUP*CP*(TPOOL - TECC) \quad ,$$

where

QMUP = heat removed, Btu/min

WMUP = (input) makeup/letdown flow rate, lb/min

CP = specific heat of water, Btu/lb F

TPOOL = temperature of water in vessel, F , and

TECC = makeup water temperature, F.

It is assumed there is heat removal but no net water addition through this flow path. Water additions are modeled as ECC flows.

The steam generator model permits (input option) cooling of either the water space or condensation of steam from the steam space. For cooling of the water space, the model first defines a heat transfer coefficient, H_1, based on the initial operating conditions,

$$H_1 = QZERO/(A\Delta T_1) \quad ,$$

where

QZERO = initial core power, Btu/min

A = steam generator heat transfer area, ft^2 , and

ΔT_1 = initial temperature difference, F.

The steam generator heat transfer rate, QSG, becomes

$$QSG = FSG*H_1*(\Delta T/\Delta T_1)^{1/3}*A*\Delta T \quad ,$$

where ΔT is the temperature difference for the current time step. The factor FSG is incorporated in the model to account for changes in the effective heat transfer area due to changes in the primary or secondary side water levels.

FSG is defined as the minimum of (WTRSG/FULSG) or a term proportional to (YLIQ-YLEG). FULSG is the initial steam generator secondary side water inventory and WTRSG is the current value. The secondary side heat and mass balance considers both boiloff and feedwater addition. The secondary safety/relief valves are not modeled and are assumed to be large enough to accommodate the calculated boiloff rate. YLIQ is the primary side liquid level and YLEG is the hotleg elevation.

At (input specified) times, the steam generator may condense steam out of the primary side steam space. For example, steam condensation may occur when cold auxiliary feedwater is being poured over the tops of the steam generator tubes. The condensation rate is modeled as

SGDWS = FS*STM/DTM*(1 - (PSG + PRES)/(2PRES)) ,

where

DTM = time step, min

FS = mole fraction steam

STM = steam mass in the primary, lb.

PSG = (input specified) secondary pressure, psia ,and

PRES = primary side pressure, psia.

The model condenses steam from the primary until the primary and secondary side steam partial pressures equilibrate. The condensed steam may (input option) be returned to the water in the reactor vessel or be assumed to remain in the bottom of the steam generator.

The reactor core in the TMI calculations was modeled in BOIL using 10 radial and 24 axial power zones. Core nodes in the mixture region are assumed to be well cooled. Nodes in the steam space are convection cooled by the steam boiling out of the mixture region. BOIL models radiation heat transfer from the top fuel nodes in the core to structures above the core and from core nodes just above the mixture to the water region. Temperature smoothing due to rod-to-rod radiation is ignored.

Part of the steam flow may be converted to hydrogen by reaction with the Zircaloy cladding. The reaction rate is modeled in BOIL as the minimum of that obtained from the steam flow, a solid-state diffusion rate law[5], or a gas phase diffusion rate[6]. Generally, the reaction is controlled by one of the first two limitations. The energy produced by the metal-water reaction is added to the fuel node heat source. The reaction may be terminated by user input options when a core node melts or a channel is plugged by core slumping. In the present TMI calculations, node melting was assumed to stop the metal-water reaction. BOIL assumes there is no metal-water reaction in nodes covered by the mixture level.

The BOIL programming assumes a single (input) value for the core melting point. Cladding and fuel are assumed to melt at the same temperature. A melting temperature of 4130°F was used in the TMI calculations. Experiments[7] indicate the core melting temperature is not single-valued and is actually a function of the concentrations of zirconium, oxidized zirconium, and uranium oxide. Melting or liquefaction temperatures between about 3500 and 5100°F are observed. The 4130°F temperature used in the TMI calculations is in the mid-range of the data.

In the BOIL models, core melting triggers four effects. One effect is that a heat of fusion term appears in the heat balances. A second effect is that the metal-water reaction is stopped (input option) in melted fuel nodes. A third effect is that steam cooling of melted nodes is stopped (input option). A fourth effect is that the core slumping models become active. It should be noted that BOIL does not alter the effective heat transfer area of melted rod nodes for purposes of calculating heat transfer within the mixture covered region. Thus, if a melted fuel node becomes covered by the mixture, either by downward fuel slumping, by ECC injection, or by increased level swell, it is assumed to be as well cooled as an intact fuel rod node. The energy released by quenching the molten node simply becomes an additional heat source term in the heat balance equation for the water region.

The original BOIL code assumed complete quenching could occur in one time step. In order to better model the conditions in the TMI accident during core recovery, the BOIL models were modified. The quenching rate in BOIL is presently calculated from the minimum obtained using either a boiling heat transfer coefficient or rates characterized by a time constant τ. Three terms are evaluated:

$$QB1 = h_B A(TROD - TPOOL)\Delta t$$

$$QB2 = MC(TROD - TE)\Delta t/\tau \quad \text{,and}$$

$$QB3 = \rho_L V_L h_{fg}(1 - \alpha)\Delta t/\tau \quad ,$$

where

h_B = boiling heat transfer coefficient, Btu/hr ft^{2}F

A = node heat transfer area, ft^2

MC = node heat capacity, Btu/F

ρ_L = water density, lb/ft^3

h_{fg} = heat of vaporization, Btu/lb

V_L = node water volume, ft^3

TROD = fuel node temperature, F

TE = equilibrium temperature, F

Δt = timestep, min

τ = time constant, min, and

α = local void fraction.

QB2 is the change in stored heat obtained in quenching to decay heat or equilibrium levels. QB3 is the heat required to vaporize the water in the coolant channel next to the fuel node. For a 1.0 minute time constant τ, the QB2 and QB3 terms are generally found to control the quenching rate in BOIL.

BOIL contains three fuel slumping or meltdown models. Models A and B assume retention of molten fuel within the core region until a user specified core melt fraction is obtained. Model A slumps molten fuel downward faster than Model B. Model C assumes molten fuel nodes drop completely out of the core into the bottom head as quickly as the nodes melt. Models A and B are believed to be more realistic than Model C. Models A and B generally produce similar results for core melt fractions less than 0.5. Model A was used in the TMI calculations.

BOIL uses the Wilson[8] correlation of bubble rise velocity, with a minimum velocity of 1 ft/sec, to calculate the core void fraction. Level swell within the core region is calculated assuming a linear variation of void fraction with elevation.

The MARCH calculated liquid level is defined as

$$YLIQ = WMASS/(\rho_L*ATOT) \quad ,$$

where ATOT is the total cross sectional area of the reactor vessel in the active core region. WMASS is the portion of the primary system water inventory located above the bottom of the core. The MARCH liquid level is an artificial parameter, except when the liquid level is actually within the active core region.

3.2 Subroutine PRIMP

Subroutine PRIMP calculates the leakage of steam and water from the primary, places an upper limit on the coolant flashing rate, and calculates a new MARCH time step size.

PRIMP leaks steam from the primary system steam space if the primary system liquid level is below the (user specified) break elevation. Water leaks from the water region if the break is below the liquid level. MARCH has no models which would permit leakage of a two-phase steam-water mixture. The user may specify that a constant two-phase flow multiplier be applied to the water leakage when the calculated water temperature is less than 20°F subcooled. However, as modeled, the multiplier affects only the water leakage rate and not the steam content of the leakage.

The steam and water leakage rates are the minimum of either an orifice flow (G) or a critical flow rate (G_c). The PRIMP flow rates for steam are

$$G = 5778\ CBRK(\rho * DP)^{0.5},\ lb/min\ ft^2 \quad ,\ \text{and}$$

$$G_c = 1642(\rho * P)^{0.5},\ lb/min\ ft^2 \quad , \qquad \text{(ref 9)}$$

where

CBRK = orifice coefficient (0.583 used)

DP = pressure differential between vessel and containment

P = vessel pressure, psia , and

ρ = gas density.

Steam leakage through the safety relief valves is calculated from

$$WVENT = CSRV * (\rho * P)^{0.5},\ lb/min\ (P \geq PSET) \quad ,$$

where CSRV is an input coefficient calculated to reproduce the rated venting rate. PRIMP will reduce WVENT if the new system pressure falls below the relief valve setpoint, PSET. For mixtures of steam and hydrogen, the gas density is defined as $\rho = (STM + H_2)/VOL$, and the steam and hydrogen leakage rates are assumed proportional to their mass fractions in the gas space. Using the relations above, the full-open PORV for TMI2 has an effective steam flow area of

$$ABRK = \frac{112{,}000(lb/hr)/60(min/hr)}{1642\sqrt{6.466*2270}} = 0.00938\ ft^2 \quad .$$

The water leakage is calculated from

$$G = 5778\ CBRK(\rho_L * DP/TPM)^{0.5},\ lb/min\ ft^2 \quad ,\ \text{and}$$

$$G_c = 1570.2(P/TPM)^{0.813},\ lb/min\ ft^2 \quad , \qquad \text{(ref 9)}$$

where

ρ_L = water density, lb/ft^3 , and

TPM = (input) two-phase multiplier .

In the TMI base case calculations, a two-phase multiplier of 1.74 was found to result in a liquid level at the top of the core at 101 minutes, the time at which the primary coolant pumps were turned off. The critical water flow rate is reduced by a factor $TMP^{0.813} = 1.57$ by use of this multiplier. It should be noted that, as used in these calculations, the two-phase flow multiplier serves simply as a correction factor which permits core uncovery at the desired time. However, a 1.74 two-phase multiplier is consistent with a low quality, moderate void fraction two-phase flow. (This multiplier corresponds to a quality of about 0.02 or a void fraction of 0.3 at 1,000 psia, assuming no slip.)

The system pressure is calculated assuming the steam and hydrogen in the gas space are ideal gases. The gas space volume, VOL, is calculated by subtracting the water volume from the total primary volume, VOLP, thus

$$VOL = VOLP - WMTOT/\rho_L$$

were WMTOT is the water weight and ρ_L is the water density evaluated at the water temperature, TPOOL. The temperature, GAST, of the gases in the steam space is calculated assuming the gases coming from the core are uniformly mixed with the gases already in the steam space. Thus,

$$GAST = \frac{(MC*GAST + DMC*TGEX)}{(MC + DMC)} ,$$

where

MC = STM*CPS + H2*CPH, Btu/F

DMC = (WS*CPS + WH*CPH)*DT, Btu/F

TGEX = temperature of gas coming from core, F

STM = weight of steam, lb

H2 = weight of hydrogen, lb

WS = flow of steam exiting core, lb/min, and

WH = flow of hydrogen exiting core, lb/min.

The new steam and hydrogen weights are

$$STM = STM + (WS - WSLK)*DT \quad , \text{ and}$$

$$H2 = H2 + (WH - WHLK)*DT \quad ,$$

where WSLK and WHLK are the steam and hydrogen leakage rates.

The steam partial pressure is

$$PS = STM/VOL * RS * (GAST + 460)/144, \text{ psia} \quad ,$$

where RS = PSAT/RHOS/(TPOOL + 460), is a steam gas constant evaluated from the steam tables at a saturation temperature TPOOL. The hydrogen partial pressure is

$$PH = H2/VOL*767*(GAST + 460)/144.$$

The total system pressure, PVSL, is the sum of PS and PH.

This algorithm for calculating the primary pressure requires the presence of a gas space (VOL>0). Thus, PRIMP cannot calculate pressures for a water solid system.

3.3 Subroutine EXITQ

EXITQ calculates the temperatures of metal structures and the gas temperatures in the exit gas flow path. A maximum of four structures are modeled. Each structure is a one-node representation of, for example, a gridplate, the upper plenum internals, or a length of primary coolant pipe. New temperatures are calculated only when gas is exiting the primary. New temperatures are not calculated when the system is closed and there is no leakage. The gas temperatures calculated in EXITQ are used to determine the enthalpy flows into the containment building, but are not used in PRIMP to modify the gas space temperature, GAST.

3.4 Subroutine PROPS

Subroutine PROPS calculates saturated steam and water properties. The routine may be called by specifying either the pressure, temperature, or steam density. Properties are determined by interpolation between 16 table values between 0.08554 and 3199 psia. Interpolations are linear or exponential, depending on the property.

3.5 Subroutine ECC

Subroutine ECC calculates pumped and core flood tank ECC injection. The pumped injection may be either 1) a constant value which is reduced to zero if the vessel pressure exceeds a specified shutoff head or 2) calculated from a pump head curve of the form

$$WECC = WMAX \sqrt{1 - \Delta P/PSO} \quad ,$$

where

ΔP = vessel/containment pressure difference, psi

WMAX = injection rate for $\Delta P = 0$, gpm , and

PSO = shutoff head, psi.

The pumps may be started and stopped at specified times. ECC water is taken from a storage tank until the tank reaches a specified level. The pump suction is then switched to the containment sump.

The accumulator injection rate is

$$WACM = WMAX \sqrt{PACM - PVSL} \quad ,$$

where PACM is the accumulator pressure. PACM is assumed to be decreased by a factor of two when the accumulator is emptied. The injected ECC water is added to the primary system water space, and uniform mixing is assumed. MARCH contains no models for removal of steam from the primary system gas space by condensation on the ECC water in the injection section.

3.6 Subroutine MACE

Subroutine MACE calculates the containment building thermal-hydraulics. MACE also controls much of the logic for the engineered safety features. MACE permits subdivision of the containment building into 10 volumes. The volumes must currently be connected in series. A maximum of 200 heat conducting nodes may be distributed between 15 structures. Steam condensation on the walls is calculated using Tagami coefficients. Ice beds, intercompartment fans, BWR suppression pools, containment sprays, and building coolers are modeled. Energy and mass inputs to MACE include the steam and hydrogen generated in the core, concrete decomposition products, volatile fission products, and hydrogen burning.

Hydrogen burning in MACE occurs when the mole fraction hydrogen in a given compartment exceeds 0.04 and the mole fraction oxygen exceeds 0.065. The start of hydrogen burning may be delayed if desired. All of the hydrogen is assumed to be burned once burning starts. The time period of the hydrogen burn is spread over one MARCH time step (generally 0.02 to 1.0 minute).

Containment overpressurization failure is triggered when the pressure exceeds an input value. The containment depressurizes by leakage through a hole whose area is input specified.

4.0 ANALYSIS OF BASE CASE ACCIDENT SEQUENCE

MARCH analyses have been performed for the first 15 hours of the TMI accident. The results of the study indicate MARCH is able to reconstruct a credible simulation of the accident scenario. The MARCH analyses are able to yield reasonable explanations of the observed primary system pressures, the surge line and pressurizer temperatures, the core and hotleg temperatures in the 2 - 4 hour period, cooling of the core, heat dissipation from the primary system, and containment phenomena. The calculations of core heatup and cladding oxidation in the 2 - 4 hour period are found to be very sensitive to modeling assumptions and the ECC injection rate. A number of MARCH calculations are discussed which illustrate this sensitivity. Briefly, the calculated accident scenario indicates partial core uncovery between 1.7 and 3.5 hours. Within the uncertainties in the boundary conditions, variations were made to yield approximately 20 percent core melting (assumed to be 4100 F in the MARCH analyses) for the base case. From examination of the data on radioactivity release to the containment, this value was felt to provide a reasonable upper limit on the fraction of the core that could have melted. It should be recognized, however, that within the uncertainties that exist in the boundary conditions, MARCH could predict very little or no core melting. By maximizing the extent of core melting in the base case, the sensitivity of the core response to the variations in the alternative cases is increased. For this degree of core melting, 15 percent clad oxidation is predicted by MARCH. Additional oxidation, which cannot be analyzed by MARCH, would have occurred in regions of the core that remained steam covered following core recovery. After 3.5 hours, the core remains covered. The bulk water temperature in the vessel is calculated to remain subcooled, and there is no bulk boiling between about 3.5 and 11 hours. Gases are calculated to be released from the primary system into the containment building when the PORV is open and the surge line connection to the hotleg is uncovered in the 3.2 - 3.8 hour, 8.3 - 9.0 hour, and 10.8 - 11.0 hour periods. After 4 hours, the primary system pressure is predicted by MARCH as the simple compression and expansion of the gases trapped in the hotlegs and steam generators.

Changes in gas volume are the result of the varying water levels produced by changing the leakage and ECC injection rates. The primary mechanism for dissipating the core decay heat in the 3.5 - 13 hour period is calculated to be by the flow of ECC water through the open PORV. Significant energy loss also occurs to the letdown/makeup cooler.

The MARCH code input and modeling are discussed below followed by separate discussions of the results in the 0 - 15 hour and 0 - 4 hour periods. A number of MARCH calculations were performed for the 0 - 4 hour period to examine the sensitivity of the results to modeling assumptions.

4.1 MARCH Code Input

Table 4.1 summarizes the input data used in the MARCH base case analysis of the TMI accident.

The ECC injection rates and times in Table 4.1 may be compared with those in Table 2.3. The rates in Table 4.1 are those which were used in MARCH to obtain reasonable agreement with the TMI data. The rates in Table 4.1 are the net makeup rates while those in Table 2.3 are total HPI flows from the BWST. Some of the differences in Table 4.1 and Table 2.3 values may be due to letdown flow. In the MARCH analysis, letdown flow is assumed to be balanced by an equal makeup flow so that there is no net water inventory change through this flow path. The makeup/letdown flow path in MARCH results only in an energy dissipation to the letdown cooler. (See Section 3.1)

Between 4.7 and 200 minutes, the NRC estimates[(10)] that the HPI injection rate is 165 gpm with 140 gpm letdown for a net makeup of 25 gpm. NSAC investigators indicate, however, that there may have been a net letdown flow during this period. The MARCH calculations assume no net makeup between 4.7 and 101 minutes. During the period of core uncovery between 101 and 200 minutes, the base case MARCH analysis uses a net makeup rate to the vessel of 90 gpm. The rationale for using a different value during the core uncovery period is as follows. At 101 minutes when the A-loop primary coolant pumps were stopped, it is assumed the water in the system

TABLE 4.1 MARCH INPUT DATA FOR TMI ACCIDENT

Time, Min.	ECC Injection, gpm
.2 - 4.7	672
4.7 - 101	0
101 - 200	90
200 - 415	565
415 - 555	300 (A)
555 - 800	80 (A)
800 - end	470 (B)

A: includes core flood tank injection
B: shutoff for pressures over 2300 psi

Time, Min.	PORV Leak Area, ft^2
0 - 142	0.00938
142 - 192	0.
192 - 198	0.00938
198 - 221	0.
221 - 318	0.00938
318 - 348	0.
348 - 459	0.005
459 - 550	0.012
550 - 600	0.
600 - 666	0.00938
666 - 756	0.
756 - 780	0.00938
780 - end	0.

Core Power:	2688 MW
Shutdown Power:	10 full-power sec. followed by ANS decay curve for 101 days at full power
Initial Pressure:	2165 psia
Initial Water Temperature:	583°F
Initial Water Inventory:	512,425 lbs.
Initial Steam Volume:	700 ft^3
Initial Steam Generator Water Inventory:	107,400 lbs.
Initial Steam Generator Temperature:	535°F
Weight UO_2:	204,820 lbs.
Weight Zircaloy:	52,500 lbs.
Core Heat Capacity:	53.71 BTU/ft^3 F

TABLE 4.1 Continued

24 Node Axial Peaking Factors, Bottom to Top:	0.55, 0.76, 0.96, 1.09, 1.21, 1.3, 1.3, 1.21, 1.16, 1.1, 1.09, 1.08, 1.1, 1.13, 1.18, 1.27, 1.21, 1.17, 1.11, 1.01, 0.81, 0.55, 0.31, 0.26
10 Node Radial Factors:	1.34, 1.291, 1.249, 1.192, 1.167, 1.083, 1.036, 0.955, 0.864, 0.637
Radial Volume Fractions:	0.00565, 0.0226, 0.0678, 0.0452, 0.113, 0.204, 0.1356, 0.0678, 0.226, 0.113
Core Flood Tank Setpoint:	600 psig
Core Flood Tank Water	124,840 lbs.
ECC Temperature:	100°F
Flood Tank Temperature:	110°F
Steam Generator Emergency Feed:	4150 lbs/min at 70°F (initiated at 8 min)
Makeup/Letdown Flow:	140 gpm until 101 minutes, 112 gpm after 3.5 hours
Pump Power:	16 MW, 0 - 74 minutes 8 MW, 74 - 101 minutes
PORV Capacity:	112,000 lbs/hr saturated steam at 2270 psia
Two-Phase Flow Multiplier:	1.74 psi/psi
Core Melt Temperature:	4130°F
Core Melting Model:	A

fell to the bottom of the steam generators and reactor vessel. The collapsed water level was apparently below the leg elevation. Under these conditions, the water level in the reactor vessel is affected only by the makeup flow and not by the letdown flow. Thus, although the net makeup to the primary system would remain the same, the net makeup to the reactor pressure vessel could potentially be as high as 165 gpm. The listed 90 gpm net makeup value to the reactor vessel between 101 and 200 minutes produces results in reasonable agreement with the available TMI data. As will be discussed in Section 4.3, the results are sensitive to the ECC injection rates in this time period.

Primary coolant pump 2B was turned on at 174 minutes. Although the pump remained on for 19 minutes, the pump flow indication was very brief. Operation of this pump may have had a significant effect on arresting the core heatup. Operation of RCP-2B is simulated in the MARCH calculations by assuming the core is temporarily covered (input option) by a two-phase level swell. The level swell is assumed to last for 10 minutes. Although some water was evidently transferred from the B steam generator to the reactor vessel, this was not simulated in the MARCH analysis.

At 200 minutes, the ECC injection rate was significantly increased. The ECC injection rate after this time was grouped into 4 time periods. The agreement between the values in Table 4.1 and Table 2.3 is reasonable, and differences that do exist can probably be accounted for by letdown flow.

Table 4.1 also lists the effective leak areas used in the base case calculations. The equivalent full-open PORV leak area in MARCH is 0.00938 ft^2. (See Section 3.2). Between 348 and 459 minutes a leak area of 0.005 ft^2 is used to simulate the valve cycling during this interval. This area is somewhat bigger than that indicated in Table 3.1 by a 40% open time. Between 459 and 550 minutes a leak area of 0.012 ft^2 is used to simulate the total leakage through the open PORV and pressurizer vent valve. As a simplifying assumption, the PORV was assumed open during the whole 756 - 780 minute time interval. With the exception of the uncertainties in the 192 - 221 minute time interval, the leakage times in Tables 4.1 and 2.1 are in agreement.

The TMI data indicate the collapsed liquid level in the reactor vessel, following shutdown of the reactor coolant pumps at 101 minutes, was approximately at the top of the top of the core. The amount of water that collapsed into the bottom of the steam generators is not known. About 47,200 pounds of water (at 583°F) is required to fill each steam generator to the hotleg elevation. Thus, 94,400 pounds is an upper limit on how much water could be left in the two loops. In the MARCH calculations it was assumed a total of 47,200 pounds was left in the loops, or about one-half the maximum amount. The pressurizer was assumed to retain 61,500 pounds (95% full). With 47,200 pounds in the bottom of the loops, 61,500 pounds in the pressurizer, and an assumed 50,000 pounds in the bottom head of the reactor vessel, complete core uncovery requires leaking 512,425 - 47,200 - 50,000 - 61,500 = 353,725 pounds (plus any ECC injection). In order to start core uncovery at 101 minutes with about 215,000 pounds remaining in the vessel, it was necessary to assume an essentially liquid blowdown modified by use of a two-phase-flow multiplier of 1.74. (See Section 3.2).

During much of the TMI transient, the tops of the steam generators on both the primary and secondary sides were full of stagnant gases. Little heat transfer is expected within these gas volumes. Condensation of steam from the primary side gas volume can occur if there is relatively cool water on the secondary side which extends up into the primary side steam space. Steam condensation may also occur when the emergency feedwater injects cold water over the tubes at the top of the steam generator. The MARCH modeling of TMI system response between 90 and 135 minutes and between 174 and about 200 minutes assumes steam is being condensed by the steam generators. In the 90 - 135 minute period, it is believed that the emergency feedwater system was being used to feed the A steam generator. This would explain the steam condensation in this period. In the 174 to 200 minute period, the TMI data indicate significant changes in the steam generator levels and pressures. These changes imply significant heat input to the steam generators. In the MARCH calculations it was assumed that the apparent steam generator heat transfer was accompanied by primary system steam condensation. The MARCH input for the base case analysis assumes there is no heat transfer to the steam generators after 200 minutes until primary coolant pump 1A is restarted at 15.8 hours.

