Analysis of the ACRR-ST-1 Experiment with the Integrated Severe Accident Analysis Code CINEMA

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1. Introduction

The maintenance of nuclear safety in power plants has always been a significant concern due to the possibility of severe accidents. As a result, several severe accident analysis codes, such as MELCOR and MAAP, were developed. These codes have been used in the Korean nuclear industry to simulate severe accidents. However, to avoid potential obstacles to future Korean nuclear reactor exports, the CINEMA (Code for Integrated Severe Accident Evaluation and Management) was developed to can simulate the entire severe accident process, including in-vessel and ex-vessel models and the behavior of fission products. The CINEMA code integrates several modules, including CSPACE for in-vessel phenomena, SACAP for ex-vessel phenomena, and SIRIUS for fission product behavior. These modules are coupled using a master program to analyze the entire severe accident process. The CSPACE module was developed by coupling the thermal hydraulic analysis code SPACE with the core meltdown analysis code COMPASS.

Due to the complexity of severe accidents, it is crucial to validate the severe accident analysis codes carefully. In this study, the CINEMA code was validated using the ACRR-ST-1 experiment. The ACRR-ST-1 experiment involved overheated fuel cooling with non-condensable gas, degradation of irradiated fuel early phase core melt progression, fission product generation within the degraded fuel. The ACRR-ST-1 experiment was modeled using the CINEMA code, and the results were compared the experimental results.

2. Modelling of ACRR-ST-1 Experiment

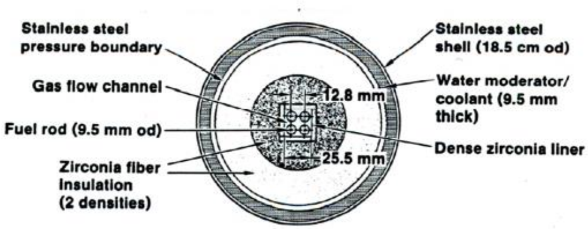

Fig. 1. Cross-sectional views of the insulated, fueled test section

The ACRR-ST experiments were designed to track the fission product release during the early phase melt progression. The experiments were conducted in the Annular Core Research Reactor which is consist of test section of 4 fuel rods with total length of 30 cm and the outer diameter of 9.5 mm. The lower part of the test section consists of fresh fuel with 15.24 cm long and the upper part of the test section consists of irradiated fuel with 15.24 cm long. The irradiated fuel pins were extracted from the BR-3 fuel rods, which had a maximum burnup of 47,000 MWd/mtU and a peak rod average power of 275 W/cm. The fuel rods were surrounded by the zircaloy cladding which fills the gap with 0.1 MPa of helium.

The pressure of the test section is 0.16 MPa and cooled by argon gas. Argon gas was injected to the test section with linear gas velocity of 98 cm/s.

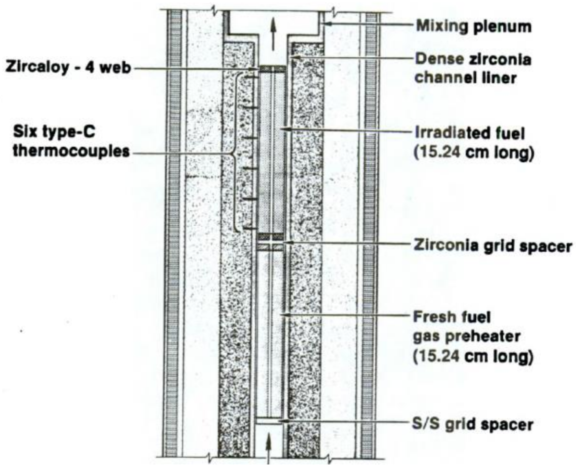

Fig. 2. Cross-sectional views of the insulated, fueled test section

In this study, ACRR-ST-1 experiment was modelled by the CINEMA. For the ACRR-ST-1 experiment, CSPACE was used to analyze the behavior within the reactor core of early transients during the severe accident progression. The test section was modeled with 18 SAM nodes which is the thermal-hydraulic structure to manage coupling between SPACE and COMPASS. Component 101 and 108 is for the argon injection and release boundary condition respectively. Since the ACRR-ST-1 test section has only 4 rods, only one ring was used during the COMPASS calculation. The shroud was divided into 17 nodes, dividing every material interface and 7 for inner and outer zirconia fiber insulation. The material properties of the shroud was averaged among all shroud materials because COMPASS does not allow individual material properties within the shroud.

