ANALYSIS OF TMI-2 BENCHMARK PROBLEM USING MAAP4.03 CODE

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The Three Mile Island Unit 2 (TMI-2) accident provides unique full scale data, thus providing opportunities to check the capability of codes to model overall plant behavior and to perform a spectrum of sensitivity and uncertainty calculations. As part of the TMI-2 analysis benchmark exercise sponsored by the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD NEA), several member countries are continuing to improve their system analysis codes using the TMI-2 data. The Republic of Korea joined this benchmark exercise in November 2005. Seoul National University has analyzed the TMI-2 accident as well as the currently proposed alternative scenario along with a sensitivity study using the Modular Accident Analysis Program Version 4.03 (MAAP4.03) code in collaboration with the Korea Hydro and Nuclear Power Company. Two input files are required to simulate the TMI-2 accident with MAAP4: the parameter file and an input deck. The user inputs various parameters, such as volumes or masses, for each component. The parameter file contains the information on TMI-2 relevant to the plant geometry, system performance, controls, and initial conditions used to perform these benchmark calculations. The input deck defines the operator actions and boundary conditions during the course of the accident. The TMI-2 accident analysis provided good estimates of the accident output data compared with the OECD TMI-2 standard reference. The alternative scenario has proposed the initial event as a loss of main feed water and a small break on the hot leg. Analysis is in progress along with a sensitivity study concerning the break size and elevation.

KEYWORDS : Severe Accident, MAAP4.03 Code, Benchmark Problem, Alternative Scenario

1. INTRODUCTION

Since the Three Mile Island Unit 2 (TMI-2) reactor accident, there have been extensive research activities to try to understand the phenomena involved [1-8], and to develop accident management strategies to mitigate the consequences from core melt accidents [9]. The experimental database on core heatup and melt progression is limited to the results of small-scale experiments, which are only partially representative of what could occur in a real reactor. As a consequence, codes are used to describe core degradation transients in nuclear reactors without a clear idea of their predictive capabilities. The accurate prediction of the TMI-2 transient relies on the proper definition of boundary conditions and plant characteristics [10-12]. Some of these data are either unknown or difficult to estimate. In particular, the data for the makeup and letdown flows were not recorded during the accident. Although these data would not bring about any improvements in the understanding of severe accident processes, their absence has necessitated significant efforts from code users who have tried to estimate them.

2. ACCIDENT SEQUENCE

Quite a few processes that occurred during the TMI-2 transient have led to the initiation of experimental programs worldwide to attain a better understanding of the physics involved. Several member countries in the Organization for Economic Cooperation and Development (OECD) are continuing to improve their system analysis codes using the TMI-2 data [12]. Seoul National University (SNU) has recently joined international endeavors to analyze the TMI-2 accident using the Modular Accident Analysis Program Version 4.03 (MAAP4.03) code [13]. MAAP4.03 is a fast running, integrated severe accident analysis code that simulates transients and specifically accounts for system events that occur during a transient, including operator actions.

The accident sequence was divided into four main phases, as depicted in Fig. 1: phase 1 between scram and stop of pumps (up to 100 minutes), phase 2 between stop of pumps and first reflooding (up to 174 minutes), phase 3 reflooding (up to 200 minutes), and phase 4 ending of the sequence (up to 300 minutes).