4.2 MARCH Results for 0 - 15 Hours

The present section gives an overview of the MARCH calculated results for the course of the TMI accident from the time of initiation until 15 hours. Shortly before 16 hours, primary coolant pump 1A was successfully turned on, and a normal cooling mode for the primary system was established. This report does not consider the accident stages beyond 15 hours. The code input for these base case calculations is listed in Table 4.1. Section 4.3 describes a number of MARCH calculations performed to examine the sensitivity of the MARCH results in the 0 - 4 hour period to several of the modeling and code input assumptions. When possible, the MARCH results are compared with the TMI data.

Figure 4.1 is a comparison of the MARCH calculated primary system pressure with the TMI data. At 6 minutes, the pressure is calculated to drop to 1350 psia. At 8 minutes, the pressure has increased to 1800 psia due to the start of boiling or flashing in the primary system. The start of emergency feedwater at 8 minutes re-establishes good heat transfer to the steam generators, and the pressure decreases below 1300 psia at 15 minutes. The system pressure remains in a plateau at about 1100 psi until the primary coolant pumps are turned off at 101 minutes. Between 90 and 135 minutes in these calculations it is assumed the emergency feedwater to the A steam generator is on. With operation of the emergency feedwater, steam is condensed from the primary gas space, and the primary pressure continues to follow the A steam generator secondary pressure. Thus, until 135 minutes in the MARCH calculations, the primary system pressure is simply following the steam generator secondary conditions. The pressure reaches a minimum value of 680 psia at 135 minutes, which is above the 600 psi setpoint for core flood tank injection, and no injection occurs. With degradation of steam generator heat transfer at 135 minutes, the primary pressure begins to increase due to steam production in the core.

MARCH calculates that the emergency feedwater condenses 42,000 pounds of steam from the primary gas space between 101 and 135 minutes. The condensed steam would fall to the bottom of the A loop steam generator.

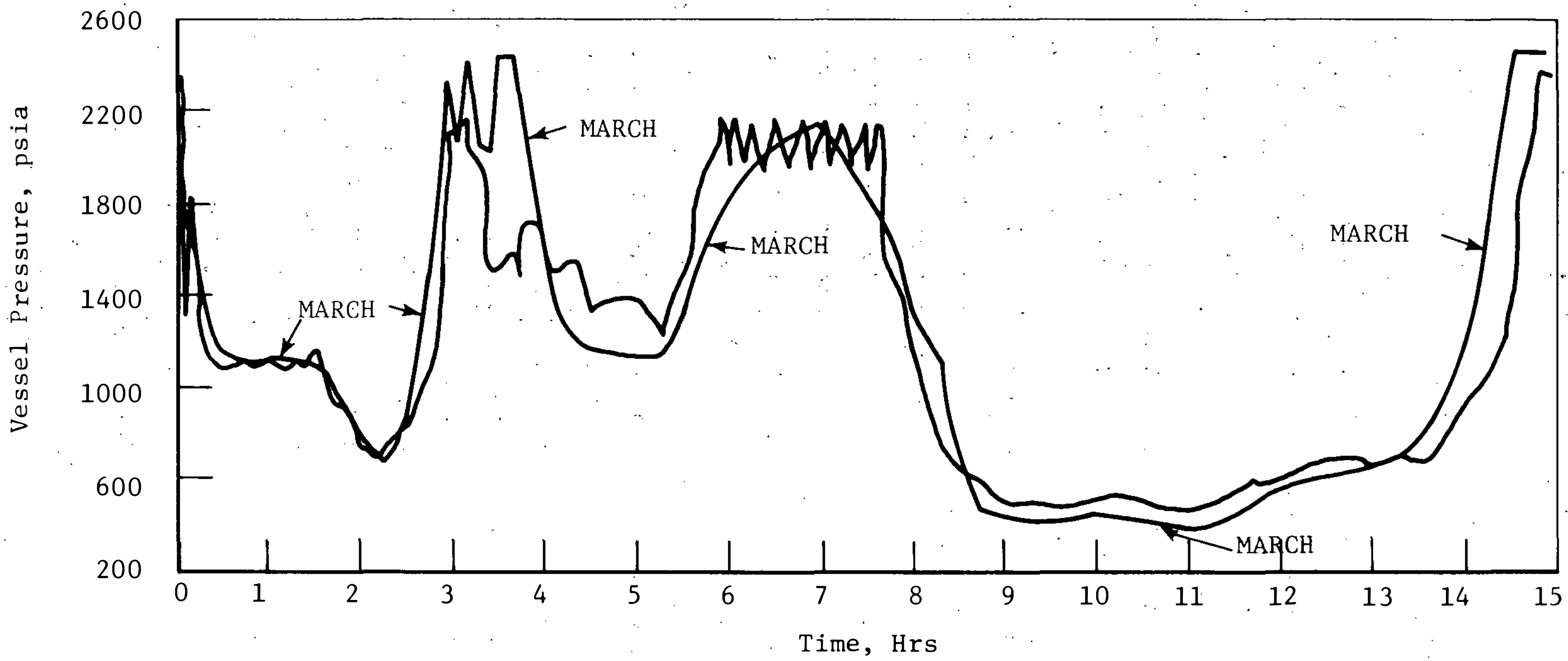

Figure 4.1 Comparison of TMI and Calculated Vessel Pressures, 0-15 Hrs

A letdown flow of 65 gm would be adequate to prevent overflow of this condensed steam into the reactor vessel prior to 135 minutes. It is assumed in the MARCH calculations that there is no overflow of this condensed steam into the reactor vessel.

Figure 4.2 shows the calculated primary coolant leak rate. By 101 minutes MARCH has leaked sufficient primary coolant to begin core uncovery. At this time the primary system contains 215,000 pounds of water of which 60,000 pounds is in the core region of the reactor vessel.

Figures 4.3 and 4.4-A show the primary coolant mixture level. (See Section 3.1 for the definition of the MARCH calculated water level.) At 101 minutes, core uncovery begins. A minimum collapsed liquid level of 5.5 ft is calculated to occur at about 135 minutes. The mixture level in the 101 - 135 minute period decreases at a rate about 50 percent larger than expected from decay heat boiloff alone. This occurs because the steam condensation in the steam generator is depressurizing the primary and causing additional flashing. After 142 minutes, when the steam condensation and PORV leakage are both stopped, the mixture level increases because the ECC injection rate (745 lb/min) exceeds the boiloff rate (approximately 400 lb/min) during this period. Core melting begins at 146 minutes at the 10 ft elevation. At about 160 minutes, the increasing mixture level intercepts the molten region of the core, which has expanded downward to the 8 ft elevation. MARCH calculates a rapid system pressurization (Figure 4.1) from 1050 psia at 155 minutes to over 1900 psia at 170 minutes from the steam produced in quenching the molten fuel. The rate of increase in water level following closing of the PORV line at 142 minutes is probably overestimated by the MARCH code. Because MARCH treats the liquid in the primary system as a single volume, some decay heat is used to maintain the primary system water at saturation during repressurization, which in the real system (in which the water was stratified) produced steam. As a result, MARCH tends to over-predict the extent of core recovery prior to 174 minutes and under-estimate the significance of the operation of reactor coolant pump 2B in cooling the core. The MARCH calculated pressure remains above 2000 psia until the PORV is again opened at 221 minutes.

Figure 4.4 shows the MARCH calculated mixture level during the uncovery period, the maximum core temperature, the fraction core which has melted, the fraction of the core currently molten (above 4130 F), and the

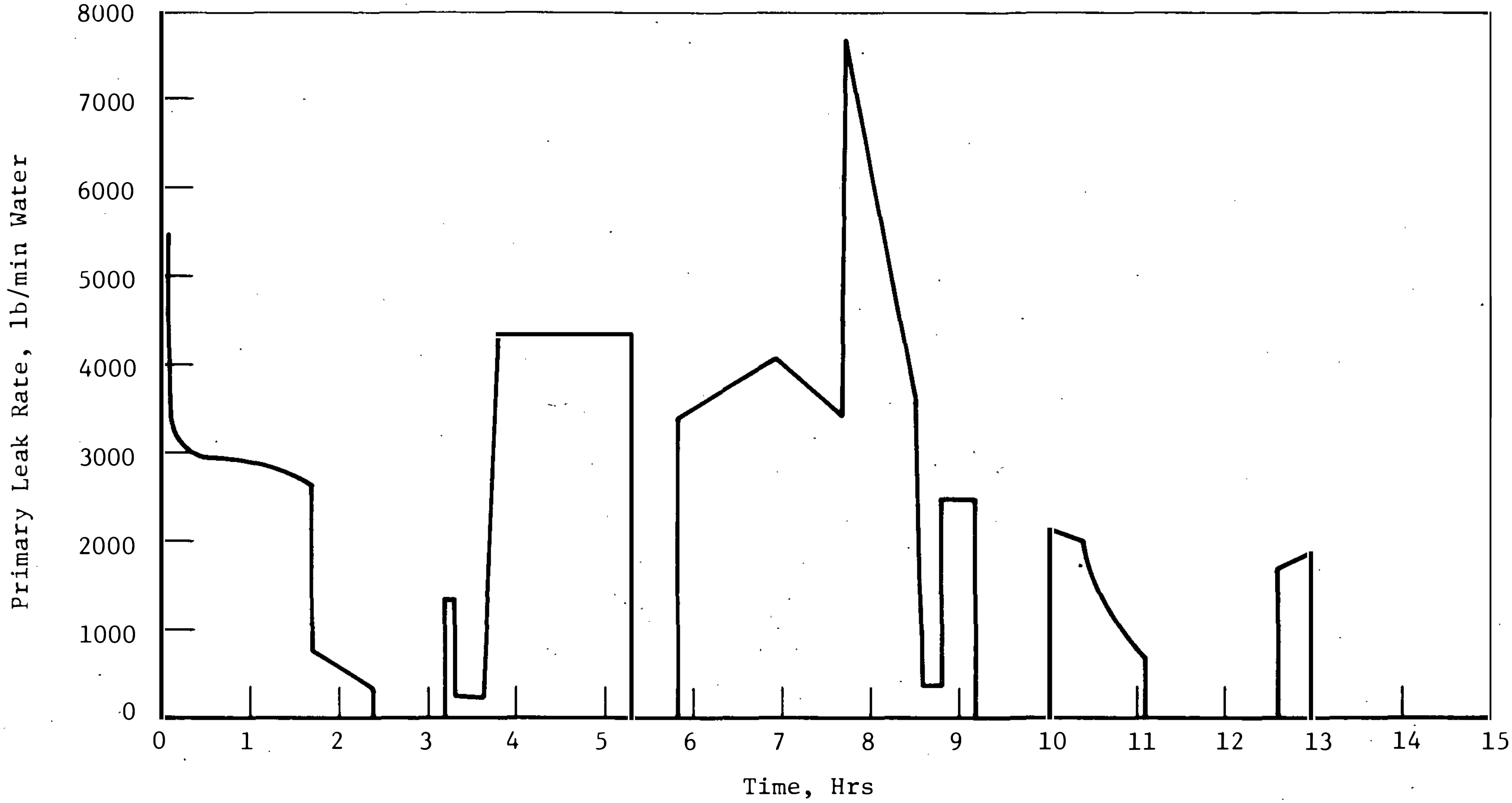

Figure 4.2 Calculated Primary Coolant Leak Rate, 0-15 Hrs

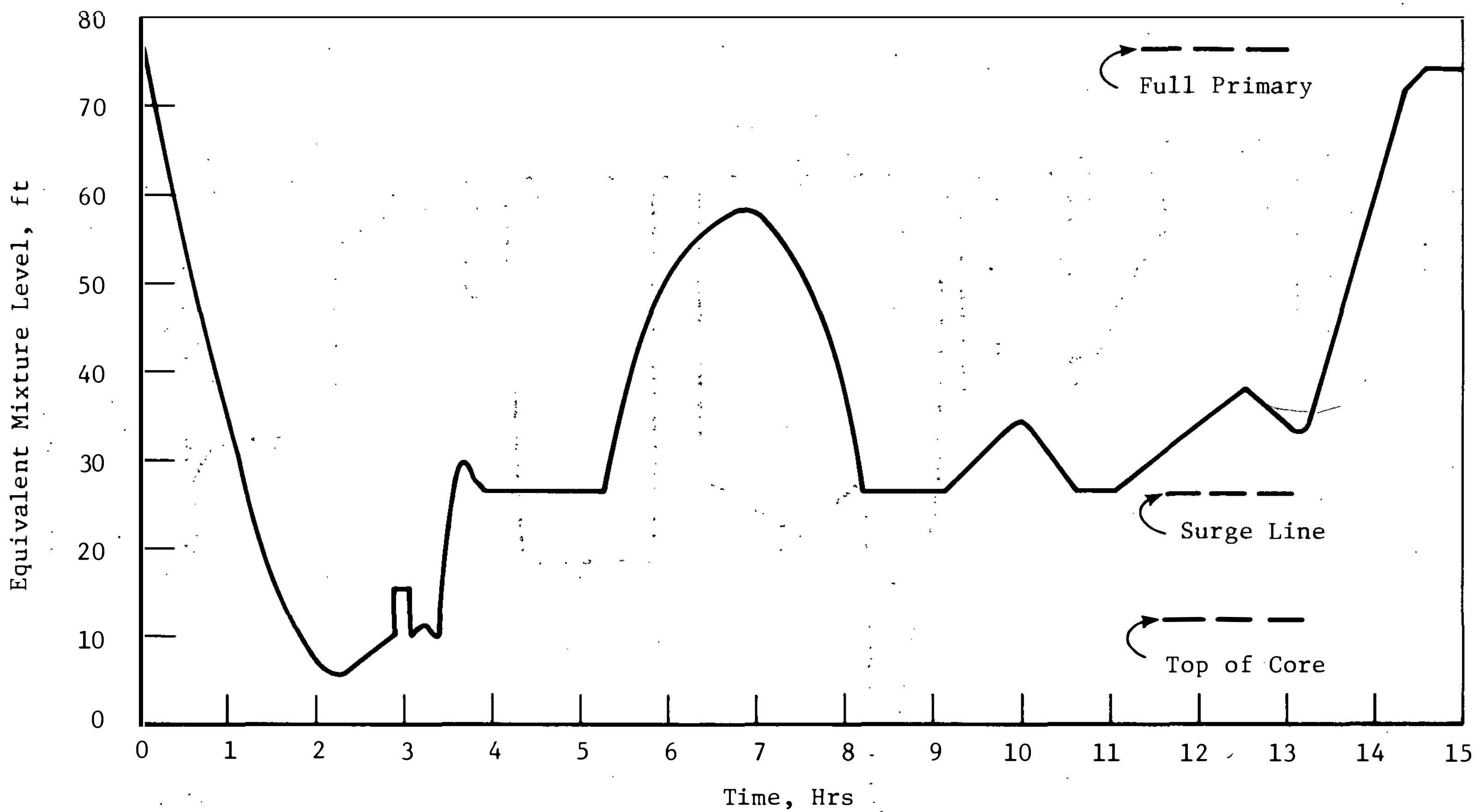

Figure 4.3 Calculated Mixture Level, 0-15 Hrs

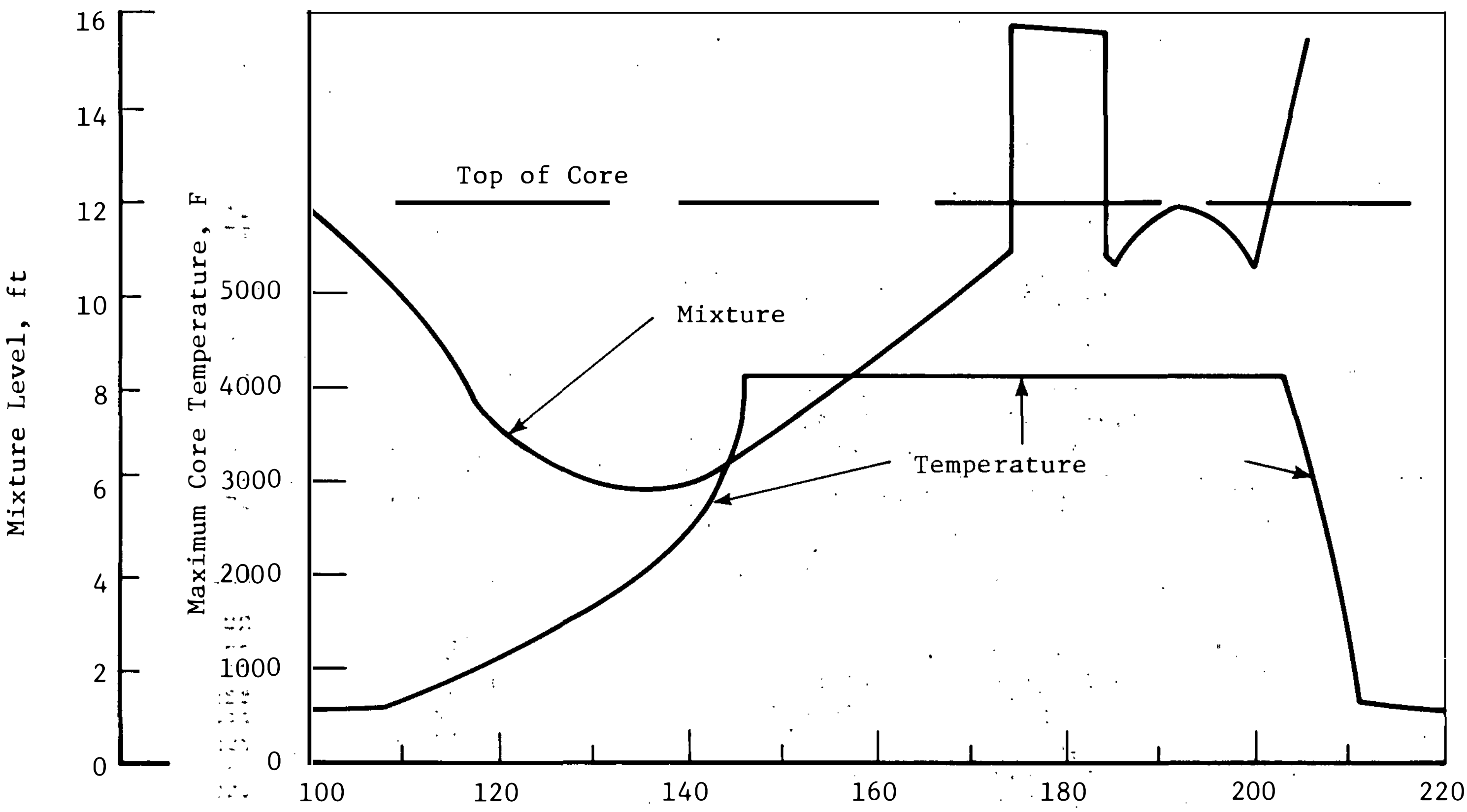

Figure 4.4-A Mixture Level and Maximum Core Temperature

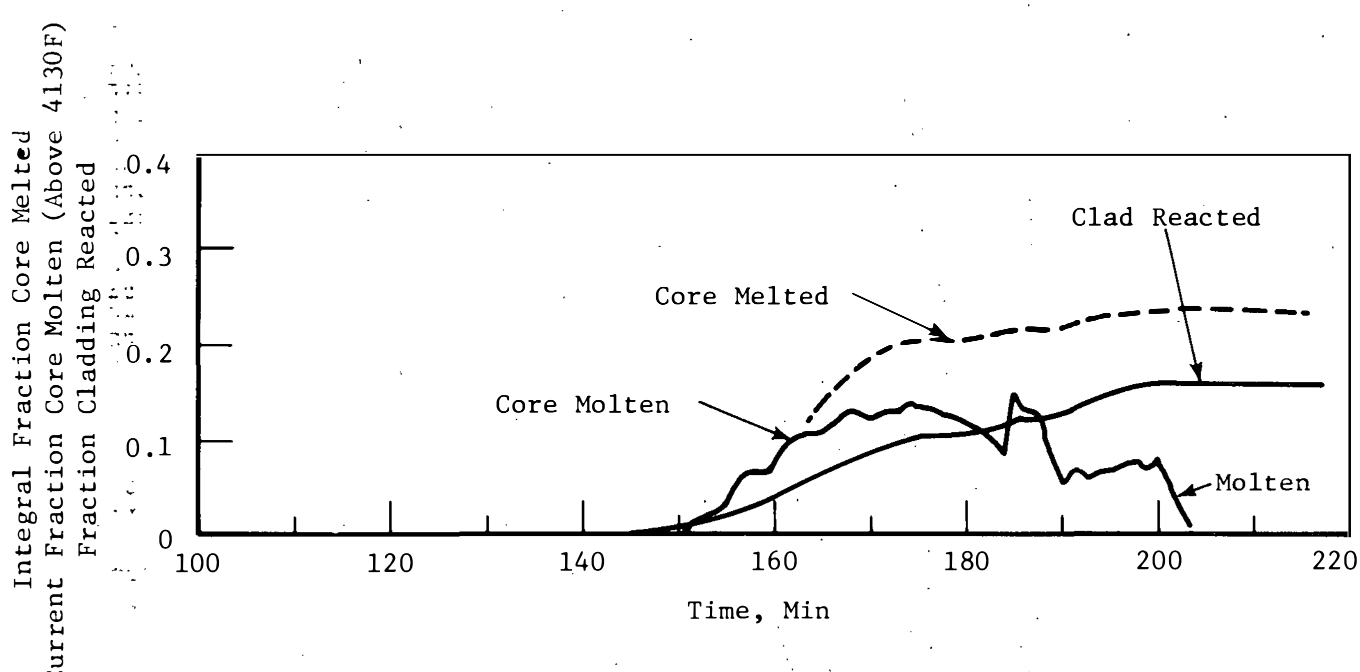

Figure 4.4-B Fraction Core Which Has Melted, Fraction Core Currently Molten (Above 4130°F), and Fraction Cladding Reacted

Figure 4.4 MARCH Calculated Results for Base Case, 100-220 Min

TABLE 4.2 CALCULATED CORE TEMPERATURE MAPS

Time = 140.2 minutes, Mixture Level = 5.92 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	5.098E+02	5.096E+02	5.093E+02	5.090E+02	5.085E+02
2.000E+00	5.114E+02	5.111E+02	5.107E+02	5.102E+02	5.095E+02
3.000E+00	5.125E+02	5.120E+02	5.116E+02	5.110E+02	5.102E+02
4.000E+00	5.120E+02	5.116E+02	5.113E+02	5.107E+02	5.099E+02
5.000E+00	5.115E+02	5.111E+02	5.108E+02	5.103E+02	5.096E+02
6.000E+00	5.295E+02	6.210E+02	6.135E+02	6.015E+02	5.855E+02
7.000E+00	1.507E+03	1.435E+03	1.373E+03	1.275E+03	1.148E+03
8.000E+00	1.986E+03	1.859E+03	1.756E+03	1.602E+03	1.412E+03
9.000E+00	2.230E+03	2.055E+03	1.922E+03	1.734E+03	1.513E+03
1.000E+01	2.371E+03	2.153E+03	1.997E+03	1.789E+03	1.553E+03
1.100E+01	1.908E+03	1.778E+03	1.675E+03	1.527E+03	1.348E+03
1.200E+01	1.546E+03	1.455E+03	1.383E+03	1.277E+03	1.145E+03

Time = 145.2 minutes, Mixture Level = 6.44 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	5.178E+02	5.175E+02	5.173E+02	5.169E+02	5.165E+02
2.000E+00	5.194E+02	5.190E+02	5.187E+02	5.182E+02	5.175E+02
3.000E+00	5.204E+02	5.200E+02	5.196E+02	5.190E+02	5.182E+02
4.000E+00	5.200E+02	5.196E+02	5.192E+02	5.186E+02	5.179E+02
5.000E+00	5.194E+02	5.191E+02	5.187E+02	5.182E+02	5.175E+02
6.000E+00	5.193E+02	5.190E+02	5.187E+02	5.181E+02	5.175E+02
7.000E+00	1.831E+03	1.731E+03	1.645E+03	1.513E+03	1.344E+03
8.000E+00	2.490E+03	2.281E+03	2.124E+03	1.905E+03	1.651E+03
9.000E+00	2.932E+03	2.558E+03	2.330E+03	2.050E+03	1.752E+03
1.000E+01	3.282E+03	2.693E+03	2.406E+03	2.089E+03	1.773E+03
1.100E+01	2.208E+03	2.030E+03	1.890E+03	1.700E+03	1.483E+03
1.200E+01	1.705E+03	1.586E+03	1.495E+03	1.371E+03	1.218E+03

