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Preliminary Analysis on Application Possibility of CINEMA Computer Code to External Reactor Vessel Cooling Evaluation for i-SMR

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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 developed for a stand-alone severe accident analysis [1]. 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 [2], SACAP (Severe Accident Containment Analysis Package) [3], and SIRIUS (SImulation of Radioactive nuclide Interaction Under Severe accident) [4], 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) [5] and SPACE (Safety and Performance Analysis CodE for nuclear power plants) models [6, 7], 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.

The i-SMR (innovative Small Modular Reactor) has been developing in Korea. The design and safety concepts were explained in reference [8]. This small reactor was adopted the IVR-ERVC (In-Vessel corium Retention through External Reactor Vessel Cooling) as a severe accident mitigation measure to prevent reactor vessel failure. This study is focused on a preliminary analysis on application possibility of CINEMA computer code to IVR-ERVC evaluation for the i-SMR. Best estimate calculations from the initiating events of the SBLOCA (Small Break LOCA) of 2-inch equivalent diameter at outside containment has been performed using the CINEMA computer code.

2. CINEMA Input Model

The input model for the CINEMA calculation of the i-SMR was a combination of the SPACE and COMPASS input models. Heat structures for the fuel rods and the lower part of the reactor vessel in the SPACE input model were replaced by the COMPASS input models. In the SPACE model, the reactor core was simulated as five channels to evaluate the thermal-hydraulic behavior in detail, and each channel was composed of five axial volumes, as shown in Fig. 1.

In the COMPASS input model of this analysis, the component numbers for the fuel and control rods were five and five. A steady state calculation was performed in order to verify the input nodalization of CINEMA for the i-SMR. The steady state results for a selected set of parameters were in very good agreement with the operating conditions of the i-SMR. The steady state conditions obtained from the simulation were used as initial conditions for the transient calculation.

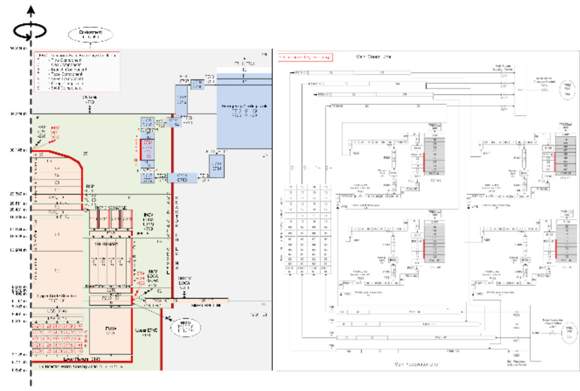


Fig. 1. CINEMA input model for NSSS (Nuclear Steam Supply System) of i-SMR.

3. CINEMA Model on the Heat transfer of the Outer Vessel Wall

In CINEMA computer code, the basic model of the downward-facing saturated pool boiling model on the outer vessel wall treats three heat transfer regimes for the ERVC analysis:

- fully-developed nucleate boiling with no dependence on the orientation of the boiling surface;
- transition boiling between the fully developed and film boiling regimes, in which the heat flux is obtained by logarithmic interpolation between the critical heat flux and the minimum heat flux, based upon the temperature difference between the surface and saturation; and
- stable film boiling, which depends upon the orientation of the boiling surface.

The boundaries between the heat transfer regimes are determined by a correlation for the critical heat flux, which separates fully developed and transition boiling, and a correlation for the minimum-stable-film-boiling heat flux, which separates transition and stable film boiling. Although heat transfer in the nucleate boiling regime is assumed to be independent of the orientation of the surface, the critical heat flux, which determines its upper limit, is dependent on surface orientation and is given by reference [9].

Similarly, the minimum-stable-film-boiling heat flux, which separates transition boiling from stable film boiling, is given as a function of θ. In the nucleate boiling regime, the heat flux, as a function of the difference between the surface temperature and the saturation temperature, is given as a function of temperature, is the simplified boiling curves from the MARCH 2.0 code [10], which can be used to calculate heat transfer coefficient. These are very similar to the MELCOR models.

The other option in CINEMA computer code for the downward-facing saturated pool boiling model on the outer vessel wall for the ERVC analysis is using a heat transfer coefficient (HTC) from the outer vessel wall to the water in containment. Heat flux on the outer downward spherical vessel wall is as follows;

$$q = h(T_w - T_c)$$

where q is heat flux, and h is heat transfer coefficient. T_w and T_c are outer vessel wall and coolant temperatures, respectively.

