Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22-23, 2025

Thermal-Structural Analysis of LAVA-4 Reactor Vessel Lower Head under In-Vessel Gap Cooling Condition

Moon Won Song[a,b*]
[a]*High-enthlapy Plasma Research Center, Jeonbuk National University*
[b]*School of Convergence Defense Industry, Jeonbuk National University*
[*]*Corresponding author: moon1song@jbnu.ac.kr*

Keywords : severe accident, in-vessel, gap cooling, thermal-structural analysis

1. Introduction

Severe accidents in nuclear power plants, such as TMI-2, Chernobyl, and Fukushima, can result in reactor vessel rupture due to the thermal load from the relocated corium in the lower head. However, in the TMI-2 accident, the reactor vessel remained intact due to the gap cooling phenomenon, which significantly reduced the thermal loads on the lower head [1]. Gap cooling occurs when water naturally infiltrates the gap between the corium and the inner wall of the lower head, not only preventing direct contact but also effectively removing heat from the relocated corium.

Due to its significance, extensive research has been conducted on gap formation and the thermal behavior associated with gap cooling. In particular, the LAVA-4 experiment provided critical data for understanding gap formation and temperature behavior in a simulated lower head environment. Furthermore, models of gap formation, water infiltration, and heat transfer were developed to predict gap size and temperature behavior, and these models were validated against LAVA-4 experimental results [2].

This study aims to analyze the thermal-structural behavior of a reactor vessel lower head under gap cooling conditions. The thermal boundary conditions, applied in the thermal-structural analysis in this study, were obtained from models validated against LAVA-4 data, incorporating gap size, water penetration rate, and temperature histories. The thermal-structural analysis results were compared with LAVA-4 experimental data and demonstrate the stress, temperature, and displacement behaviors of a reactor vessel lower head under the in-vessel gap cooling conditions.

2. Temperature-Displacement Coupled Analysis

A finite-element solver, Abaqus 2024 learning edition, was utilized for the thermal-structural analysis of LAVA-4 reactor vessel lower head. A simple axisymmetric 2D model was applied as shown in Figs. 1 (a) and (b). Fig. 1(a) represents configurations of corium simulant(Al_2O_3), reactor vessel lower head, and penetrating water into gap between the corium simulant and the reactor vessel. Fig. 1(b) presents the elements of the LAVA-4 reactor vessel lower head and boundary conditions for the finite-element analysis. As can be seen in Fig. 1(b) the thermal boundary conditions are divided into two regions based on the front of the penetrating water between the corium simulant and the inner wall of the reactor vessel lower head. The region ahead of the water front receives heat flux from the corium simulant, while the convective boundary was applied on the behind of the water front. The following parameters were adopted from time-dependent calculation results validated in previous studies;

- gap size [3]
- water penetration rate [4]
- heat flux ahead of the water front [5]
- heat transfer coefficient behind the water front [5]

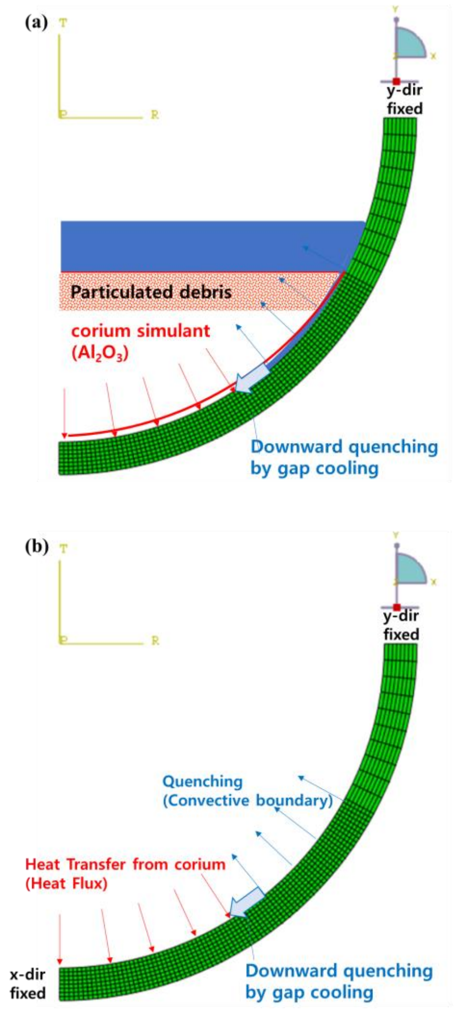


Fig. 1. LAVA-4 reactor vessel lower head;
(a) schematic representation with corium simulant
(b) thermal boundary conditions

Considering the symmetry of the 2D axisymmetric model, fixed boundaries in the y- and x-directions were applied to the elements tangent to the x-axis and y-axis, respectively.