Time = 150.2 minutes, Mixture Level = 7.11 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	5.344E+02	5.342E+02	5.340E+02	5.336E+02	5.331E+02
2.000E+00	5.360E+02	5.357E+02	5.353E+02	5.348E+02	5.342E+02
3.000E+00	5.370E+02	5.366E+02	5.362E+02	5.356E+02	5.348E+02
4.000E+00	5.366E+02	5.362E+02	5.359E+02	5.353E+02	5.345E+02
5.000E+00	5.361E+02	5.357E+02	5.354E+02	5.349E+02	5.342E+02
6.000E+00	5.360E+02	5.356E+02	5.353E+02	5.348E+02	5.341E+02
7.000E+00	1.085E+03	9.993E+02	9.308E+02	8.449E+02	7.664E+02
8.000E+00	3.118E+03	2.782E+03	2.554E+03	2.255E+03	1.925E+03
9.000E+00	4.000E+03	3.159E+03	2.788E+03	2.393E+03	2.010E+03
1.000E+01	4.130E+03	3.346E+03	2.849E+03	2.402E+03	2.004E+03
1.100E+01	2.559E+03	2.302E+03	2.114E+03	1.878E+03	1.622E+03
1.200E+01	1.928E+03	1.747E+03	1.620E+03	1.470E+03	1.294E+03

Time = 165.2 minutes, Mixture Level = 9.71 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	6.044E+02	6.041E+02	6.039E+02	6.036E+02	6.031E+02
2.000E+00	6.059E+02	6.056E+02	6.052E+02	6.047E+02	6.041E+02
3.000E+00	6.069E+02	6.065E+02	6.061E+02	6.055E+02	6.047E+02
4.000E+00	6.065E+02	6.061E+02	6.057E+02	6.052E+02	6.045E+02
5.000E+00	6.060E+02	6.056E+02	6.053E+02	6.048E+02	6.041E+02
6.000E+00	6.059E+02	6.055E+02	6.052E+02	6.047E+02	6.041E+02
7.000E+00	6.061E+02	6.057E+02	6.054E+02	6.049E+02	6.042E+02
8.000E+00	1.515E+03	7.670E+02	6.060E+02	6.054E+02	6.046E+02
9.000E+00	4.130E+03	4.130E+03	4.130E+03	4.130E+03	1.638E+03
1.000E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	3.349E+03
1.100E+01	4.130E+03	4.130E+03	4.130E+03	3.511E+03	2.282E+03
1.200E+01	4.130E+03	3.908E+03	3.084E+03	2.443E+03	1.774E+03

TABLE 4.2 CONTINUED

Time = 174.2 minutes, Mixture Level = 15.8 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	6.546E+02	6.544E+02	6.542E+02	6.538E+02	6.534E+02
2.000E+00	6.561E+02	6.558E+02	6.555E+02	6.550E+02	6.544E+02
3.000E+00	6.571E+02	6.567E+02	6.563E+02	6.557E+02	6.550E+02
4.000E+00	6.567E+02	6.563E+02	6.560E+02	6.554E+02	6.547E+02
5.000E+00	6.562E+02	6.558E+02	6.555E+02	6.550E+02	6.544E+02
6.000E+00	6.561E+02	6.558E+02	6.554E+02	6.550E+02	6.543E+02
7.000E+00	6.563E+02	6.560E+02	6.556E+02	6.551E+02	6.545E+02
8.000E+00	6.570E+02	6.566E+02	6.562E+02	6.556E+02	6.549E+02
9.000E+00	1.450E+03	1.840E+03	1.109E+03	1.134E+03	6.545E+02
1.000E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	3.949E+03
1.100E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	3.718E+03
1.200E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	2.090E+03

Time = 180.4 minutes, Mixture Level = 15.8 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	6.490E+02	6.488E+02	6.486E+02	6.482E+02	6.478E+02
2.000E+00	6.505E+02	6.502E+02	6.499E+02	6.494E+02	6.488E+02
3.000E+00	6.515E+02	6.511E+02	6.507E+02	6.501E+02	6.494E+02
4.000E+00	6.511E+02	6.507E+02	6.504E+02	6.498E+02	6.491E+02
5.000E+00	6.506E+02	6.502E+02	6.499E+02	6.494E+02	6.488E+02
6.000E+00	6.505E+02	6.501E+02	6.498E+02	6.494E+02	6.487E+02
7.000E+00	6.507E+02	6.504E+02	6.500E+02	6.495E+02	6.489E+02
8.000E+00	6.513E+02	6.509E+02	6.506E+02	6.500E+02	6.493E+02
9.000E+00	3.047E+02	1.295E+03	6.585E+02	6.613E+02	6.490E+02
1.000E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	3.586E+03
1.100E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	3.536E+03
1.200E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	2.090E+03

Time = 185.4 minutes, Mixture Level = 10.7 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	6.479E+02	6.477E+02	6.475E+02	6.471E+02	6.467E+02
2.000E+00	6.494E+02	6.491E+02	6.488E+02	6.483E+02	6.477E+02
3.000E+00	6.504E+02	6.500E+02	6.496E+02	6.490E+02	6.483E+02
4.000E+00	6.500E+02	6.496E+02	6.492E+02	6.487E+02	6.480E+02
5.000E+00	6.495E+02	6.491E+02	6.488E+02	6.483E+02	6.477E+02
6.000E+00	6.494E+02	6.490E+02	6.487E+02	6.483E+02	6.476E+02
7.000E+00	6.496E+02	6.492E+02	6.489E+02	6.484E+02	6.478E+02
8.000E+00	6.502E+02	6.498E+02	6.495E+02	6.489E+02	6.482E+02
9.000E+00	6.502E+02	6.885E+02	6.491E+02	6.486E+02	6.479E+02
1.000E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	2.987E+03
1.100E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	4.130E+03
1.200E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	2.207E+03

Time = 195.6 minutes, Mixture Level = 11.6 ft

Z	TRO(1)	TRO(3)	TRO(5)	TRO(7)	TRO(9)
1.000E+00	6.560E+02	6.558E+02	6.556E+02	6.553E+02	6.548E+02
2.000E+00	6.575E+02	6.572E+02	6.569E+02	6.564E+02	6.558E+02
3.000E+00	6.584E+02	6.580E+02	6.577E+02	6.571E+02	6.564E+02
4.000E+00	6.580E+02	6.576E+02	6.573E+02	6.568E+02	6.561E+02
5.000E+00	6.575E+02	6.572E+02	6.569E+02	6.564E+02	6.558E+02
6.000E+00	6.574E+02	6.571E+02	6.568E+02	6.564E+02	6.557E+02
7.000E+00	6.577E+02	6.573E+02	6.570E+02	6.565E+02	6.559E+02
8.000E+00	6.583E+02	6.579E+02	6.576E+02	6.570E+02	6.563E+02
9.000E+00	6.578E+02	6.575E+02	6.572E+02	6.567E+02	6.560E+02
1.000E+01	1.933E+03	1.933E+03	1.933E+03	1.933E+03	6.623E+02
1.100E+01	3.654E+03	3.661E+03	3.705E+03	3.547E+03	2.839E+03
1.200E+01	4.130E+03	4.130E+03	4.130E+03	4.130E+03	4.130E+03

fraction cladding reacted. Table 4.2 shows core temperature maps during the calculated core uncovery period. Just prior to closure of the PORV block valve at 142 minutes, the MARCH calculations indicate peak fuel rod temperatures in excess of 1400 F for nearly the whole core. Thus, cladding failures and release of gaseous fission products from the cladding gap and into the containment building would be expected prior to closure of the PORV. This is in agreement with measurements of containment radiation levels which began to increase at about 134 minutes. The self powered neutron detectors were also indicating core temperatures in excess of 1000 F in the top half of the core prior to PORV closure.

As seen in Figure 4.4, core melting (liquid eutectic formation) is predicted to begin at 146 minutes. Less than one percent of the cladding has oxidized at this time. MARCH calculated little hydrogen (approximately 1.7 lb) is released to the containment prior to PORV closure at 142 minutes. The fraction of the core above the 4130 F melting temperature increases to 0.10 at 162 minutes. After 162 minutes, portions of the molten core are quenched by the rising mixture level and new regions begin to melt. Between 174 and 184 minutes, operation of RCP-2B is assumed to cover the core with a two-phase steam-water mixture. This is simulated in the MARCH calculations by artificially imposing (by MARCH input) a level swell for 10 min. No water was assumed to be added to the vessel by RCP-2B. During the 10 min. period of operation of RCP-2B, the core melt fraction is calculated to decrease from 0.14 to 0.085. The solidified nodes quickly remelt, however, when the mixture level collapses after 184 min. Continued makeup at 90 gpm to the vessel eventually increases the mixture level to the top of the core. Fuel quenching is predicted to occur rapidly after initiation of HPI at a rate of 565 gpm at 200 min.

Figure 4.4-B indicates 15.4 percent of the cladding reacted during the core uncovery period. It was noted earlier that MARCH assumes no metal-water reaction in the mixture covered regions following core recovery.

In the MARCH calculations for the time period between 174 and 202 minutes, it was assumed that steam was being condensed from the primary system gas space. Without condensation in the steam generators, MARCH would predict pressures high enough to lift the safety relief valves. During this time period, the molten core regions are beginning to be recovered and cooled in the MARCH analyses. The fraction of the core above 4,130 F (the assumed melting temperature) decreases from 0.14 at 174 min to 0.08 at 200 min. The change in the core stored heat over this period is equivalent to about 80 percent of the core decay heat. About 64,000 lb of steam is produced in this cooling process. The amount of steam generation is very sensitive to the extent of core overheating, vessel water level, and modelling of the cooling process. It was assumed that the condensed steam after falling to the bottom of the steam generators was returned (refluxed) to the reactor vessel by overflowing through the cold legs. The observed increase of the B steam generator secondary pressure indicates that heat transfer was occurring in this period.

Figure 4.5 compares the TMI hot leg temperatures with the MARCH calculated gas space temperature and hot leg metal temperature. The MARCH gas space temperature increases rapidly after core uncovery starts at 101 minutes. The gas space temperature is calculated to reach a peak of 1520 F at 173 minutes and decrease to 940 F by 200 minutes as cooler steam boils out of the nearly recovered core. Hot leg metal temperatures are calculated to reach a temperature of 680 F by an accident time of 4.5 hours. The TMI hot leg temperatures generally fall between the MARCH gas space and hot leg metal temperatures. (The limitations of the MARCH modeling of the gas space and hot leg metal temperatures are discussed in Section 3.3.)

From Figure 4.1, it is seen that the calculated system pressure remains above 2100 psia after 174 minutes until the PORV is reopened at 221 minutes. The TMI data show a sharp decrease in pressure at 200 minutes in apparent coincidence with the initiation of high HPI flow. The sharp pressure decrease at 200 minutes is believed to be due to condensation of stean in the TMI cold legs. The MARCH calculated water level at 205 minutes is about 15 ft and the cold legs are predicted to be empty. Steam in the upper plenum and hot legs could, therefore, flow through the core barrel vent valves to be condensed on the cold ECC water. This effect is not modelled in MARCH.

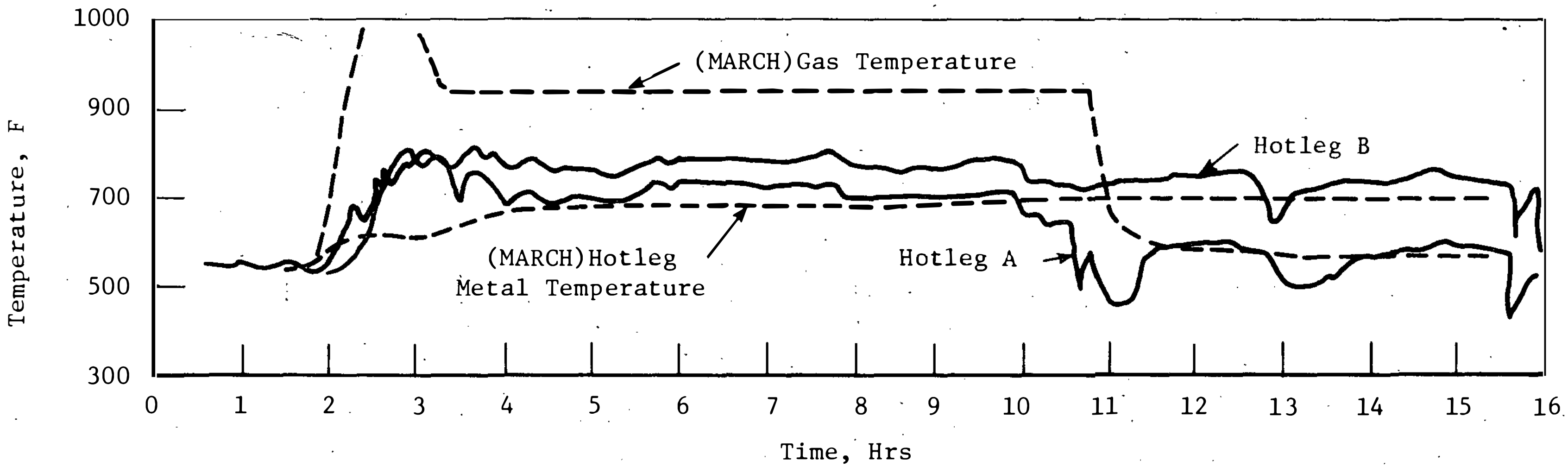

Figure 4.5 Comparison of TMI Hotleg Temperatures with Calculated Temperatures, 0-16 Hrs

Following the 220 minute period, the MARCH calculated system pressure in Figure 4.1 shows reasonable agreement with the TMI data. The pressure in the 4 - 14 hour period is generally responding to the simple compression and expansion of the gas space above the rising and falling water level. The water levels are shown in Figure 4.3. The assumed ECC injection rates and the PORV opening and closing times are listed in Section 4.1. In the MARCH calculation for the base case there is no steam generator heat transfer after 202 minutes. Figure 4.6 compares the integrated decay heat with the heat loss to the steam generators for the 15 hour period.

The MARCH calculation assumes significant heat loss to the makeup/letdown cooler. This heat loss occurs through the assumed continuous operation of a makeup/letdown flow of 112 gpm after 3.5 hours. Figure 4.7 compares the decay heat with the heat loss to the letdown cooler. The heat loss to the letdown cooler generally ranges between 20 and 50 percent of the decay heat in the MARCH calculation. The modeling of the heat loss to the letdown cooler probably results in an over-estimation. This is because it is assumed in MARCH that the temperature of the letdown flow is the bulk water temperature in the primary system. Since the letdown flow comes from a cold leg location, it was actually cooler than assumed in much of the accident.

The majority of the decay heat generated in the 4 - 15 hour period was removed in the MARCH calculation by the flow of ECC water through the core and out the PORV. Evidence that this was the primary flow path for heat removal is the agreement shown in Figure 4.8 between the TMI surge line/pressurizer temperature and the MARCH calculated bulk water temperature. Following the substantial decrease in ECC flow rate at 9 1/4 hours, the water in the primary system is predicted to heat up. At 11 hours in the TMI accident, bulk boiling started in the primary system. This is evidenced by the convergence of the saturation and pressurizer temperatures.

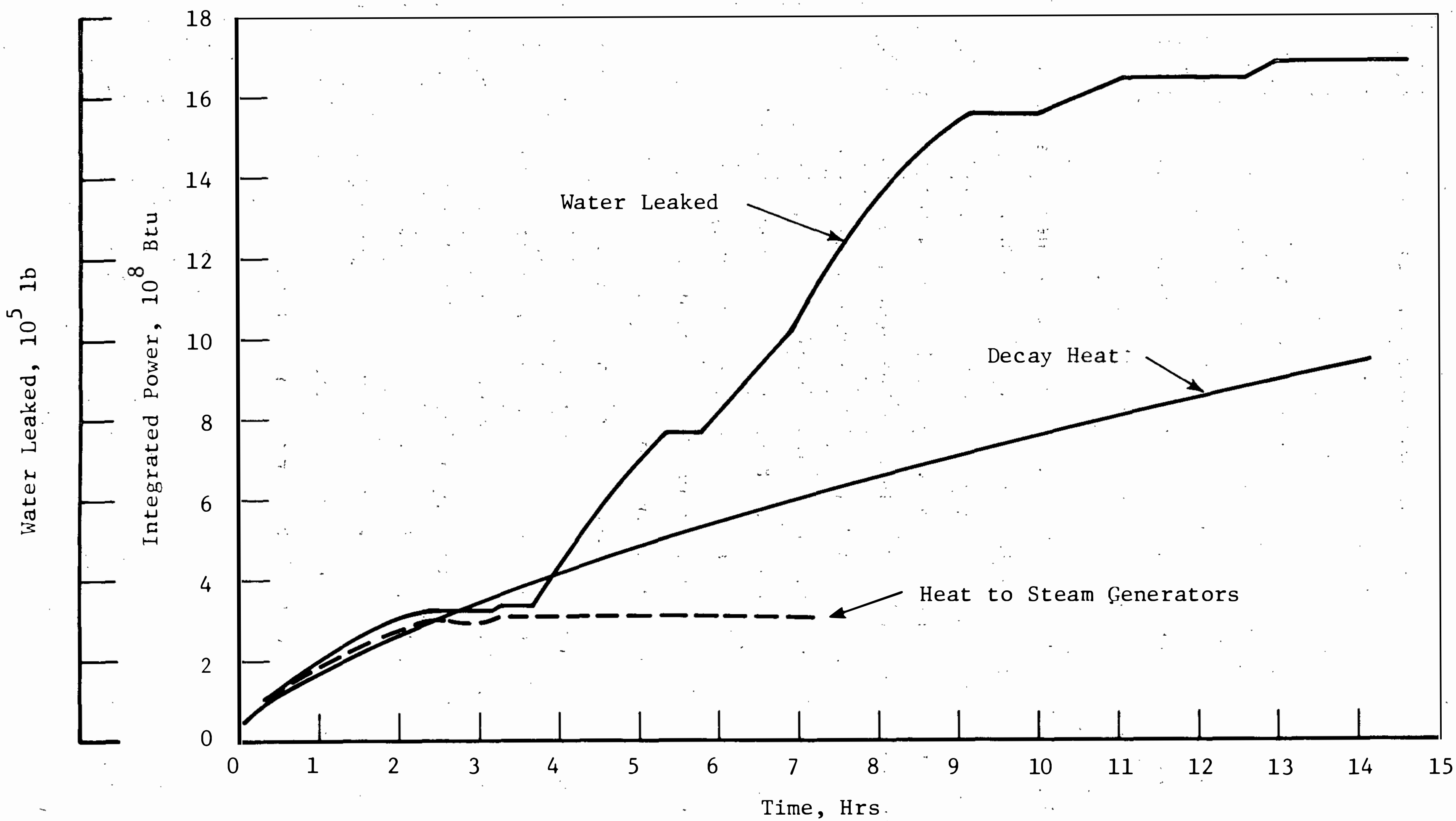

Figure 4.6 Calculated Heat Loss to Steam Generators and the Total Water Leakage to the Containment, 0-15 Hrs

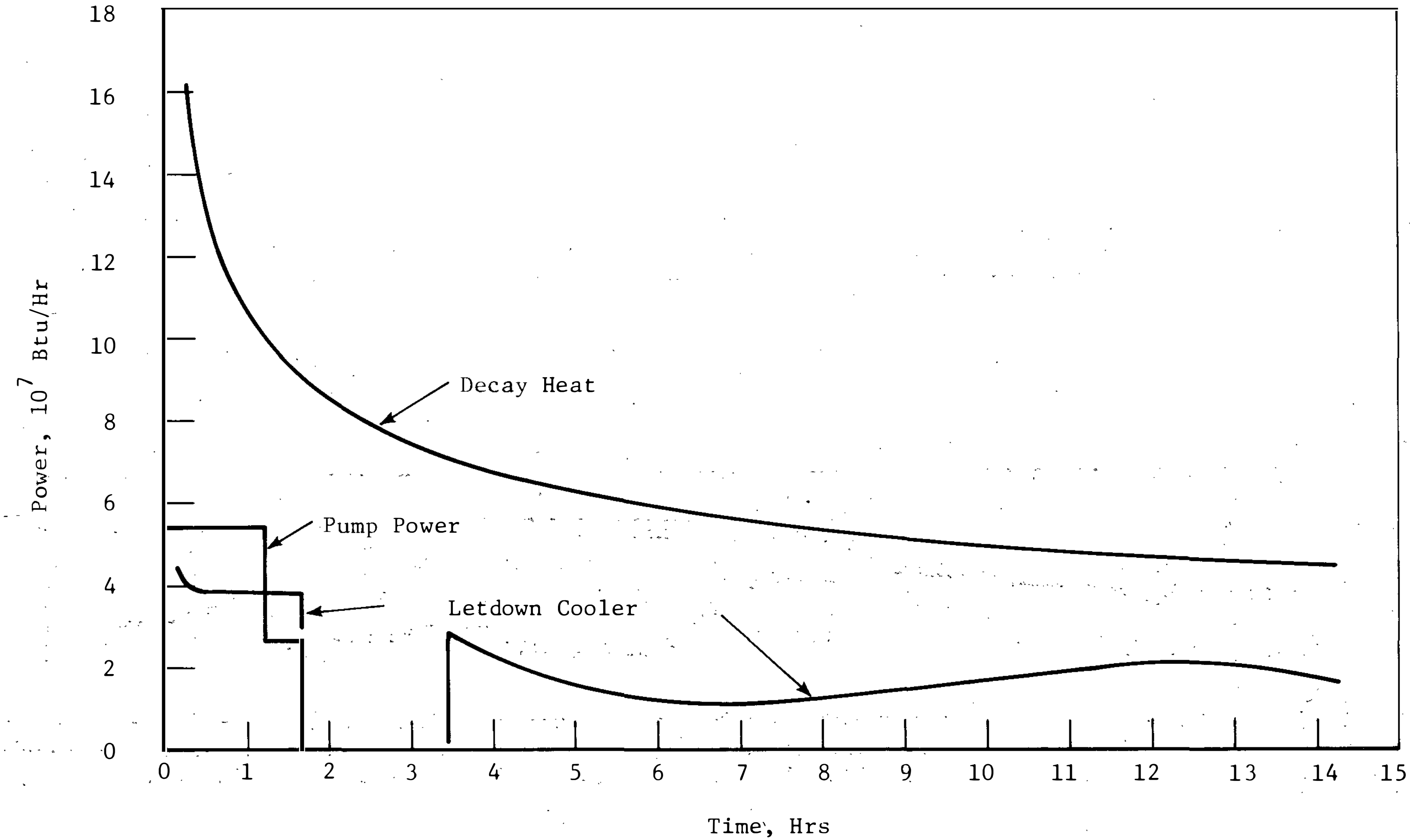

Figure 4.7 Comparison of Decay Heat, Calculated Heat to Letdown Cooler, and Primary Coolant Pump Power, 0-15 Hrs

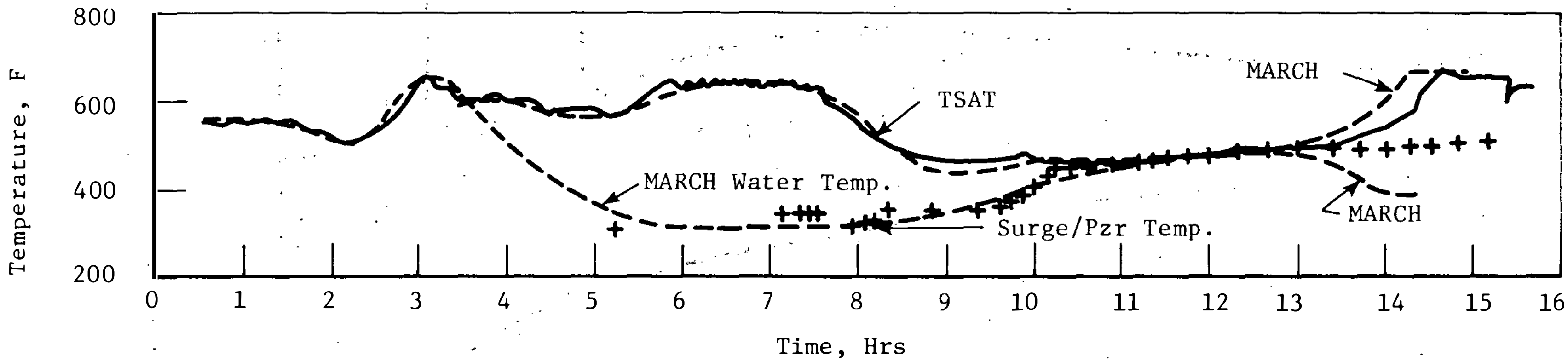

Figure 4.8 Comparison of TMI and MARCH Calculated Saturation and Surge Line/Pressurizer Temperatures, 0-16 Hrs

During the core uncovery between 101 and 210 minutes, MARCH calculates that 15.4% of the cladding reacted, and 353 pounds of hydrogen were produced. During the course of the accident, some of this hydrogen was vented from the primary system into the containment building. Figure 4.9 shows the distribution of hydrogen between the primary and containment as predicted by MARCH. Hydrogen is released in the MARCH calculations when the PORV is open and the surge-line connection to the hot leg is uncovered. MARCH predicts that 74 percent of the hydrogen is released before 4.5 hours. An additional 18 percent is released at 8.3 hours, and 5 percent at 10 hours. These projections assume all of the cladding reaction occurs in the 101 to 200 minute time interval as in the MARCH calculations. In reality, hydrogen would have been produced in the damaged core after the MARCH models predict the core is recovered and the fuel quenched.