Heat transfer coefficient on the outer downward spherical vessel wall is approximately $10^5 – 4x10^5$ W/m^2 from the Fig. 2. For this reason, h is from 3,333 till 13,333 W/m^2.K using T_w of 403 K and T_c of 377 K. In this study, 6 HTCs are used as follows;

Case 0: 0, Cases 1: 500 W/m^2.K, Case 2: 1,000 W/m^2.K, Case 3: 2000 W/m^2.K, Case 4: 5,000 W/m^2.K, Cases 5: 15,000 W/m^2.K

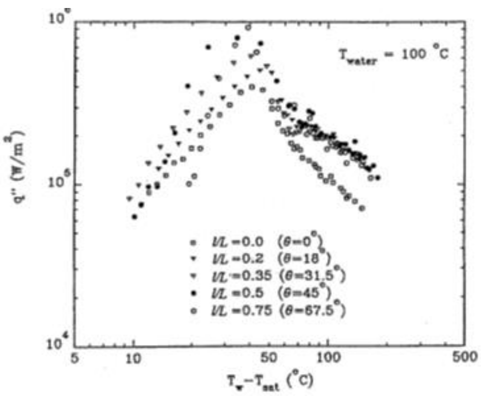


Fig. 2. Boiling curves at different locations along the heating surface by F.B.Cheung (saturated boiling water at 100 ºC).

4. Results and Discussion

Best estimate calculations from the initiating event of the SBLOCA at outside containment was performed using the CINEMA computer code. In this sequence, one EDV (Emergency Depressurization Valve) and one ERV (Emergency Recirculation Valve) were opened by low pressurizer level signal, which resulted in the ERVC condition. The transient calculation was performed for 345,600 sec (4 days). The accident was initiated by producing 2-inch break in the outer containment. The reactor and the RCP (Reactor Coolant Pump) s were assumed to be tripped at an accident initiation time. The in-vessel water inventory rapidly decreased and a boiling started in the core. The fuel began to heat up when the core was uncovered. Oxidation of the fuel cladding began when the cladding surface temperature reached 1,000 K and produced an oxidation heat. The fuel cladding was failed by ballooning. When the cladding surface temperature reached 1,700 K, oxidation of the zircaloy was accelerated as the steam was supplied from the bottom of the reactor vessel.

At about 2,100K of the cladding surface temperature, the zircaloy inside the oxide shell began to liquefy and the outer portion of the fuel pellets was dissolved. The relatively thin ZrO_2 shell ruptured at about 2,390 K because the shell strength decreased with the temperature increase. The bottom of the core dried out because a hot mixture of liquefied fuel and cladding had relocated downward. The debris formed at the bottom of the fuel rods, where the liquefied mixture had resolidified. The melting temperature of the zirconium dioxide is 2,390 K, and that of the uranium dioxide is 2,400 K in these calculations. The melted core material had relocated to the lower plenum of the reactor vessel. Finally, the reactor vessel was failed by a creep through a melt thermal attack without ERVC. However, the reactor vessel did not fail by ERVC.

Table I shows CINEMA results on the major events. In the base case using MELCOR model, the reactor vessel did not fail by ERVC. If the heat transfer coefficient from the outer vessel wall to the coolant is less than 500 W/m^2.K, the reactor vessel was failed in spite of ERVC condition. However, if the heat transfer coefficient is higher than 1,000 W/m^2.K, the reactor vessel did not fail by ERVC condition.

Table I: CINEMA results on the major parameters

	Base Case	Case 0	Case 1	Case 2	Case 3	Case 4	Case 5
Heat Transfer Coefficient (W/m^2.K) or Model	MELCOR Model (approx. 5,000)	0	500	1000	2000	5000	10000
Reactor Vessel Failure (Failure time, sec)	x	O (74,370)	O (120,970)	x	x	x	x
Corium Relocation Time (sec)	127,000	55,800	70,900	95,700	113,000	113,000	120,000
Corium Mass in lower Plenum(Metal, Oxide)(ton)	49.7 (24.8, 24.9)	-	-	49.2 (24.6, 24.6)	49.0 (24.9, 24.1)	49.0 (24.9, 24.1)	49.2 (24.4, 24.8)
Minimum Vessel Thickness (cm)	7.28	-	1.26	3.64	5.46	5.46	6.45
Maximum Heat Flux (W/m^2)	3.7x10^5	0	5.23x10^5	5.1x10^5	5.01x10^5	5.03x10^5	4.76x10^5

Fig. 3 shows CINEMA results inside and outside reactor vessel pressure. After the SBLOCAs occur at 0 sec, the inside reactor vessel pressurizer pressure rapidly decreases to the saturation pressure. As the EDV and the ERV valves were opened, inside pressure of the reactor vessel pressure is equivalent to the outside pressure. As the coolant began to boil, the expansion of the coolant caused by a boiling was able to compensate for the break flow, and the pressure maintained a saturation pressure. The volumetric flow out through the break is greater than the coolant expansion caused by a boiling, and the pressure began to decrease again. Fig. 4 shows CINEMA results on the water level inside containment. The water level in the containment is very high by coolant inside reactor vessel from the EDV valve opening. This water level maintained by outer water injection to achieve the ERVC condition.