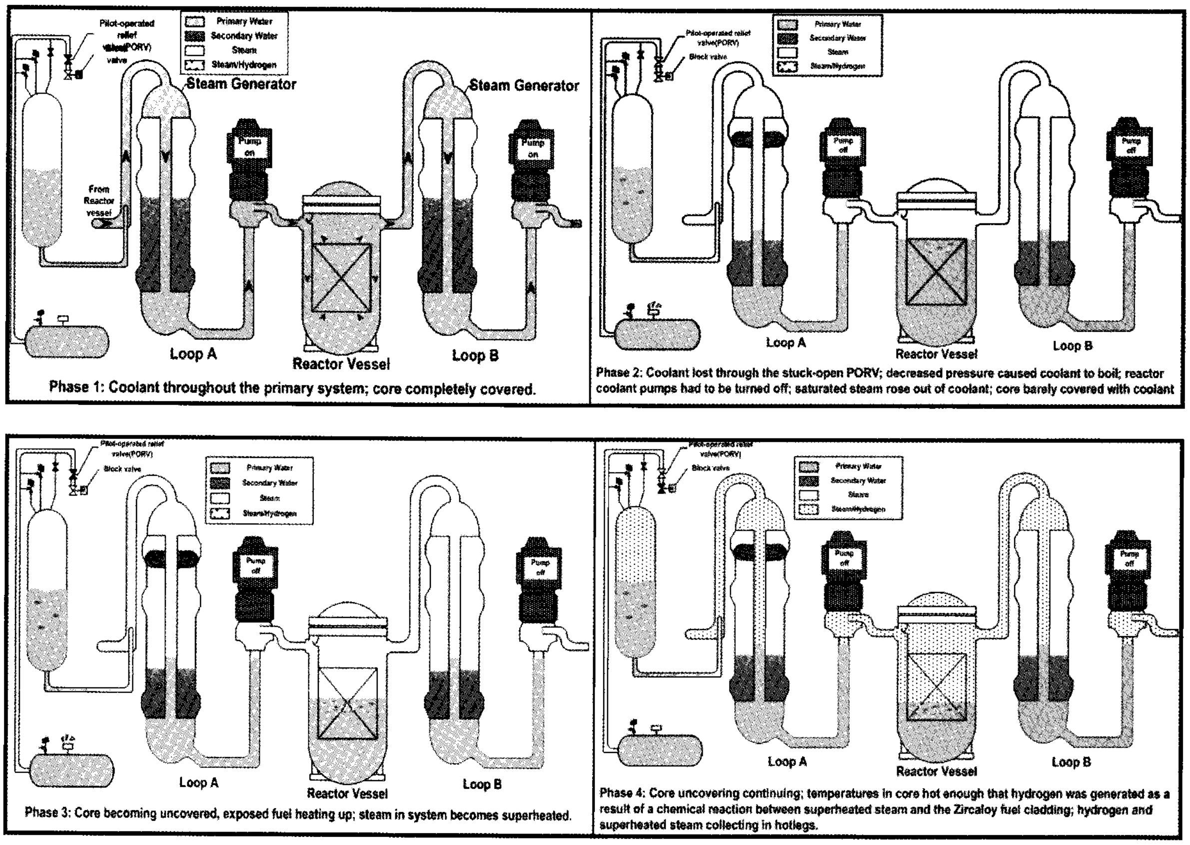

Fig. 1. TMI-2 Accident Progression

The first 300 minutes of the TMI-2 accident have been simulated. Models are included for all the important phenomena that might occur during accident sequences leading to degraded cores. MAAP4.03 requires a parameter file describing the plant geometry, system performance, controls, and initial conditions, as well as an input deck. A parameter file based upon data supplied as part of the TMI-2 analysis exercise package was constructed. An input deck is used to define boundary conditions and operator actions during the accident. Not all the plant data for the secondary side are available as boundary conditions for the primary system. For instance, motor-driven auxiliary feedwater flow rates were not recorded during the accident. An input deck was created based upon the identified sequence and operator actions during the accident. A more detailed presentation of the accident sequence and important phenomenology are provided in the accident scenario [14].

3. CODE DESCRIPTION

The MAAP4.03 primary system nodalization for tracking these quantities in a Babcock &Wilcox type plant is shown in Fig. 2. The reactor consists of four volumes: the core, downcomer, upper plenum, and reactor dome. All loops except one, the broken loop, are lumped together, and the broken loop is treated separately. The broken loop refers to the loop that may contain a primary system break. The user selects whether the pressurizer is in the broken or unbroken loop. The A-loop is taken as the broken loop for analysis of the TMI-2 accident.

The reactor vessel is nodalized in the form of heat sinks and control volumes. For the core region, the number of radial rings and axial rows is specified. The radial peaking factor and volume fraction are fixed for each ring, and an axial peaking factor is assigned for each row. Seven rings and thirteen rows are used to nodalize the TMI-2 core for simulation of the accident.

Mass and energy rates of change for core materials are calculated for each core node. Steam and hydrogen are assumed to flow along the uncovered and unblocked flow channels, and the mass flow rates and enthalpies in each channel are determined by accounting for the generation and consumption at each axial level. The core water pool is treated as a lumped mass and energy volume.