4.3 Sensitivity Analyses for the Time Period 0 - 4 Hours

Several MARCH calculations were performed to examine the sensitivity of the results to modeling and input assumption during the first few hours of the TMI accident. These calculations were directed principally at an assessment of the core damage predictions. The extent of core damage is found to be particularly sensitive to the ECC makeup rates during core uncovery. A smaller sensitivity is seen to variations in the core melting point and meltdown model assumptions. One calculation was performed to examine the sensitivity of the results to changes in the steam generator heat transfer in the 90 - 135 minute time interval. Results are also presented for cases in which the heat transfer coefficient to steam is modified by incorporation of a radiation heat transfer model.

Makeup Rate

The net ECC makeup rates during the core uncovery period are listed in Table 4.1 for the base case calculation. Table 4.3 lists the

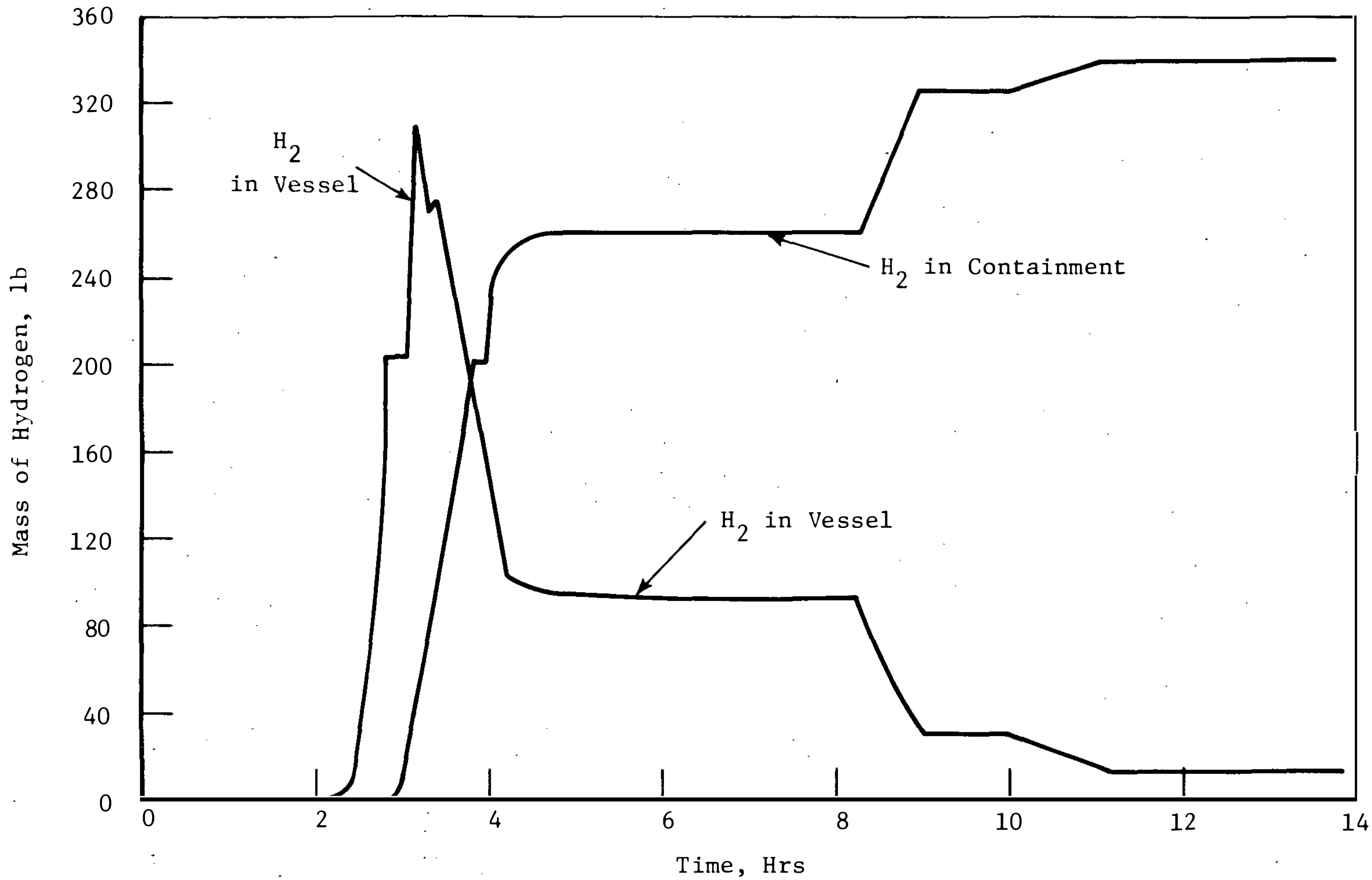

Figure 4.9 Calculated Distribution of Hydrogen Produced in Initial Core Uncovery Between Containment and Vessel, 0-14 Hrs

fraction core melted and the fraction cladding reacted for several variations of the ECC makeup in the 101 - 200 minute time interval. The MARCH results indicate changing the makeup rate by ±20 gpm during the uncovery period can make the difference between essentially a no-melt, no-cladding reaction prediction to a near doubling of the base case predictions. The uncertainty in the makeup flow rates during this period of the TMI accident is greater than ±20 gpm.

TABLE 4.3. EFFECT OF MAKEUP RATE

Case	Net Makeup*, gpm	Fraction Core Melted	Fraction Clad Reacted
low ECC	70	0.42	0.24
base case	90	0.24	0.15
high ECC	110	0.012	0.017

*Makeup rates during the 101 - 200 minute time interval.

Melting Assumptions

Predictions of core melting and cladding reaction might be expected to be sensitive to the BOIL meltdown model, the core melting temperature, and the degree of steam penetration or blockage of damaged core regions. See Section 3.1 for a discussion of the BOIL meltdown models. Table 4.4 lists MARCH results assessing the effects of these parameters. None of the parametric variations in Table 4.4 affect the results as significantly as the makeup rate to the vessel. The sensitivities indicated in Table 4.4 are particular to the TMI scenario. For other scenarios much greater sensitivities can occur.

TABLE 4.4. EFFECT OF MELTING ASSUMPTIONS

Case	TMELT, F	Meltdown Model	Reaction in Melted Nodes	Fraction Core Melted	Fraction Clad Reacted
Base	4130	A	no	0.24	0.15
High TMELT	5000	A	no	0.19	0.14
Low TMELT	3500	A	no	0.22	0.09
Model B	4130	B	no	0.19	0.12
No Blockage	4130	A	yes	0.20	0.16

Water Level and Steam Generator Heat Transfer

A MARCH calculation was also performed for a case in which no steam generator heat transfer after 101 minutes and a much lower water level than in the base case were assumed. The water level was allowed to drop to 2 feet at 120 minutes by leaking more fluid out the PORV. In the base case, the system pressure began to decrease at about 100 minutes to a minimum of 680 psi at 135 minutes. The variation case shows a pressure decrease to 920 psia at 140 minutes. The pressure decrease is obtained because of the small boiloff rate resulting from the very low water level. At 146 minutes for the present case, 7.5 percent of the core is molten with melting occurring at the 3 and 7-9 ft elevations. MARCH predicts rapid core slumping and essentially complete meltdown follows shortly later. It is concluded that the case described here is not consistent with the observed TMI behavior. Thus, good steam generator heat transfer probably continued beyond the 101 minute period of the TMI accident, and water levels could not have fallen as low as the 2 foot level for an extended period of time without substantially more core damage occurring.

Radiation to Steam

Marino[11] has pointed out the importance of radiation heat transfer to steam during the core uncovery period. MARCH calculations discussed elsewhere in this report considered only convective cooling of rods and did not include a model for calculating radiation heat transfer from the fuel rods to steam. In order to determine the magnitude of this effect, a model was incorporated into subroutine BOIL to calculate rod-to-steam radiation heat transfer. The BOIL radiation heat transfer coefficient is[12]

$$\mathrm{HRAD} = 17.3 \times 10^{-10}(\varepsilon_{SR}T_R^{\,4} - \varepsilon_{SS}T_S^{\,4})/(T_R - T_S) * (1. + \varepsilon_R)/2 \quad ,$$

Btu/hr ft^2 F

where ε_{SR} = steam emissivity at temperature T_R

ε_{SS} = steam emissivity at temperature T_S

ε_R = fuel rod emissivity

T_R = rod temperature, R

T_S = steam temperature, R.

For comparison, Marino used the RELAP correlation

$$\mathrm{HRAD} = 17.3 \times 10^{-10}\ \varepsilon(T_R^{\,4} - T_S^{\,4})/(T_R - T_S)$$

with a constant emissivity $\varepsilon \sim 0.23$.

The steam emissivities in the BOIL model are evaluated at an optical thickness, PD_H, where P is the local partial pressure of steam in the flow channel and D_H is the hydraulic diameter. At low pressures, such as would occur in a large pipe break LOCA, the results of McAdams[13] indicate the optical thickness and the steam emissivities are small, and radiation heat transfer to steam can be neglected. Likewise, if the gas in the flow channel is all hydrogen, radiation heat transfer is negligible. For high system pressures, such as those prevailing in the TMI accident, experimental data on steam emissivities could not be found. However, if the low pressure data of McAdams are extrapolated to high pressures, large steam emissivities are predicted. In this case, the radiation heat transfer coefficients may be an order of magnitude larger than the convective coefficients.

Table 4.4 shows the effect on the core damage predictions of including radiation heat transfer to steam in the MARCH calculations. For the base case makeup rate during core uncovery of 90 gpm, the inclusion of radiation heat transfer reduces the MARCH predicted core melt fraction from 0.24 to 0.05.

TABLE 4.5 EFFECT OF ROD-TO-STEAM RADIATION HEAT TRANSFER

Case	Radiation Heat Transfer	Net Makeup, gpm	Fraction Core Melted	Fraction Clad Reacted	Min. Mixture Level, ft
Base ECC	no	90	0.24	0.15	5.8
Base ECC	yes	90	0.05	0.04	5.6
Low ECC	no	70	0.42	0.15	-
Low ECC	yes	70	0.22	0.14	4.9

For a 70 gpm makeup rate, the predicted core melt fraction is reduced from 0.42 to 0.22. Thus, inclusion of radiation heat transfer to steam significantly reduces the predicted core damage. As before, the calculations exhibit a large sensitivity to the makeup rate. If radiation heat transfer is considered, it is apparent the "base case" core damage prediction requires reducing the corresponding makeup rate during core uncovery from 90 gpm to 70 gpm. According to Table 4.5, such a change would be accompanied by a reduction in the predicted minimum coolant mixture level from 5.8 to 4.9 ft, which is in better agreement with most interpretations of the TMI SPND data.

5.0 MARCH ANALYSES OF ALTERNATIVE ACCIDENT SEQUENCES

MARCH analyses were performed for ten alternative accident sequences. These alternatives sequences involve variations in ECC flow, delay in initiation of emergency feedwater, timing for the closure of the PORV block valve, operation of the primary coolant pumps, and loss of all electric power. The ten alternative cases are:

1) The high pressure injection is run continuously in the ECC injection mode rather than throttled back.
2) The high pressure injection is run continuously in the ECC injection mode, but emergency feedwater flow is not initiated until 60 minutes.
3) Emergency feedwater is delivered at 40 seconds rather than being delayed until 8 minutes.
4) Delivery of emergency feedwater is delayed until 60 minutes.
5) The PORV block valve is closed at 25 minutes.
6) The PORV block valve is not closed until 3.3 hours.
7) All primary coolant pumps are stopped concurrently with reactor trip.
8) All electric power is lost sometime between 0.5 and 5 hours.
9) The PORV block valve remains closed after 142 minutes and HPI is not initiated at 200 minutes.
10) High pressure injection is not initiated at 200 minutes.

The MARCH calculations are performed assuming all factors, other than those directly associated with the alternative condition, are the same as in the base case. The base case calculation is discussed in Section 4.0. The calculations for the alternative accident sequences were stopped after the course of the accident was clearly established. For some of the sequences the course and timing of the accident is significantly different from the base case. For most of the alternatives modeling of the accident with MARCH was straight-forward. However, for some cases (possibly cases 4, 6, and 7) the nature of the flow through the open PORV, the operability of RCP-2B at 174 minutes, and the steam generator heat transfer may be significantly different from the base case. MARCH does not contain a detailed noding of the primary system, a detailed steam generator model, or models for calculating the quality of the break flow. These models are generally controlled by MARCH input assumptions. Thus, for some alternative cases the simplified MARCH models may not be adequate for accurate calculation of the system response. More than one calculation has been performed for some of the cases in order to assess the effects of modeling uncertainties.

The MARCH results for the alternative accident sequences are discussed below.

5.1 Case 1. The high pressure injection is run continuously in the ECC injection mode.

In the TMI transient, high makeup flow was initiated at about 13 seconds by the operators and at 2 minutes the ECC mode of the HPI system was activiated as the pressure dropped below the 1640 psig setpoint. After about 4.6 minutes the HPI flow was assumed to be reduced to a zero net makeup rate in the base case calculations. For the Case 1 MARCH calculation, two HPI pumps were assumed to operate continuously after initiation at 2 minutes. The total injection rate for the two pumps was calculated from a fit to the TMI high pressure injection head curve

$$WECC = 1200 \sqrt{1 - \Delta P/2890}, \text{ gpm} \quad ,$$

where ΔP is the difference between the primary system and containment pressures in psi. For the base case calculations, constant injection rates independent of ΔP were used. Injection rates of 665 gpm at 2000 psi and 970 gpm at 1000 psi are obtained from this equation. As seen in Figure 5.1, the MARCH calculated injection rate approaches an asymptotic value of about 770 gpm (6400 lb/min) after about 40 minutes. The primary coolant pumps are assumed to continue to operate during the entire transient. Thus, good steam generator heat transfer is obtained following operation of the auxiliary feedwater at 8 minutes.

The MARCH results in Figure 5.1 for this case indicate an initial system depressurization to about 1500 psia at 5 minutes and 1300 psia at 20 minutes followed by a gradual repressurization to about 1650 psia after 40 minutes.

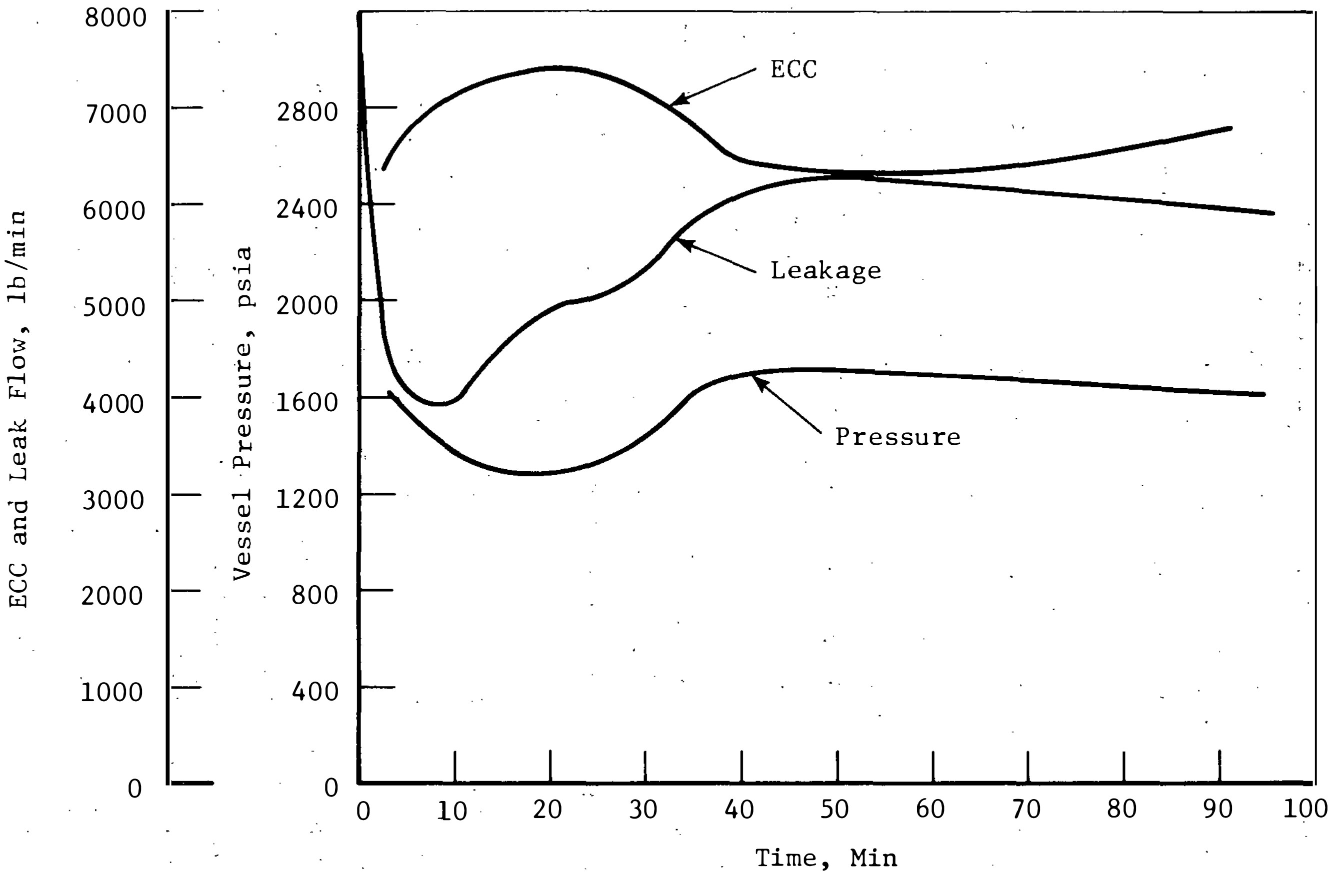

Figure 5.1 Calculated Results for Alternative Case 1
(Full HPI Flow), 0-100 Min

Following this time, the leak rate and ECC injection rate are balanced at about 750 - 800 gpm. The system remains full during the transient.

After about 30 minutes, the MARCH calculations indicate all of the decay heat is being removed by ECC water flowing through the core and out the leaking PORV. The surge line temperature is predicted to decrease from 497°F at 40 minutes to 456°F at 100 minutes. Termination of the transient was assumed to be accomplished by eventual closure of the block valve at 142 minutes. The MARCH calculation was stopped at 100 minutes.

5.2 Case 2. The high pressure injection is run continuously. Emergency feedwater is delayed until 1.0 hour.

MARCH predicts similar results for Cases 1 and 2. This occurs because after the first few minutes of the transient, most of the decay heat can be removed by the ECC injection flowing through the core and out the leaking PORV. During the first minute of these two transients MARCH predicts that 40% of the initial steam generator water inventory is boiled off dissipating 97% of the integrated core power. However, after 15 minutes for Case 2, the ECC flow-through is sufficient to dissipate about 90% of the decay heat. Thus, after the first few minutes of the Cases 1 and 2 transients, the steam generator does not play a dominant role. At 60 minutes the system pressures for Cases 1 and 2 are within about 50 psia of each other. Initiation of auxiliary feedwater at 60 minutes in Case 2 causes a temporary pressure decrease to about 1350 psia at 80 minutes. However, after 100 minutes, MARCH predicts surge line temperatures within 30°F, system pressures within 30 psi, and water inventories within 2.8% of each other for the two cases. Despite the extended delay in feedwater operation, no fuel damage is expected for this case.

5.3 Case 3. Emergency feedwater is delivered to the steam generator at 40 seconds rather than being delayed until 8 minutes.

Core uncovery in the TMI accident is believed to have occurred at 100 minutes when the final reactor coolant pumps were stopped and the water in the primary system fell to the low points in the reactor vessel and steam generators. The MARCH code is not capable of representing the effect of a pump on maintaining flow through the system or of calculating the actual water distribution throughout the primary under the conditions of the accident. The blowdown period in MARCH is simulated by isolating the fraction of the primary system water that would remain in the system after the pumps are shut off from the mass that must be leaked out prior to core uncovery. In this manner, MARCH is able to simulate the time of core uncovery reasonably accurately. Changing the time of delivery of the emergency feedwater from 8 minutes to 40 seconds has only a small effect on the MARCH heat transfer calculations. It is assumed to have no effect on the pump flow or quality of the coolant leakage. For the base case MARCH predicts core uncovery starts at 100.1 minutes. With emergency feedwater delivery at 40 seconds, MARCH predicts core uncovery is delayed until 105.8 minutes. Thus, the more rapid delivery of feedwater with all other factors unchanged would have an insignificant effect on the course or timing of the accident.

