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Analysis of Pressurizer Surge Line Flow Effect on TMI-2 Severe Accident Progression

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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 severe accident sequence analysis from an initiation event to a containment failure. 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.

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. As a part of a validation of the CINEMA computer code, the TMI-2 (Three Mile Island Unit 2) severe accident has been analyzed. 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 corium relocation into the lower plenum of the reactor vessel.

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 significant cladding oxidation, a partial melting of the core material, 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. In the TMI-2 severe accident, the break location was the PORV (Pilot Oprated Relief Valve), which was located top of the pressurizer. For this reason, a flow path through the the surge line between the hot leg and the pressurizaer was generated and the water level of the pressurizer was very high, which resulted in the melt progression in the core. In this study, the surge line modeling effect on the core melt progression in the TMI-2 severe accident was analyzed using the CINEMA computer code.

2. Detailed Description of the TMI-2 Severe Accident

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

- Phase 1: From 0 to 6,000s. 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 6,000 to 10,440s. 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 10,440 to 12,000s. This represents the first recovering and major quenching of the core by a short operation of the main coolant pump at 10,440s and a continued core heat up and damage, even when the core is recovered again by an operation of the high pressure safety injection system after 12,000s.
- Phase 4: From 12,000 to 18,000s. This represents the initiation of the HPI (High Pressure Injection). The central region of the partially molten core material was not coolable by HPI even through the water level reached the level of the hot legs by 12,420s, because the corium pool was generated in the core. Between 13,440s and 13,560s, the crust encasing and supporting the molten core region is believed to have failed, allowing molten material to relocate to the lower plenum. However, the molten material was quenched by coolant in the reactor vessel at this phase.

Table I shows detailed main events in the TMI-2 severe accident. Turbine and main feedwater pump were tripped by a total loss of feed water at 0 seconds, which resulted in the opening of the pressurizer PORV by a high pressure opening set point of 15.5 MPa. The reactor was tripped by a high pressurizer pressure signal. The pressurizer PORV was not closed when the pressurizer pressure was reached 15.2 MPa, which was the closed set point. This resulted in the SBLOCA. The 1 (of 3) makeup pump 1B was operated at 41s The HPI was operated by high pressurizer pressure signal at 122s.

Table I: Detailed main events in the TMI-2 severe accident

Time (s)	Main Events
0	Turbine and main feedwater pump trip (Total Loss of Feed Water)
3	Pressurizer PORV opening (15.5 MPa)
8	Reactor scram on high pressure signal
13	No Pressurizer PORV closing (15.2 MPa) (SBLOCA)
41	Operation of 1 (of 3) makeup pump 1B
122	HPI operation
278	Stop of HPI
480	Auxiliary feedwater startup
552	Core boiling begins
4,440	Shutdown B-loop RCP (end of phase 1)
6,000	Shutdown A-loop RCP
6,184	Core uncovery
7,742	Cladding oxidation begins (T= 1,000K)
7,719	Cladding failure (T=1,117K)
8,340	Close of the PORV line block valve
9,014	Fuel melting
10,440	Restart one B-loop RCP (end of phase 2)
11,580	Shutdown of the B-loop RCP
12,000	Start of primary system feed and bleed
13,440	Core material slumping (end of phase 3)
18,000	General emergency declared (end of phase 4)

However, this pump was stopped by the operator, because of a misreading of the high level of the pressurizer. The auxiliary feedwater was startup at 480s. Boiling was occurred in the core at 552s. The B-loop and A-loop RCPs were stopped at 4,440s and 6,000s, respectively. The core was uncovered at 6,184s, which resulted in an increase of the fuel cladding temperature. The fuel cladding oxidation begun at 7,442s. The cladding was failed by overstrain at 7,719s. The operator closed the PORV line block valve at 8,340s, which meant the close of the SBLOCA. However, the fuel was melted at 9,014s. One B-loop RCP was operated to supple the coolant to the core at 10,440s. This pump was stopped at 11,580s. The feed and bleed operation of the primary system was started to cooldown the core at 12,000s. However, the melted core material was relocated to the lower plenum at 13,440s, because of the molten pool formation in the core. General emergency declared at 18,000s.

Fig. 1 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.

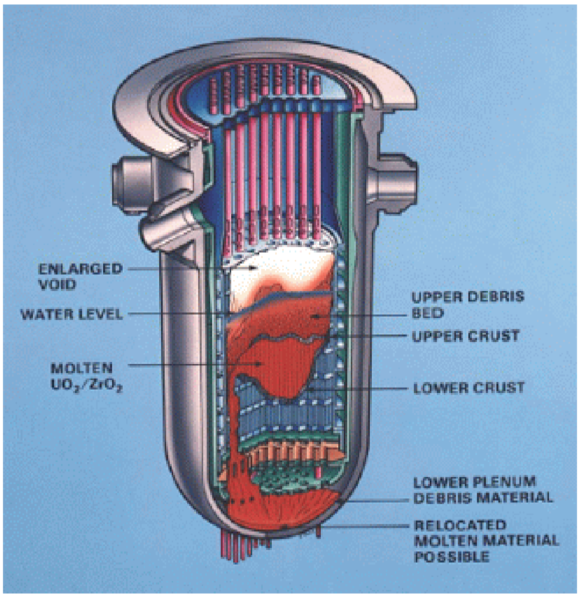


Fig. 1. End state of TMI-2 severe accident.

