

Preliminary Analysis on Thermal Load for IVR-ERVC of i-SMR using CINEMA Computer Code

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1. Introduction

The i-SMR (innovative Small Modular Reactor) has been developing in Korea. The design and safety concepts were explained in reference [1]. 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. To achieve this, thermal load from the corium pool to the reactor vessel wall is lower than the maximum heat removal of the CHF (Critical Heat Flux) at the outer reactor vessel wall. For this reason, it is necessary to analyze the thermal load for the IVR-ERVC of the i-SMR. This study is focused on a preliminary analysis on thermal load for the IVR-ERVC of the i-SMR using CINEMA (Code for INtegrated severe acciDent Management Analysis) computer code [2].

The CINEMA computer code is composed of CSPACE [3], SACAP (Severe Accident Containment Analysis Package) [4], and SIRIUS (SIMulation of Radioactive nuclide Interaction Under Severe accident) [5], 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 [6, 7, 8], 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.

2. CINEMA Model for Thermal Load Analysis

Fig. 1 shows a conceptual schematic of two-layered melt pool configuration. The upper layer is assumed to be a light metallic layer of Fe-Zr and the lower to be an 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 study, other configurations such as three-layer system are not considered. The steady state governing equations for a given configuration are formulated in each layer, that is, molten oxidic layer, upper oxidic crust layer, downward oxidic crust layer, light metallic layer, and vessel wall, by considering energy and the heat transfer between each layer.

The thermal load analysis 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. A numerical model in the CINEMA computer code was developed for a thermal load response to the outer RPV during a severe accident. The model is based on a simple mechanistic model using an energy balance equation. The governing equations were solved using a non-linear Newton-Raphson method. The details are in Reference [2].

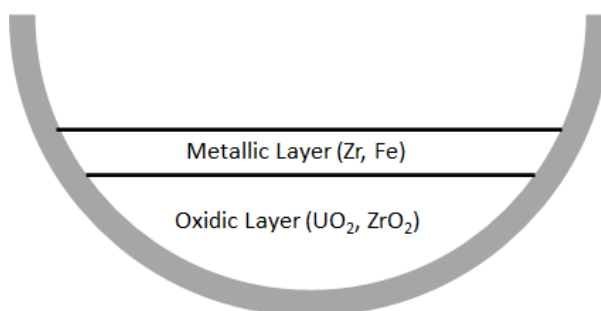


Fig. 1. General two-layer formation in the lower plenum.

3. CINEMA Input Model

The input model for the CINEMA calculation of the i-SMR was a combination of the SPACE and COMPASS input models. Fig. 2 shows CINEMA input model for IVR-ERVC analysis of i-SMR. This is general standard input for the thermal load analysis. The total masses of fuel, cladding ZrO_2 are 23.4 ton, 4.6 ton, and 2.64 ton, respectively. As shown in Fig.1, one SPACE volume is modelled for ERVC analysis. The inner and outer radiuses are 1.576 m and 1.763 m, respectively. Lower vessel thickness is 0.187 m. The lower reactor vessel is divided into 10 radial and 10 axial nodes. Annular volume between the outer vessel wall and the inner containment wall is modelled by SPACE volumes. The

PCCS (Passive Containment Cooling System) is modelled to control the pressure inside the containment.

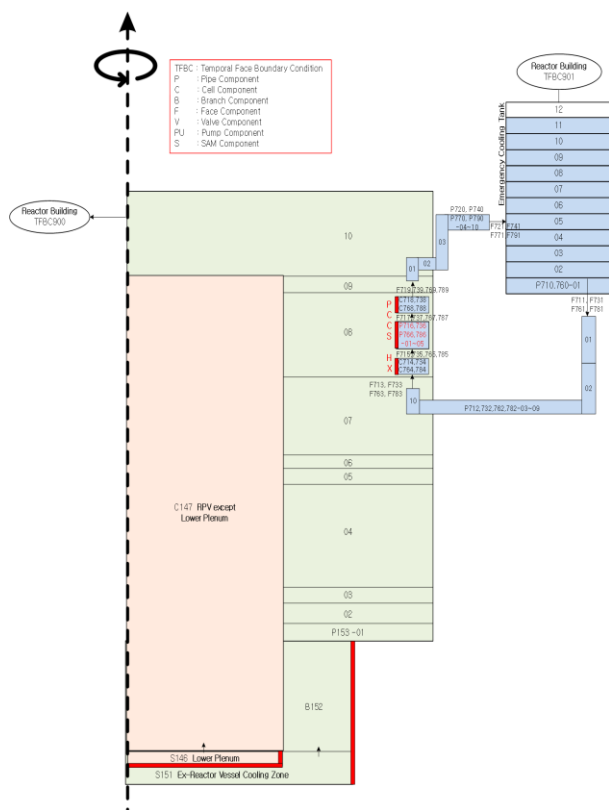


Fig. 2. CINEMA input model for IVR-ERVC analysis of i-SMR.