5.4 Case 4. Emergency feedwater to the steam generators is delayed until 1.0 hour.

MARCH calculations indicate the course and timing of the accident would be significantly altered from the base case if emergency feedwater had been delayed until 1.0 hour. The MARCH results for this case are shown in Figure 5.2. MARCH predicts the primary system will repressurize to the safety valve setpoint of 2450 psi at about 27 minutes. When the safety valves lift, the MARCH models predict an increase in leakage rate from about 5500 lb/min to 8800 lb/min. Continued leakage of coolant at this high rate with the safety valves open results in the beginning of core uncovery at 45 minutes. It was assumed in the MARCH calculations that the primary coolant pumps would continue to operate as in the base case calculations, until core uncovery began. Thus, it was assumed the primary pumps would cavitate and be stopped when the primary system void fractions were the same in the two cases. As in the base case, when the primary pumps are stopped, the liquid

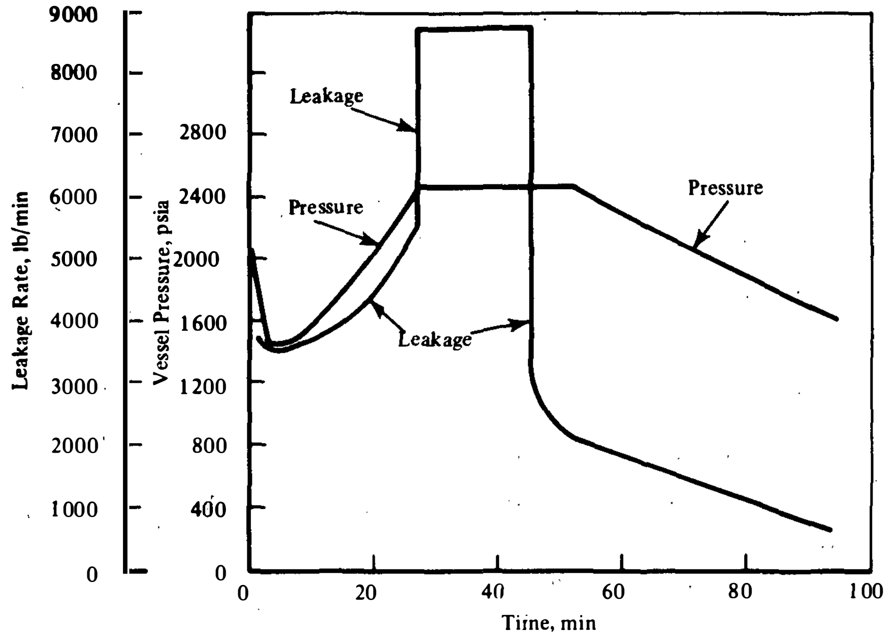

Figure 5.2-A. Coolant Leakage and Vessel Pressure

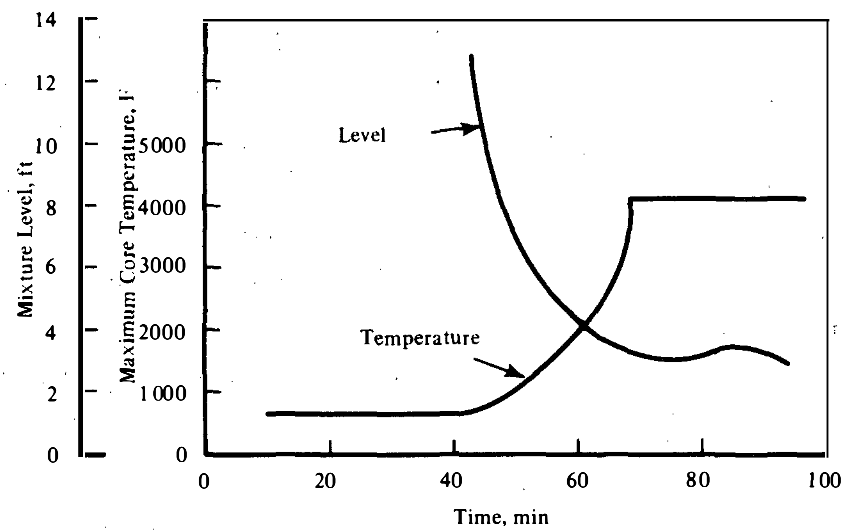

Figure 5.2-B. Mixture Level and Maximum Core Temperature

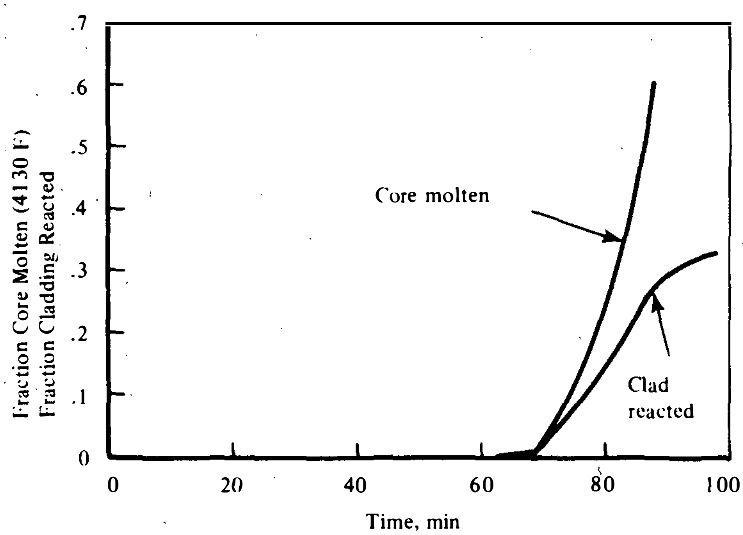

Figure 5.2-C. Fraction Core Currently Molten (Above 4130 F) and Fraction-Cladding Reacted

FIGURE 5.2. CALCULATED RESULTS FOR ALTERNATIVE CASE 4 (EMERGENCY FEED WATER DELAYED TO 1.0 HR), 0-100 MIN

is presumed to collapse into the bottom of the reactor vessel and steam generators. The ECC and makeup flows and the pump heat were the same as for the base case calculations. The blowdown quality is assumed to be the same as for the base case until the safety valves lift. During operation of the safety valves the MARCH models predict a combined liquid-steam blowdown composed of about 35 weight percent steam.

Figure 5.2 shows plots of the calculated primary system pressure, coolant leak rate, core mixture level, peak core temperature, fraction core molten (above 4130 F), and fraction cladding reacted. At about 45 minutes, the primary pumps stop and the core uncovers. The coolant leakage rapidly decreases from about 8800 lb/min to below 2000 lb/min as the break flow changes from primarily liquid to steam. The safety valves reclose at 52 minutes. The primary system pressure begins to decrease at 52 minutes due to core uncovery and the start of emergency feedwater delivery to the steam generators at 60 minutes. The emergency feedwater condenses steam on the primary side and reduces the system pressure. The core mixture level has decreased to four feet at this point, and cladding failures may be expected as peak core temperatures exceed 1600 F. Significant metal-water reaction and core melting are predicted to begin just before 70 minutes. Large core melt fractions were obtained by 100 minutes, when the MARCH calculation was stopped.

It should be recognized that operator action could have differed in this case from the response to the actual accident conditions. In particular, the lifting of the safety valves and the resulting high leakage rates are likely to have been recognized and resulted in different operator response. Additionally, since the modeling of the leakage quality has a significant effect on the course of the accident, the simplified models in MARCH may not be adequate for this case.

5.5 Case 5. The PORV leakage is stopped at 25 minutes.

The MARCH calculations for this case assume that closure of the PORV at 25 minutes will permit continued operation of the primary coolant pumps. At 25 minutes, MARCH predicts that the primary system inventory is 436,000 lb. For comparison, all of the primary pumps remained operational until 74 minutes when the primary inventory was 290,000 lb. in the base case. Thus, it is reasonable to assume the pumps could remain operational indefinitely if the primary inventory available at 25 minutes can be maintained. With operation of the primary pumps, good steam generator heat transfer will continue, and further loss of primary coolant due to boiloff would be stopped.

MARCH predicts a system pressure of about 1180 psi at 25 minutes when the PORV is closed. Following closure of the PORV, MARCH predicts the system pressure increases to between 1300 and 1450 psi, depending on the ECC injection rate, the heat rejection to the letdown cooler, and the assumed pump power.

The MARCH calculation for this case was stopped at 100 minutes with the system pressure at 1400 psia.

5.6 Case 6. PORV leakage continues until 3.3 hours.

The first 2.3 hours of this case are the same as for the base case. MARCH calculations were performed for two possible, but significantly different scenarios for accident times beyond 2.3 hours. The difference in the scenarios results from whether or not actuation of the core flood tanks occurs shortly after 3.3 hours. The analyses indicate that significant core melting would occur if the core flood tanks do not dump, assuming the ECC injection after 2.3 hours is the same as that assumed in the base case. However, if the core flood tanks are actuated, MARCH predicts the core heat up is arrested at a core melt fraction of 0.018.

There are two aspects to the question of whether or not the core flood tanks operate. One aspect concerns the position of the core flood tank isolation valves. It has been suggested (NUREG - 0600, p. I-4-28) that the isolation valves may have been closed so that no injection would have been possible regardless of the system pressure after 2.3 hours. The second aspect concerns whether or not the system pressure would have continued its apparent downward trend after 2.3 hours, assuming the PORV is open, to a value below 600 psi so that injection could occur. The majority of the evidence indicates that continued depressurization below the 680 psi minimum obtained in this phase of the TMI transient would not have occurred. The MARCH analyses indicate that good steam generator heat transfer is required to explain the system pressure observed in the first 135 minutes of the TMI transient. During the first 100 minutes of the transient, operation of the primary coolant and pumps produced good thermal coupling to the steam generators. Between 90 and 135 minutes, emergency feedwater was apparently being injected into the top of the A steam generator.[2] The feedwater cooled the steam generator tubes, and condensed the steam on the primary side. The downward trend in pressure reversed and started an upward trend a few minutes before the PORV was closed at 142 minutes. Presumably this upward trend indicates that the amount of heat being transferred to the steam generator had been reduced and that the pressure would have continued to rise after 2.3 hours even with the PORV open. In this event the core flood tanks would not have actuated.

The MARCH results fcr the case with no core flood tank injection are shown in Figure 5.3. Core melting begins at 146 minutes. Assuming operability of RCP-2B and the same ECC injection as in the base case, MARCH predicts the core heatup is arrested at a core melt fraction of 0.33. Since the actions of the operators which affected the pump operation were probably influenced by the course of the accident, it is not clear that the operation of RCP-2B at 174 minutes and the increase in ECC flow to 565 gpm at 200 minutes are valid assumptions for the present case. Apparently, however, the extent of damage to the core would have been more severe if the PORV had not been closed at 2.3 hours.

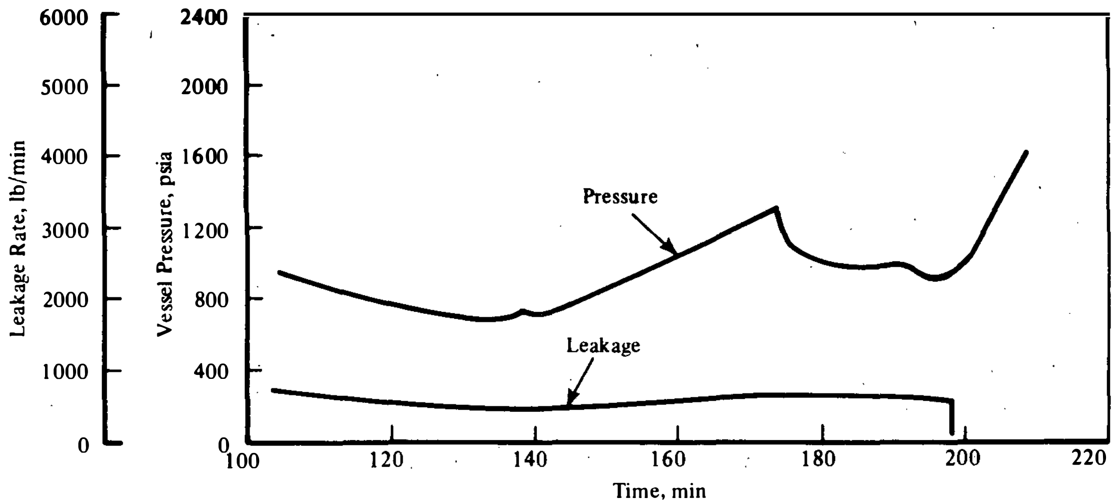

Figure 5.3-A. Coolant Leakage and Vessel Pressure

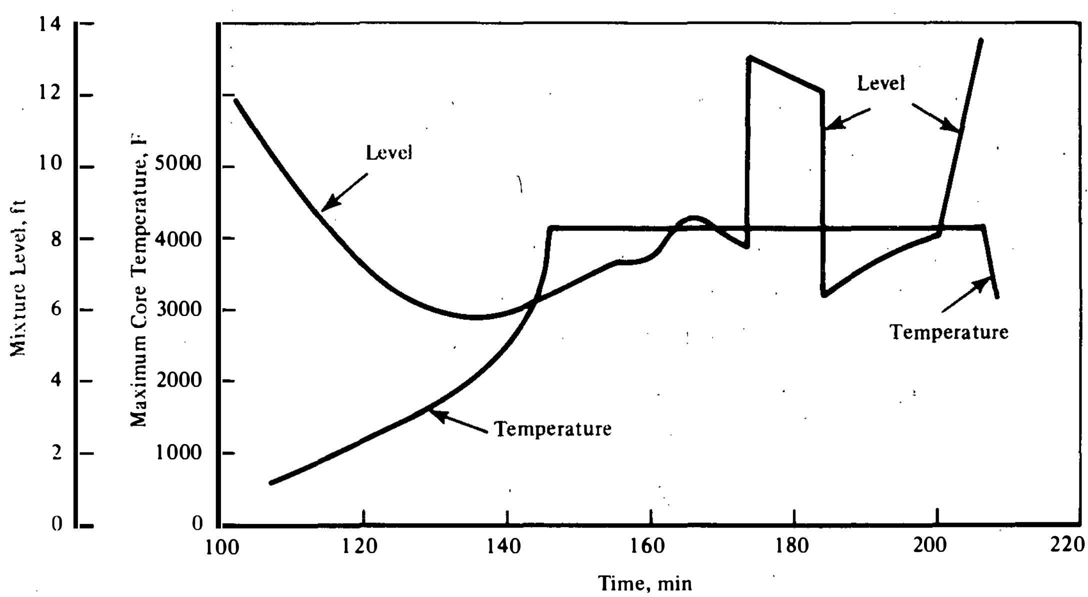

Figure 5.3-B. Mixture Level and Maximum Core Temperature

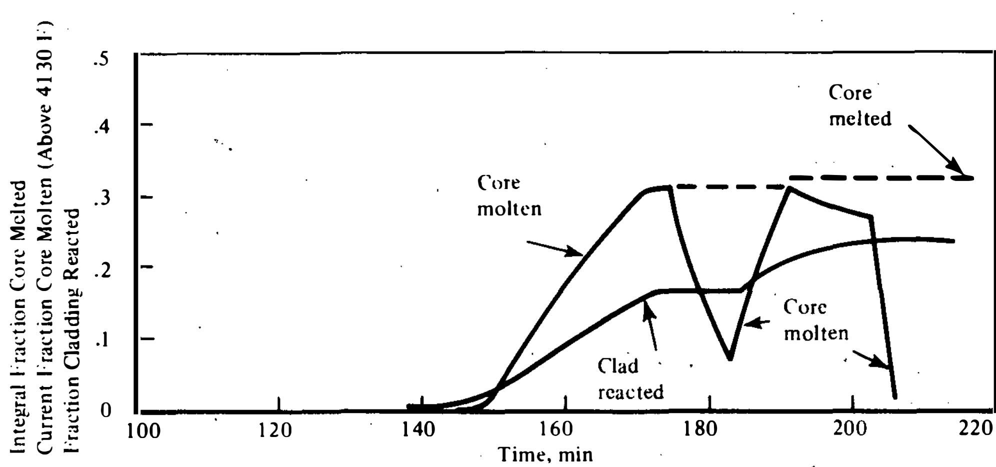

Figure 5.3-C. Fraction Core Which Has Melted, Fraction Core Currently Molten (Above 4130 F), and Fraction Cladding Reacted

FIGURE 5.3. CALCULATED RESULTS FOR ALTERNATIVE CASE 6 (PORV Closure at 198 Min), 100-200 Min

MARCH calculations were also performed for a variation of this case in which the core flood tanks were activated. In this case the emergency feedwater flow was assumed to continue until 150 minutes rather than being stopped at 135 minutes as in the base case. 150 minutes would have been a reasonable time to stop the feedwater, since the A steam generator level would have reached 100% on the operating range. In this calculation, the primary system pressure decreases below 600 psi, and the core flood tanks are assumed to inject. The MARCH core heat transfer and core flooding models predict that the tanks will inject about 21,000 lb of water between 145 and 150 minutes before the system pressure increases and stops the flow. The injection increases the vessel mixture level from about 5.3 to 9.3 feet at the time when core melting is beginning and effectively arrests core heatup. With continued 90 gpm makeup to the vessel and the assumed operation of RCP-2B, the core melt fraction remains below 2%. It is concluded that if core flood tank injection had occurred, there would have been little core melting. However, the balance of the evidence indicates core flood tank injection would not have occurred and that more extensive core damage would have resulted than in the base case.

5.7 Case 7. All primary coolant pumps are stopped concurrently with reactor trip.

Stopping the primary coolant pumps concurrently with reactor trip may be expected to produce a significantly different system response from that obtained in the actual TMI transient. Stopping the primary coolant pumps has two effects.

One effect is that after void formation in the primary system, heat transfer from the primary system into the steam generators is expected to be substantially reduced. Good steam generator heat transfer from the steam space is anticipated only when the emergency feedwater is on, cooling the top of the steam generator tubes and condensing steam on the primary side. During periods of poor steam generator heat transfer, the primary system is predicted by MARCH to repressurize to values somewhat below the safety relief valve setpoint.

A second effect of turning off the primary coolant pumps is that the liquid and steam will separate. That is, the liquid will fall to the low regions in the primary system, and the steam will rise to the high points. With the primary pumps operating, an essentially homogenous mixture of water and steam is circulated through the system. Separation of the steam and water should have an important effect on the steam content of the coolant leaking through the open PORV. In the general case, two different break flow scenarios are plausible.

Following the primary coolant pump trip, a period of essentially liquid blowdown through the PORV is expected. Under some conditions (perhaps, with high pressurizer heater power and low ECC injection) it may be possible to reestablish a void volume in the pressurizer, and quickly turn the leakage from liquid to steam. If a steam blowdown is produced, the blowdown period will be greatly extended.

A different leak scenario seems more likely. In this scenario, the PORV leakage remains primarily liquid as in the base case until sufficient coolant inventory leaks from the primary to uncover the surge line connection to the A-loop hot leg. Steam leakage occurs after the surge line connection uncovers. The sequence of events in the actual TMI accident during the 3 - 16 hours period supports this latter leak flow scenario.

At the time of surge line uncovery, the pressurizer level would be expected to drop. This is an important consideration since the operators would be expected to increase HPI flow to maintain adequate pressurizer level. The MARCH analyses indicate about a 25-minute interval between surge line uncovery and core uncovery.

More detailed system modeling than MARCH provides is required for evaluation of the break flow scenario. MARCH contains no models which would explicitly predict the effect of primary pump flow on steam generator heat transfer or the liquid/steam content of the coolant leakage. However, MARCH can be used to examine the effects on system performance of different assumptions about steam generator heat transfer and flow through the PORV. A number of MARCH calculations were performed to examine the various possibilities.

Two MARCH calculations were performed in which it was assumed that steam blowdown occurred after leaking 14 percent of the coolant inventory as a liquid blowdown. For these cases, MARCH predicts no core uncovery prior to closure of the PORV at 2.3 hours. In the calculations a period of primarily liquid blowdown lasts about 20 minutes, followed by a longer period of steam leakage. If good steam generator heat transfer is assumed even though the primary pumps are off, the primary system water inventory falls to 443,000 pounds at 2.3 hours. If steam generator heat transfer is assumed to degrade after about 20 minutes, the water inventory falls to about 317,000 pounds at 2.3 hours. About 210,000 pounds of water will fill the bottom of the steam generators and the reactor vessel to the top of the core. Thus, no core uncovery is predicted for this case in the first 2.3 hours.

Figure 5.4 shows MARCH results for variations in this case in which the liquid blowdown continues until the surge line uncovers. As illustrated in Figure 5.4-A, the surge line uncovers and steam blowdown begins at 49 minutes. The cold legs uncover at about 68 minutes, and core uncovery begins at 74 minutes. The system pressure increases, due to degraded steam generator heat transfer, to the safety relief valve setpoint at 45 minutes. Shortly before the start of emergency feedwater at 90 minutes, the pressure begins to decrease due to a reduction in steam generation as the core uncovers. At 90 minutes, the emergency feedwater flow is assumed to start, as in the base case, and steam begins to condense from the steam space. Since the primary coolant pumps are off, both steam generators were filled to the top of the cold legs at 68 minutes. The average condensation rate calculated by MARCH between 90 and 135 minutes is 2485 lbs/min.

Between 90 and 109 minutes the amount of water condensed in the A-steam generator (at a rate of 2485 lbs/min) balances the quantity lost through letdown (at a rate of 1160 lbs/min) from 68 minutes to this time. After 109 minutes, the steam condensed in the A-steam generator will thus cause an overflow back into the reactor vessel. In the MARCH calculations, this delay in refill of the A-loop was neglected, and it was assumed all condensate was in effect immediately returned to the reactor vessel. A net makeup of 90 gpm to the vessel was also assumed. As in the base case, the low vessel water level effectively decouples the letdown and makeup rates.

Core melting begins at 106 minutes. The core mixture level remains in the 8 - 11 feet region for most of the time after melting starts. About 24 percent core melting is predicted prior to 200 minutes. This is about the same result as for the base case. The MARCH calculation was stopped at this time.

These MARCH results indicate that turning off the primary coolant pumps could have either resulted in significantly reduced core damage or similar damage depending on the behavior of the flow out the PORV. More sophisticated analyses of system hydraulic behavior are required to resolve these uncertainties further.

5.8 Case 8. All AC electric power is lost sometime between 0.5 and 5 hours.

MARCH calculations were performed assuming loss of AC electric power at 2 hours. It is assumed loss of AC power prevents closure of the block valve at 2.3 hours, no ECC is available, and steam generator emergency feedwater is unavailable.

The MARCH results for the first 2 hours of the transient are the same as for the base case. MARCH results beyond 100 minutes are shown in Figure 5.5. At 120 minutes emergency feedwater flow to Steam Generator A is stopped and the steam condensation on the primary side ceases. The primary system repressurizes to about 1100 psi at 135 minutes. At this time, the mixture level in the core falls below 4 feet, and the leakage through the open PORV is balanced by the steam and hydrogen generation in the core. Core melting starts at 141 minutes. The vessel pressure remains in the 1100-1200 psia range until the core collapses into the bottom head at 173 minutes. Thus, unless operator action is taken to activate emergency power, core meltdown could be expected within an hour.