Figure 3 shows the containment nodalization and the flow paths used to track materials in the containment model. The cavity refers to the volume below the reactor vessel, the lower compartment to the volume below the operating deck and inside the crane wall, the annular

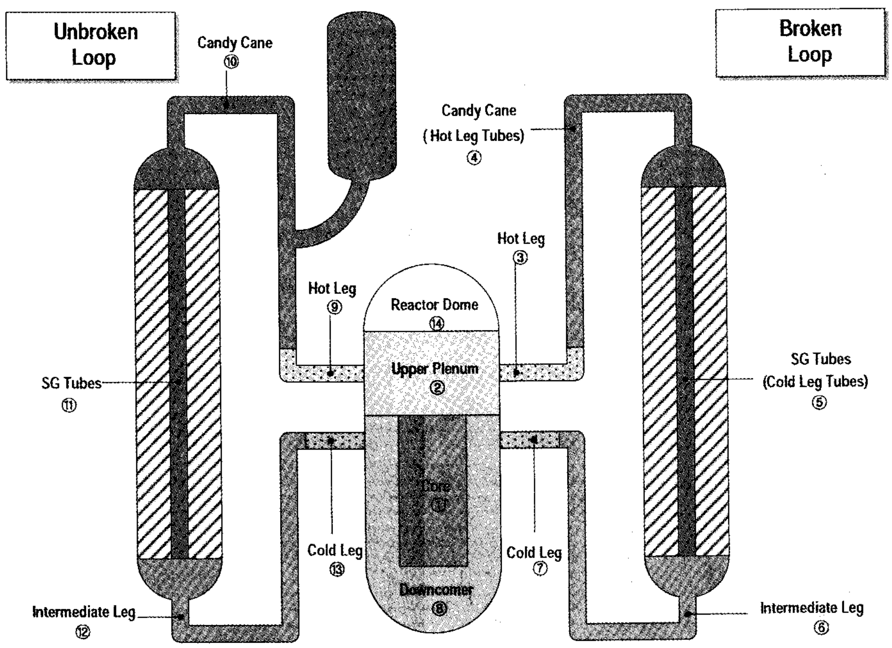

Fig. 2. Primary System Nodalization

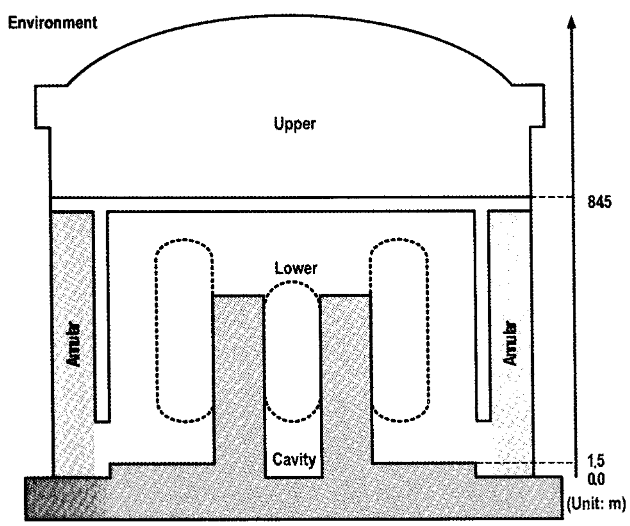

Fig. 3. Containment Nodalization

compartment to the volume outside the crane wall below the operating deck, and last, the upper compartment to the volume above the operating deck. Within each of these volumes, the code tracks the thermofluid characteristics of steam, air, hydrogen, noncondensable gases, and fission products. Within the lower compartment and the reactor cavity, corium and water are also accounted for.

4. RESULTS AND DISCUSSION

An accurate prediction of the TMI-2 transient relies on proper definition of boundary conditions and plant characteristics. Having demonstrated the boundary conditions for the primary system, plant data for the primary system can be compared against the code results.

The following data can be used for comparison with predictions: primary system pressure, pressurizer water level, broken steam generator pressure, A-loop steam generator water level, and B-loop steam generator water level. From the perspective of the TMI-2 accident simulation, the primary system pressure turns out to be a key parameter for comparison. The data provide a continuing measure of the energy balance among the core, primary system, and the two steam generators. As such, the pressure reflects the correctness of the boundary conditions as well as the adequacy of a code's thermal hydraulic models.