4. Results and Discussion

In this calculation, it is assumed that all molten core material was relocated to the lower plenum of the reactor vessel at initial 10 sec for 40,000 sec. Fig. 3 shows CINEMA results on pressures inside and outside reactor vessel. The containment pressure of lower part of the reactor vessel is higher than that of the inside reactor vessel by hydrostatic of ERVC water. Fig. 4 shows water level inside containment. The water level is maintained, because the boiled water is condensed in the PCCS.

Fig. 5 shows CINEMA results on corium mass in lower plenum. The masses of oxidic corium and metallic corium are 26.0 ton and 14.5 ton, respectively. The mass of oxidic corium is a litter higher than 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 structural material.

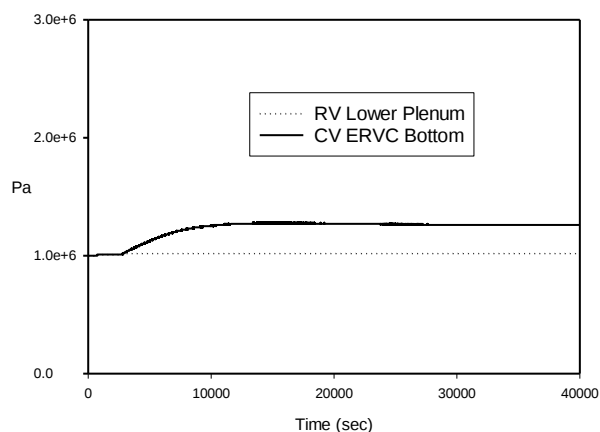


Fig. 3. CINEMA results on pressures inside and outside reactor vessel.

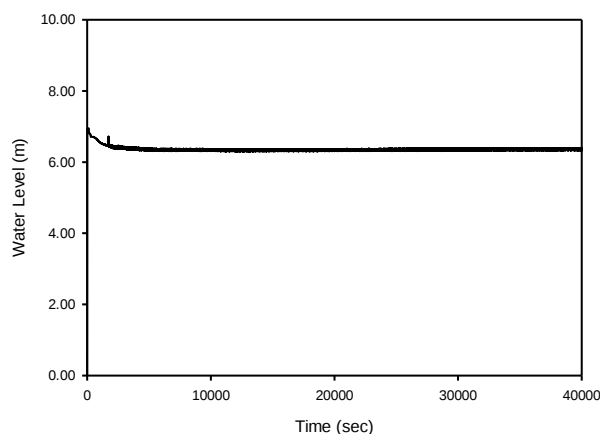


Fig. 4. CINEMA results on water level inside containment.

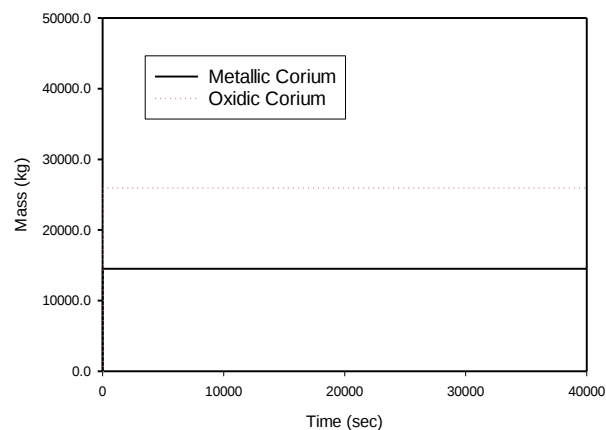


Fig. 5. CINEMA results on corium mass in lower plenum.

Fig. 6 shows CINEMA results on layer height of the corium in lower plenum. Heights of the oxidic and metallic layers are 0.813m and 0.297m, respectively, which means the 61° and 72° from the bottom of the reactor vessel.

