# Preliminary Analysis on IVR-ERVC for i-SMR using CINEMA Computer Code

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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) and SPACE (Safety and Performance Analysis CodE for nuclear power plants) models [5, 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 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 IVR-ERVC for the i-SMR using CINEMA computer code. Best estimate calculations from the initiating events of the SBLOCA (Small Break LOCA) of 2-inch equivalent diameter at outside containment in the i-SMR has been performed to evaluate the maintenance possibility of the reactor vessel integrity by ERVC.

## 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, respectively. The total masses of fuel and cladding are 21.4 ton and 6.4 ton, respectively. As shown in Fig.1, one SPACE volume is modelled for ERVC analysis. The lower reactor vessel is divided into 5 radial and 10 axial nodes. Lower vessel thickness is 0.182m.

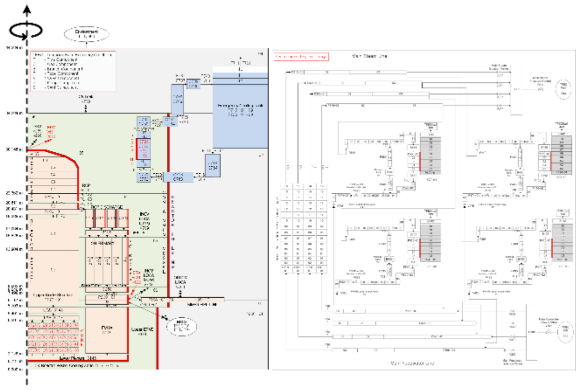

Fig. 1. CINEMA input model for NSSS (Nuclear Steam Supply System) of i-SMR.

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.

## 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, which can be used to calculate heat transfer coefficient.

## 4. Results and Discussion

Best estimate calculations from the initiating event of the SBLOCA at outside containment was performed using the CINEMA computer code for 96 hours(4 days, 345,600 sec). 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.

Table I shows CINEMA results on the major events. The accident was initiated by producing 2-inch break in the outer containment at 0 sec. 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 at 4,860 sec. 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 at 5,700 sec. 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.

The bottom of the core dried out at 26,600 sec. 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 at 32,900 sec . 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 at 128,000 sec. In this calculation, the reactor vessel did not fail by ERVC. The total hydrogen generation mass was estimated as 233.2 kg.

Table I: CINEMA results on the major events.

| Major Events | SBLOCA (sec) |
|---|---|
| Transient Initiated | 0 |
| One EDV and one ERV Opening | 20 |
| Core Uncovery | 4,860 |
| Start of FP Gap Release ($T_{clad}$=1,173 K) | 5,700 |
| Core Dryout | 26,600 |
| Fuel Melting and Relocation | 32,970 |
| Corium Relocation into Lower Head | 128,000 |
| Reactor Vessel Failure | No Failure |
| End of Calculation | 345,600 |
| Total Hydrogen Generation Mass | 233.2 kg |

Fig. 2 shows CINEMA results on pressures inside and outside reactor vessel. 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 by valve opening. 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.

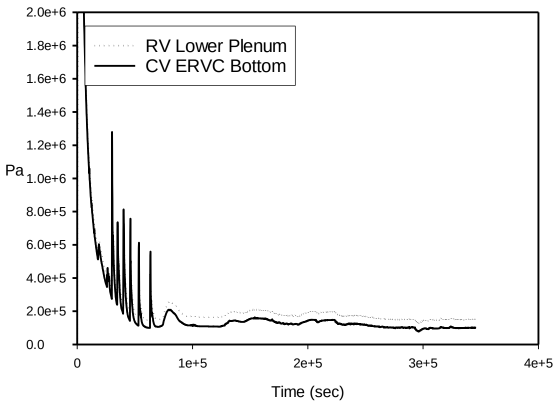


Fig. 2. CINEMA results on pressures inside and outside reactor vessel.

Fig. 3 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 to achieve the ERVC condition by PCCS (Passive Containment Cooling System) operation.

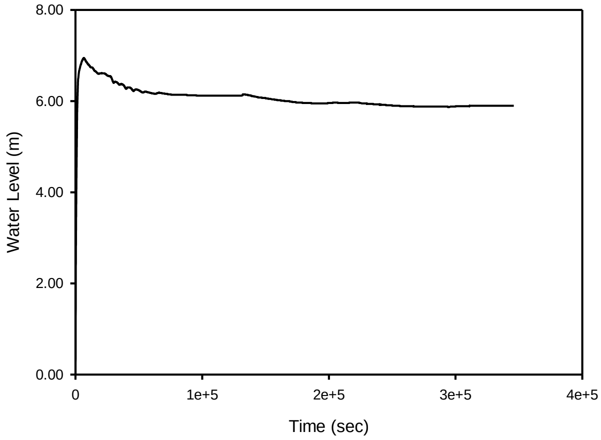


Fig. 3 CINEMA results on water level inside containment.

