JAERI-M
91-193

THERMAL-HYDRAULIC ANALYSIS OF THE THREE MILE ISLAND UNIT 2 REACTOR ACCIDENT WITH THALES CODE

October 1991

Kazuichiro HASHIMOTO and Kunihisa SODA

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

JAERI-M レポートは，日本原子力研究所が不定期に公刊している研究報告書です。
入手の問合わせは，日本原子力研究所技術情報部情報資料課（〒319-11茨城県那珂郡東海村）あて，お申しこしください。なお，このほかに財団法人原子力弘済会資料センター（〒319-11 茨城県那珂郡東海村日本原子力研究所内）で複写による実費頒布をおこなっております。

JAERI-M reports are issued irregularly.
Inquiries about availability of the reports should be addressed to Information Division Department of Technical Information, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken 319-11, Japan.

編集兼発行　日本原子力研究所
印　　刷　いばらき印刷㈱

Thermal-hydraulic Analysis of the Three Mile Island Unit 2 Reactor Accident with THALES Code

Kazuichiro HASHIMOTO and Kunihisa SODA

Department of Fuel Safety Research
Tokai Research Establishment
Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken

(Received October 14, 1991)

The OECD Nuclear Energy Agency (NEA) has established a Task Group in the Committee on the Safety of Nuclear Installations (CSNI) to perform an analysis of Three Mile Island Unit 2 (TMI-2) accident as a standard problem to benchmark severe accident computer codes and to assess the capability of the codes. The TMI-2 Analysis Exercise was performed at the Japan Atomic Energy Research Institute (JAERI) using the THALES (Thermal-Hydraulic Analysis of Loss-of-Coolant, Emergency Core Cooling and Severe Core Damage) - PM1/TMI code. The purpose of the analysis is to verify the capability of THALES-PM1/TMI code to describe accident progression in the actual plant. The present paper describes the final result of the TMI-2 Analysis Exercise performed at JAERI.

Keywords: TMI-2, Thermal-hydraulic, THALES, Analysis Exercise, Severe Accident, Safety, Hydrogen, PORV

THALESコードによるTMI－2事故熱水力挙動解析

日本原子力研究所東海研究所燃料安全工学部
橋本和一郎・早田　邦久

(1991年10月14日受理)

TMI－2事故を標準問題として扱い，これを各国で開発されているシビアアクシデント解析コードで解析し，コード間の比較・評価を行うためのタスクグループがOECD/NEA/CSNIに設置された。原研は，このタスクグループに参加し，THALES－PM1/TMI コードを用いて解析を行った。この解析の目的は，実炉の事故進展に対する同コードの適用性を確認することである。本報告は，原研で行ったTMI－2標準問題の最終結果をまとめたものである。

東海研究所：〒319-11　茨城県那珂郡東海村白方字白根2-4

Contents

目　　次

I. Introduction

The OECD Nuclear Energy Agency (NEA) has established a Task Group in the Committee on the Safety of Nuclear Installations (CSNI) to perform an analysis of Three Mile Island Unit 2 (TMI-2) accident as a standard problem to benchmark severe accident computer codes and to assess the capability of the codes. The TMI-2 Analysis Exercise was performed at the Japan Atomic Energy Research Institute (JAERI) using the THALES (Thermal-Hydraulic Analysis of Loss-of-Coolant, Emergency Core Cooling and Severe Core Damage) - PM1/TMI code. The THALES-PM1/TMI code is a modified version of the THALES-PM1[1)] developed at JAERI for the analysis of core meltdown accidents of light water reactors. The purpose of the analysis is to verify the capability of THALES-PM1/TMI code to describe accident progression in the actual plant.

In the analysis, the initial and boundary conditions were based on the TMI-2 Standard Problem data base[2,3)] which was used by the OECD/NEA/CSNI in performing the TMI-2 Analysis Exercise. All components in the primary cooling system were modeled in the calculation, but the secondary side was modeled by one volume for each loop. Major parameters calculated include the primary system pressure, the coolant mass flow rate through the pilot-operated relief valve (PORV), the coolant levels in each component of the reactor cooling system (RCS), the fuel temperature at different radial and axial positions in the core, the hydrogen generation rate, and the heat transfer in the steam generator.

The object of the TMI-2 Analysis Exercise was the first 300 minutes of the TMI-2 accident in which most of important phenomena concerning thermal-hydraulic and core degradation took place. This term was divided into four phases and the Analysis Exercise was performed by participants step by step from phase 1 to phase 4. The analysis result for the initial two phases performed at JAERI using THALES-PM1/TMI was published in Nuclear Technology.[4] The present paper describes the final result of the TMI-2 Analysis Exercise covering phases 1 to 4 performed with THALES-PM1/TMI code.

II. Code Description

The THALES-PM1/TMI code is a modified version of the THALES-PM1[1] which is part of THALES code system. The THALES code system was developed to describe the physical processes governing the progression of core meltdown accidents, including initial blowdown, core heatup and meltdown, pressure vessel melt-through, debris/concrete interaction, and containment failure.

In the THALES code system, the THALES-PM1 calculates the thermal-hydraulic behavior in the primary cooling system, core heatup, melt-progression in the pressure vessel and melt-through of the vessel for pressurized water reactors (PWRs).

THALES-PM1/TMI includes a system model, a hydraulic model, and mass and heat transfer model. The primary cooling system of a PWR is modeled by control volumes and junctions. Each control volume is divided into gas and liquid regions. It is assumed that the pressure in the primary system is uniform and that thermal equilibrium is maintained in each control volume. THALES-PM1 minutes of the TMI-2 accident in which most of important phenomena concerning thermal-hydraulic and core degradation took place. This term was divided into four phases and the Analysis Exercise was performed by participants step by step from phase 1 to phase 4. The analysis result for the initial two phases performed at JAERI using THALES-PM1/TMI was published in Nuclear Technology.[4] The present paper describes the final result of the TMI-2 Analysis Exercise covering phases 1 to 4 performed with THALES-PM1/TMI code.

II. Code Description

The THALES-PM1/TMI code is a modified version of the THALES-PM1[1] which is part of THALES code system. The THALES code system was developed to describe the physical processes governing the progression of core meltdown accidents, including initial blowdown, core heatup and meltdown, pressure vessel melt-through, debris/concrete interaction, and containment failure.

In the THALES code system, the THALES-PM1 calculates the thermal-hydraulic behavior in the primary cooling system, core heatup, melt-progression in the pressure vessel and melt-through of the vessel for pressurized water reactors (PWRs).

THALES-PM1/TMI includes a system model, a hydraulic model, and mass and heat transfer model. The primary cooling system of a PWR is modeled by control volumes and junctions. Each control volume is divided into gas and liquid regions. It is assumed that the pressure in the primary system is uniform and that thermal equilibrium is maintained in each control volume. THALES-PM1 calculates a system pressure, mixture levels, gas and liquid temperatures, and flow rates between control volumes.

The core heatup model in THALES-PM1/TMI treats the one-dimensional heat transfer along the fuel rod, debris, and coolant. Fuel rods within the core are divided into several groups, and each fuel rod is further divided into several axial segments. For the fuel segments below the mixture level, the heat transfer to the coolant is calculated neglecting the temperature rise of the coolant. Above the mixture level, the heat transfer between the fuel segments and the steam is considered for each fuel rod group. The steam is distributed to each fuel rod group so as to be proportional to the number of fuel rods in each group.