The calculated and TMI-2 standard reference primary system pressures are compared in Fig. 4. Generally good agreement with the reference is obtained during most of the simulated period. Particularly good agreement is observed from the start of the accident until 170 minutes, while relatively large deviations are noted thereafter.

During the first 100 minutes the pressure response was dominated by the loss of feedwater and primary loop pump trip. All of these transients are well simulated with MAAP4.03. The good agreement with the standard reference from 100 to 170 minute suggests that the heat transfer rate across the steam generator and the heat

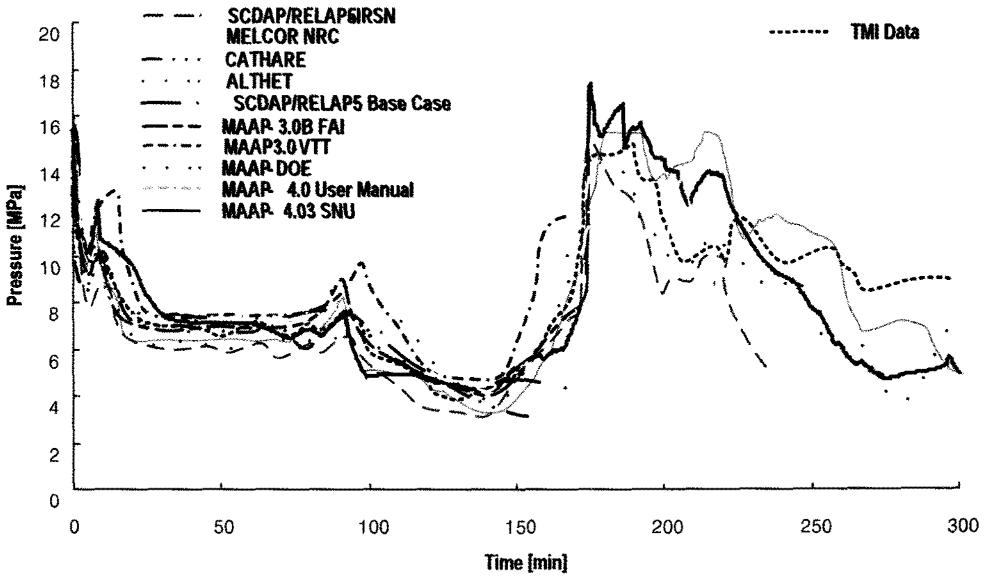

Fig. 4. Primary System Pressure

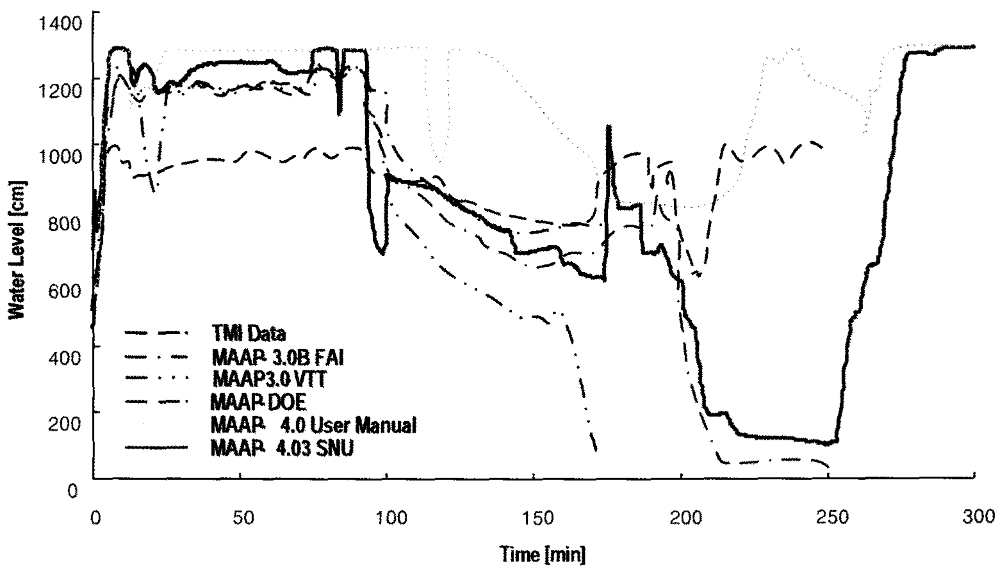

Fig. 5 Pressurizer Level

generated in the core during this period are adequately modeled. Of particular importance is the finding that the code is able to correctly calculate the pressurization of the primary system after the Pilot Operated Relief Valve (PORV) is closed. The sharp rise in the MAAP4.03 results after 170 minutes appears to deviate from the standard reference. This is caused by the digitized interpretation of the pressure response data during the accident, which is not correct. The ensuing fast pressure rise should have started only after 170 minutes in response to the initiation of core quenching caused by the brief operation of the B-loop pump. The primary system pressure decreased following the High Pressure Injection (HPI) at 200 minutes. The results demonstrate that MAAP4.03 predictions diverge on their estimates of pressure on account of persisting uncertainties and unidentified physical processes.