5.9 Case 9. PORV remains closed after closure at 2.3 hours, and HPI not initiated at 3.3 hours.

If the PORV block valve remains closed after being closed at 142 minutes, the MARCH calculations for this case indicate the system will

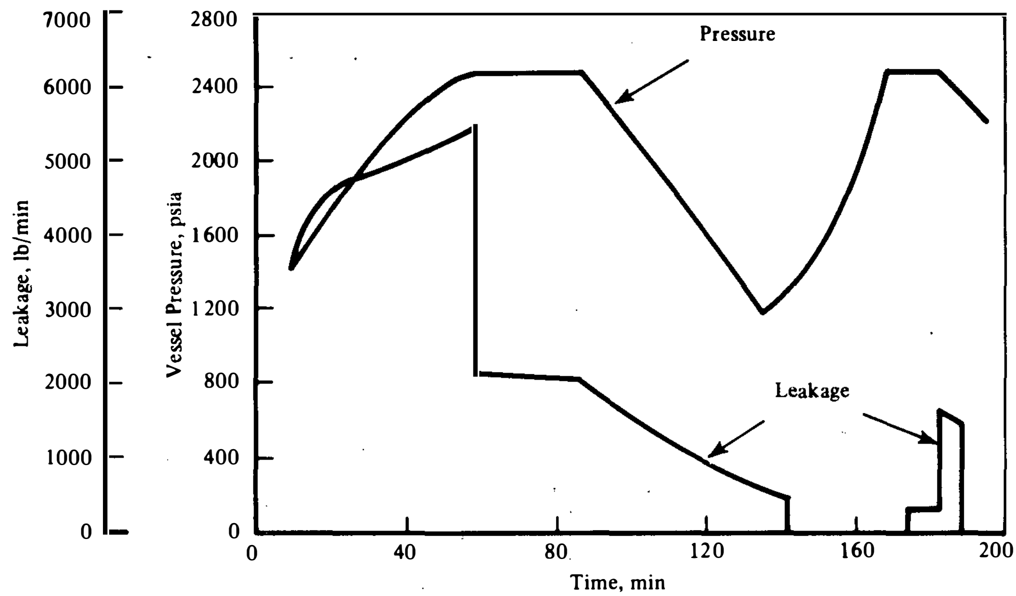

Figure 5.4-A. Coolant Leakage and Vessel Pressure

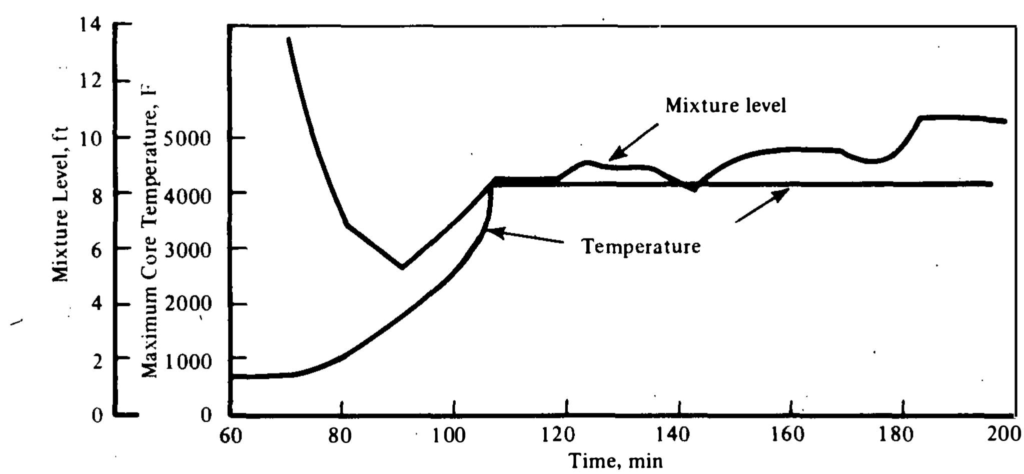

Figure 5.4-B. Mixture Level and Maximum Core Temperature

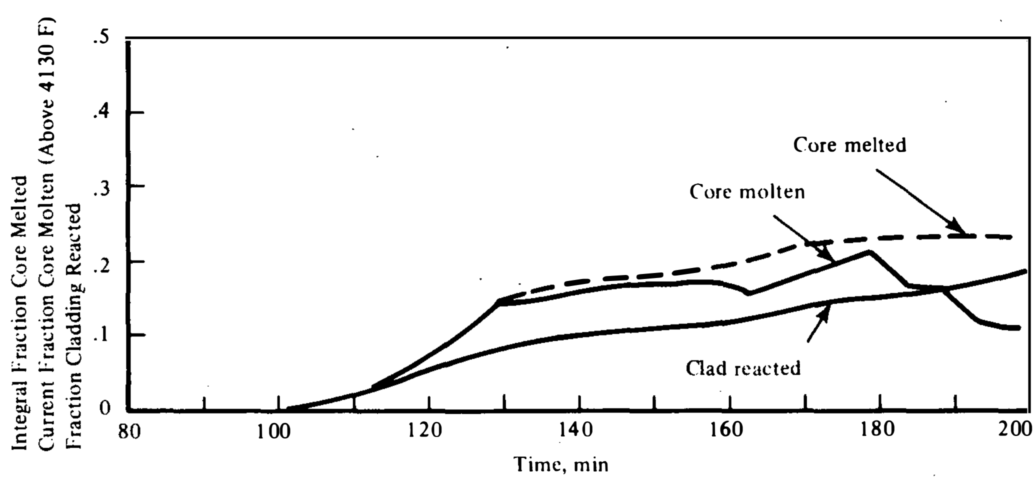

Figure 5.4-C. Fraction Core Which Has Melted, Fraction Core Currently Molten (Above 4130 F), and Fraction Cladding Reacted

FIGURE 5.4. CALCULATED RESULTS FOR ALTERNATIVE CASE 7 (NO RCP'S), 0-220 MIN

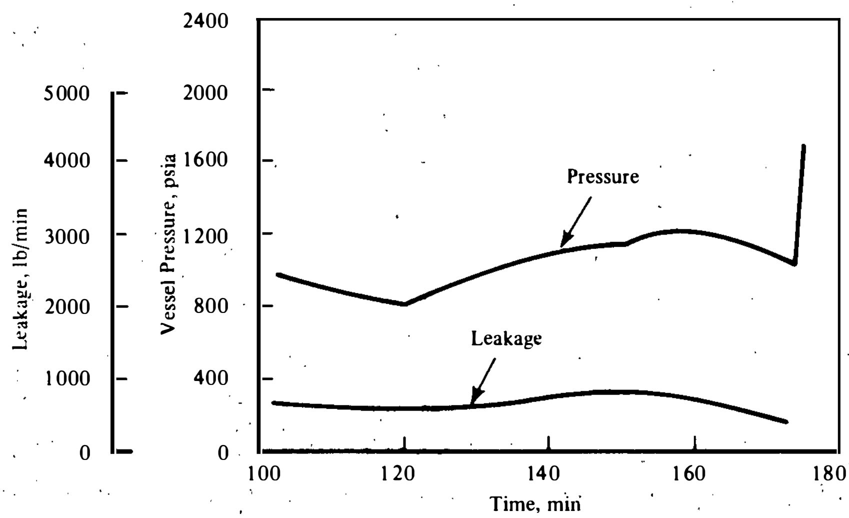

Figure 5.5-A. Coolant Leakage and Vessel Pressure

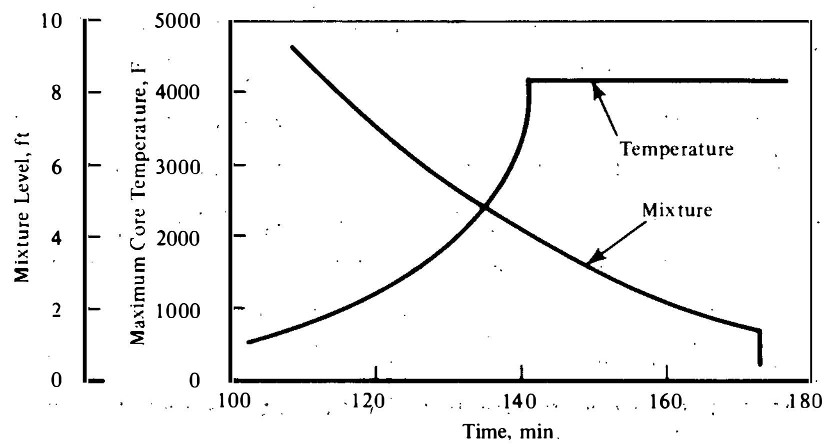

Figure 5.5-B. Mixture Level and Maximum Core Temperature

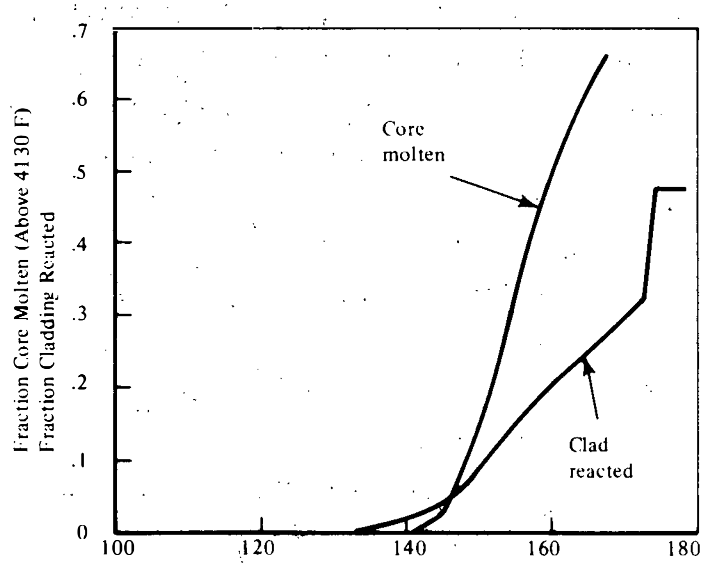

Figure 5.5-C. Fraction Core Currently Molten and Fraction Cladding Reacted

FIGURE 5.5. CALCULATED RESULTS FOR ALTERNATIVE CASE 8 (LOSS OF POWER AT 120 MIN), 100-180 MIN

repressurize and begin venting through the safety relief valves. Because the system pressure does not fall to the point of automatic actuation of emergency core cooling, it is also postulated that the HPI is not initiated at 200 minutes. The MARCH base case calculation indicates substantial progress in recovering the core and quenching the damaged region had been made prior to initiation of the HPI at 200 minutes. However, if the HPI is not initiated at 200 minutes, a second period of core heatup is calculated to begin. The precise timing of events in the second core heatup period is found to be sensitive to the calculated prior history of the accident and to assumptions made after 200 minutes regarding steam generator heat transfer, the refluxing of the steam condensed by the steam generators, and the magnitude of the continuing coolant makeup to the reactor vessel.

Significant uncertainties exist regarding the amount of heat transfer that can occur in the steam generators during this time period. Firstly, good heat transfer would only be expected when the water level on the secondary side of the steam generator is higher than the level on the primary side or when feeding the steam generators with the emergency feedwater system. Secondly, the hydrogen generated in the earlier phase of the accident could impede the flow of steam through the hot legs. Steam Generator B is isolated on the secondary side during this time period. Steam condensed in the A steam generator is being withdrawn through the letdown line either preventing or reducing the potential for refluxing of condensate back into the vessel.

In the MARCH analysis for Case 9 it is assumed that no steam generator heat transfer and no refluxing occurs after 184 minutes.

As in the base case, a 90 gpm makeup rate to the vessel is assumed during the core uncovery period. For this case, the safety relief valves lift at 190 minutes, and the coolant mixture level falls from 9.9 feet at 185 minutes to 7.7 feet at 200 minutes. About 18 percent of the core is calculated to be molten (above 4130 F) at this time. The coolant mixture level is calculated to gradually fall to 6.5 feet at 300 minutes. At 200 minutes, most of the core nodes above the 10-foot core elevation are above 4130 F, core nodes in the 6.5 - 10 feet regions are generally below 1000 F, and those below 6.5 feet remain covered by the mixture level. At about 260 minutes, the core nodes in the 6.5 - 10 feet region begin to be heated above 4130 F and melt. By 300 minutes, when the MARCH calculation was stopped, the core melt fraction is calculated to reach 0.45 and the fraction cladding reacted is 0.28.

Although there are a number of major assumptions made for the analysis of this case, the implication of the calculations is that the condition of the core would have been significantly worse than for the base case. Eventual progression to complete core meltdown would occur for this case assuming the HPI is not initiated.

55.10 Case 10. HPI not initiated at 3.3 hours.

The MARCH results for Cases 9 and 10 are similar and exhibit the same sensitivities to boundary conditions and modeling assumptions. The primary difference between Cases 9 and 10 is that the second core uncovery period occurs sooner and produces somewhat more severe core damage for Case 10 than for Case 9. The reason for this is that, with the PORV reopened as in the actual TMI accident, coolant is lost from the system more rapidly. Consequently, after 200 minutes the primary coolant level is generally lower for Case 10 than for Case 9. By an accident time of 300 minutes, MARCH predicts the core melt fraction has increased to 0.57. As for Case 9, the core conditions are significantly worse than for the base case. Under the assumptions made in the analysis, complete meltdown of the core would be expected to eventually result.

6.0 MARCH ANALYSIS OF CORE MELTDOWN SCENARIOS

MARCH calculations were performed to evaluate the timescale and course of hypothetical scenarios in which complete core meltdown occurs. Two scenarios were selected for analysis. One scenario produces an early core meltdown and the other a meltdown sequence starting at 3 days.

The early meltdown scenario duplicates the TMI transient until the start of core uncovery at 101 minutes. A complete meltdown is produced by assuming there is no makeup after 101 minutes, RCP-2B does not operate (at 174 minutes), the PORV remains open, and there is no steam generator heat transfer beyond 135 minutes.

The delayed core meltdown scenario starts with the conditions existing at TMI at 3 days. At this time, the primary system was filled with water at a temperature of 300 F and a pressure of 1000 psia. The steam generators were also assumed to be full of water at a temperature of 250 F. It was assumed that the cladding in the top 4 feet of the core was 90 percent oxidized. The hydrogen produced in this reaction was assumed to have been burned at 10 hrs. The calculations do not account for a corresponding oxygen depletion, however. The decay heat in the upper 4 feet of the core was also reduced, corresponding to a 50 percent release of volatile fission products. Core meltdown for this case was produced by assuming failure of all feedwater to the steam generators, failure of the ECC, and no heat rejection to the letdown cooler.

In general, containment failure in meltdown accidents may result from failure to close (isolate) the normal containment penetrations, generation of missiles produced in steam explosions or at the time of reactor vessel failure, melt-through of the concrete floor of the containment, over-pressurization due to excessive generation of steam and non-condensables, and over-pressurization due to hydrogen burning. With the building coolers working, rapid burning of large amounts of hydrogen presents the major mechanism for an early above ground failure of the containment.

Uncertainties in the calculation of the containment pressures obtained in hydrogen burning are (1) the amount of hydrogen produced during the meltdown accident, (2) the timing of the burn, and (3) whether the burn is localized or involves the whole containment. The various MARCH meltdown models produce about a factor of two range in the amount of hydrogen produced during the period of core meltdown within the pressure vessel. The timing of the burn can become important, if it is assumed burning is delayed beyond the time when flammable mixtures are first obtained. If the hydrogen is slowly released from the primary system and burned as it is released, relatively low pressures are obtained if containment safeguards are operational. However, if large amounts of hydrogen are released from the primary at one time (such as at head failure) or if the containment stores hydrogen in excess of the flammability limit, large pressures can be obtained. Complete burning at one time of the hydrogen potentially available from oxidation of 100 percent of the core cladding could produce containment failure by overpressurization.

MARCH calculations generally predict that 30 to 60 percent of the core cladding is oxidized during core meltdown, prior to the time the core collapses into the bottom head of the reactor vessel. The range of the cladding oxidation results from uncertainties in modeling assumptions and on the type of meltdown scenario being analyzed. Additional cladding oxidation may occur when the molten core collapses into the water in the bottom head of the reactor vessel. Calculations performed using MARCH subroutine HOTDROP (Section 3.0) indicate significant additional cladding oxidation in the bottom head requires very fine particulation of the debris. Cladding oxidation is also enhanced if the remaining metallic zirconium is separated from the core debris rather than being mixed or alloyed with the ZrO_2 and UO_2. If the core particulates into 1 cm or larger particles, HOTDROP predicts little additional cladding oxidation during core slump into the bottom head. However, if the particles are sufficiently small (e.g., 0.01 cm) and there is direct contact between zirconium metal and the water, complete oxidation can occur. The particle sizes required for complete zirconium oxidation are in the range associated with the degree of fragmentation observed in vapor explosions. The HOTDROP calculations and the data of Baker[6] indicate that, if a steam explosion does not happen, little additional cladding oxidation would be expected during the core slump into the water in the bottom head.

In order to scope the uncertainties in hydrogen production due to metal-water reaction in the bottom head, two MARCH calculations were performed (for the early meltdown case). In one calculation, no additional reaction in the bottom head was assumed and, in the other case, complete reaction was assumed.

A second area of uncertainty in the MARCH calculation results from the modeling of the interaction of the core debris with water in the cavity below the reactor vessel following bottom head failure. As in the modeling of the interaction in the bottom head, the extent of debris fragmentation is uncertain. However, for the reactor cavity processes, the size of the particles is of less concern than whether or not fragmentation occurs at all.* This is because the rate of particle quenching and steam production is relatively unimportant. The containment pressurization is affected primarily by the total amount of steam produced rather than the rate. For particle sizes less than a few inches in diameter, about the same peak pressures are predicted.

Normally, the reactor cavity would not be expected to contain water at the time of vessel head failure. For the early meltdown case, the TMI reactor cavity is assumed to be dry. However, at the time of head failure, the primary system will depressurize and the core flood tanks will dump water into the cavity. Thus, for the early meltdown case, it is assumed that the core flood tank water (124,000 lb) can be vaporized in the reactor cavity.

For the case with meltdown delayed until after 3 days, the reactor cavity was believed to be flooded by the water on the containment floor. Thus, a much larger source of water is potentially available.

---

*This statement is strictly valid only if it is assumed the question of cladding oxidation has been previously settled while the core debris is still in the reactor vessel. Some of the same fragmentation uncertainties about cladding oxidation exist for the reactor cavity.

However, vaporization of the core flood tank water (124,000 lb) cools the core debris to below 2000 F. Vaporization of this quantity of water increases the TMI containment pressure about 25 psi. With the coolers on, the containment would not be threatened. Vaporization of additional water would increase the pressure but imply complete quenching of the debris, so the boiloff rate would become controlled by the decay heat. With the coolers on, boiloff of water by decay heat is easily handled. For the delayed meltdown case, it was assumed in the MARCH calculations that only 124,000 lb water would be rapidly vaporized by the core debris in the reactor cavity.

Tables 6.1 and 6.2 list the calculated event times in the meltdown scenarios. Figures 6.1, 6.2, and 6.3 are plots of the amounts of hydrogen and the pressures in the containment as a function of time. For the early meltdown case, two calculations were performed assuming the building coolers work. One calculation maximizes the hydrogen production during meltdown (complete reaction of all the cladding) and the steam generation in the reactor cavity. The other calculation assumes no metal-water reactor in the bottom head and assumes no fragmentation in the reactor cavity. For the second case, models in the INTER subroutine (Section 3.0) assume simultaneous attack of the concrete containment floor and boiling of water from the top surface. The INTER models also oxidize the Zircaloy in the core debris and release hydrogen to the containment. The calculations were repeated assuming the coolers do not work.

Early Meltdown, Minimum Hydrogen Production

Figure 6.1 shows MARCH results for the early meltdown and minimized hydrogen production. During core meltdown within the vessel, 40 percent of the cladding is oxidized producing 921 lb of hydrogen. As seen in Figure 6.1-A, hydrogen begins to be released to the containment in significant amounts just prior to core collapse into the head. All of the hydrogen is released to the containment when the head fails

TABLE 6.1 EARLY CORE MELTDOWN

Event	Time, minutes	
	Case A*	Case B**
start core uncovery	101	101
start core melt	133	133
core collapse into head	165	165
head failure	190	167
start concrete attack	190	220

* Case A: no metal-water reaction in the bottom head, and no debris particulation in reactor cavity.

** Case B: 100% metal-water reaction in the bottom head, and debris particulation in reactor cavity are assumed.

TABLE 6.2 DELAYED CORE MELTDOWN*

Event	Time, minutes
steam generator 90% dry	4626 (3d, 5.1 hr)
safety valves lift	4695 (3d, 6.3 hr)
start core uncovery	5495 (3d, 19.6 hr)
start core melt	5585 (3d, 21.1 hr)
core collapse into head	5798 (3d, 24.1 hr)
head failure	5808 (3d, 24.8 hr)
start concrete attack	5821 (3d, 25.0 hr)

* Event starts at 3 days. Case B assumptions (Table 6.1) made.

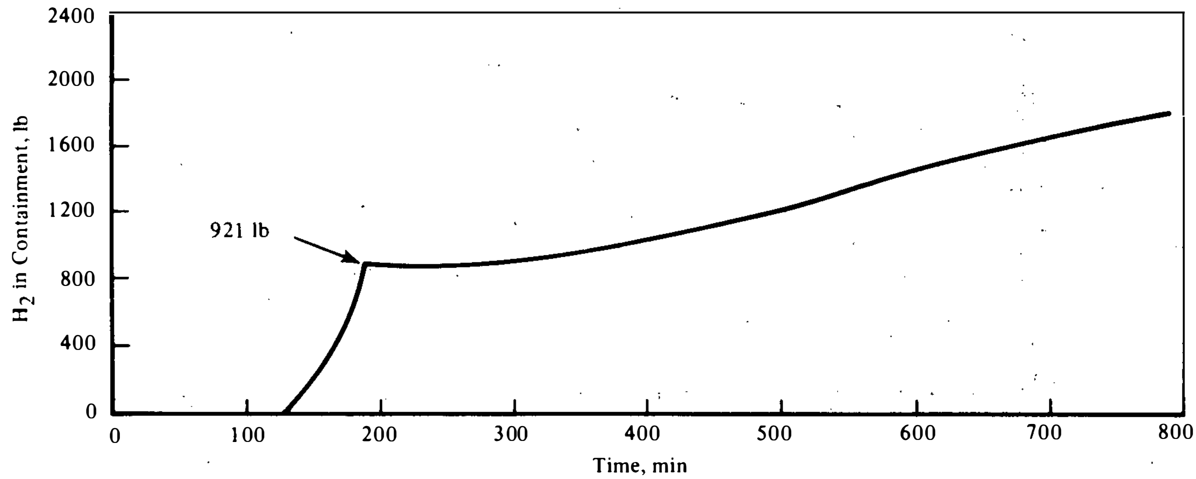

Figure 6.1-A. Containment Building Hydrogen Inventory

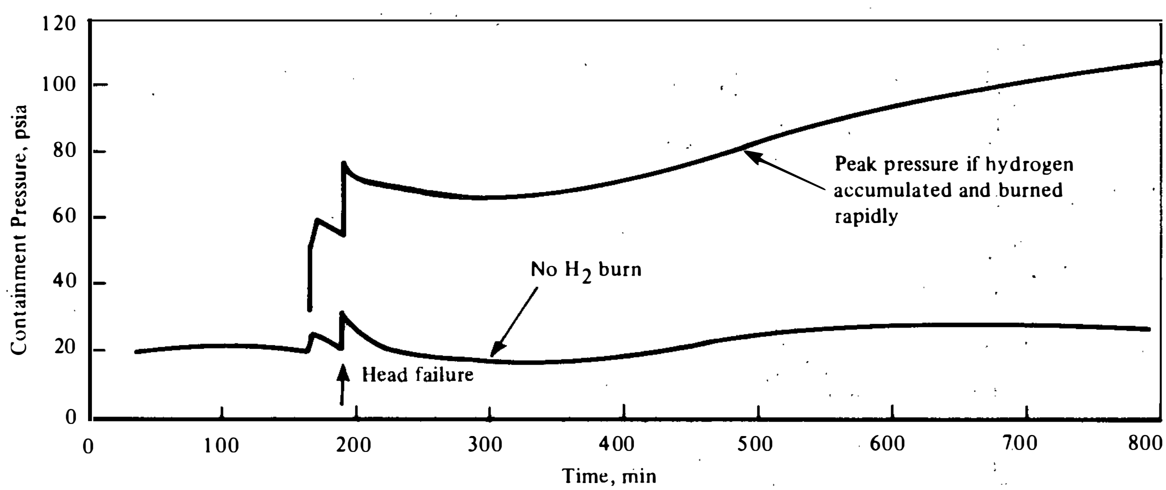

Figure 6-1-B. Containment Pressure With Building Coolers Operating

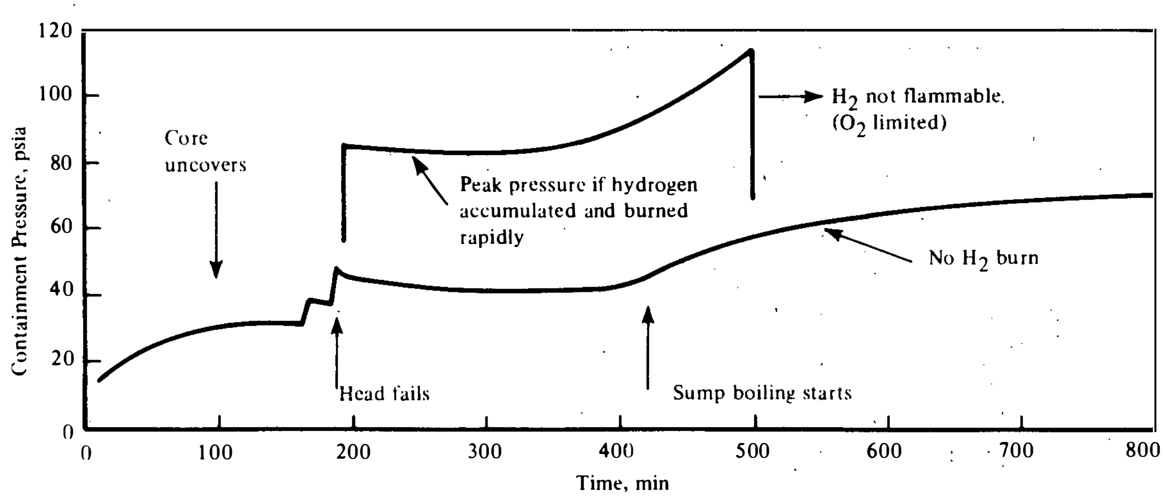

Figure 6-1-C. Containment Pressures With No Building Coolers

FIGURE 6.1. MARCH RESULTS FOR ASSUMED EARLY CORE MELTDOWN, 40 PERCENT CLADDING REACTION, NO RAPID STEAM PRODUCTION IN REACTOR CAVITY