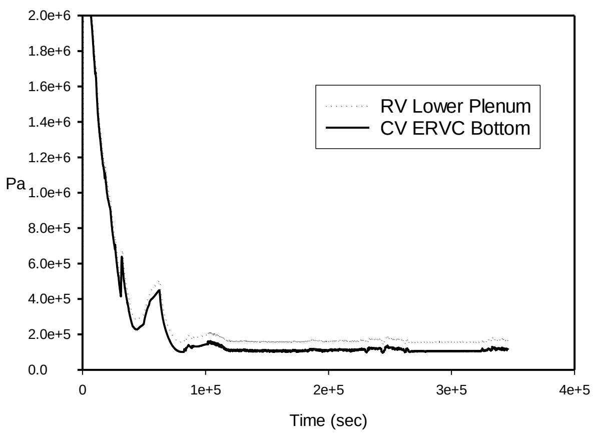


Fig. 3. CINEMA results on inside and outside reactor vessel pressures.

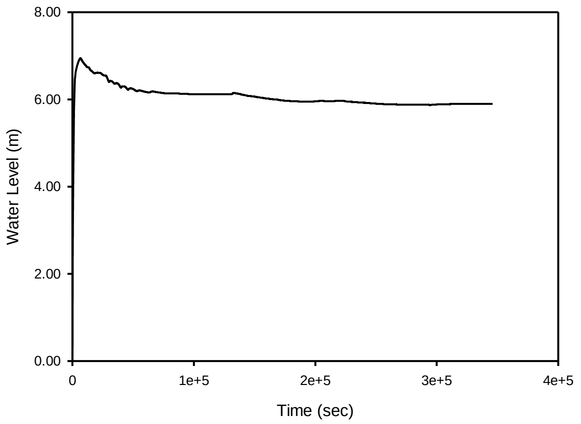
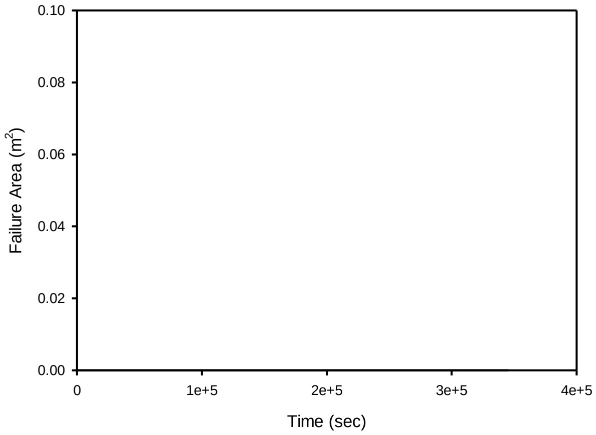
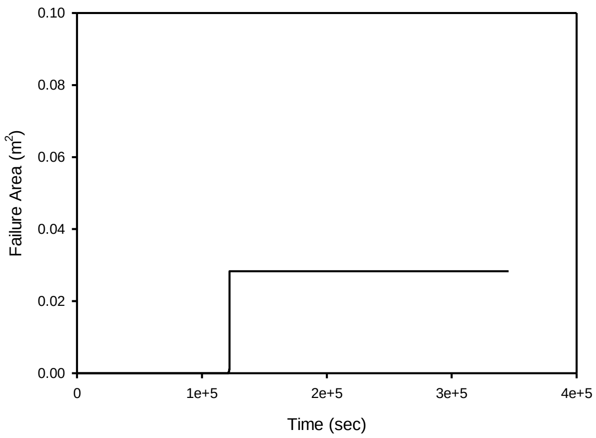


Fig. 4. CINEMA results on water level in the containment.