THALES-PM1/TMI incorporates the Baker-Just model for the metal-water reaction after the initiation of core uncovery. Since the Baker-Just model gives a conservative value, the present calculation may overpredict the rate of temperature rise and hydrogen generation. Concerning the initiation of melt progression, THALES-PM1/TMI assumes two different failure criteria: fuel damage due to melting and fuel fragmentation due to quench.

To simulate the TMI-2 accident, the original THALES-PM1 code is limited in the heat transfer and pump models. In the heat transfer model, THALES-PM1 assumes a fixed value for the heat transfer coefficient for the gas and liquid phases in the core and in the steam generator. However, in the TMI-2 accident the heat transfer mechanism was more complicated. For the pump model, the original THALES-PM1 code always assumes phase separation in each volume even when the reactor cooling pump is operated. However, in the TMI-2 accident, it was estimated that the core was covered with two-phase mixture while the pump was operated.

Modifications to the THALES-PM1 code were made in the heat transfer and pump models to generate THALES-PM1/TMI. In THALES-PM1/TMI these two models have the following characteristics:

1. Heat transfer model. THALES-PM1/TMI employs the heat transfer model in which the heat transfer coefficient between the fuel rods and the coolant or the steam in the core or between the primary and the secondary side in the steam generator is determined from correlations used in RELAP4/MOD5[5] as shown in Table I.

2. Pump model. The pump model in THALES-PM1/TMI takes the two-phase mixing effect into account except for the pressurizer. When the pump is in operation, a quasi-steady state momentum balance is assumed in each loop and the loop flow rate is determined.

III. Assumptions

Figure 1 shows the system model of TMI-2 by THALES-PM1/TMI. The primary system consists of a core, downcomers, a pressurizer, hot legs, and cold legs. The secondary system is modeled by one volume for each of the A- and B-loops, respectively. In this figure, the cold leg is defined as single volume including pump suction and the primary side of the steam generator.

each volume even when the reactor cooling pump is operated. However, in the TMI-2 accident, it was estimated that the core was covered with two-phase mixture while the pump was operated.

Modifications to the THALES-PM1 code were made in the heat transfer and pump models to generate THALES-PM1/TMI. In THALES-PM1/TMI these two models have the following characteristics:

1. Heat transfer model. THALES-PM1/TMI employs the heat transfer model in which the heat transfer coefficient between the fuel rods and the coolant or the steam in the core or between the primary and the secondary side in the steam generator is determined from correlations used in RELAP4/MOD5[5] as shown in Table I.

2. Pump model. The pump model in THALES-PM1/TMI takes the two-phase mixing effect into account except for the pressurizer. When the pump is in operation, a quasi-steady state momentum balance is assumed in each loop and the loop flow rate is determined.

III. Assumptions

Figure 1 shows the system model of TMI-2 by THALES-PM1/TMI. The primary system consists of a core, downcomers, a pressurizer, hot legs, and cold legs. The secondary system is modeled by one volume for each of the A- and B-loops, respectively. In this figure, the cold leg is defined as single volume including pump suction and the primary side of the steam generator.

The fuel rods are equally divided into three groups: central, intermediate, and peripheral. Each fuel rod is axially divided into 24 nodes, and the peaking factor is given for each axial node. Therefore, the core is divided into 72 regions, and the heat transfer between the cladding and the coolant is calculated at each of the 72 regions.

It was assumed that a node in the core becomes a debris node as the temperature of the node reaches 2800^{o}C. Before reaching this temperature it was assumed that cladding melting took place at 1800^{o}C. THALES-PM1/TMI considers seven relocation mechanisms, as shown in Fig. 2. In the present calculation, we assumed that the debris node moves down to the adjacent lower node (relocation pattern 2 in Fig. 2). This assumption is based on the final core configuration obtained through the core bore examination of the TMI-2 in which the large debris region that is supported by standing rods was formed in the middle of the core, as shown in Fig. 3.[6)]

Concerning the cladding/steam interaction, we assumed that one-tenth of the steam generated in the core interacts with zirconium. This assumption is based on the fact that THALES-PM1/TMI does not consider the effect of steam starvation due to the blockage in the core. If the core blockage is significant, the steam starvation effect should be considered by specifying the value of the interaction rate between the steam and the cladding. Although the percentage of steam that interacts with zirconium should be changed as the core geometry changes, the present version of the code does not have such capability. Thus, the value of 10% was assumed to be an average value for the nominal case in the present analysis considering the steam starvation due to core geometry change. The discharge coefficient for the PORV of the pressurizer was specified as 0.7.

Initial and boundary conditions were based on the TMI-2 Standard Problem data base.[7,8] All the data obtained during the TMI-2 accident were collected and qualified in the data base. However, some data such as makeup flow rate were greatly uncertain. Therefore, in such a case, the best-estimate value was used when data were given in a certain range. For the makeup/high-pressure injection (HPI) and letdown flow rates, time dependent values shown in Figs. 4 and 5 were used as boundary conditions that were given in the Standard Problem data base as the best estimate values. For the auxiliary feed water injection rate which was not recorded during the accident, the best estimate value was used for phases 2 to 4, but the value was modified for phase 1. The reason for the modification is that THALES-PM1/TMI does not have the capability to simulate upper part injection of the once-through steam generator (OTSG). The modified auxiliary feed water flow rate in phase 1 shown in Figs. 6 and 7 by broken line was determined to realize the actual primary system pressure history during early phase of the accident.

IV. Results and Discussions

Major parameters calculated include primary system pressure, coolant mass flow rate through the PORV, coolant levels in each component of the RCS, fuel temperatures at different radial and nominal case in the present analysis considering the steam starvation due to core geometry change. The discharge coefficient for the PORV of the pressurizer was specified as 0.7.

Initial and boundary conditions were based on the TMI-2 Standard Problem data base.[7,8] All the data obtained during the TMI-2 accident were collected and qualified in the data base. However, some data such as makeup flow rate were greatly uncertain. Therefore, in such a case, the best-estimate value was used when data were given in a certain range. For the makeup/high-pressure injection (HPI) and letdown flow rates, time dependent values shown in Figs. 4 and 5 were used as boundary conditions that were given in the Standard Problem data base as the best estimate values. For the auxiliary feed water injection rate which was not recorded during the accident, the best estimate value was used for phases 2 to 4, but the value was modified for phase 1. The reason for the modification is that THALES-PM1/TMI does not have the capability to simulate upper part injection of the once-through steam generator (OTSG). The modified auxiliary feed water flow rate in phase 1 shown in Figs. 6 and 7 by broken line was determined to realize the actual primary system pressure history during early phase of the accident.

IV. Results and Discussions

Major parameters calculated include primary system pressure, coolant mass flow rate through the PORV, coolant levels in each component of the RCS, fuel temperatures at different radial and axial positions in the core, and hydrogen generation rate.

Figure 8 shows the primary system pressure calculated by THALES-PM1/TMI compared with the actual data. The overall trend is in reasonable agreement between the calculated value and the actual data. In phase 1 (1-100 min), the calculated value slightly underestimated the actual data. The difference between calculated and actual data becomes a little bit larger in the first half of phase 2 (100-126 min). However the calculated value well agrees with the actual data in the second half of phase 2 (126-174 min) in which the fuel temperature escalated due steam/cladding reaction after core uncovery. The sudden increase of the pressure at 174 min (beginning of phase 3), due to the B-loop transient is realized in the calculation. In phase 4 (200-300 min), the calculated value gradually underestimates the actual data. This underestimation might be due to underestimation of heat transfer from the debris to the coolant in the calculation after an extensive core degradation.