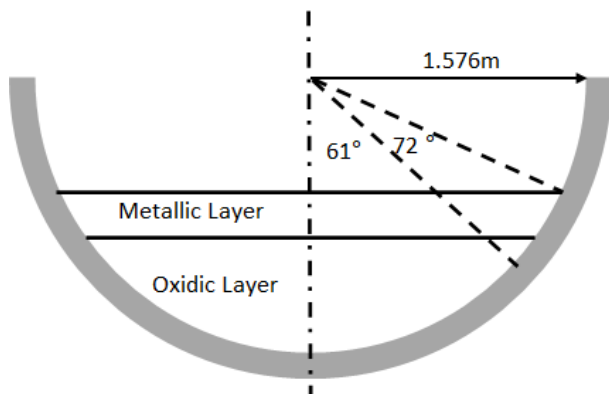


Fig. 6. CINEMA results on configuration of the layer height in lower plenum.

Fig. 7 shows CINEMA results on heat flux from the outer vessel wall to water inside containment. Radial Nodes 1 to 10 are 0.2 m, 0.4 m, 0.6 m, 0.8 m, 1.0 m, 1.2 m, 1.384 m, 1.45 m, 1.506 m, 1.576 m from the center point of the circle. Nodes 1 to 7 are oxidic layer, and nodes 8 and 9 are metallic layer. Node 10 is no corium in the lower plenum. Maximum heat flux is approximately 0.42 MW/m² at Node 8, which is a metallic layer. This maximum value comes from the focusing effect of the metallic layer. Fig. 8 shows CINEMA results on reactor vessel temperature at radial Node 8. Axial Node 1 and Node 10 are inside and outside reactor vessel, respectively. Maximum temperatures of Axial Nodes 1 to 5 are melting temperature, which means the melting of reactor vessel inside.

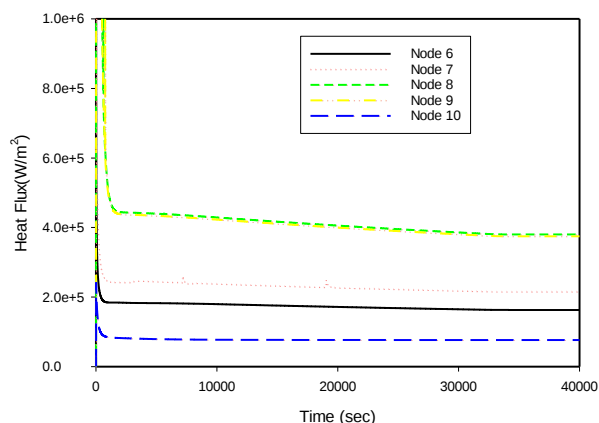


Fig. 7. CINEMA results on heat flux from the outer vessel wall to water inside containment.

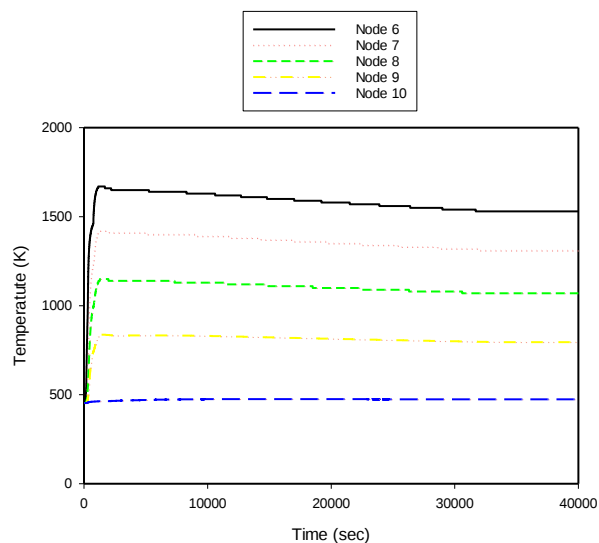
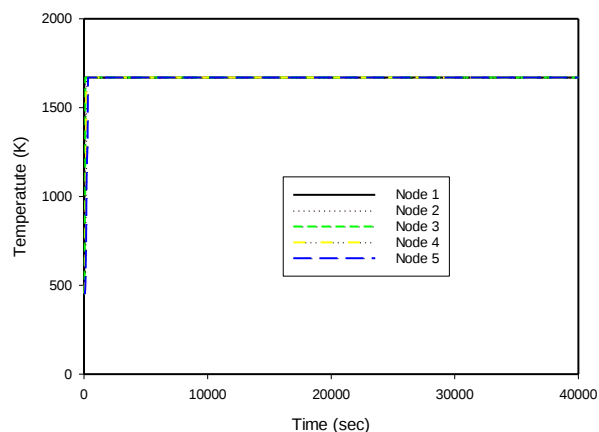


Fig. 8. CINEMA results on reactor vessel temperature.