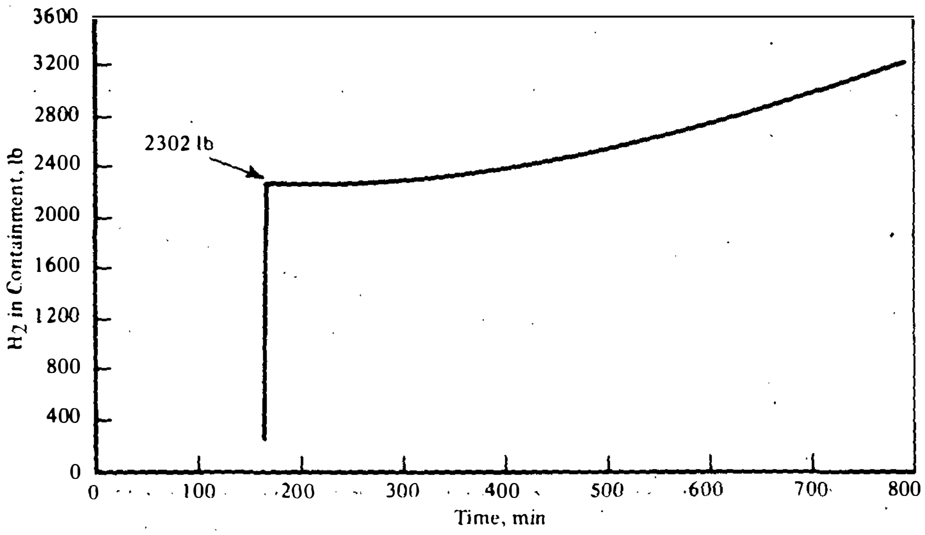

Figure 6.2-A. Containment Building Hydrogen Inventory

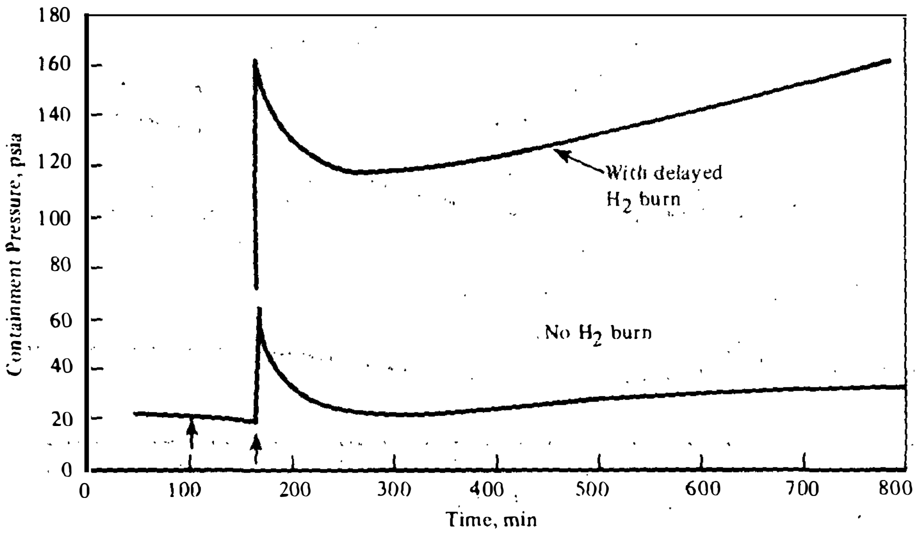

Figure 6.2-B. Containment Pressure With Building Coolers Operating

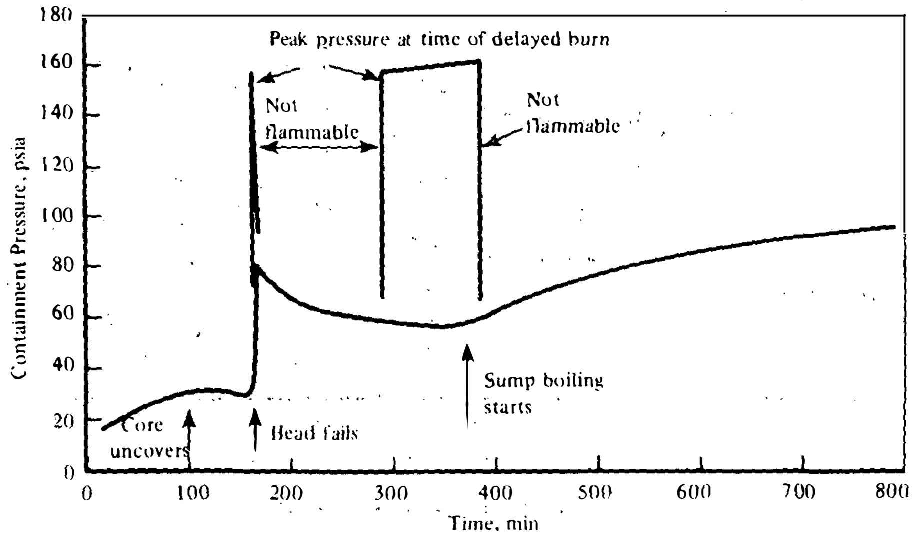

Figure 6.2-C. Containment Pressures With No Building Coolers

FIGURE 6.2. MARCH RESULTS FOR ASSUMED EARLY CORE MELTDOWN, 100 PERCENT CLADDING REACTION, WITH RAPID STEAM GENERATION IN REACTOR CAVITY

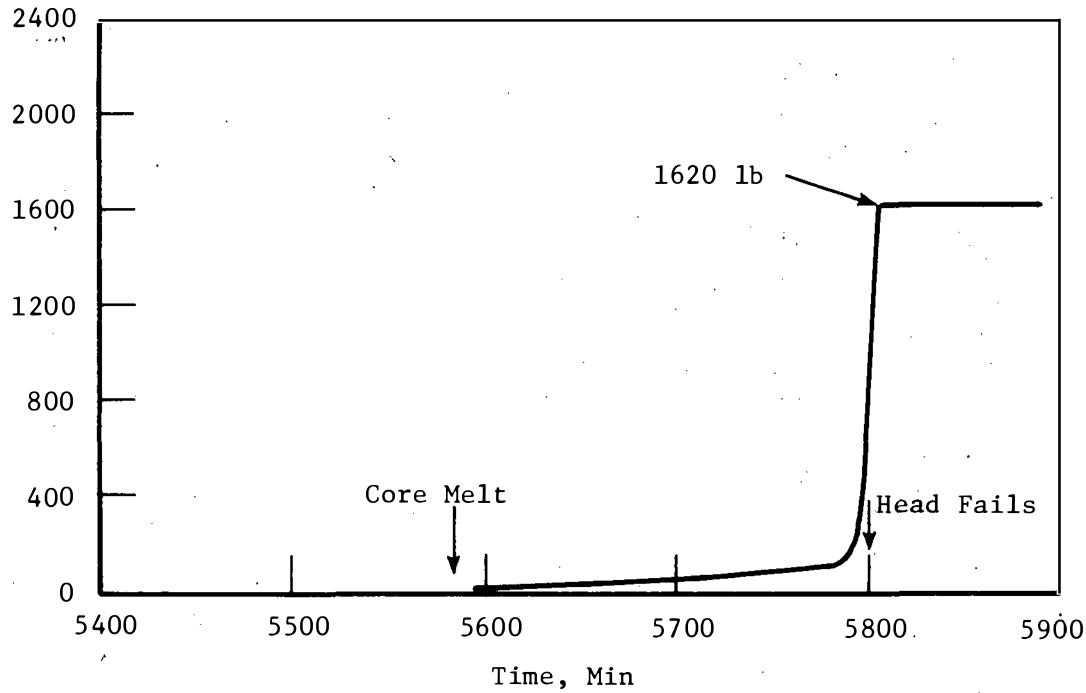

Figure 6.3-A Containment Building Hydrogen Inventory

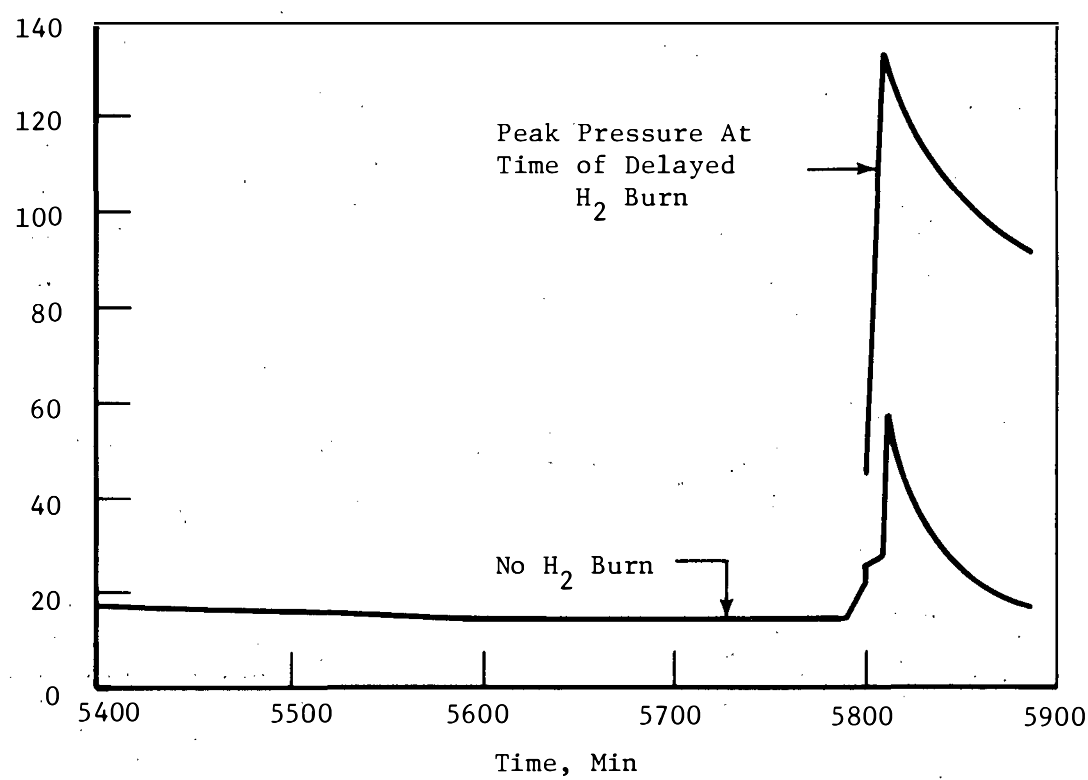

Figure 6.3-B Containment Pressure

Figure 6.3 MARCH Results for Assumed Meltdown Accident Sequence Starting at 3 Days, 70 Percent Cladding Reaction, With Rapid Steam Generation in Reactor Cavity at 190 minutes. The hydrogen release to the containment approximately doubles over the next 10 hours as additional cladding is oxidized during the concrete attack phase of the accident. Figure 6.1-B shows the containment building pressures with the coolers on for two cases: (1) if there is no hydrogen burn and (2) if all the hydrogen in the containment is burned at the indicated time. Figure 6.1-C shows results with the coolers off. Note that in order to obtain these pressures it is assumed no hydrogen is burned prior to the indicated time. At a containment pressure of 20 psia, a hydrogen mole fraction of 0.04 (the flammability limit) corresponds to about 460 lb hydrogen. Rapid burning of 460 lb of hydrogen would produce about a 25 psia pressure spike. Thus, continual burning at the flammability limit with the coolers operating would not lead to excessive pressures. If hydrogen burning is delayed until 10 hours after head failure, containment burning could begin to threaten the containment. For the TMI design pressure of 60 psi, a nominal failure pressure of about 135 psia would be expected. For the case in which the building coolers do not operate, containment failure (at 135 psia) by overpressurization would occur at 45 hours without hydrogen burning.

<u>Early Meltdown, Maximized Hydrogen Production</u>

Figure 6.2 shows results for the early meltdown case in which containment pressures are maximized by assuming 100 percent cladding reaction in addition to debris fragmentation in the reactor cavity. Figures 6.2-B and 6.2-C show that the combination of (delayed) hydrogen combustion and rapid steam generation in the reactor cavity could lead to containment failure. The pressures in Figures 6.2-B and 6.2-C assume hydrogen burning is delayed beyond the time flammable mixtures are first obtained. A MARCH calculation, in which hydrogen burning at a mole fraction of 0.04 was assumed, produced a peak containment pressure of 138 psia for the case in Figure 6.2-B with the coolers on. This occurs because most of the hydrogen burning is coincident with head failure. If the building coolers don't operate, containment failure (at 135 psia) is predicted at 37 hr.

Delayed Meltdown

Figure 6.3 shows results for the case with core meltdown starting after 3 days. Pressures are maximized by making the same assumptions as those made for Figure 6.2. However, for the present case it was assumed that 30 percent of the cladding was oxidized and the hydrogen previously burned. This is consistent with the hydrogen burn actually occurring at 10 hours in the TMI accident. Thus, 1620 lb of hydrogen are released to the containment when the head fails. Figure 6.3-B shows that the containment pressure could approach the failure pressure with combined hydrogen combustion and rapid steam generation. Until a better understanding is developed of ignition source requirements for hydrogen/steam/air mixtures and of molten fuel coolant interactions, it must be assumed that containment failure by this mode is possible.

Base Pad Melt-Through

Figure 6.4 shows the vertical penetration of the molten core debris into the concrete base of the containment building. These results were calculated by subroutine INTER (Section 3.0). The results show a high penetration rate (10-20 cm/hr) for the first 10 hr of concrete attack followed by a much lower (0.5-1.0 cm/hr) penetration rate. The penetration rate during the first 10 hr is dominated by the loss of stored heat in the debris. After 10 hr the debris approaches equilibrium temperatures just above the melting points of the metal (∿2450 F) and oxide (∿2100 F) debris layers. The heat loss to the concrete and, hence, the penetration rate become controlled by the decay heat.

The results of the INTER calculations indicate that penetration of the concrete basemat would not occur. However, the models in INTER are not adequate to predict the long-term penetration behavior of the molten core with enough confidence to determine whether or not penetration would eventually result.

Case	Building Coolers	Fraction Clad React	Debris Frag.	Cont. Pressure
A	Off	1.0	Yes	Fig. 6.2-C
B	On	1.0	Yes	Fig. 6.2-B
C	On	0.4	No	Fig. 6.1-B
D	Off	0.4	No	Fig. 6.1-C

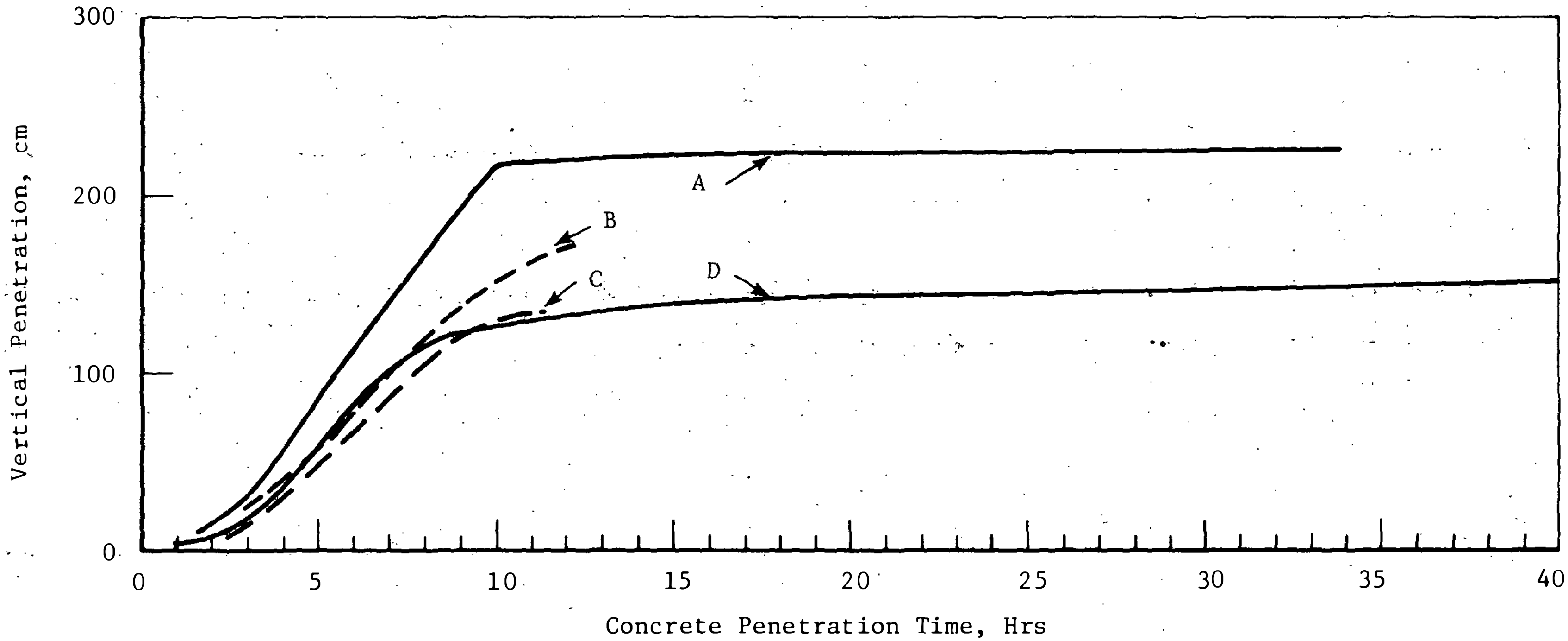

Figure 6.4 Bottom Pad Melt-Through Calculations With Subroutine INTER

Summary

Since the implication of some of the "what if" alternative accident sequences which were examined was that core meltdown could result, a number of core meltdown scenarios were analyzed with the MARCH code. It should be emphasized that the magnitude of release of radioactivity to the environment in a core meltdown accident is very sensitive to the mode and timing of containment failure. As shown in WASH-1400 the most severe consequences occur for an above-ground failure of the containment shortly following core melting. If the integrity of the containment boundary above-ground remains intact or if the failure of the containment is delayed a number of hours, the release of radioactivity to the atmosphere will be reduced by orders of magnitude. This is particularly true for containment designs such as at Three Mile Island 2 in which containment sprays can rapidly reduce the airborne concentration of radioactivity.

The purpose of the analyses that were performed was to explore mechanisms that are believed to represent the greatest threat to containment integrity: rapid steam generation from the quenching of hot fuel debris and the combustion of hydrogen produced in steam-metal reactions. Containment failure by overpressurization resulting from failure of the containment heat removal system was also examined. This mode of failure was found to be delayed significantly in time and the consequences would be much less severe than those associated with early failures, e.g., those due to hydrogen burning. A number of uncertainties regarding the behavior of the system under core meltdown conditions make it impossible to predict with confidence whether or not containment integrity will be retained in a meltdown accident. Some of the important uncertainties that can only be resolved by more research are:

1. The extent of metal-water reaction taking place during the core slumping and reactor vessel meltthrough phases of the accident.
2. The degree of core fragmentation and steam generation when hot fuel interacts with water in the reactor cavity.

3. The magnitude of ignition source required to produce significant burning as a function of atmosphere composition for various hydrogen/air/steam mixtures.
4. The pressure level and mode of failure of reactor containment.

As illustrated in Figure 6.2-B, under the most adverse conditions of metal-water reaction and steam generation in the reactor cavity, the quasi-equilibrium pressure following rapid hydrogen combustion could be well in excess of the estimated failure pressure. Even at pressures below the nominal failure pressure, the combination of the quasi-static loading and potential shock waves generated in an explosion could result in containment failure. In the scenarios considered, most of the hydrogen produced during meltdown is predicted to be rapidly released to containment at the time of reactor vessel failure; thus, the possibility of obtaining a high hydrogen concentration prior to ignition appears to be significant. With our present state of knowledge, it is not possible to rule out the possibility of containment failure shortly following penetration of the reactor vessel head.

7.0 INTERPRETATION OF THERMAL AND HYDRAULIC CONDITIONS

During the first sixteen hours of the accident, a complex sequence of changes occurred in the thermal and hydraulic conditions of the primary system. Some measured data from the accident are available to aid in explaining the behavior of the system. However, to an extent, the data are ambiguous and consistent with a range of possible conditions. Furthermore, the flow rates and other data required to establish the boundary conditions for the problem are not known with sufficient accuracy to predict the thermal and hydraulic behavior well. In the following discussion, a qualitative description of the hydraulic behavior of the primary system is presented. This description is essentially the same as developed by the MARCH computer code for this time period. Although the agreement between the MARCH results and the measured hydraulic and thermodynamic data is good, it should be recognized that the MARCH code provides an over simplified representation of primary system hydraulics. Additionally, variations were made within the range of uncertainty in the boundary conditions to obtain good agreement. For this reason, we have attempted in the reconciliation with measured data to use examples of hand calculations which are more easily understood and checked by the reader than comparison with MARCH calculations. The times reported in the discussion are relative to the beginning of the accident. Most of the event times are known within an uncertainty of a few minutes.

7.1 General Description of Accident Conditions

Because of the failure of the electromatic relief valve to reclose following the initial surge of pressure, reactor coolant was continuously leaked through the valve greatly in excess of the makeup rate for approximately 140 minutes until the block valve (another valve in the same line) was closed. Although the data indicate a high water level was maintained in the pressurizer, the quantity of liquid in the reactor primary system decreased throughout this period. While the reactor coolant pumps were in operation, a mixture of steam and liquid water was pumped through the core effectively cooling it. When the last set of reactor coolant pumps was shut off at 1:41, however, the liquid and steam phases separated, with the liquid phase apparently falling to the level of the top of the core. For the next half hour, steam generated by decay heat in the core was partially released to the pressurizer and out the open valve and partially condensed in the "A" steam generator. The water level in the primary side of the steam generator was not high enough, however, to permit the condensed water to flow back into the reactor vessel to resupply the core. For this reason, the water level in the core continued to drop to approximately four to six feet from the bottom of the core.

At 2:18 the block valve in the relief line was closed and the loss of water from the system was stopped. The water level in the core apparently began to rise slowly over the next half hour at which time one of the reactor coolant pumps in the B loop was turned on for 19 minutes. During the first few minutes of operation sufficient water was pumped to fill the annular downcomer region in the vessel and to force some additional water into the core. Although a part of the core remained uncovered following the operation of the reactor coolant pump, some quenching of the core occurred at this time.

The high pressure injection system was actuated for a few minutes at 3:20, apparently recovering the core. High pressure injection was again actuated at 3:56 for a short time period. Following this time, the core was probably never uncovered again, although some severely damaged regions of the core remained very hot and steam blanketed for approximately four days. The steam released from the hot regions was condensed in water in the upper plenum before reaching the hot legs.

At 4:27, significant makeup flow to the primary system was established from makeup pumps B and C and maintained until 9 hours. The flow through the core during this time period was high enough that all of the decay heat in the core could be removed without boiling the water. After leaving the core, the heated water flowed through the pressurizer and out the relief valve to the containment building. In this time period, the upper portions of the two hot legs and steam generators were blocked to steam flow by hydrogen which had been produced earlier from reaction of steam with zirconium. Because the hot legs and steam generators are well insulated, the temperatures measured at the tops of the hot legs remained nearly constant for a number of hours at approximately 750 F, the temperature to which they had been heated during the period of core uncovery.

Shortly before 16 hours the 1A reactor coolant pump was turned on, establishing flow through the loops and allowing the decay heat from the core to be removed by the steam generator. This concluded the initial transient period.

7.2 Reconciliation with Measured Data

Time Period 0 to 2:18

The source range monitor measures neutron flux external to the reactor vessel. In the first few hours of the TMI accident, the source range monitor provided a measure of the height of water in the annular downcomer outside of the reactor core region. In a sense, the water in the downcomer acted as a shutter which obscured the detector from the source of neutrons in the core. A high reading on the source range monitor should, therefore, be interpreted as a low water level in the downcomer and core. In addition, as the water temperature increases and the density of water in the downcomer decreases, the output of the source range monitor should increase.

Following the throttling of the makeup pumps to a low flow rate at 4 minutes, the water added to the system was not adequate to balance the loss of water discharging through the open relief valve. As the density of fluid (liquid and steam) being pumped through the system decreased, the neutron flux measured external to the reactor vessel by the source range monitor began to increase. At 1:41, when the last operating reactor coolant pumps were shutoff, separation of the two phase mixture of steam and water resulted. In addition, the head of water, which had been maintained above the core by operation of the pump, settled back into the core and downcomer. This effect is seen as a dip in the source range monitor output.

The water level at this time must have been approximately at the top of the core. Within a few minutes, the hot leg temperature readings began to rise rapidly above the saturation temperature of the water indicating that steam had been superheated by uncovered fuel. The source range monitor shows that the water level in the core (and downcomer) continued to decrease until approximately 2:20.