Figure 9 shows the release rate of coolant from the PORV of the pressurizer and its integrated value. Since the PORV is located at the top of the pressurizer, the coolant release rate from the PORV largely depends on the water level in the pressurizer. Since the water level was full in the pressurizer from 0 to 100 min, the coolant release rate from the PORV largely depended on the primary system pressure. The maximum coolant release rate was calculated by a critical flow model in which the Moody's correlation was applied. From 50 to 100 min, the coolant release rate was almost constant, because the primary system pressure was constant and the pressurizer was full of coolant.

After 100 min, the coolant flow rate significantly decreased due to a rapid decrease of the water level in the pressurizer. The amount of water released from the PORV during phases 1 and 2 was calculated to be 1.11 x 10^5 kg; this agrees with the best-estimate value in the TMI-2 Standard Problem data base, i.e., 1.03 to 1.10 x 10^5 kg. At 139 min, the PORV was closed in the accident, and it resulted in termination of coolant release. The PORV was again opened during 191 - 194 min and 197 - 198 min in phase 3, and during 220 - 260 min and 276 - 318 min in and after phase 4. These operation were taken into account in the calculation with input. As a result of these PORV operation, additional coolant release occurred in phases 3 and 4, as shown in Fig. 9.

Figure 10 shows the water level in the core and in the pressurizer. The core uncovery occurred after 100 min in the present calculation. The water level in the core gradually decreased during phase 2 until three-forths of the core was exposed to the steam phase at the end of phase 2. This minimum water level coincides with the length of the standing rods on the bottom of the core of the TMI-2 reactor. The water level in the core recovered due to high-pressure injection after during phase 4. The water level in the pressurizer during phase 2 was <50% of the full level due to the suspension of the reactor cooling pumps in both of A- and B-loops. In the calculation, a dryout in the pressurizer occurred at 167 min, though such dryout did not occur in the TMI-2 accident. This discrepancy may be due to the failure in simulating the flow in the surge line of the pressurizer. The water level in the pressurizer also recovered during phase 4.

Figure 11 shows the fuel temperature at the upper, middle, and lower part of the rods in the central region. The upper part of the fuel rods in the central region reached 2800^{o}C at 155 min into the accident. In THALES-PM1/TMI, a fuel node is identified as a "debris node" after the temperature of the node reached the specified dissolution temperature, i.e., 2800^{o}C in this calculation. Therefore, the debris node in this code includes both of molten and solid material formed by the cooling of molten material. In the code, the debris node is treated as uniform material that is composed of UO_2 and zirconium. The material property of the debris node is determined in the code from the enthalpy of materials using a temperature-enthalpy curve and the weight ratio of components. The upper node in the central region became a debris node after 155 min. At the end of phase 2, the upper 32 nodes of a total of 72 nodes became debris nodes in this calculation. As noted in Sec. III, it was assumed that debris nodes moved down to the adjacent lower node. A comparative calculation made by assuming that debris nodes fell to the bottom of the pressure vessel resulted in a violent pressure increase in primary system after debris formation that was not recorded in the actual data. This was due to much larger heat transfer from debris to water in the core.

Figure 12 shows the hydrogen generation rate and the integrated hydrogen production from 0 to 200 min obtained in the present analysis. In the calculation most of the hydrogen was generated between 130 and 180 min into the accident. The total amount of hydrogen was calculated to be 460 kg. In the past, a variety of values for total hydrogen generation during the TMI-2 accident have been reported by different analyst. The present analysis was in close agreement with the value obtained by Henrie and Postma[9] as the total amount of hydrogen generated. However the hydrogen generation during the B-loop transient estimated by Henrie and Postma is not modeled in the present version of THALES-PM1/TMI.

V. Conclusions

In the present analysis, the auxiliary feed water injection rate during phase 1 in the TMI-2 Standard Problem data base was modified. Since the primary side of the OTSG during phase 1 was either liquid or two phase flow condition, the auxiliary feed water flow rate has a large effect on the thermal-hydraulic condition in the primary system. The auxiliary feed water injection rate during phases 2 to 4 was not modified. The calculated thermal-hydraulic behavior with the modified auxiliary feed water injection rate was reasonable in comparison with the actual data and post accident data. Concerning the hydrogen generation, the present result gave reasonable value of total amount of hydrogen generated.

The following conclusions were obtained from the present analysis.

1. The analytical results generally agrees with the actual behavior, indicating that the physical models employed in the code are reasonable.
2. The feed water injection rate which was not actually

accident have been reported by different analyst. The present analysis was in close agreement with the value obtained by Henrie and Postma[9] as the total amount of hydrogen generated. However the hydrogen generation during the B-loop transient estimated by Henrie and Postma is not modeled in the present version of THALES-PM1/TMI.

V. Conclusions

In the present analysis, the auxiliary feed water injection rate during phase 1 in the TMI-2 Standard Problem data base was modified. Since the primary side of the OTSG during phase 1 was either liquid or two phase flow condition, the auxiliary feed water flow rate has a large effect on the thermal-hydraulic condition in the primary system. The auxiliary feed water injection rate during phases 2 to 4 was not modified. The calculated thermal-hydraulic behavior with the modified auxiliary feed water injection rate was reasonable in comparison with the actual data and post accident data. Concerning the hydrogen generation, the present result gave reasonable value of total amount of hydrogen generated.

The following conclusions were obtained from the present analysis.

1. The analytical results generally agrees with the actual behavior, indicating that the physical models employed in the code are reasonable.
2. The feed water injection rate which was not actually

recorded was modified in phase 1, and it gave a reasonable boundary condition on the analysis.

3. The hydrogen generation rate was calculated in that most of the hydrogen was generated between 130 and 180 min into the accident. The total hydrogen generation obtained in the present analysis closely agreed with the value obtained by Henrie and Postma.

4. Better results were obtained concerning the core degradation behavior in the early phase of the transient by the present analysis in which the debris node was assumed to fell to the adjacent lower node. However,the physical models for the fuel relocation and debris formation need to be improved further to be consistent with accident progression in the later phase of the transient.

5. The calculated primary pressure during phase 4 underestimates the actual data due to underestimation of heat transfer from debris to the coolant after an extensive core degradation.

The present version of THALES-PM1/TMI has limitations in some degree in modeling the surge line of pressurizer and heat transfer from debris to the coolant after an extensive core degradation. Since THALES code system has been developed to use probabilistic risk assessment, some model such as debris relocation model are not based on mechanistic model. This point is another limitation on the analysis of TMI-2 accident. However through sensitivity analyses on the TMI-2 Standard Problem, we could presume what was the reasonable model. For example, concerning the debris relocation model we selected Model 2 in which a debris node moved down to the adjacent lower node, and this model gave a reasonable thermal-hydraulic behavior.

References

1. K. ABE, N. WATANABE, M. IDA, M. NISHI, T. NOGUCHI, J. WAKABAYASHI AND H. NARIAI,"Overview of development and application of THALES code system for analyzing progression of core meltdown accident of LWR's," Proceedings of second international topical meeting on nuclear power plant thermal hydraulics and operations, 6/49, Tokyo, Atomic Energy Soc. of Japan (1986).