80 elements were generated along the inner wall's circumferential direction in the reactor vessel, matching the number of volumes used in the previous study [5]. In addition, the heat flux and the heat transfer coefficient applied to each element were set to be the same as the results of the previous study [5].

The LAVA-4 reactor vessel has an outer radius of 250 mm and a thickness of 25 mm. The LAVA-4 reactor vessel is made of SA516 Gr. 70. The thermal conductivity, specific heat, thermal expansion coefficient, and plastic strain of the LAVA-4 reactor vessel were defined based on the temperature-dependent properties of SA516 Gr. 70. To simplify the analysis, bilinear model was applied to the plastic deformation region.

3. Results and Discussion

To verify the correct implementation of the thermal boundary conditions to the FE model, the temperature history from the Abaqus FE model was compared with that of a previous study using the same boundary conditions [ ], as shown in Fig. 2. The results confirmed the successful implementation of the thermal boundary conditions from previous study to the FE model.

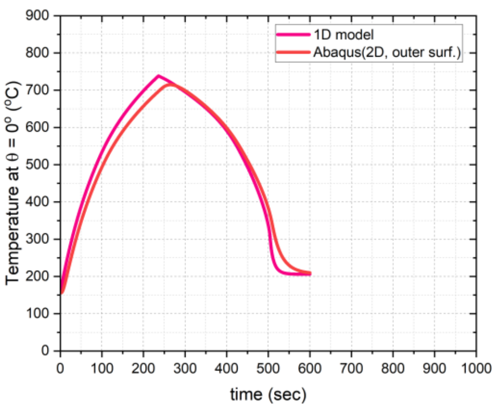


Fig. 2. Temperature histories of the LAVA-4 reactor vessel in the previous study(1D) and Abaqus FE model(2D)

Fig. 3 and 4 present the temperature distribution, displacement distribution, Mises stress distribution, and principal stress (circumferential) distribution at 236 sec, when the maximum temperature occurs, and at 221.2 sec, respectively. The maximum values of temperature, displacement, and were 835.1 ℃ and 2.336 mm, respectively. The maximum Mises stress occurs at 600 seconds, after the heating and cooling of the reactor vessel, with a value of 286.9 MPa as shown in Fig. 5.

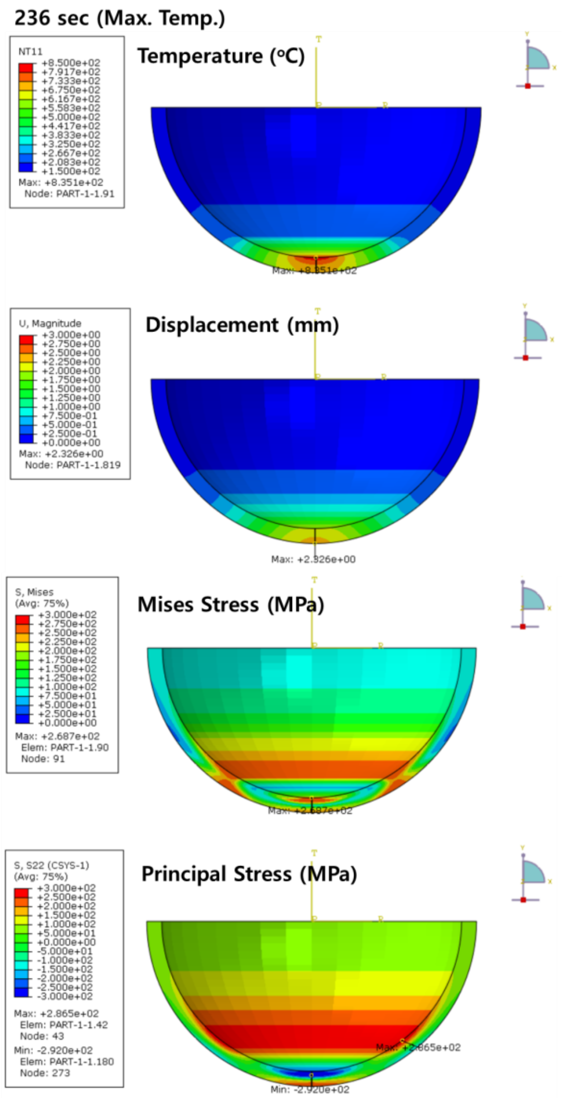
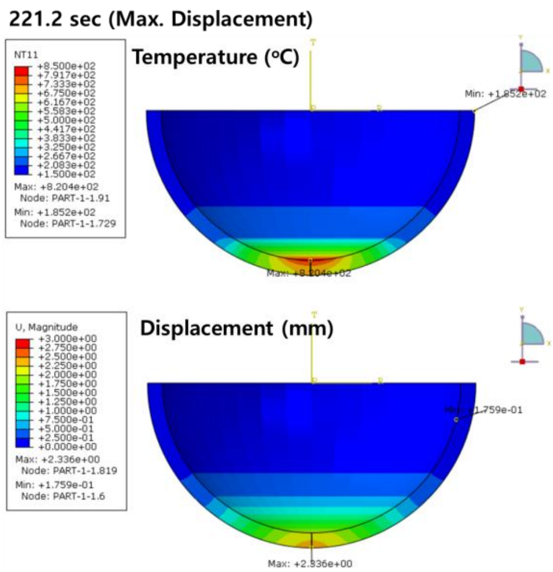