The collapsed pressurizer level and the calculated level response are compared to the TMI-2 standard reference in Fig. 5. During the two phase discharge period through the PORV, the algorithm used in MAAP4.03 to calculate the pressurizer void fraction is iterative and produces oscillatory results. Therefore, the indicated level calculated

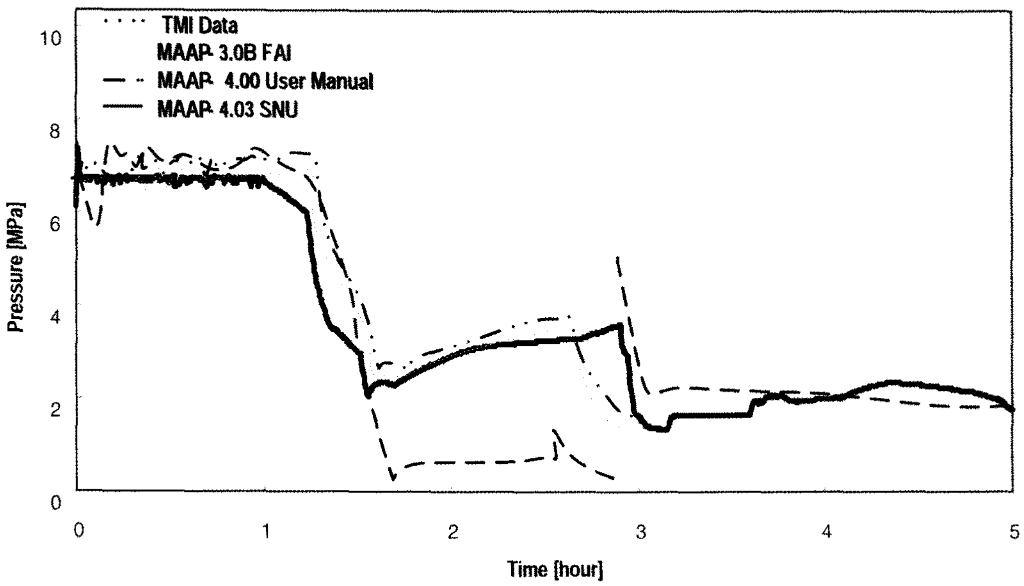

Fig. 6. A-Loop Steam Generator Pressure.

for the period also fluctuates. However, as depicted in Fig. 5, this oscillatory behavior results in an average behavior that is in good agreement with the standard reference obtained throughout most of the transient. Filling of the pressurizer in the first few minutes is correctly calculated. The initial discharge from the PORV is properly simulated. Despite the calculated void oscillations between 10 to 94 minutes, the maximum void fraction, which corresponds to the oscillation amplitude, also concurs with the standard reference. After 100 minutes the calculated level is seen to decrease whereas the standard reference indicates that this decrease is delayed to 120 minutes. This is a rather minor discrepancy in the pressure drop compared to the primary system pressure, which was decreasing at this point in time. Several codes predicted the drainage of the pressurizer after closure of the PORV, which led to core quenching.