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. 2 shows an input nodalization of the CINEMA computer code for the TMI-2.

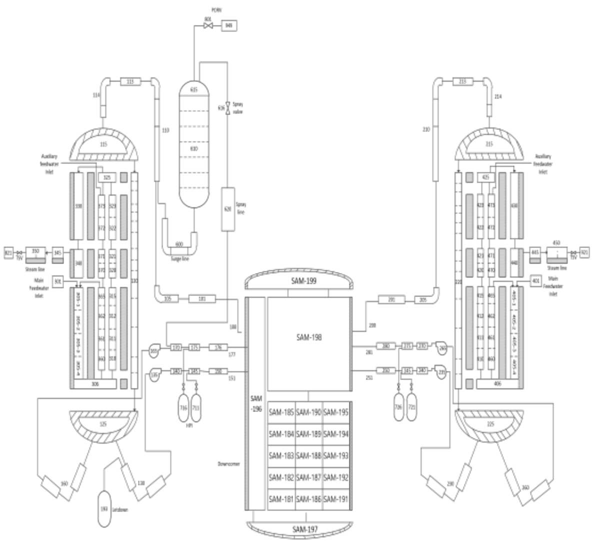


Fig. 2. CINEMA nodalization for TMI-2.

All primary and main secondary systems are modeled including the pressurizer, PORV, 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

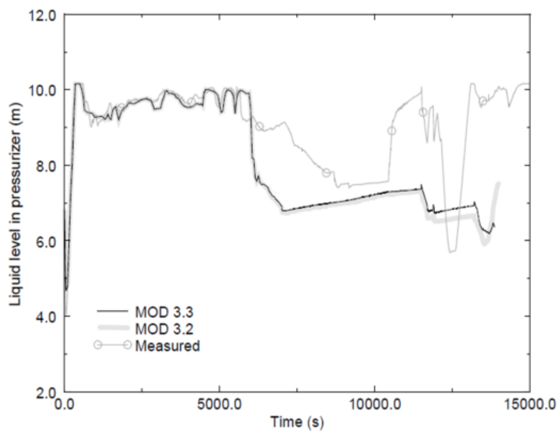
A steady state calculation was performed to verify the input nodalization of CINEMA for TMI-2 severe accident. The steady state conditions obtained from the simulation were used as initial conditions for the transient calculation.

Fig. 3, 4, and 5 shows the CINEMA results on liquid level in the pressurizer, pressurizer pressure, and fuel cladding surface temperature, respectively, to compare with the measured data and SCDAP/RELAP5 results. In Fig. 5, 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. In preliminary calculation, the pressurizer water drained completely after the PORV block valve was closed at 8,340s, which effectively affected core heat up, as shown in Fig. 5. This is a old SPACE input case in Figs 3 and 4. More accurate representations of the surge line flow and pressurizer might eliminate some of the problems encountered in preliminary TMI-2 analyses. For example, the junction connecting the surge line to hot leg A should be oriented horizontally rather than vertically (to reflect its true alignment) and the CCFL (Counter Current Flow Limitation) model should be activated at the junction connecting the surge line to the pressurizer, rather than at the hot leg junction. For this reason, the CCFL input parameters affect the pressurizer water drain to the core.

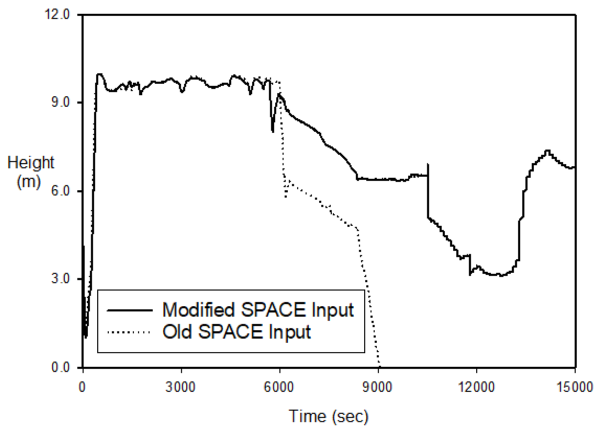
For this problem in the preliminary analysis, the SPACE input model on the CCFL was modified as strong as possible in the secondary analysis (modified SPACE input case), which resulted in the small water drain to the core as shown in Fig. 3. As shown in Fig. 4, the CINEMA result in strong CCFL input model in the modified SPACE input of the secondary case is very similar to the TMI-2 data in general.