2. D. W. GOLDEN and A. J. WEISS, "Update on standard problem, data base, and uncertainties," NUREG/CP-0082, Vol.6, 55(1987).

3. R. W. BROWER,"Summary of TMI-2 Data Base," EGG-TMI-7844(1987).

4. K. HASHIMOTO, K. SODA AND H. SEKIYA,"Thermal-Hydraulic Analysis of the Initial Phase of the Three Mile Island Unit 2 Accident," Nucl. Technol. 87, 4, 1058(1989).

5. U.S. NUCLEAR REGULATORY COMMISSION,"RELAP4/MOD5: A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems/ User's Manual/Volume II/Program Implementation," ANCR-NUREG-1335(Vol.2)(1976).

concerning the debris relocation model we selected Model 2 in which a debris node moved down to the adjacent lower node, and this model gave a reasonable thermal-hydraulic behavior.

References

1. K. ABE, N. WATANABE, M. IDA, M. NISHI, T. NOGUCHI, J. WAKABAYASHI AND H. NARIAI,"Overview of development and application of THALES code system for analyzing progression of core meltdown accident of LWR's," Proceedings of second international topical meeting on nuclear power plant thermal hydraulics and operations, 6/49, Tokyo, Atomic Energy Soc. of Japan (1986).

2. D. W. GOLDEN and A. J. WEISS, "Update on standard problem, data base, and uncertainties," NUREG/CP-0082, Vol.6, 55(1987).

3. R. W. BROWER,"Summary of TMI-2 Data Base," EGG-TMI-7844(1987).

4. K. HASHIMOTO, K. SODA AND H. SEKIYA,"Thermal-Hydraulic Analysis of the Initial Phase of the Three Mile Island Unit 2 Accident," Nucl. Technol. 87, 4, 1058(1989).

5. U.S. NUCLEAR REGULATORY COMMISSION,"RELAP4/MOD5: A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems/ User's Manual/Volume II/Program Implementation," ANCR-NUREG-1335(Vol.2)(1976).

6. J. M. BROUGHTON, P. QUAN, D. A. PETTI and E. L. TOLMAN,"A Scenario of the Three Mile Island Unit 2 Accident," Nucl. Technol. Vol. 87, No. 1, pp.34-53. (1989).

7. R. D. MCCORMICK, J. L. ANDERSON AND D. W. GOLDEN,"TMI-2 Data Summary Report," EGG-TMI-7843(1987).

8. J. L. ANDERSON,"Recommended HPI (High Pressure Injection) Rates for the TMI-2 Analysis Exercises (0 to 300 minutes)," EGG-TMI-7833(1987).

9. J. O. HENRIE AND A. K. POSTMA,"Lessons Learned from Hydrogen Generation and Burning during the TMI-2 Event," GEND-061(1987).

Table I Heat Transfer Correlations Adopted in THALES-PM1/TMI

STATE	HEAT TRANSFER CORRELATION
Subcooled Liquid Forced Convection	Dittus and Boelter
Nucleate Boiling	Thom
Forced Convection Vaporization	Schrock and Grossman
Transition Boiling	McDonough, Milich and King
Stable Film Boiling	Groeneveld
Low Flow Film Boiling	Modified Bromley
Superheated Vapor Forced Convection	Dittus and Boelter
Low Pressure Flow Film Boiling	Dougall and Rohsenow

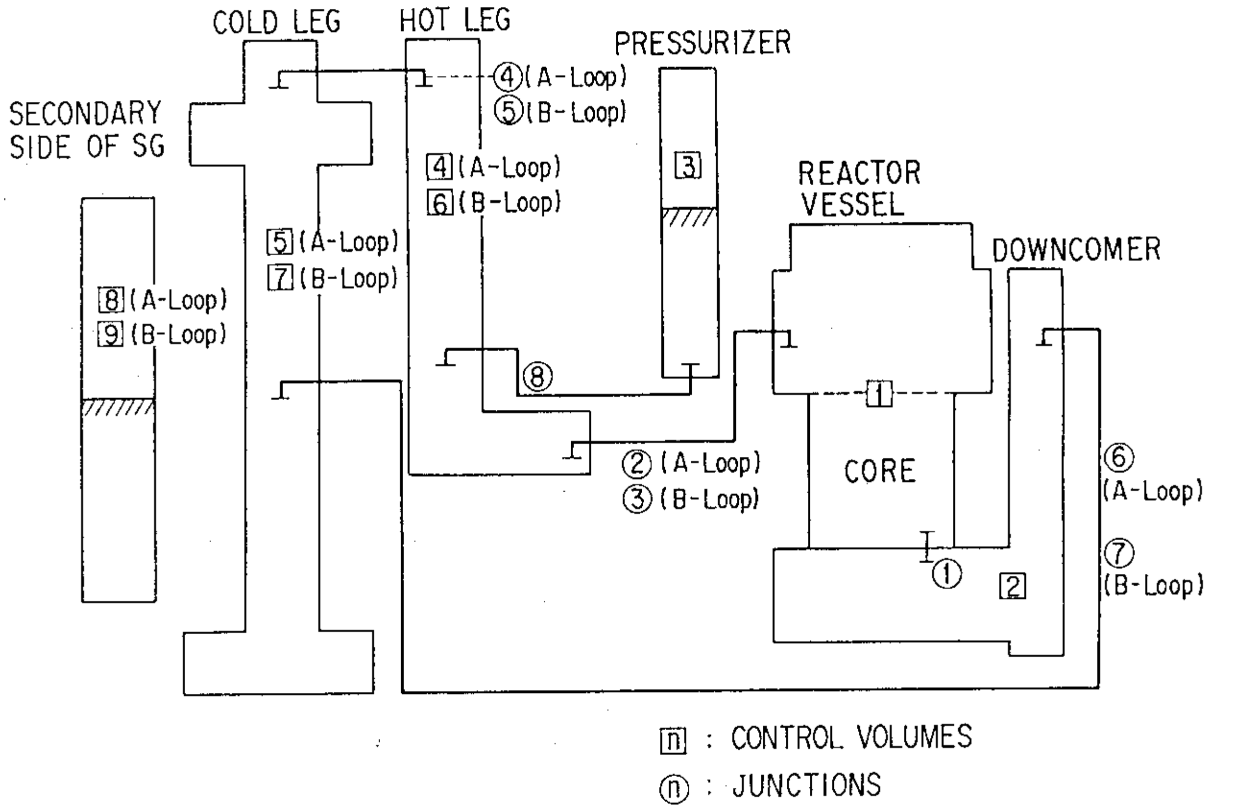

Fig. 1 System model of TMI-2 by THALES-PM1/TMI.

