

Analysis on IVR through ERVC for Integrity Maintenance of Reactor Vessel in i-SMR using CINEMA

Rae-Joon Park ^{a*}, Seokgyu Jeong ^a, Jaehyun Ham ^a, Donggun Son ^a, Sang Ho Kim ^a

^aKorea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-Gu, Daejeon, Korea

*Corresponding author: rjpark@kaeri.re.kr

***Keywords :** innovative small modular reactor, in-vessel retention, external reactor vessel cooling, CINEMA

1. Introduction

As an integrated severe accident computer code development in Korea, CINEMA (Code for INtegrated severe acciEnt 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 design and safety concepts of the i-SMR (innovative Small Modular Reactor) 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 an 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 effectiveness of ERVC in maintaining the reactor vessel integrity.

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. In the COMPASS input model of this analysis, the component numbers for the fuel and control rods were five and five, respectively. One SPACE volume is modelled for ERVC analysis. The lower reactor vessel is divided into 10 radial and 10 axial nodes. More details are in reference [8].

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 IVR-ERVC Analysis

Fig. 1 shows general CINEMA model on the corium behavior in the lower plenum for IVR-ERVC analysis. The CINEMA models upper coolant layer, debris bed, low density metallic layer, and high density oxidic layer. The metallic layer is composed of light Fe-Zr and the oxidic layer of $\text{UO}_2\text{-ZrO}_2$. Since the metallic layer is assumed to contain no uranium, the heat generation is totally provided by the lower oxidic layer. In this model, other configurations such as three-layer system are not considered. The steady state governing equations for a given configuration are formulated in each layer.

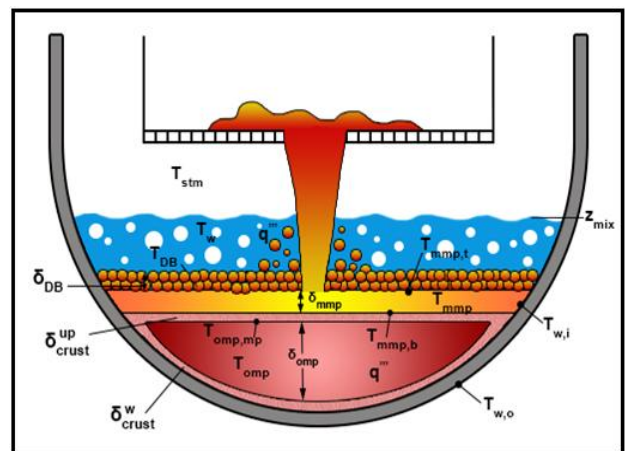


Fig. 1. General CINEMA model on the corium behavior in the lower plenum.

The thermal load analysis for IVR-ERVC evaluation is concentrated on the heat flux distribution in consideration of a focusing effect in the thin metallic layer. This effect of the metallic layer is mainly determined by the molten pool configuration in the lower plenum of the reactor vessel. The melt pool configurations inside the lower plenum affect the initial thermal load to the outer reactor vessel and play a key role in determining the integrity of the reactor vessel.

Unlike the APR1400 and SMART100, which are equipped with reactor vessel insulation, the i-SMR has no insulation. Consequently, while co-current flow, where water and steam flow in the same direction, occurs on the outer wall of the reactor vessel where insulation is installed. Upper coolant is injected into the containment vessel, and steam generated on the outer wall of the reactor vessel moves upward, forming a counter-current flow in the i-SMR without reactor vessel insulation. In the event of a severe accident in the i-SMR, the pressure inside and outside the reactor vessel can rise to approximately 50 bars. To apply critical heat flux correlations developed for atmospheric pressure to high pressure, it is necessary to accurately predict the critical heat flux as a function of pressure under high-pressure conditions.

CINEMA allows the use of pool boiling curves for heat transfer from the outer wall of the reactor vessel to the coolant. In other words, the CINEMA code allows users to input heat transfer coefficients and utilize heat transfer correlations embedded within the system. 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].

CINEMA's structural integrity assessment model for the lower hemisphere of the reactor vessel includes penetration failure and creep failure models utilizing the Larson-Miller factor. The i-SMR has no penetrations in the lower hemisphere of the reactor vessel.

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 tripped at 10 seconds. 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. 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 debris formed at the bottom of the fuel rods, where the liquefied mixture had resolidified. The melted core material had relocated to the lower plenum of the reactor vessel at 23,710 sec. In this calculation, the reactor vessel did not fail by ERVC.

Table I: CINEMA results on the major events

Major Events	SBLOCA (sec)
Transient Initiated	0
Reactor Scram	10
RCPs Off	10
Core Uncover	14,010
Corium Relocation into Lower Head	23,710
Reactor Vessel Failure	No Failure
End of Calculation	345,600

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. The pressure of bottom inside containment vessel is a little higher than that of reactor vessel lower plenum due to the hydraulic pressure.

