Preliminary Analysis of TMI-2 Severe Accident using CINEMA Computer Code

Rae-Joon Park [a*], Dong Gun Son [a], Jun Ho Bae [a], Kwang Soon Ha[a]
[a]*Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-Gu, Daejeon, Korea*
[*]*Corresponding author: rjpark@kaeri.re.kr*

1. Introduction

As an integrated severe accident computer code development in Korea, CINEMA (Code for INtegrated severe accidEnt Management Analysis) has been developing for a stand-alone severe accident analysis. The basic goal of this code development is to design a severe accident analysis code package by exploiting the existing domestic DBA (Design Basis Analysis) code system for the severe accident analysis. The CINEMA computer code are composed of CSPACE, SACAP (Severe Accident Containment Analysis Package), and SIRIUS (SImulation of Radioactive nuclide Interaction Under Severe accident), which are capable of core melt progression with thermal hydraulic analysis of the RCS (Reactor Coolant System), severe accident analysis of the containment, and fission product analysis, respectively, as shown in Fig. 1.

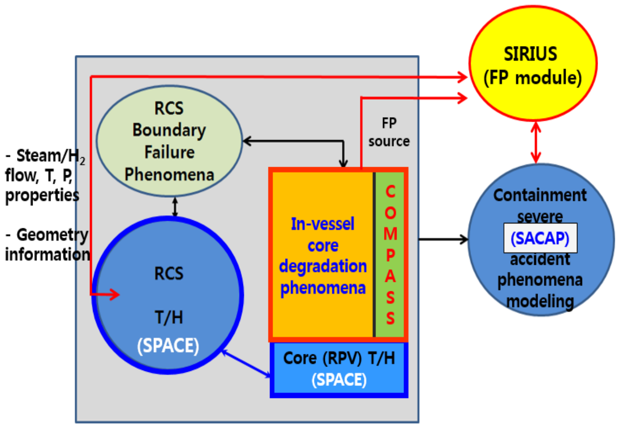


Fig.1 CINEMA code structure.

The CSPACE is the result of merging the COMPASS (COre Meltdown Progression Accident Simulation Software) and SPACE (Safety and Performance Analysis CodE for nuclear power plants) models, which is designed to calculate the severe accident situations of an overall RCS thermal-hydraulic response in SPACE modules and a core damage progression in COMPASS modules. For the purpose of CINEMA validation, the TMI-2 (Three Mile Island Unit 2) severe accident has been analyzed in this study. This analysis has been performed to estimate the efficiency of the CINEMA computer code and the predictive qualities of its models from an initiating event to a severe accident. Preliminary CINEMA results are compared with TMI-2 data, such as, a pressurizer pressure.

2. TMI-2 Severe Accident

On March 28, 1979, the TMI-2 pressurized water reactor underwent a prolonged, a total loss of feed water with a SBLOCA (Small Break Loss Of Coolant Accident) that resulted in a partial melting of the core, significant cladding oxidation, and a significant release of fission products from the fuel. The progression of the TMI-2 accident was mitigated by an injection of the emergency cooling water.

The TMI-2 accident scenario [1] can be divided into four phases, beginning with a reactor scram, as follows:

- Phase 1: From 0 to 100 minutes. This represents the part of the accident where some or all of the main coolant pumps were operating, forcing convective two phase coolant through the core.
- Phase 2: From 100 to 174 minutes. During this time span, all the main pumps were shut down, and a boiling off of the water in the reactor vessel resulted in a progressive uncovering of the core, causing major and very severe core damage.
- Phase 3: From 174 to 224 minutes. This represents the first recovering and major quenching of the core by a short operation of the main coolant pump at 174 minutes and a continued core heatup and damage, even when the core is recovered again by an operation of the high pressure safety injection system after 200 minutes.

Fig. 2 shows the end state of TMI-2 severe accident. During the TMI-2 severe accident, approximately 62 tons of core material was melted and 19 tons was relocated to the lower plenum of the reactor vessel. However, the reactor vessel did not fail.