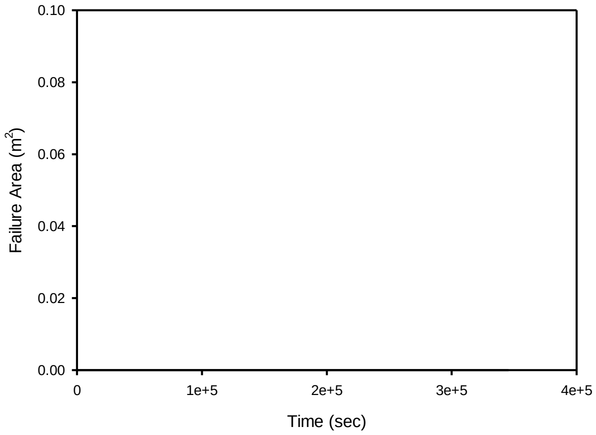
Figures 5 and 6 show CINEMA results on reactor vessel failure size and reactor vessel thickness, respectively. The initial reactor vessel thickness is 0.182 m.



(Base Case)

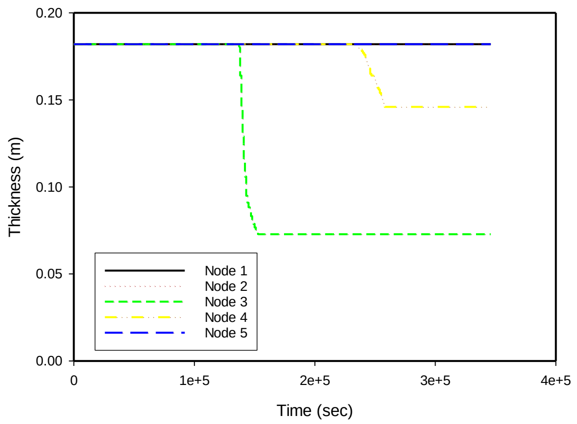


(HTC=500)

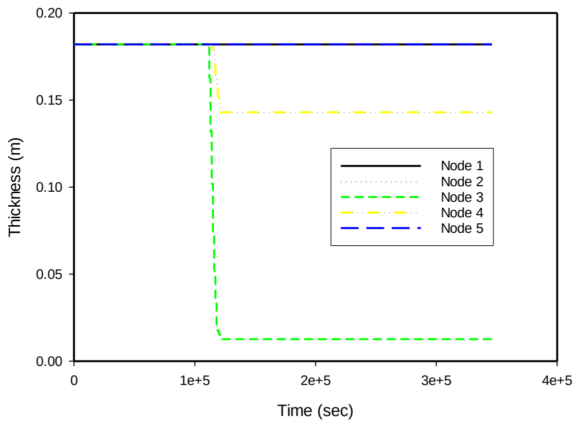


(HTC=1000)

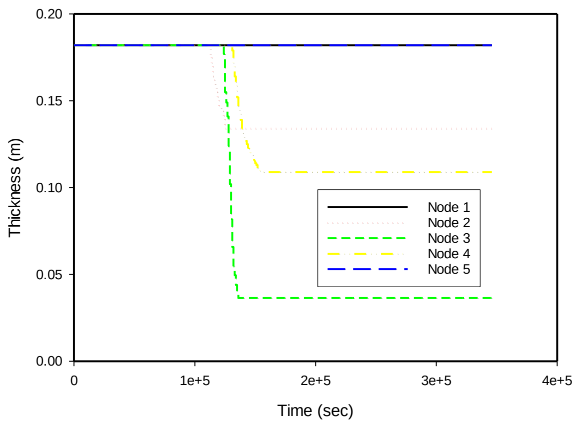
Fig. 5. CINEMA results on reactor vessel failure size.



(Base Case)



(HTC=500)



(HTC=1000)

Fig. 6. CINEMA results on reactor vessel thickness.

In the base case using MELCOR model, the reactor vessel did not fail by ERVC. However, the reactor vessel was melted to 60 % of the initial thickness. If the heat transfer coefficient from the outer vessel wall to the coolant is 500 W/m^2.K, the reactor vessel was failed in spite of ERVC condition. However, if the heat transfer coefficient is 1,000 W/m^2.K, the reactor vessel did not fail by ERVC condition, in spite of some melting of the reactor vessel.

5. Conclusion

A preliminary analysis on application possibility of CINEMA computer code to IVR-ERVC evaluation for the i-SMR. Best estimate calculations from the initiating event of the SBLOCA of 2-inch equivalent diameter at outside containment has been performed using the CINEMA computer code. In the base case using MELCOR model, the reactor vessel did not fail by ERVC in spite of some melting of the reactor vessel. If the heat transfer coefficient from the outer vessel wall to the coolant is less than 500 W/m^2.K, the reactor vessel was failed, in spite of ERVC condition. However, if the heat transfer coefficient is higher than 1,000 W/m^2.K, the reactor vessel did not fail by ERVC condition. More detailed analysis of the ERVC for the i-SMR is necessary to evaluate the IVR-ERVC.

ACKNOWLEDGMENTS

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