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. 4 shows CINEMA results on corium mass in lower plenum. The melted metallic core material had relocated to the lower plenum of the reactor vessel initially at 23,710 sec. At the end of calculation, the masses of oxide corium and metallic corium are 21.1 ton and 12.3 ton, respectively.

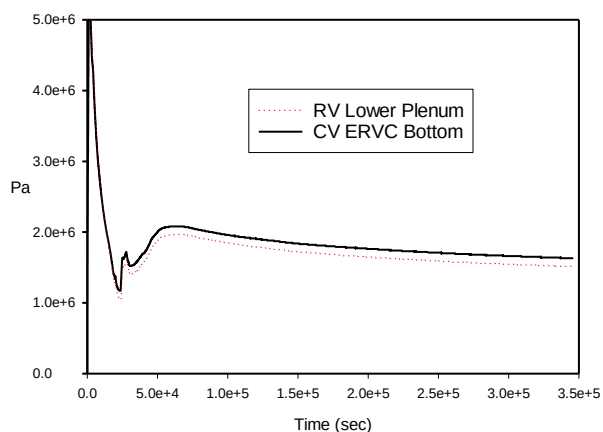


Fig. 2. CINEMA results on pressures inside and outside reactor vessel.

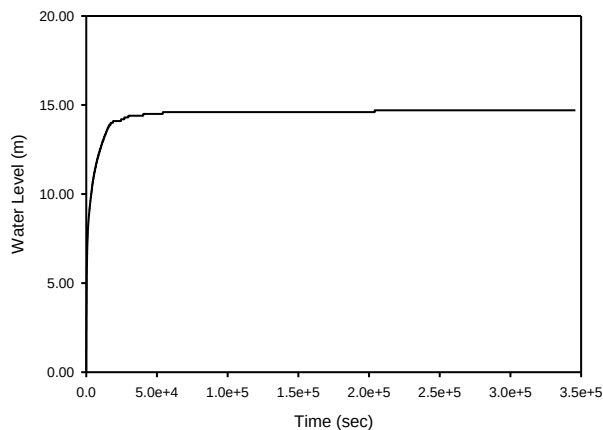


Fig. 3. CINEMA results on water level inside containment.

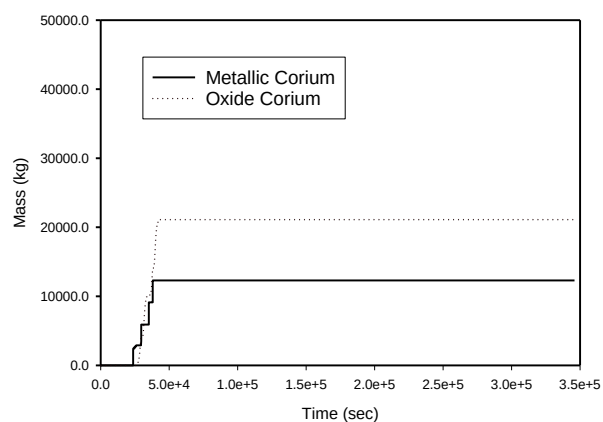


Fig. 4. CINEMA results on corium mass in lower plenum.

Fig. 5 shows CINEMA results on layer height of the corium in lower plenum. At the initial stage, all height as

water. After the corium in the core initially relocates to the lower plenum at 23,710 sec, corium heights of oxide and metallic and debris increases, but the water height rapidly decreases. Fig. 6 show CINEMA results on reactor vessel thickness. Radial Node 1 is the bottom of the reactor vessel and Radial Node 10 is the top. Radial Node 1 to 6 and Radial Node 10 are not melted by ERVC. However, the reactor vessel of Radial Node 8 was melted to 70 % of the initial thickness. Radial Node 8 is located at the metallic layer in the corium pool. From the CINEMA results, the reactor vessel did not fail by ERVC.

Fig. 7 show CINEMA results on reactor vessel temperature at Radial Node 8. Axial Node 1 is inside of the reactor vessel and Axial Node 10 is outside. Axial Node 1 to 7 maintain melting temperature of the reactor vessel. However, the outside of the reactor vessel maintains approximately 500 K by water contact.