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, which resulted in initiating the SBLOCA. 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 7MPa. After a pump termination at 10,000s, 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.

As shown in Fig. 5, the fuel cladding was quenched rapidly by water injection into the core at 10,440s, which was occurred by B-loop RCP operation in the secondary case. This results is not agree with TMI-2 data. For this reason, it is necessary to modify the CINEMA computer code on the quench model of the melted core material.



(Measured Data and SCDAP/RELAP5 Results)

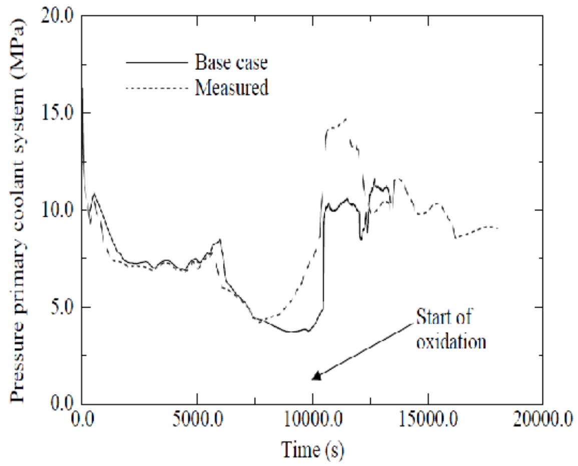


(CINEMA Results)

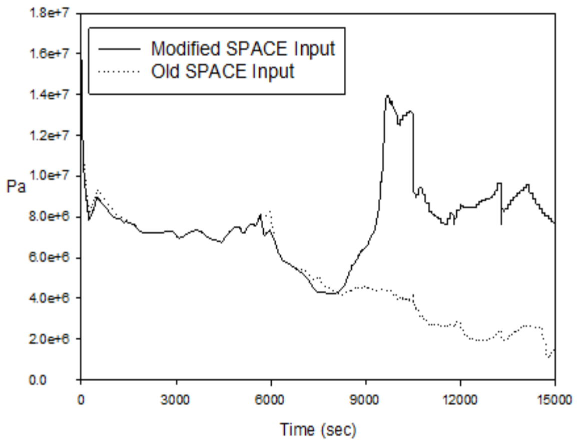
Fig. 3. CINEMA results on liquid level in pressurizer.

5. Conclusions

The surge line flow modeling effect on the core melt progression in the TMI-2 severe accident was analyzed using the CINEMA computer code. The CCFL input parameters in CINEMA affect the pressurizer water drain to the core through the pressurizer surge line. The CINEMA results on strong CCFL model are very similar to the TMI-2 data in general. More CINEMA analysis for a melted fuel relocation and quenching process in the core and lower plenum are necessary to simulate the TMI-2 severe accident.

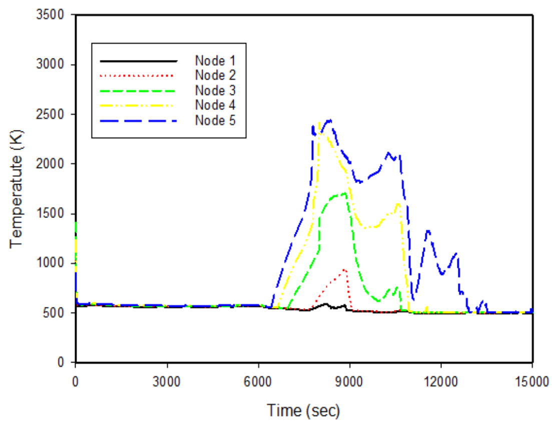


(Measured Data and SCDAP/RELAP5 results)

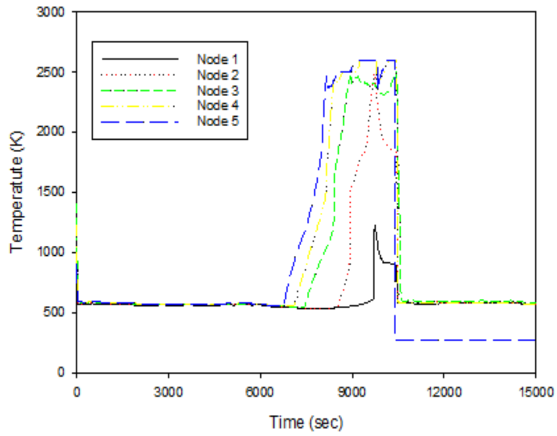


(CINEMA Results)

Fig. 4. Pressurizer pressure history in TMI-2 severe accident



(Old SPACE Input)



(Modified SPACE Input)

Fig. 5. CINEMA results on fuel cladding surface temperature.

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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, pp 34, 1989.