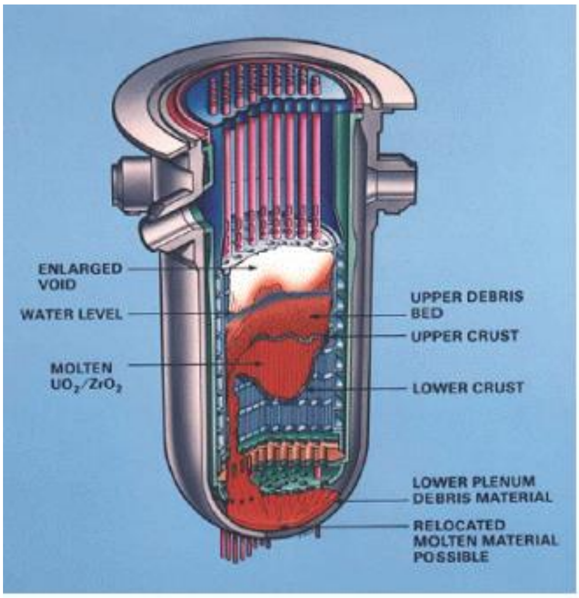


Fig. 2. End state of TMI-2 severe accident.

3. CINEMA Input Model

TMI-2 was designed and manufactured by Babcock & Wilcox, Inc. The core contained 177 fuel assemblies. The reactor coolant system (RCS) consisted of the reactor vessel, two vertical one-through steam generators, four reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system was arranged with two heat transport loops, each with two RCPs and one steam generator. Fig. 3 shows an input nodalization of the CINEMA computer code for the TMI-2.

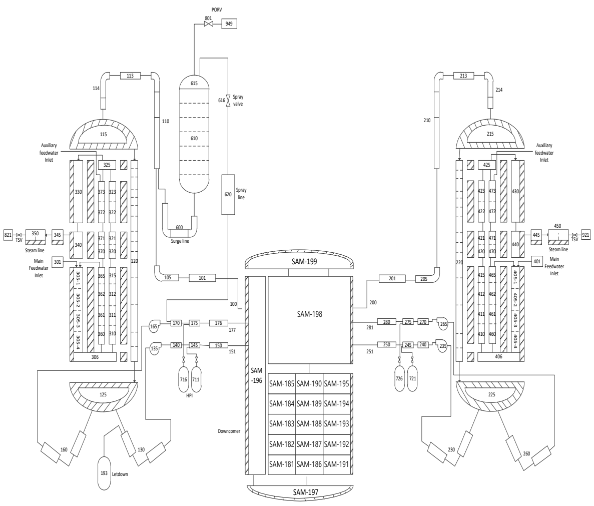


Fig. 3. CINEMA nodalization for TMI-2

All primary and main secondary systems are modeled including the pressurizer, PORV (Pilot-Operated Relief Valve), and safety injections. In core input model, 3 radial and 5 axial nodes are used. Fuel and control rod are connected to the fluid volumes in the core.

4. CINEMA Results and Discussion

A steady state calculation was performed to verify the input nodalization of CINEMA for TMI-2. Table I shows a comparison of the TMI-2 operating condition with the calculation results. The steady state results of the CINEMA calculation for a selected set of parameters were in very good agreement with the TMI-2 operating conditions. The steady state conditions obtained from the simulation were used as initial conditions for the transient calculation

Table I: Comparison of CINEMA steady state results with TMI-2 operating condition.

Parameter	Plant Operating Condition	CINEMA Results
Reactor Power (MW)	2700.0	2700
Primary System Pressure (MPa)	15.2	15.3
Cold Leg Temperature 1A (K)	561.0	571.0
Cold Leg Temperature 2A (K)	548.0	571.0
Hot Leg Temperature Loop A (K)	592.0	598.0
Hot Leg Temperature Loop B (K)	592.0	598.0
Feedwater Temperature (K)	513.0	513.0
SG A Pressure (MPa)	7.31	5.85
SG B Pressure (MPa)	7.24	5.85
SG A Steam Temperature (K)	586.0	578.0
SG B Steam Temperature (K)	585.0	579.0

Fig. 4 shows the preliminary CINEMA results on the pressurizer pressure with a comparison of SCDAP/RELAP5 results, which is very similar to the TMI-2 data. A reduction feed water to the steam generator caused the coolant to expand and initially increased the RCS pressure. The pressurizer PORV opened when the pressure reached 15.7 MPa. The PORV failed to close as the RCS pressure decreased, initiating a small break loss of coolant accident. Emergency core cooling was reduced by operators who thought that the pressurizer liquid level indicated a nearly full RCS, while coolant continued to be lost from the PORV. After an initial decrease in the RCS pressure, the pressurizer pressure remained at approximately 7 MPa. After a pump termination at 10,000 seconds, the liquid level in the reactor vessel decreased, which resulted in a core uncovery. Continued core degradation with a coolant boiling caused the pressurizer pressure to increase.