The calculated broken steam generator pressure and the TMI-2 standard reference are presented in Fig. 6. The secondary side pressures were approximately equal to the secondary side relief valve set point during most of the first 60 minutes of the accident. Thereafter, the atmospheric dump valves were opened and used to control the secondary pressure. This was simulated in MAAP4.03 by allowing the effective valve opening area for each steam generator to be changed at selected times between 60 and 174 minutes. This can be corrected by increasing the dump valve opening area in the broken loop side during this period of time. Due to the limited heat transfer across the tubes of this steam generator during the period, little effect on the primary system behavior is expected. Mechanistic modeling of heat transfer to the steam generator secondary side, when the primary loop is stagnant, would improve the capability of MAAP4.03 to simultaneously calculate the correct pressure in the broken loop steam generator.

The resulting steam generator water levels are illustrated in Figs.7 and 8. The steam generator subroutines in MAAP4.03 calculate the thermodynamic properties and mass and energy rates of change for the water and gas volumes on the steam generator secondary side. The A-loop steam generator is located in the broken loop and thus it may have thermodynamic and process variables that are different from those of the B-loop steam generator in the unbroken loop. The thermodynamic properties and rates of change are calculated based upon the available steam flow area and the volume of noncondensable gases in the steam generator. Assuming the steam rate to be constant, MAAP4.03 considers the derivative of the level during a period of increasing level less the steaming rate in order to estimate the auxiliary feedwater mass flow rates. Figures 7 and 8 indicate that the steam generator water levels are in good agreement with the TMI-2 standard reference.

The extent of hydrogen produced from several code calculations is shown in Fig. 9. Although no accurate observation is available for comparison with the calculated results, the divergence of results obtained by the various codes indicates that the status of the codes cannot be considered mature. As shown, the MAAP4.03 calculation results in approximately 300 kg of total hydrogen produced. It is generally considered that about 500 kg of hydrogen was produced in the TMI-2 accident [15-18]. The calculation underestimates the amount of hydrogen generation due to enough heat being removed in the steam generators to cause the primary system pressure to decrease sufficiently. The pressurizer would consequently have drained into