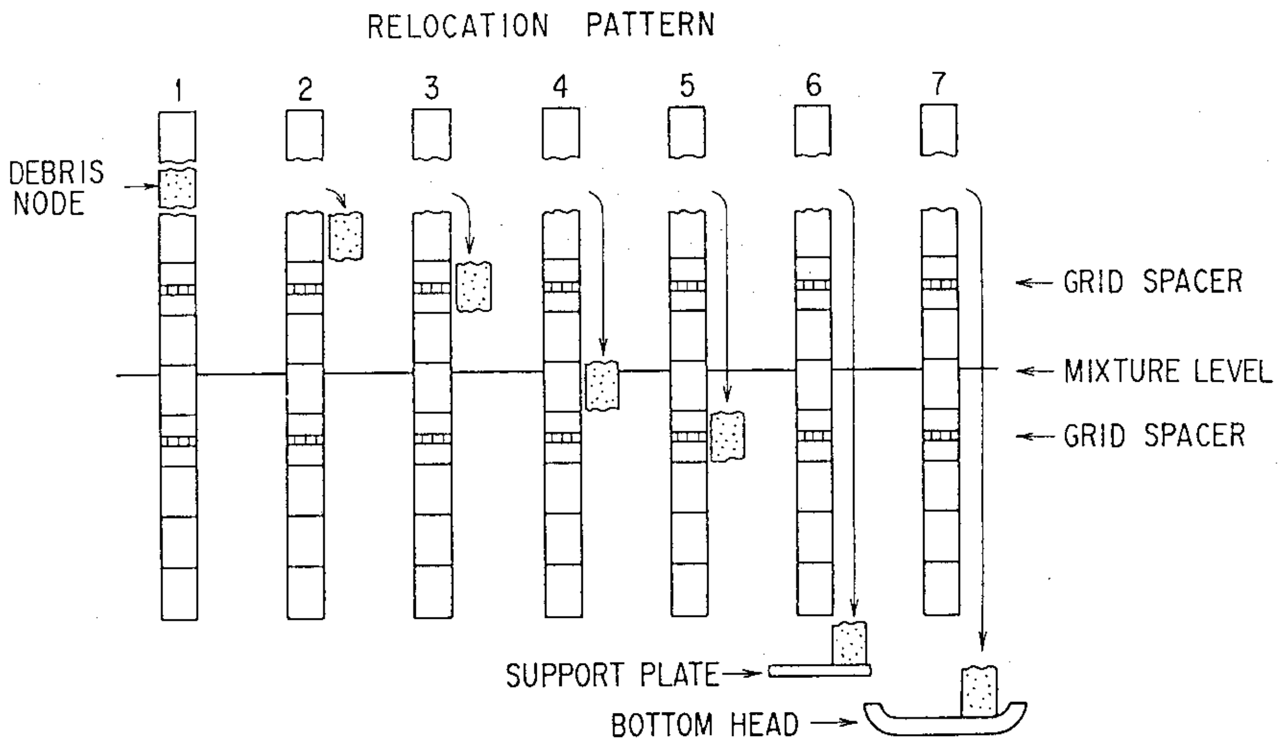

Fig. 2 Fuel relocation patterns considered in THALES-PM1.

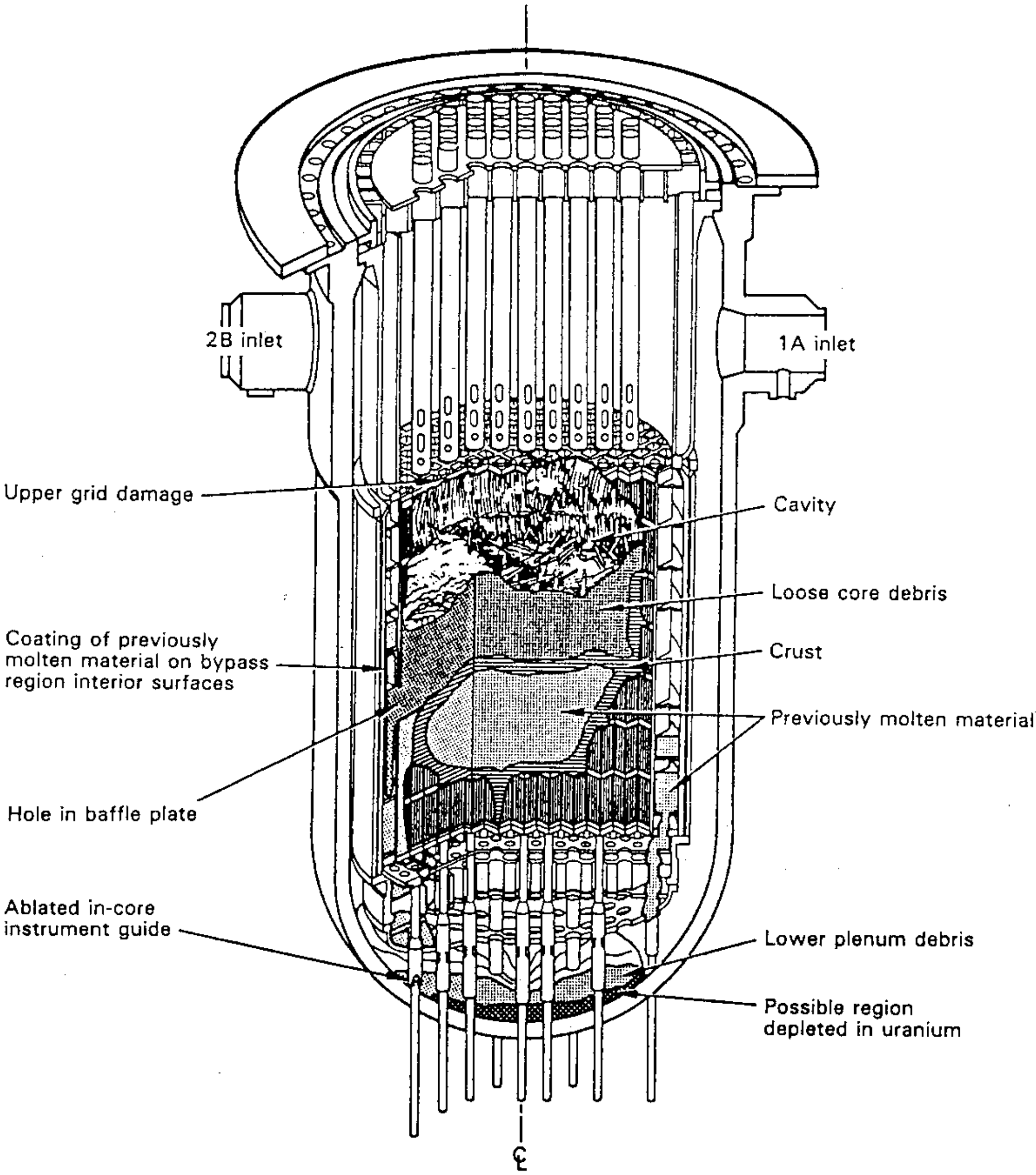

Fig. 3 End-state configuration of the TMI-2 reactor vessel and core.[6)]