Fig. 4 shows CINEMA results on corium mass in lower plenum. At the end of calculation, the masses of oxide corium and metallic corium are 24.9 ton and 24.8 ton, respectively. The mass of oxide corium is equivalent to the metallic corium, which is different from the general PWR (Pressurized Water Reactor). Mass of metallic corium is higher than that of general PWR, because of higher initial mass.

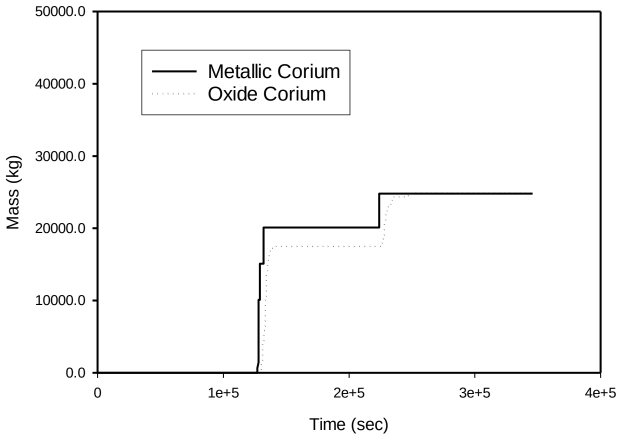


Fig. 4. CINEMA results on corium mass in lower plenum.

Fig. 5 shows CINEMA results on layer height of the corium in lower plenum. At the initial stage, all height was water. After the corium in the core initially relocates to the lower plenum at 128,000 sec, corium heights of oxide and metallic and debris increases, but the water height rapidly decreases. The two-phase mixture level is not zero, because of the condensed water injection through the ERV from the PCCS.

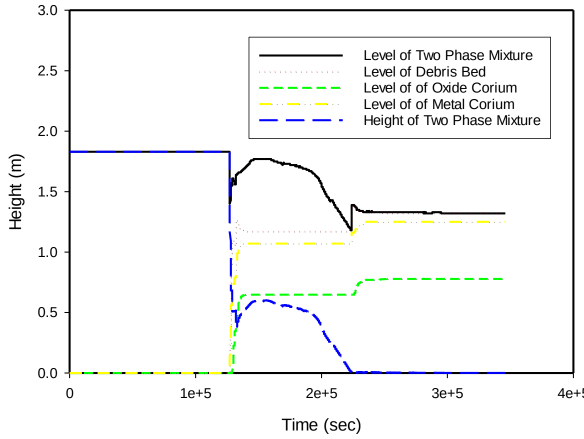


Fig. 5. CINEMA results on layer height of corium in lower plenum.

Fig. 6 shows CINEMA results on heat flux from the outer vessel wall to water inside containment. Maximum heat flux is approximately 0.4 MW/m$^2$. Fig. 7 shows CINEMA results on reactor vessel temperature. Node 1 and Node 10 are inside and outside reactor vessel, respectively. Maximum temperatures of Nodes 1 to 6 are melting temperature, which means the melting of reactor vessel inside.

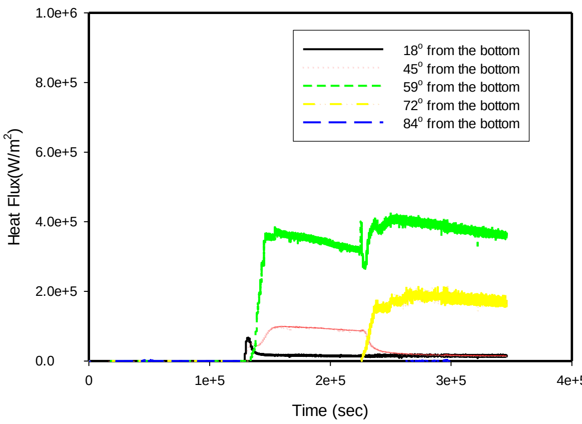


Fig. 6. CINEMA results on heat flux from the outer vessel wall to water inside containment.

Fig. 8 show CINEMA results on reactor vessel thickness. The initial reactor vessel thickness is 0.182 m. The reactor vessel did not fail by ERVC. However, the reactor vessel was melted to 60 % of the initial thickness.