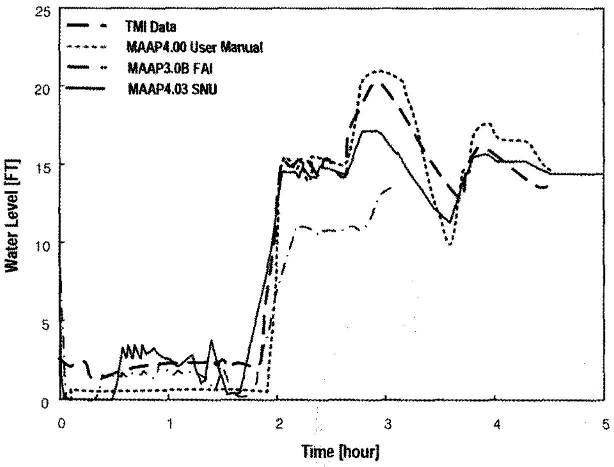

Fig. 7. A-Loop Steam Generator Water Level

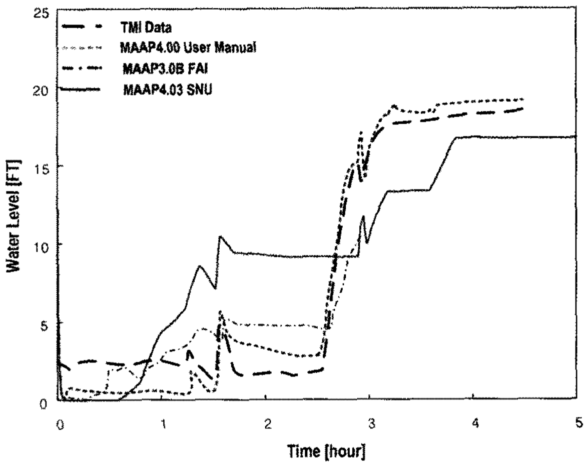

Fig. 8. B-Loop Steam Generator Water Level

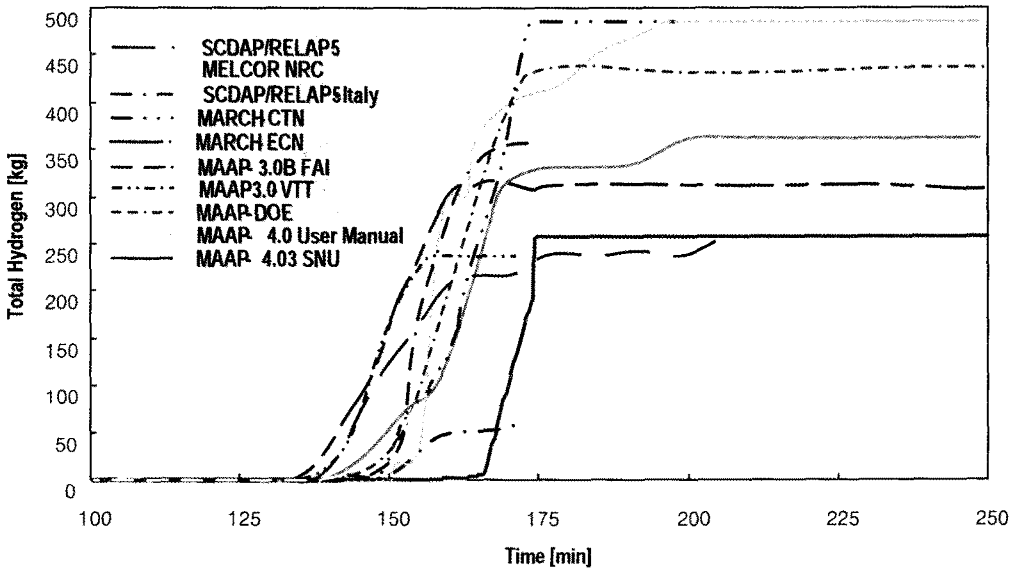

Fig. 9. Total Hydrogen Generation

the core and prematurely terminated the accident. Due to geometry degradation in the core, the calculated rate of hydrogen generation started to decrease. Once a core node is molten, heat transfer and hydrogen generation cease in that node, hastening core melting but underpredicting the amount of hydrogen produced.

The TMI-2 analysis involves more than a few unknown boundary conditions that codes cannot be meaningfully validated against. In this regard, benchmark reactor calculations constitute a pertinent means of obtaining an estimation of current codes capabilities. The OECD Nuclear Energy Agency (NEA) presented an "alternative" scenario for the TMI-2 reactor and proposed the initial event as a loss of main feed-water and opening of a small break on hot leg A, as presented in Fig. 10. The TMI-2 description and initial steady state condition are listed in Table 1. The main objective was to calculate an alternative accident scenario with simplified boundary conditions in order to minimize the impact of the thermal hydraulic initial transient and to focus on the severe accident progression [12]. To prepare for this new benchmark, the first step would be to check the consistency between the boundary conditions and hydraulic components modeling chosen by all the code users. Regarding reactor modeling, it was noted that the secondary system boundary conditions corresponded to an imposed outlet pressure and a control of the water level in the steam generator by regulation of the feedwater.