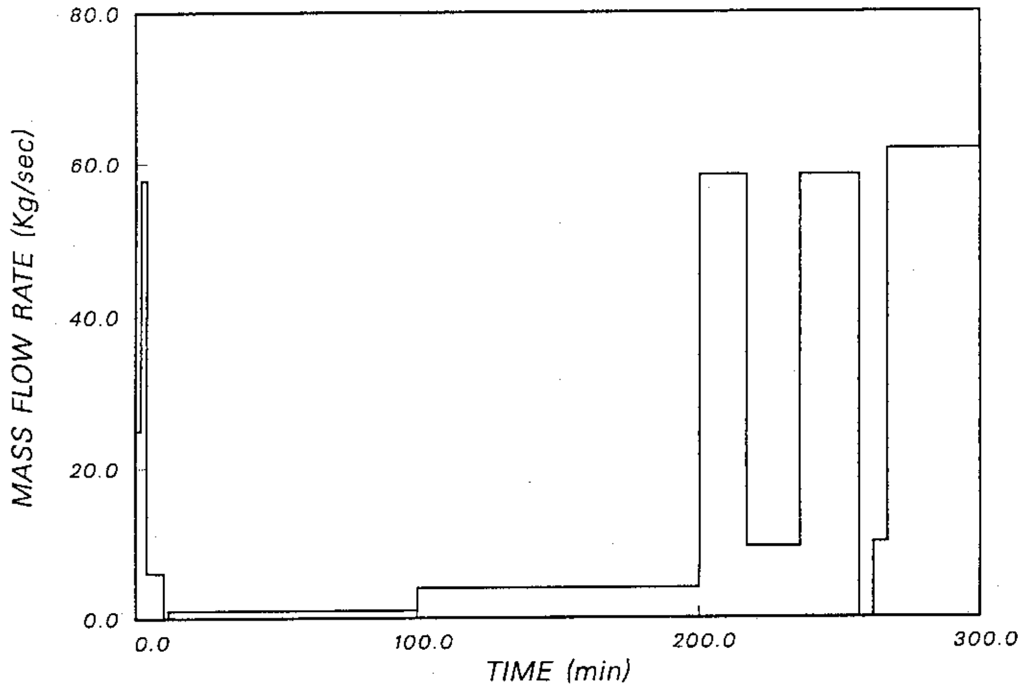

Fig. 4 Makeup/HPI flow rate for boundary condition.

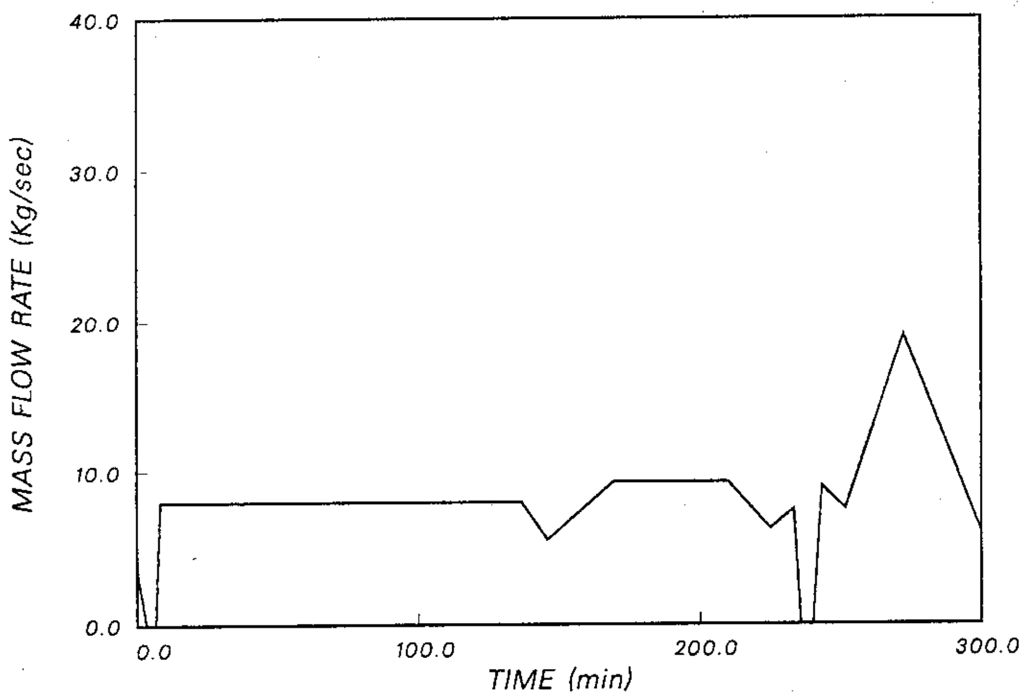

Fig. 5 Letdown flow rate for boundary condition.

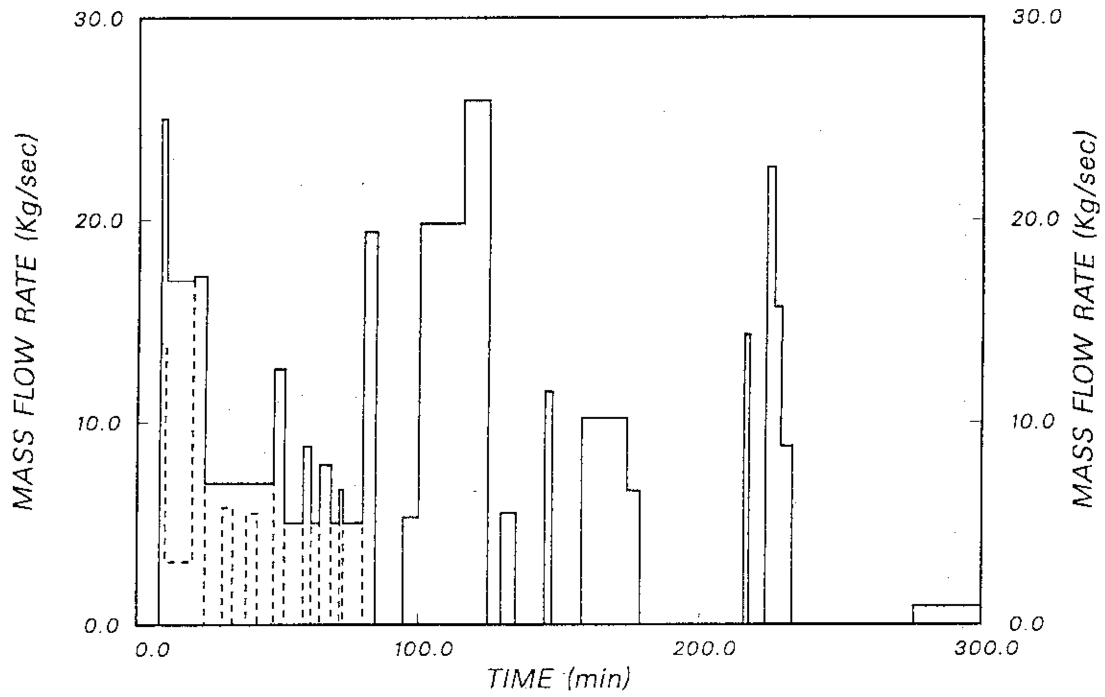

Fig. 6 Auxiliary feed water flow rate for boundary condition (A-loop).