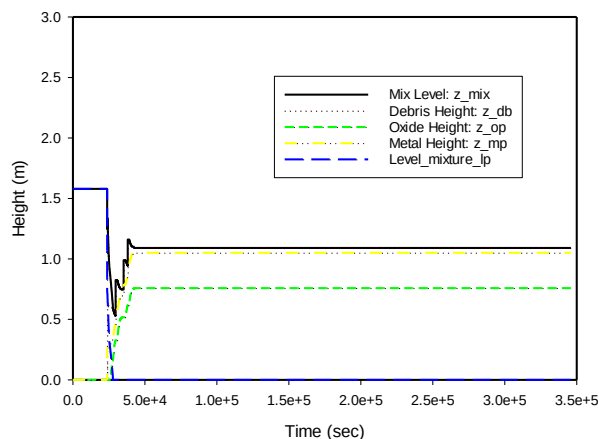
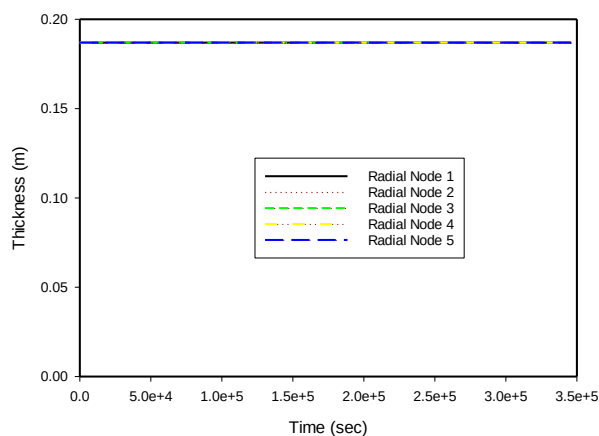


Fig. 5. CINEMA results on layer height of corium in lower plenum.



5. Conclusion

An IVR-ERVC analysis for the i-SMR using CINEMA computer code has been performed to evaluate the effectiveness of ERVC in maintaining the reactor vessel integrity. Best estimate calculations from the initiating event of the SBLOCA of 2-inch equivalent diameter at outside containment has been performed. The reactor vessel was melted to 70 % of the initial thickness at the metallic layer in the corium pool. However, the reactor vessel did not fail by the ERVC in spite of some melting of the reactor vessel. More detailed analysis on bounding approach for the IVR-ERVC 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

This work was supported by the Innovative Small Modular Reactor Development Agency grant funded by the Korea Government(MSIT) (No. RS-2023-00259516).

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.

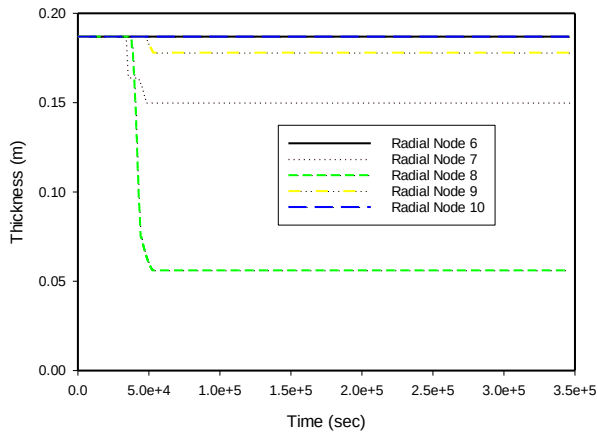


Fig. 6. CINEMA results on reactor vessel thickness.

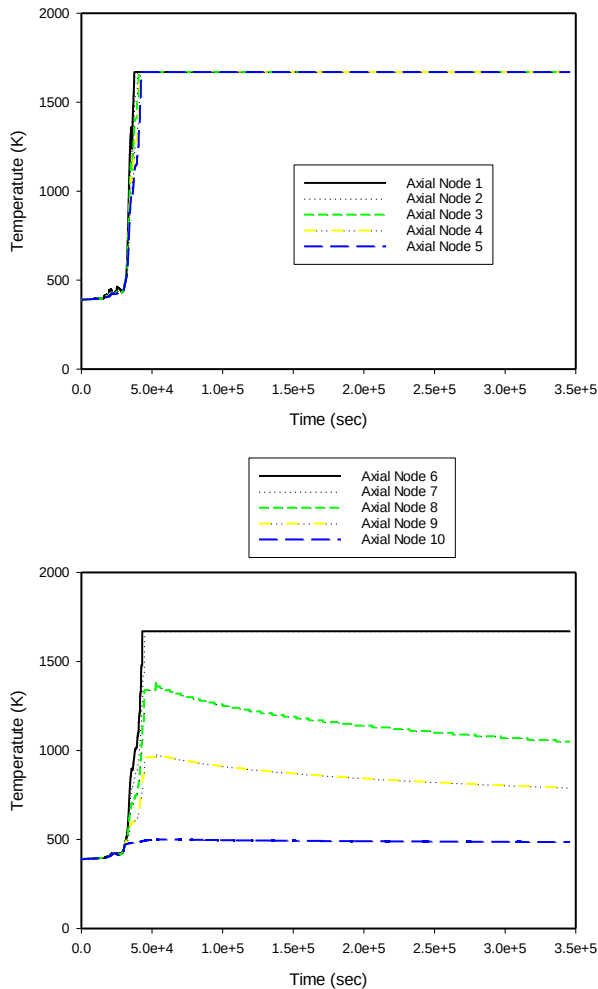


Fig. 7. CINEMA results on reactor vessel temperature at Radial Node 8.