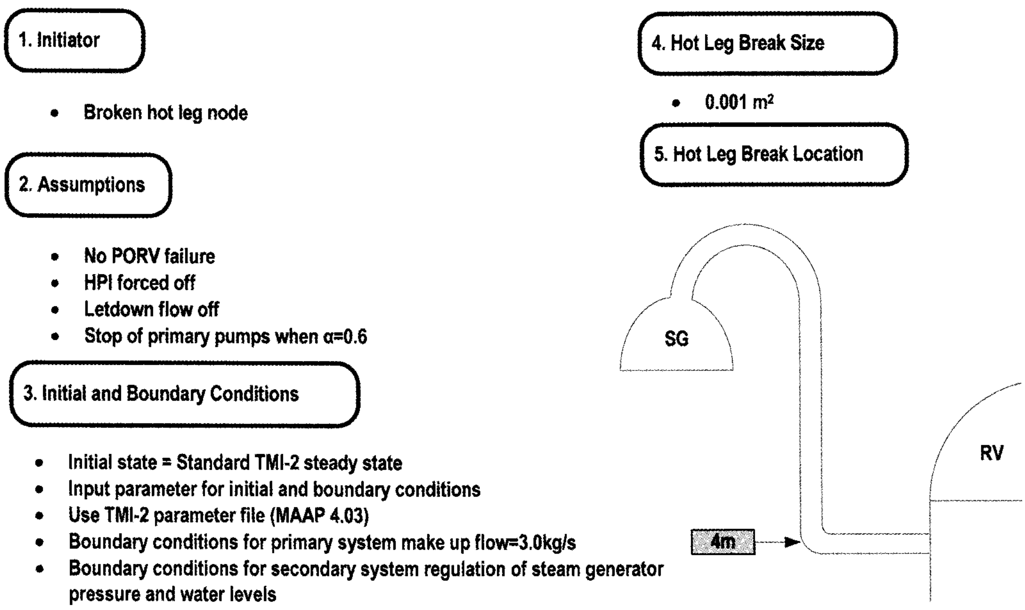

Fig. 10. TMI-2 Alternative Scenario

Table 1. TMI-2 Plant Description and Initial Steady-State Conditions

Parameter	Unit	ENEA (ASTEC)	GRS (ATHLET-CD)	DIMNP (MELCOR)	SNL (MELCOR)	IRSN (IC/CAT V2)	IVS (ASTEC)	SNU (MAAP4.03)	TMI-2 (Turbine Trip)
Reactor power	MW	2700	2663	2780	2770	2700	2700	2568	**2700**
Primary pressure	MPa	15.2	14.9	15.2	14.9	14.95	14.84	15.2	**15.2**
Temp. hot leg A	K	588	592.2	589.9	592	592.7	593	579	**592**
Temp. hot leg B	K	588	592.2	589.9	592	592.7	593	579	**592**
Temp. cold leg A	K	560	564.8	561.1	564	564.8	565.2	559	**548-561**
Temp. cold leg B	K	559	564.8	561.1	564	564.1	565.5	559	**565**
Mass flow rate - loop A	kg/s	8290	8638	8883	8791	8386	8494.6	-	**8280**
Mass flow rate - loop B	kg/s	8560	8675	8883	8809	8372	8490.6	-	**8560**
Pressurizer level	m	5.78	5.38	5.01	5.77	7.08	5.84	6.78	**5.77**
Total primary mass	kg	227600	222500	230880	231273	228800	233340	211170	-
Pressure SG A	MPa	6.43	6.34	6.41	6.42	6.38	6.92	6.38	**7.31**
Pressure SG B	MPa	6.3	6.2	6.41	6.27	6.24	6.91	6.38	**7.24**
Steam temp. SG A	K	571	568	569	568	583.7	578.4	-	**586**
Steam temp. SG B	K	572	568	569	567	583.8	577.9	-	**586**
Collapsed level SG A	m	3.49	4.03	8	5.6	5.26	3.78	3.28	-
Collapsed level SG B	m	3.19	3.54	8	5.1	5.23	3.57	3.28	-
Liquid mass SG A	kg	18630	14630	15670	18700	17276	23290	17506	-
Liquid mass SG B	kg	17510	13140	15670	16900	16959	21620	17506	-
Feedwater flow SG A	kg/s	737	743.7	761.6	750	713	749.9	756	**723**
Feedwater flow SG B	kg/s	778	738.5	761.6	770	729	744.1	756	**717**
SG feedwater temp.	K	513	503	510.9	513	513	513	508	-

The benchmark exercise aims to compare the different code results obtained for the specified transient, to draw conclusions from the comparison, and to estimate the efficiency of codes to predict safety margins and the predictive qualities of models.

5. CONCLUSION

The present analysis showed general agreement with the standard reference of the TMI-2 accident. While the uncertainties in the boundary conditions render it difficult to draw unique quantitative conclusions regarding the core and the primary system behavior during a severe accident, understanding of the system trends and many other insights were gained from this analysis exercise. Many of the codes have difficulties in simulating the late phase of the TMI-2 accident. The insights gained from this analysis were related to the role of hydrogen generation on the heat removal from the steam generators, the pressurizer response, and the impact of core geometry on oxidation. The OECD NEA presented an alternative scenario for the TMI-2 reactor using simplified boundary conditions in order to minimize the impact of the thermal hydraulic initial transient and to focus on the severe accident progression. A sensitivity study is in progress for the TMI-2 alternative scenario. The aim is not to reproduce the TMI-2 accident involving scores of uncertainties, but rather the sequence of events should remain close to the TMI-2 scenario in order to be considered as a validation of codes as well as a benchmark.

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