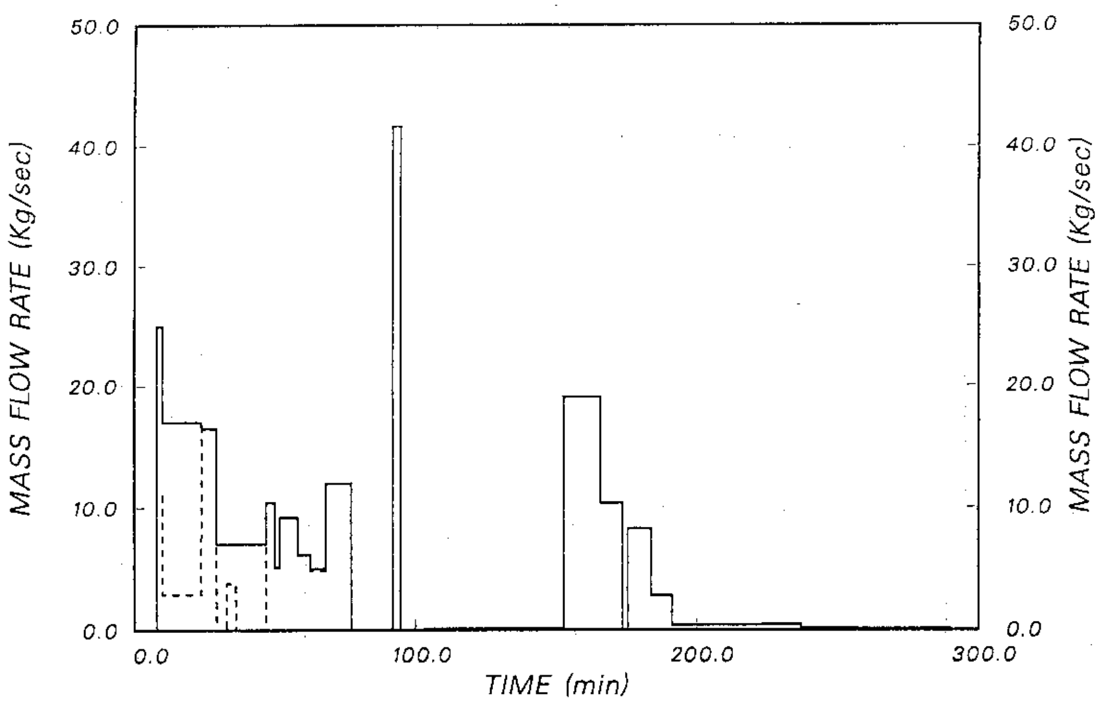

Fig. 7 Auxiliary feed water flow rate for boundary condition (B-loop).

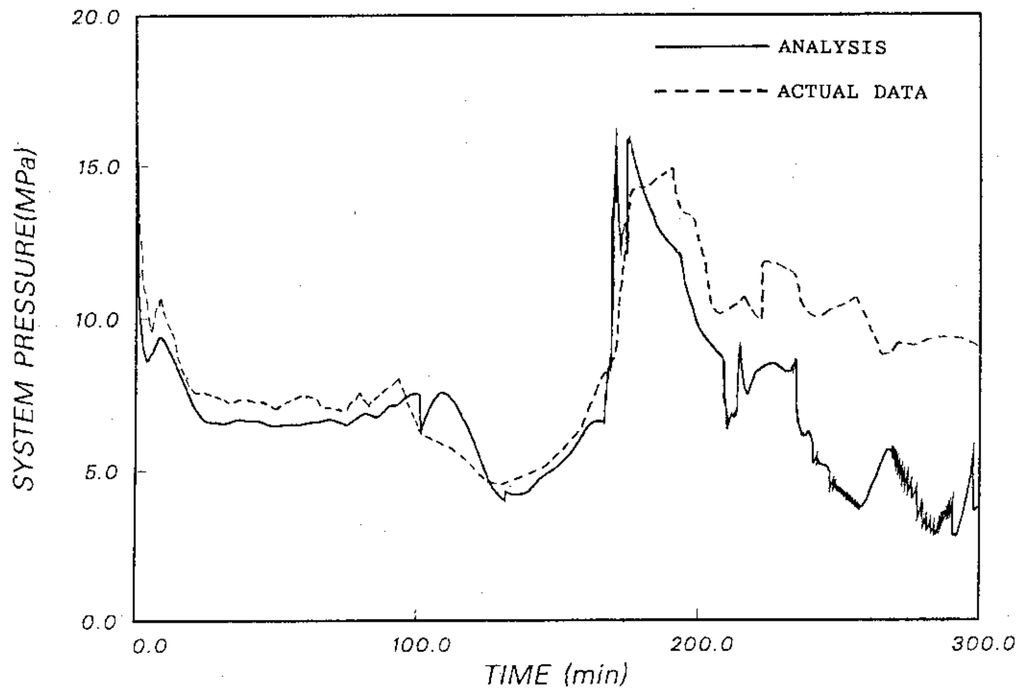

Fig. 8 Primary system pressure during TMI-2 accident calculated by THALES-PM1/TMI in comparison with the actual data.

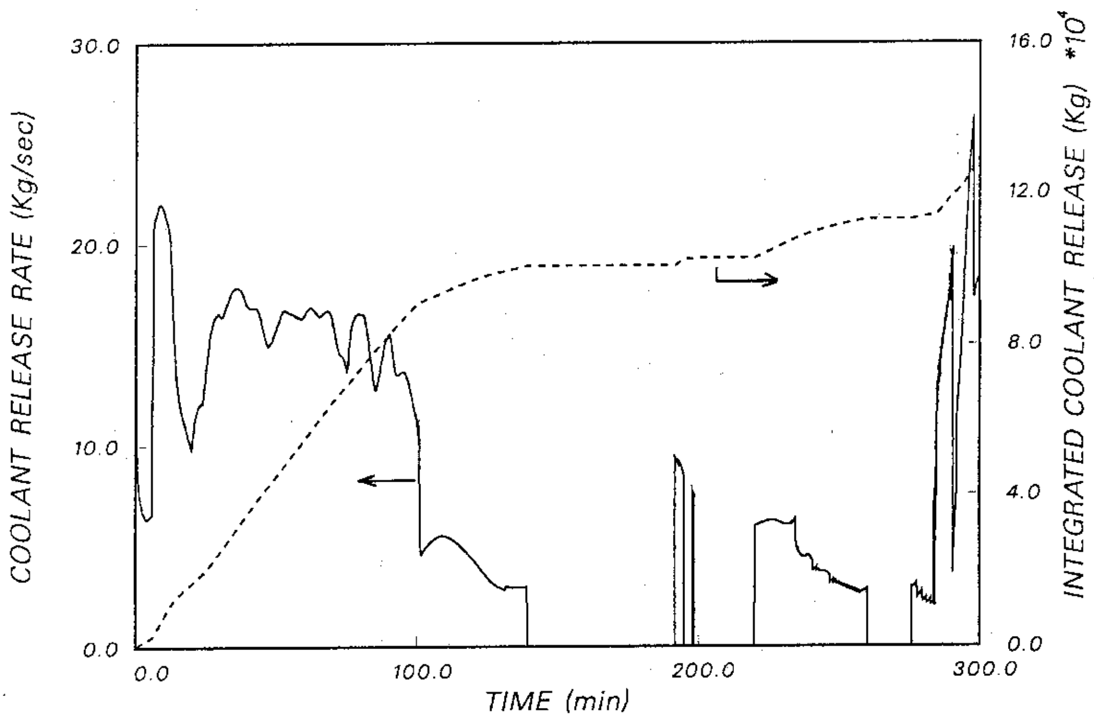

Fig. 9 Calculated release rate of coolant from the PORV of the pressurizer and its integrated value.