Good heat transfer occurred to the A steam generator from 1:30 to 2:15. During this period the primary system pressure closely followed the saturation temperature of the A steam generator. The water condensed on the primary side of the steam generator was not able to flow back into the reactor vessel, however, until the liquid level reached the elevation of the cold legs.

Time Period 2:18 to 3:20

At 2:18 the block valve in the line of the electromatic relief valve was closed and the pressure in the primary system began to rise. The water level in the core also began to increase during this period as indicated by the decrease in the source range monitor. At 2:54 the reactor coolant pump 2B was activated for 18 minutes. Enough flow occurred during the operation of the pump to fill the downcomer and to partially quench the hot core. Filling of the downcomer can be seen in the behavior of the source range monitor.

The increase in pressurizer level from 2:45 to 3:00 is believed to be the result of condensation in the pressurizer. Assuming that the water in the pressurizer remains saturated, the change in level corresponds to an increase of 2,200 lbs. The mass of steam that could potentially be condensed in heating the water in the pressurizer to saturation at 2100 psia is 12,600 lbs. Thus, condensation can easily account for the observed level increase.

Although the top of the core remained uncovered for a period of time following operation of the 2B reactor coolant pump, the partial quenching that resulted from operation of the pump was probably important in limiting the maximum heatup of the core.

Time Period 3:20 to 5:15

At 3:20 and at 3:56 the high pressure injection system was turned on for a few minutes and then reduced in flow rate. The rapid pressure drop from 2000 to 1500 psia which occurred at 3:20 with the block valve closed is apparently the result of steam in the system flowing through the core barrel check valve and condensing on the emergency core cooling water. As shown by the source range monitor, the core was rapidly refilled following the initial actuation of the high pressure injection system and probably remained covered for the rest of the accident. Refilling of the pressurizer after 3:30 is probably an indication that the water level had increased to the surge line at this time. Some severely damaged regions of the core remained very hot for a number of days following the initial transient as shown by thermocouple reading above the core. Although superheated jets of steam from the damaged regions would be expected to penetrate into the upper plenum, condensation and mixing would have occurred before the fluid reached the hot legs.

A significant event happened at 3:45 which produced a large quantity of steam. A simultaneous rapid increase occurs in the recorded values of reactor pressure, cold leg temperatures and the source range monitor. The EPRI/NSAC group explains this response as the result of the slumping and quenching of embrittled core material.(2) The increase in cold leg temperatures was the result of backflow of steam through the core barrel check valves. The offset in the reading of the source range monitor is interpreted as a change in the fuel configuration.

At 4:27 the makeup flow to the vessel was increased to a level which was able to remove the entire decay heat from the core without boiling. The injected water flowed into the cold legs, through the core, through the hot leg in the A loop to the pressurizer surge line, and out the electromatic relief valve. The temperatures measured in the surge line and in the pressurizer in this time period show that this water was subcooled. The injection rate from the borated water storage tank of 640 gpm reported in NUREG 0600 is probably higher than was typical for this time period because the average includes two periods of high injection rate. However, at 5:45 for a decay heat level of 6.2×10^7 Btu/hr, a flow rate of 640 gpm and an inlet temperature of 110 F, the core outlet temperature would be 300 F. The temperature measured at the pressurizer surge line at this time is 310 F. The agreement is good evidence that the above explaination of the hydraulics in this time period is correct.

Time Period 5:15 to 7:39

At 5:15 the block valve in the line of the electromatic relief valve was closed and the system was repressurized to 2000 psia. The pressure was maintained at this level by a series of openings and closings of the valve. As in the previous time period the core decay heat was removed by the makeup flow passing through the core and out the pressurizer. With the water in the system subcooled, the primary system pressure in this period was determined by the compression of the non-condensible gases trapped in the upper regions of the hot legs and steam generators. Assuming a net makeup flow rate of 565 gpm (based on NUREG 0600) and a perfect gas, a gas volume of 2540 ft^3 (2000 psia) can be inferred from the system pressurization rate during the periods of pressure increase. A possible breakdown of this gas volume could have been: the reactor coolant pump volume (400 ft^3), 1/2 the volume of the cold legs (476 ft^3), 1/2 the volume of the hot legs (469 ft^3), 1/2 the volume of the upper head (254 ft^3) and 500 ft^3 in each steam generator. Although it is difficult to accurately predict the distribution of gas among the different volumes in the primary system, the gas volume inferred from the pressurization rate is reasonable for this time period.

Earlier in the accident when the core was uncovered, some of the hydrogen generated from zirconium water reaction flowed into the hot legs and upper portions of the steam generators. The presence of the hydrogen in the legs effectively blocked the flow of steam from the core to the steam generators. Because the primary system is well insulated (the characteristic thermal decay period for the walls is approximately 150 hours), the hot legs which had been heated to 750-800 F during core uncovery remained hot for a number of hours. Even the flow of subcooled water through the A loop hot leg into the pressurizer surge line was ineffective in cooling the upper portion of the hot leg. The thermal conductance along the pipe is too small to have reduced the wall temperature significantly. Furthermore, the hydraulic regime of hot fluid above cold fluid is thermally stable and would not have induced convective cooling.

Time Period 7:39 to 10:20

At 7:39 the block valve was opened and the system was depressurized. As the volume of gases expanded to the level of the pressurizer surge line some of the hydrogen in the hot legs was probably released through the pressurizer to the containment. At 9:04 the makeup flow rate was decreased and by 10:20 the water temperature in the pressurizer reached saturation. Based upon a decay heat level of 5×10^7 Btu/hr, the net makeup flow (including the discharge of flooding tanks) must have been less than 270 gpm to result in saturated conditions at the core outlet. This is consistent with the operation of one makeup pump in this time period.

Time Period 10:20 to 13:20

While the water in the A loop hot leg was subcooled, there was little concentration gradient to diffuse steam through the hydrogen in the hot legs to condense in the steam generator. A simple one-dimensional analysis of the countercurrent diffusion of steam and hydrogen in the hot leg indicates that the mass flow rate of steam through the hydrogen would be very small. However, with the mass of hydrogen depleted by the preceding depressurization and with the water in the hot leg near saturation, the driving force for diffusion increases. At 10:30 flow of steam began in the A hot leg as can be seen in the trace of the hot leg temperature. The pressure increase on the secondary side of the steam generator also indicates flow in the hot leg at this time.

From 11:00 to 11:20 approximately 640 ft^3 of water appears to have drained from the pressurizer. If the pressurizer level reading is correct, a consistent hydraulic picture must be able to explain where this large quantity of water could have gone. A plausible explanation is that the A loop cold legs and pumps also contained hydrogen following core uncovery. This may also explain the reason the 1A pump could not be operated at 4:10. The existence of hydrogen in the A cold legs would not only make that volume potentially available to receive water flow from the pressurizer, but would also introduce an offset in the water levels between the A steam generator and the hot leg. The maximum possible offset would be 29 ft corresponding to the elevation difference between the cold legs and the bottom of the steam generator. If the hydrogen in the 2A cold leg were relieved to the pressurizer through the open pressurizer spray line, the water in the pressurizer would flow back through the core (or the core barrel check valves) into the cold legs. Equalization of pressure heads would force the hydrogen out of 1A cold leg and would equalize the levels in the A steam generator and hot leg. Since the cross sectional area of the steam generator is greater than that of the hot leg, an additional 15.5 ft^3 of liquid volume becomes available for every foot of change in level. The total volume potentially available in the A loop is therefore the reactor coolant pumps (200 ft^3), the cold legs (476 ft^3), and through equalization of the levels in the steam generator and hot leg (225 ft^3) for a total of 900 ft^3. A fraction of this potentially available volume could therefore have accommodated the liquid drained from the pressurizer. Consistent with this explanation, the temperatures in the two A loop cold legs rose rapidly at 11:15 indicating back flow of hot water into the legs.

The rate of rise of the pressurizer level at 11:30 is consistent with a net makeup rate of 80 gpm which is within a factor of two of the average makeup rate in this time period from NUREG 0600.

Time period 13:20 to 16:00

At 13:23, the makeup flow rate was again increased. Since the block valve was closed, the injected water acted to pressurize the non-condensible gas regions in the hot legs. As expected, pressurizer temperature during this time period became subcooled as a result of the pressurization of the non-condensible gases. The temperature did not decrease, however, because with the block valve closed, the flow leaving the core would not enter the pressurizer. At 15:50, the 1A pump was operated resulting in the flow of water through the A hot leg to the steam generator and effectively terminating the transient.

8.0 ANALYSIS OF HYDROGEN BURNING DURING THE TMI-2 ACCIDENT

On March 28, 1979, approximately 9 hours and 50 minutes after the turbine trip that initiated the TMI-2 accident a pressure spike of approximately 28 psig was observed within the containment building. The cause of this pressure rise is believed to be hydrogen burning. The purpose of this analysis is to assess the nature and extent of hydrogen burning required to produce such a pressure increase and to consider its potential implications.

8.1 Consideration of the Observations at TMI-2

For the purpose of this analysis the containment atmosphere conditions prior to the turbine trip were taken to be 14.7 psia and 120°F, with a relative humidity of 50 percent. The containment free volume is given as 2.1×10^6 ft^3 in the TMI-2 FSAR. These values lead to an initial dry air inventory in the containment of 4,675.5 lb-moles; this was used as the starting point of the subsequent analyses. The composition of the dry air was taken as 21 v/o O_2 and 79 v/o N_2.

A key question with regard to the pressure spike is its spacial extent, i.e., did the entire containment volume experience this pressure rise or was it localized. The 28 psig pressure spike was measured by the reactor containment building pressure monitors as well as being reflected in the reference pressures of both the steam generators. Since the two steam generators are widely separated it may be inferred that the 28 psig pressure rise was indeed seen throughout the containment volume. This need not necessarily preclude the existence of locally higher pressures.

Just prior to the time of the pressure spike inside the containment the temperature and pressure were measured to be about 128°F and 1.5 psig, respectively. This temperature may possibly not be representative of the entire containment atmosphere, though it was assumed to be so in this analysis. Assuming the quantity of air to be unchanged from that initially in the containment and keeping the total pressure at 1.5 psig, various quantities of hydrogen were added to the atmosphere; as the amount of hydrogen was increased the quantity of water vapor was decreased, since the total moles of gas are constant. The pressure inside the containment was then calculated for a number of such atmosphere compositions, assuming uniform mixing and constant volume, adiabatic burning of all the hydrogen. From these calculations it was determined that the combustion of 564 lb of hydrogen would be required to raise the pressure to 42.7 psia (28 psig). The inferred composition of the atmosphere just prior to the hydrogen burn is 86.7 v/o air, 5.2 v/o hydrogen, and 8.1 v/o water vapor. This composition is on the edge of the flammability region and well away from commonly accepted detonable limits.

The first post accident analyses of the containment building atmosphere were performed on March 31, 1979, and gave the following average results on a dry gas basis: 1.7 v/o hydrogen, 16.1 v/o oxygen, and 82.2 v/o nitrogen. Using the results of these analyses together with the previously determined initial air inventory, an oxygen depletion of 258.5 lb-mole is determined. This corresponds to the burning of 517.0 lb-mole (1034 lb) of hydrogen. If this quantity of hydrogen as well as that remaining at the time of the analyses were present in the containment atmosphere just prior to the bydrogen burn, the inferred atmosphere composition would have been 86.7 v/o air, 11.0 v/o hydrogen, and 2.3 v/o water vapor. This composition is well into the flammable region but not near the accepted detonable limits. If the quantity of hydrogen burned determined from the oxygen depletion analyses is assumed to be uniformly distributed throughout the containment volume and undergoes constant volume burning the resulting containment pressure would be about 62 psia. This is, of course, substantially higher than the measured peak pressure.

The foregoing analyses were based on the assumption of uniform hydrogen distribution throughout the containment volume. There is, of course, a wide variety of inhomogenous hydrogen-air distributions possible. Some limiting cases were considered and are discussed below. If all the hydrogen that burned was concentrated in a localized volume together with the stoichiometric quantity of air required for the reaction, this hydrogen-air "bubble" would only occupy some fraction of the total containment volume.

Using the hydrogen burned as determined from the oxygen depletion as the reference, this bubble was found to occupy about one-third of the total volume. If the hydrogen-air bubble is assumed to burn at constant volume and then expand into the cold or unburned gas until the pressure throughout containment equalizes, the resulting pressure is 46 psia. The pressure in the reacted gas prior to expansion is 127 psia; this pressure would not be seen by the entire containment, however. The foregoing assumptions of constant volume burning followed by adiabatic expansion were made for analytical convenience; physically, these conditions could only be approached if the hydrogen-air mixture underwent detonation. In a deflagration the pressures throughout the containment volume would remain equalized. We have not calculated the detonation parameters for the above configuration; based on similar analyses in the past, however, we would expect that the Chapman-Jouget pressure, if detonation did take place, would be approximately twice the above quasi-equilibrium pressure in the burned gas. While the above highly idealized approximation to the burning of a non-homogeneous hydrogen-air mixture may or may not represent what actually took place, the calculated final pressure is sufficiently close to the measured value to lend plausibility to an explanation of this type.

The quantity of hydrogen burned as determined from the measured oxygen depletion is substantially at variance with that required to raise the containment pressure to 28 psig under homogeneous burning assumptions. A possible scenario for reconciling the high quantity of hydrogen burned as indicated by the oxygen depletion with the observed pressure increase was presented. Some further comments relating to this apparent dichotomy are noted below.

If the quantity of hydrogen burned was limited to that required to produce the observed pressure increase under the assumption of uniform distribution, i.e., 564 lb, then the inferred containment atmosphere just prior to the burn would have been: 86.7 v/o air, 5.2 v/o hydrogen, and 8.1 v/o water vapor. Such a composition would be on the borderline of the flammability region and, if ignited, would not be expected to result in complete reaction of the hydrogen. If the residual hydrogen as indicated by the March 31 containment atmosphere analyses were included, the resulting atmosphere composition would be: 86.7 v/o air, 6.8 v/o hydrogen, and 6.5 v/o water vapor. This composition is still near the edge of the flammability region and would not be expected to result in a complete reaction; hydrogen concentrations in excess of about 8 v/o are believed to be required to approach complete reaction. While there is undoubtedly some containment atmosphere composition which would result in the burning of the above quantity of hydrogen, such a composition would require a significantly greater initial quantity of hydrogen than the above 564 lb and imply a residual hydrogen content susbstantially greater than that measured.

The quantity of hydrogen burned as inferred from the oxygen depletion analyses is substantially greater than the quantity required to explain the measured containment pressure increase. This discrepancy can apparently be explained by assuming a nonuniform distribution of the hydrogen. Implicit in this explanation are the following:

1) The measured 28 psig pressure increase existed throughout the containment volume.
2) Much higher pressures could have existed locally.
3) If such higher pressures did indeed take place, the instrumentation that recorded the containment pressure response either did not experience these localized pressures or was not capable of recording them.

8.2 Effect of Containment Design

Consideration has also been given to the effect of containment design on the possible implications of hydrogen burning of the magnitude occurring during the TMI-2 accident, i.e., the same hydrogen burning event was assumed to take place in different containment designs. In this way some insight could be developed on the relative vulnerability of different containment designs to this type of accident.

The principal threat to the TMI-2 containment was associated with the 28 psig pressure spike from hydrogen burning. At all other times during the course of the accident the containment pressure remained below 5 psig.

Since the design pressure of the containment structure is 60 psig, no real threat to the integrity of the containment is believed to have existed at any time during the accident.

Table 8-1 gives some of the key characteristics for a selection of designs representative of the spectrum of containments used for large commercial reactors in this country. Examination of the characteristics of large dry containments indicates that these are comparable to the TMI-2 containment; thus, it would be expected that other containments of this type would not have been threatened by the hydrogen burning experienced at TMI-2.

The containments classified as being of the pressure suppression type are characterized by low design pressure or small free volume. Some of these may be more vulnerable to damage due to hydrogen burning than are the large dry containment types. Each type of pressure suppression containment is discussed below.

The ice condenser containment design is included in one of several reactors being evaluated in the Reactor Safety Study Methodology Applications Program. For the specific ice condenser containment considered in this program a nominal failure pressure of 45 psia has been determined. This failure pressure was based on static loading conditions. As applied in the Reactor Safety Study Methodology Applications Program, at the nominal failure pressure there is a 50 percent probability of failure. This probability increases above the nominal level and decreases below it. If the 28 psig pressure rise observed at TMI-2 were applied to the ice condenser containment, a significant likelihood of failure would be expected. If the 1034 lb of hydrogen burned, as determined from the oxygen depletion analyses, were uniformly distributed and burned in this ice condenser containment, a pressure of 96 psia would be predicted. At this level, failure of the structure would have to be considered a virtual certainty.

A Mark I BWR containment was one of the two designs evaluated in the Reactor Safety Study. That particular design had a design pressure of 56 psig and a predicted nominal failure pressure of 160 psig (175 psia). That containment was inerted, thus hydrogen burning was not a consideration

TABLE 8.1. TYPICAL CONTAINMENT DESIGN PARAMETERS

Containment Type	Example Plant	Free Volume (ft^3)	Design Pressure (psig)
Large Dry Containment			
- Prestressed Concrete	TMI-2	2×10^6	60
- Free Standing Steel	St. Lucie	2.5×10^6	44
- Sub-atmospheric, reinforced concrete	Surry	1.8×10^6	45
- Spherical Steel Shell	Perkins	3.3×10^6	47
Pressure Suppression			
- Ice Condenser	Sequoyah	1.2×10^6	12
- BWR Mark I	Peach Bottom	2.8×10^5	56
- BWR Mark II	Zimmer	3.9×10^5	55
- BWR Mark III	Grand Gulf	1.7×10^6	15

in the Reactor Safety Study analysis for this plant. In what follows it is assumed that the atmosphere of the BWR Mark I containment is not inerted. Due to the small free volume of this type of containment the quantity of hydrogen that could be burned will be limited to about 280 lb-mole (560 lb) by the amount of oxygen available. This is roughly half the quantity of hydrogen burned in TMI-2, as inferred from the oxygen depletion analysis. Assuming an initial pressure of about one (1) atmosphere, the burning of this amount of hydrogen would produce a pressure of 120 - 130 psia in the Mark I containment. At this level only a small probability of failure would be predicted. However, as the initial pressure in the containment is increased, due to further addition of hydrogen or other factors, the final pressure from hydrogen burning could reach and exceed the predicted failure level (even though the quantity of hydrogen burned is the same as previously). Thus, by virtue of its higher expected failure pressure, the Mark I containment could have some margin of protection from hydrogen burning, but this margin could be removed as a result of potentially higher initial pressures arising from the small free volume.

The BWR Mark II containment design is characterized by a somewhat larger free volume and a similar design pressure to the Mark I design. It is constructed of pre-stressed concrete rather than steel as in the Mark I. As in the Mark I, the quantity of hydrogen that could be burned in the Mark II containment would be limited by the available oxygen to less than the quantity inferred to have burned in the TMI-2 accident. For an assumed initial pressure of about one atmosphere the burning of a stoichiometric air-hydrogen mixture would lead to pressures of 120 - 130 psia. While no specific analyses have been performed for the Mark II containment, these pressures would be expected to be near the expected failure pressure. With increasing initial pressure the resulting final pressure would also increase. Thus it is suggested that the Mark II design could be vulnerable to hydrogen burning of the magnitude experienced at TMI-2, particularly since these designs are not normally inerted.

The BWR Mark III design is roughly comparable in free volume and design pressure to the ice condenser design. This comparability in characteristics suggests that the Mark III containment would have a similar vulnerability to hydrogen burning; that is, it appears unlikely that the Mark III containment would survive hydrogen burning similar to that occurring in TMI-2.

The above observations on the potential vulnerability to hydrogen burning of various containment designs were limited to consideration of the extent and characteristics of burning as they have been inferred for the TMI-2 accident. This is not meant to suggest that substantially different burning events are impossible. Clearly greater as well as lesser quantities of hydrogen generation can be postulated; also, other modes of burning are possible. Also, this discussion has not touched on other potential modes of containment falure; these may be either more or less likely, as well as more or less significant than hydrogen burning.

9.0 REFERENCES

(1) "Investigation Into the March 28, 2979 Three Mile Island Accident by Office of Inspection and Enforcement", NUREG-0600 (August, 1979).

(2) "Analysis of the Three Mile Island Unit 2 Accident", NSAC-1 (July, 1979).

(3) "Reactor Safety Study, Appendix VIII, "Physical Processes in Reactor Meltdown Accidents", WASH-1400 (August, 1974). Also, Wooton, R. O., "BOIL 1, A Computer Program to Calculate Core Heatup and Meltdown in a Coolant Boiloff Accident", Battelle's Columbus Laboratories (March, 1975).

(4) Murfin, W. B., "A Preliminary Model for Core/Concrete Interaction", Sandia Laboratories, SAND77-0370 (August, 1977).

(5) Cathcart, J. V., "Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program Sponsored by the NRC Division of Reactor Safety Research for Oct-Dec 1976", Oak Ridge National Laboratory, ORNL-NUREG-TM-87.

(6) Baker, L., and Just, L. C., "Studies of Metal-Water Reactions at High Temperatures III. Experimental and Theoretical Studies of the Zirconium-Water Reaction", Argonne National Laboratory, ANL-6548 (May, 1962).

(7) Peeks, M., Hassmann, K., and Hagen, S., "Analysis of a Hypothetical Core Meltdown Accident of a Pressurized-Water Reactor", Siemens Forsch.-u Entwickl.-Ber. Bd 8, No. 2, pp 114-121 (1979).

(8) Wilson, J. F., et al., "The Velocity of Rising Steam in a Bubbling Two Phase Mixture", ANS Transactions, Vol. 5, p 151 (June, 1962). Also, see Slifer, B. G., and Hench, J. E., "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors", p B-12, NEDO-10329 (April, 1971).

(9) "The Reactor Handbook, Engineering", Vol. 2, AECD-3649, p 77 (May, 1955).

(10) Letter from L. S. Tong to M. Cunningham, Attachment III, "Best Estimate Initial and Boundary Conditions for TMI-2 Analyses", US NRC (July 26, 1979).

(11) Marino, G. P., and Marks, J. M., "Calculation of Fuel Rod Temperatures Reached in the Three Mile Island-2 Incident", file memorandum, Fuel Behavior Research Branch, NRC (October 25, 1979).

(12) Krieth, F., "Principles of Heat Transfer", 3rd ed., p 280, Intext Publishers (1973).

(13) McAdams, W. H., "Heat Transmission", 3rd ed., McGraw-Hill Book Co. (1954).

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG/CR-1219
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)* Analysis of the Three Mile Island Accident and Alternative Sequences.	2. *(Leave blank)*
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) R. O. Wooton, R. S. Denning, and P. Cybulskis	5. DATE REPORT COMPLETED MONTH December YEAR 1979
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Battelle, Columbus Laboratories 505 King Avenue Columbus, Ohio 43201	DATE REPORT ISSUED MONTH January YEAR 1980
	6. *(Leave blank)*
	8. *(Leave blank)*
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission Washington, D. C. 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO. NRC-04-76-293-08
13. TYPE OF REPORT Topical Report	PERIOD COVERED *(Inclusive dates)*
15. SUPPLEMENTARY NOTES	14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

Analyses were performed with the MARCH computer code to examine a number of variations in system operation in the TMI accident to evaluate their effect on the extent of core damage. The results indicate that:

1) The throttling of HPI had a major effect on core damage. If the system had been permitted to operate at high flow, the core would not have been uncovered.
2) Earlier closure of the block valve would have prevented core damage. An additional delay of one hour in closing the valve would have resulted in severe core damage and possibly core meltdown.
3) The delay in operation of the emergency feedwater system had little effect on the extent of core damage. However, a delay of one hour in the delivery of emergency feedwater would probably have resulted in more severe core damage and possibly core meltdown.

In addition to the analysis of alternative accident sequences: a consistent interpretation of the thermal and hydraulic behavior during the first sixteen hours has been developed; potential containment failure modes for accidents involving complete core meltdown have been examined; and the vulnerability of different types of containment designs to hydrogen explosion has been evaluated.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
Three Mile Island 2 Core Meltdown Hydrogen	

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19. SECURITY CLASS *(This report)*	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS *(This page)*	22. PRICE S

NRC FORM 335 (7-77)

*U.S. GOVERNMENT PRINTING OFFICE : 1980 0-308-887/6764

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

U.S. MAIL