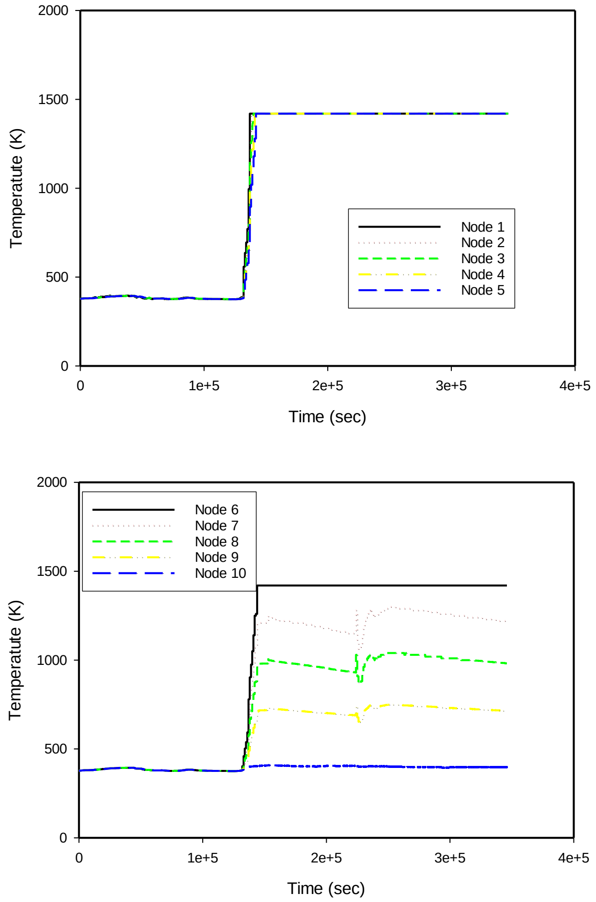


Fig. 7. CINEMA results on reactor vessel temperature.

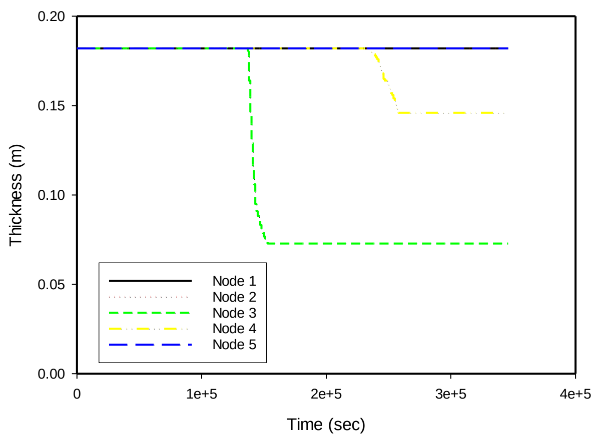


Fig. 8. CINEMA results on reactor vessel thickness.

## 5. Conclusion

A preliminary analysis on the IVR-ERVC for the i-SMR using CINEMA computer code. Best estimate calculations from the initiating event of the SBLOCA of 2-inch equivalent diameter at outside containment has been performed to evaluate the maintenance possibility of the reactor vessel integrity by the ERVC. The reactor vessel did not fail by the ERVC in spite of some melting of the reactor vessel. The reactor vessel was melted to 60 % of the initial thickness of 0.182m. Maximum heat flux from the outer reactor vessel wall to the water inside containment is approximately 0.4 $MW/m^2$. More detailed analysis of the ERVC for the i-SMR is necessary to evaluate the IVR-ERVC. In addition, bounding approach is necessary, such as thermal load analysis from the corium to the reactor vessel, the maximum heat removal analysis of the CHF(Critical Heat Flux) at the outer vessel wall, and reactor vessel structure analysis.

## ACKNOWLEDGMENTS

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## REFERENCES

[1] KHNP, KAERI, FNC, KEPCO E&C, CINEMA User Manual, S11NJ16-2-E-TR-7.4, Rev. 0, 2018.
[2] J.H. Song, D.G. Son, J.H. Bae, S.W. Bae, K.S. Ha, Chung, B.D., Choi, Y.J. CSPACE for a Simulation of Core Damage Progression during Severe Accidents, Nuclear Engineering and Technology, 53, 2021.
[3] FNC, SACAP User Manual, S11NJ16-2-E-TR-7.4, Rev. 0., 2017.
[4] K.S. Ha, S.I. Kim, H.S. Kang, D.H. Kim, SIRIUS: a Code on Fission Product Behavior under Severe Accident, Transactions of the Korean Nuclear Society Spring Meeting, Jeju, Korea, 2017.
[5] S.J. Ha, Development of the SPACE Code for Nuclear Power Plants, Nuclear Engineering and Technology, 43, 2011.
[6] KHNP, KEPCO E&C, KAERI, SPACE User Manual, S 06NX08-K-1-TR-36, Rev. 0, 2017.
[7] KHNP, KEPCO E&C, KAERI, SPACE Theoretical Manual, S06NX08-K-1-TR-36, Rev. 0, 2017.
[8] J.H. Ham, S.H. Kim, S.K. Jeong, Preliminary severe accident analysis of INCV-LOCA in i-SMR using CINEMA code, Trans. of KNS Spring Meeting, Jeju Korea, Mar 9-10, 2024.
[9] M. S. El-Genk and Z. Guo, Transient Critical Heat Flux for Inclined and Downward Facing Flat Surfaces, ANS Proceedings, HTC-6 Volume 6, 1992, National Heat Transfer Conference, San Diego USA, August 9-12, 1992.