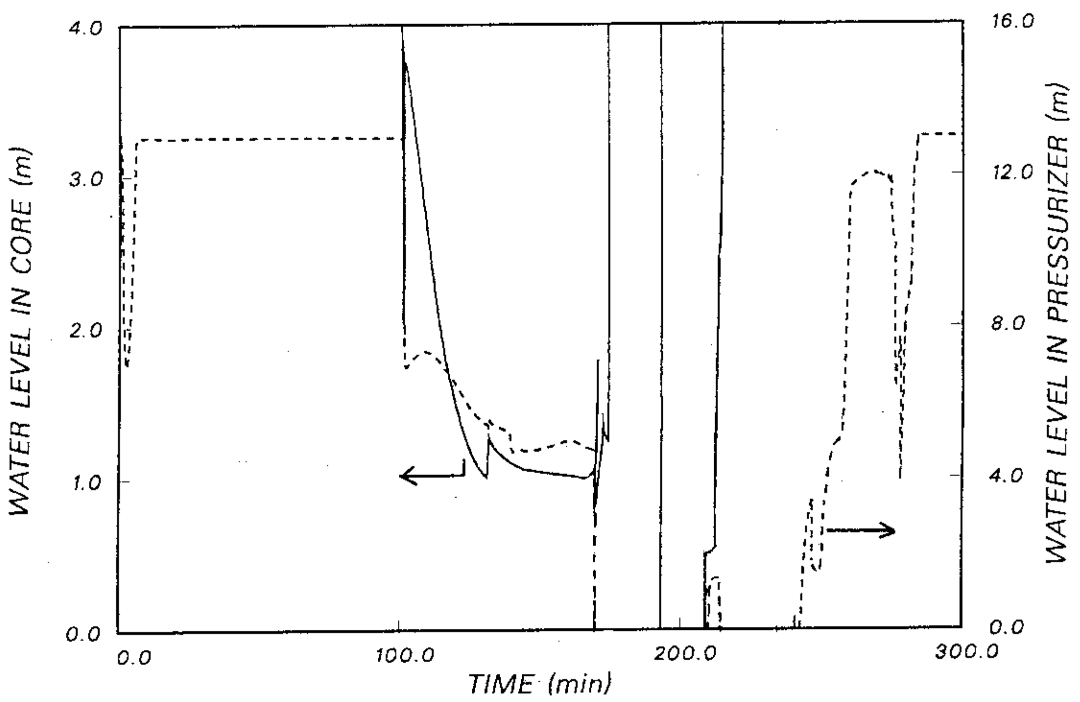

Fig. 10 Calculated water levels in the core and in the pressurizer.

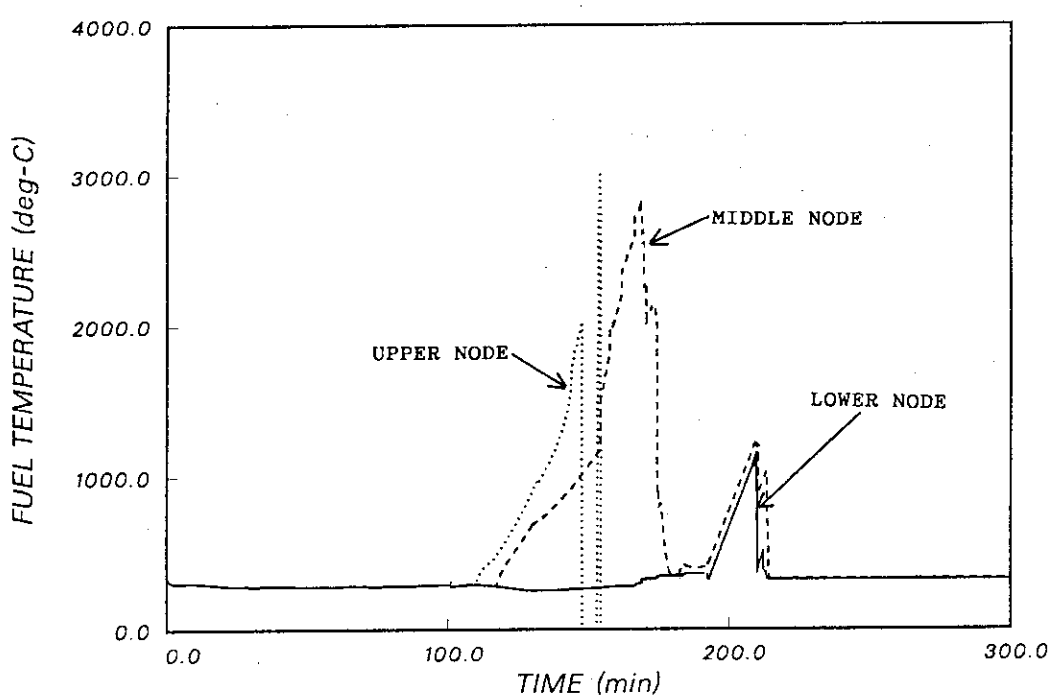

Fig. 11 Calculated fuel temperature at upper, middle and lower part of the rods in the central region.

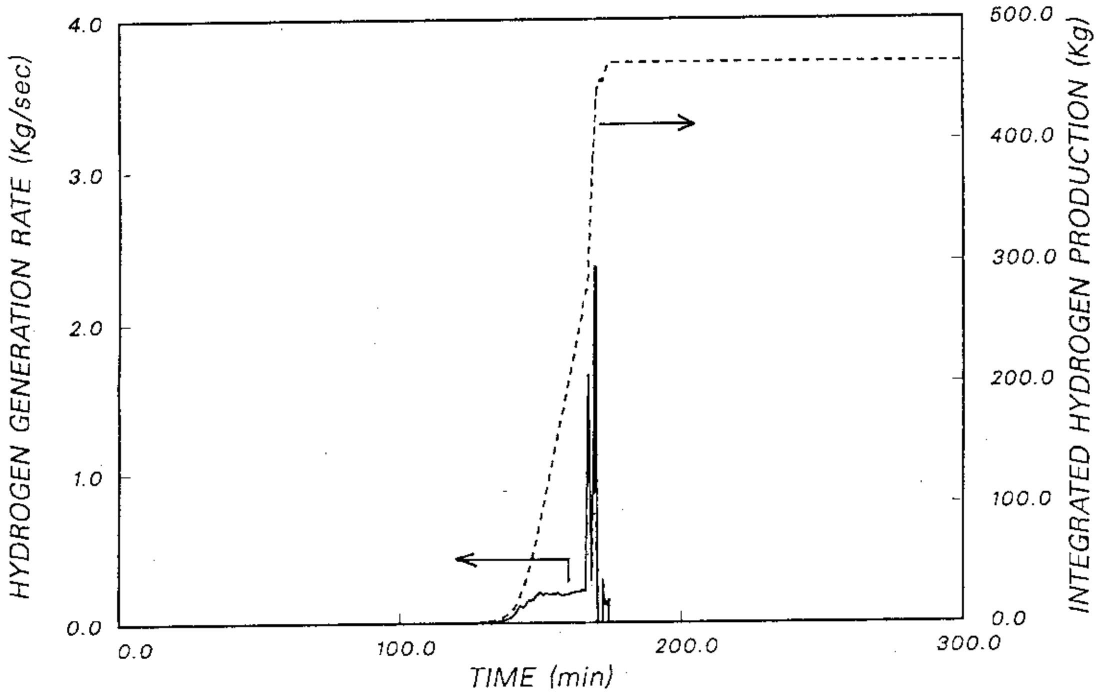

Fig. 12 Calculated hydrogen rate and the integrated hydrogen production.