Fig. 3. Distributions of temperature, displacement, Mises Stress, and principal stress at 236 sec



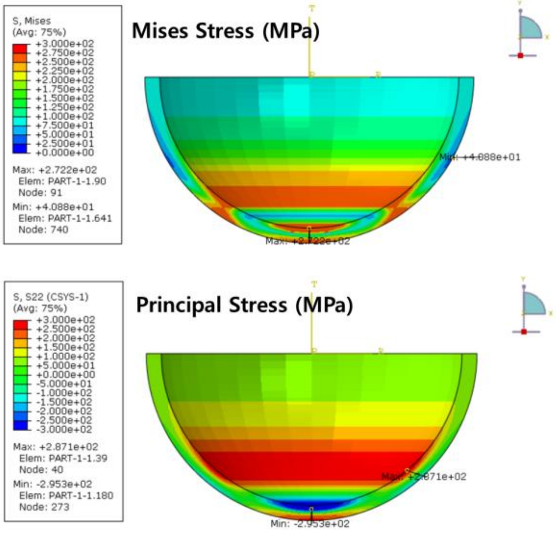


Fig. 4. Distributions of temperature, displacement, Mises Stress, and principal stress at 221.2 sec

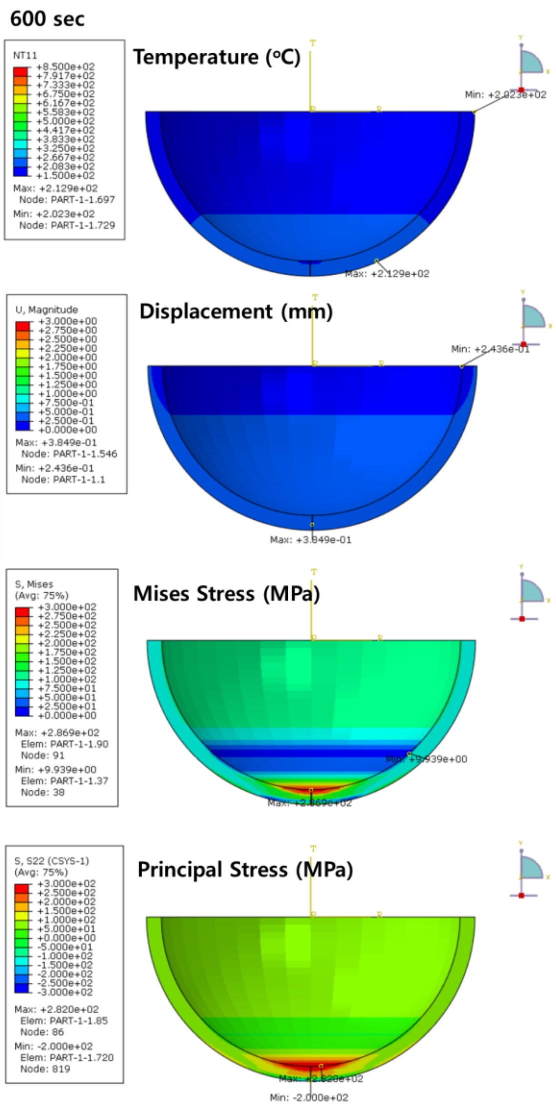


Fig. 5. Distributions of temperature, displacement, Mises Stress, and principal stress at 600 sec