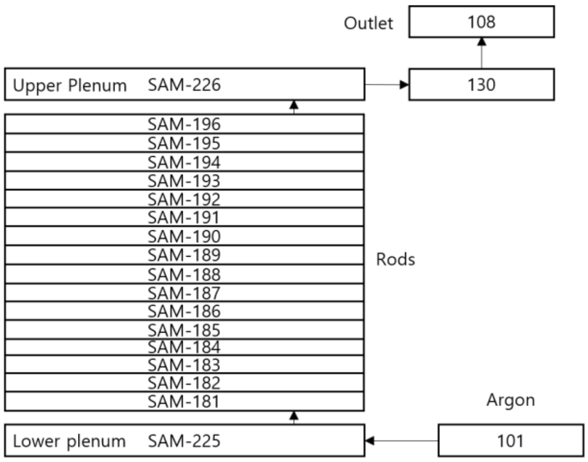

Fig. 3. CINEMA model for the ACRR-ST-1 experiment

3. CINEMA Results and Discussion

Fuel rod temperature at the top, middle, bottom of the irradiated fuel region was evaluated.

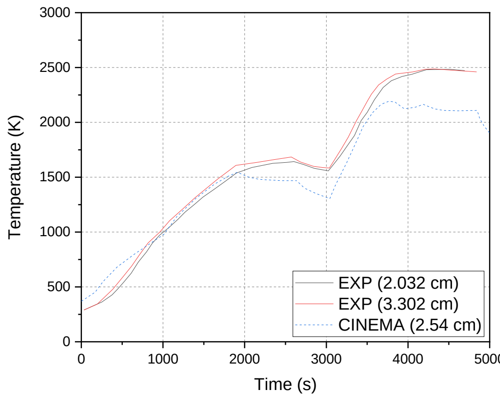

Fig. 4. Temperatures in the bottom of the irradiated fuel

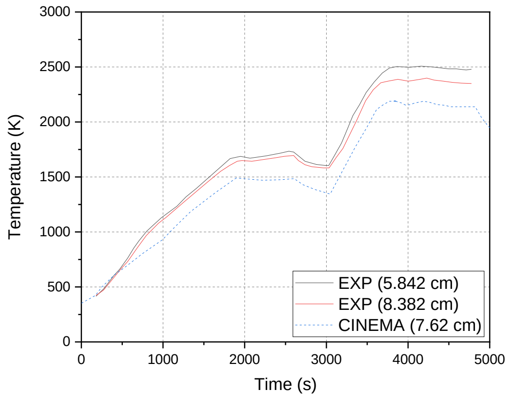

Fig. 5. Temperatures in the middle of the irradiated fuel

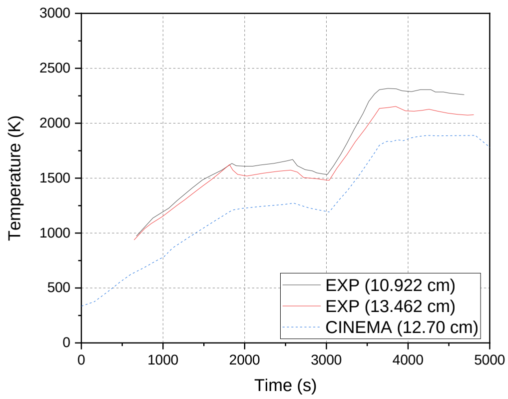

Fig. 6. Temperatures in the top of the irradiated fuel

CINEMA underpredicted the rod temperature overall. Especially, the temperature at the top of the test section was severely underestimated. The following causes were involved in the error.

1. Underestimation of the axial heat transfer through the test section.
2. Limitation of the COMPASS modelling over the ACRR-ST-1 experiment. COMPASS was designed to model the full-size core which divide the core into several rings through the radial direction. However, ACRR-ST-1 experiment only had 4 fuel rods which makes impossible to divide the ring. The modules to calculated radiative and convective heat transfer between each ring and coolant are useless for the case. Also, ACRR-ST-1 experiment does not have any control rod while COMPASS demand the condition of control rods.

4. Conclusions and Further Work

CINEMA code is validated by the ACRR-ST-1 experiment to verify calculation capability over early phase melt progression of the reactor core. The calculation results underpredicted the rod temperature overall but the results can be modified with accurate modelling of the shroud and axial heat loss. This can be easily achieved with addition of the radial mesh separating function into the COMPASS. Although the current results underestimated the bundle temperatures in the upper part of the test section, the tendency of the temperature is well predicted. For the further work, the fission product and aerosol mass tracking will be conducted by SIRIUS module to validate the prediction performance of CINEMA code over the fission product release.

5. Acknowledgments

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