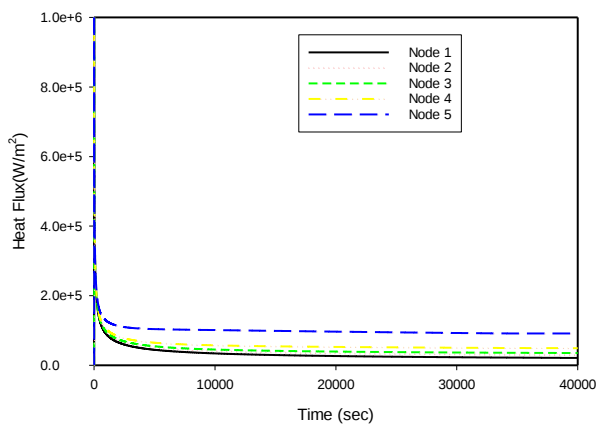


Fig. 9 show CINEMA results on reactor vessel thickness. The initial reactor vessel thickness is 0.187 m. The reactor vessel did not fail by ERVC. However, the reactor vessel was melted to 50 % of the initial thickness at Nodes 8 and 9 of the metallic layers.

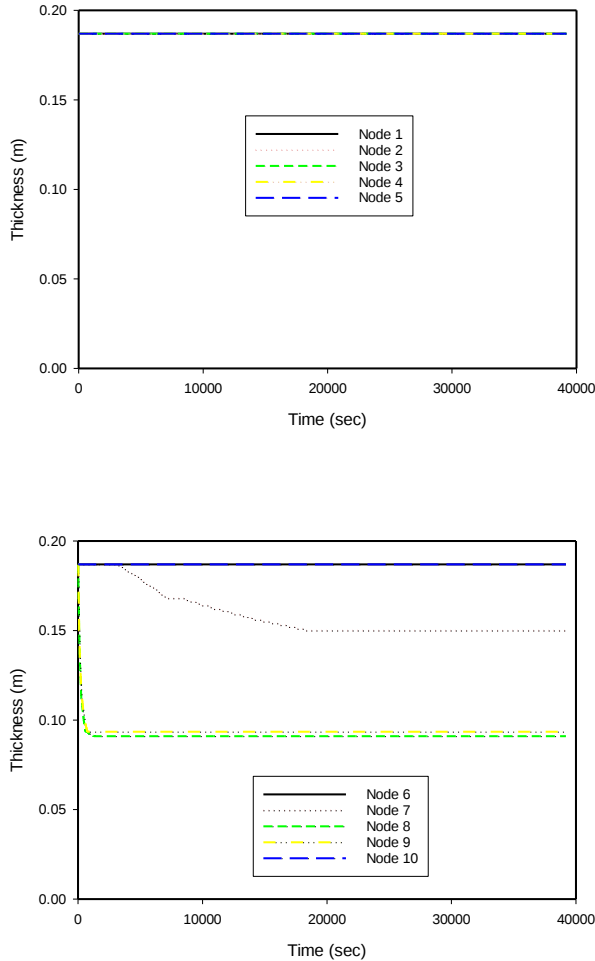


Fig. 9. CINEMA results on reactor vessel thickness.

5. Conclusion

A preliminary analysis on the thermal load from the corium pool to the reactor vessel wall for the IVR-ERVC of the i-SMR has been performed using CINEMA computer code. The reactor vessel did not fail by the ERVC in spite of some melting of the reactor vessel. The reactor vessel was melted to 50 % of the initial thickness of 0.187m. Maximum heat flux from the outer reactor vessel wall to the water inside containment is approximately 0.42 MW/m^2 , due to the focusing effect of the metallic layer. More detailed analysis on the thermal load is necessary. For the evaluate the IVR-ERVC of the

i-SMR, the maximum heat removal analysis of the CHF(Critical Heat Flux) at the outer vessel wall and reactor vessel structure analysis will be formed.

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REFERENCES

- [1] 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.
- [2] KHNP, KAERI, FNC, KEPCO E&C, CINEMA User Manual, S11NJ16-2-E-TR-7.4, Rev. 0, 2018.
- [3] 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.
- [4] FNC, SACAP User Manual, S11NJ16-2-E-TR-7.4, Rev. 0., 2017.
- [5] 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.
- [6] S.J. Ha, Development of the SPACE Code for Nuclear Power Plants, Nuclear Engineering and Technology, 43, 2011.
- [7] KHNP, KEPCO E&C, KAERI, SPACE User Manual, S 06NX08-K-1-TR-36, Rev. 0, 2017.
- [8] KHNP, KEPCO E&C, KAERI, SPACE Theoretical Manual, S06NX08-K-1-TR-36, Rev. 0, 2017.