Fig. 5 shows the preliminary CINEMA results on the fuel cladding surface temperature and fuel cladding mass in center region of the core, respectively. Position of nodes 1,2,3,4,5 are 0.36 m, 1.09 m, 1.82m, 2.55 m, 3.28 m from the bottom of the fuel rod, respectively. Fuel cladding mass of only one fuel rod is in this Figure. The CINEMA results showed the fuel melting and relocation to the lower part of the core. The CINEMA results are very similar to the TMI-2 data in general, with the exception of a rapid increase in the pressure at approximately 10,000 seconds, which is a result from deficiency of CINEMA model on a melted fuel relocation and quenching process

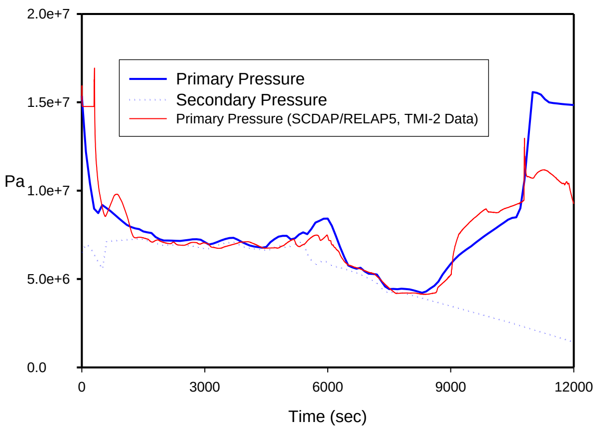
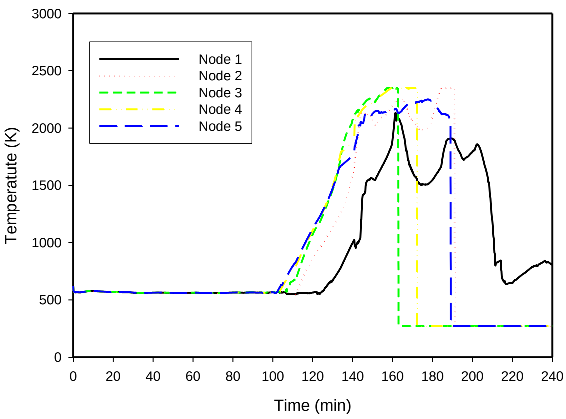
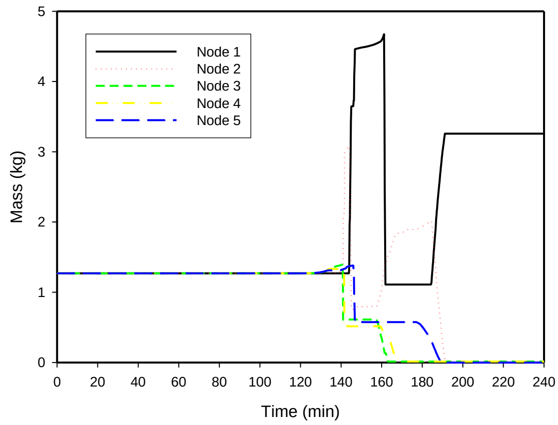


Fig. 4 Pressure history in TMI-2 severe accident



(Fuel Cladding Surface Temperature)



(Fuel Cladding Mass of One Fuel Rod)

Fig. 5 Preliminary CINEMA results on TMI-2 severe accident

4. Conclusions

The CINEMA results on steady state are very similar to the TMI-2 operating condition. The preliminary CINEMA results on transient are very similar to the TMI-2 data in general with the exception of melt relocation and quenching. More CINEMA model development and analysis for a melted fuel relocation and quenching process in the core and lower plenum are necessary to simulate the TMI-2 severe accident.

ACKNOWLEDGEMENT

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[1] J.M. Broughton, P. Kuan, D.A. Petti, E.L. Tolman, et al., A Scenario of the Three Mile Island Unit 2 Accident, Nuclear Technology, Vol. 87, p. 34, 1989.