In the LAVA-4 experiment, displacement was measured at the bottom of the reactor vessel lower head (θ = 0°). Figure 6 presents a comparison between the measured displacement history and the displacement history calculated by the FE model. The comparison shows that the FE model underestimates displacement by approximately 1 mm compared to the experiment. Since the temperature history was well validated, as shown in Fig. 7, the reason of the underestimation is not improper application of the thermal boundary conditions. Therefore, further studies incorporating thermal creep effects and refined FE modeling are needed to improve the agreement between the FE model and experimental displacement data.

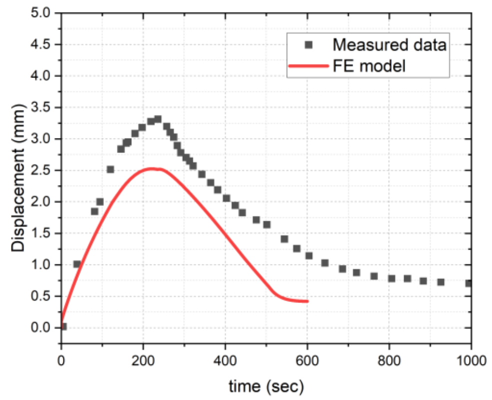


Fig. 6. Displacement histories of measured data and FE model

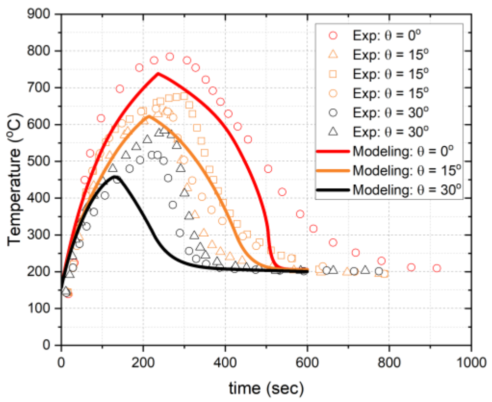


Fig. 7. Temperature histories of measured data and FE model

4. Conclusions

This study analyzed the thermal-structural behavior of a reactor vessel lower head under in-vessel gap cooling conditions using an FE model validated against LAVA-4 experimental data. The temperature history showed good agreement with the reference data, confirming the proper implementation of thermal boundary conditions. However, the FE model underestimated displacement by approximately 1 mm, likely due to the absence of thermal creep effects. The maximum Mises stress occurred at 600 seconds, reaching 286.9 MPa, indicating significant stress redistribution during cooling. To enhance accuracy, future research should incorporate thermal creep effects and refine FE modeling to improve displacement predictions and ensure a more precise assessment of reactor vessel behavior under severe accident conditions.

REFERENCES

[1] J. R. Wolf, J. L. Rempe, L. A. Stickler, G. E. Korth, D. R. Diercks, L. A. Neimark, D. W. Akers, B. K. Schumetz, T. L. Shearer, S. A. Chavez, G. L. Thinnes, R. J. Witt, M. I. Corradini, J. A. Kos, TMI-2 vessel investigation project integration report, NUREG/CR-6197, 1994.
[2] S. B. Kim et al., Experimental Study on In-veseel Debris Coolability during Severe Accident; Experimental Study on In-vessel Debris Coolability during Severe Accident, KAERI/RR-2229/2001, 2002.
[3] M. W. Song et al., Modeling of annular gap thickness formed by interaction between corium and water in lower head of reactor vessel, Nuclear Engineering and Design, 369, 110841, 2020
[4] M. W. Song, H. C. NO, Classification and modeling of flooding in vertical narrow rectangular and annular channels according to channel-end geometries, Nuclear Engineering and Design 361, 110539, 2020
[5] M. W. Song et al, Modeling of in-vessel gap cooling and its validation against LAVA, ALPHA, and LMP200 Experiments, Nuclear Engineering and Design, 374, 111028, 2021