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Source terms analysis in the PHEBUS FPT-1 experiment using MELCOR 2.2 code

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1. Introduction

Integrated severe accident analysis codes are used to predict the behavior of radioactive source terms inside containment building in severe accidents. The integrated severe accident analysis codes such as MELCOR, MAAP, ASTEC include uncertainties in predicting the release and behavior of source terms. To improve the accuracy and reliability of the analysis of source term behavior, it is necessary to assess and improve the uncertainties caused by severe accident analysis code. In this study, we conducted a benchmark study of the PHEBUS FPT-1 experiment, a representative experiment focusing on the behavior of fission product, using MELCOR. We compared calculated variables representing core degradation, such as the temperature of fuel and cladding, hydrogen generation, and the release fraction of fission products to the containment with experimental data. Based on the results, the capability of the MELCOR code in predicting the release of source terms was evaluated.

2. Methodology

2.1. PHEBUS FPT-1 experiment

The main objective of the PHEBUS experiment is to study the release, transport and retention of fission products in an in-pile facility under conditions representative of a severe accident in a Light Water Reactor (LWR) [1]. The PHEBUS facility is scaled down by a factor 5000 relative to a 900 MW PWR [2]. Figure 1 shows the schematic of PHEBUS facility. In the FPT-1 experiment, 20 fuel rods contain UO_2 with an enrichment of 3.5%. Among the fuel rods, 18 fuel rods were irradiated with a mean burn-up of approximately 23.4 GWd/tU and two were fresh fuel rods. One silver-indium-cadmium (AIC) control rod was placed at the center of the fuel assembly. The FPT-1 experiment was conducted under steam-rich conditions, with steam flow rate varying over time within the range of 0.5 to 2.2 g/s [2]. Figure 2 shows the steam flow rate over time.

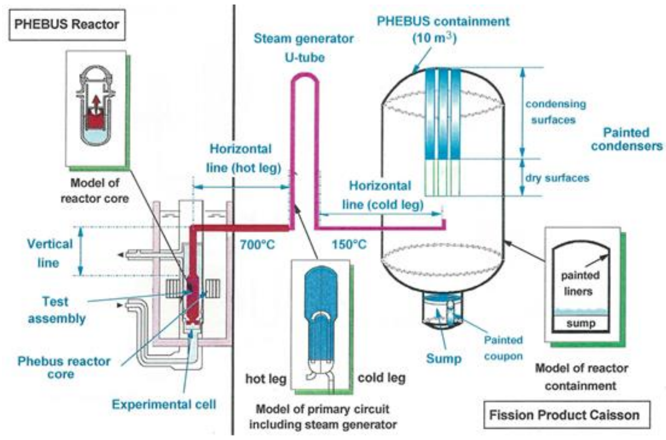

Figure 1. Schematic of PHEBUS facility [2]

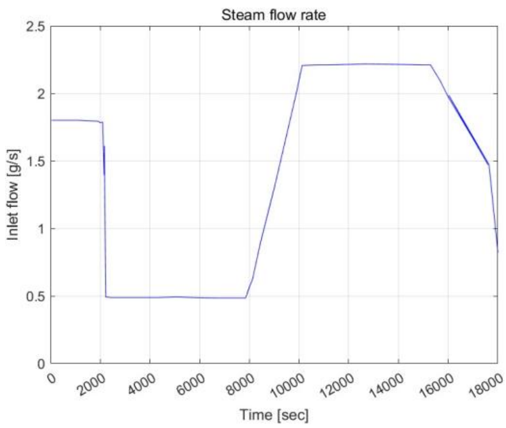

Figure 2. Steam flow rate of PHEBUS FPT-1

2.2. Input modeling

The benchmarking conducted in this study utilized the MELCOR code version 2.2 for the PHEBUS FPT-1 experiment. The input model was prepared based on the input deck of FPT-3 test developed by Bae et al [3]. Figure 3 shows the nodalization of the PHEBUS facility. The experimental facility consisted of a core, two horizontal tubes simulating the hot leg and cold leg, a reverse U-shaped tube simulating the steam generator, and a containment vessel. Figure 4 shows the nodalization of the core. The core was modeled via 2 radial rings with 27 axial nodes. The active core including fuel ranges from axial nodes 5 to 26.

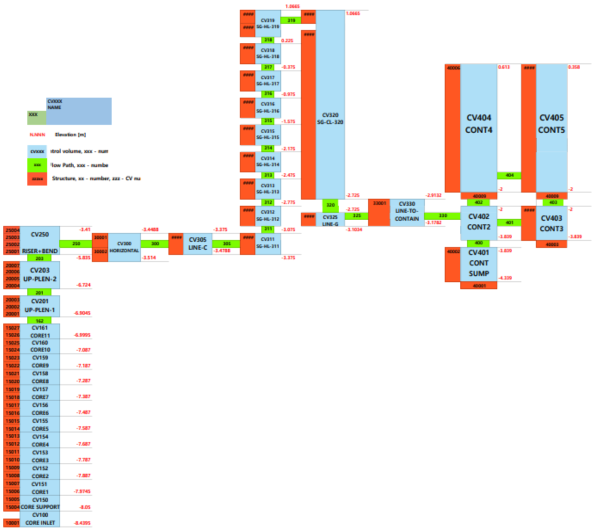

Figure 3. Nodalization of PHEBUS facility

active core

Level	Ring1	Ring2	Elevation	
			-6.9045	
27			-6.987	CORE11
26			-6.9995	
25			-7.037	CORE10
24			-7.087	
23			-7.137	CORE9
22			-7.187	
21			-7.237	CORE8
20			-7.287	
19			-7.337	CORE7
18			-7.387	
17			-7.437	CORE6
16			-7.487	
15			-7.537	CORE5
14			-7.587	
13			-7.637	CORE4
12			-7.687	
11			-7.737	CORE3
10			-7.787	
9			-7.837	CORE2
8			-7.887	
7			-7.937	CORE1
6			-7.9745	
5			-7.987	CORE
4			-8.0405	Support
3			-8.05	CORE
2			-8.1315	INLET
1			-8.4395	

Figure 4. Nodalization of core

3. Result and Discussion

The MELCOR results were compared with experimental data [2, 4]. First of all, we conducted an analysis of the fuel and cladding temperature, and hydrogen generation related to core degradation. Figure 5 and 6 show the temperature of fuel at 300 mm and 400 mm from the bottom of the active core. The fuel temperature at 300 mm and 400 mm increased similar to the experimental data, but the temperature peak appearing at 11,000 seconds due to heat generation by the metal-water reactor was not observed in the calculation result at 400 mm. Figure 7 and 8 show the cladding temperature at 600 mm and 700 mm. The MELCOR results were also close to the experimental data, and predicted the temperature peaks well around 11,000 seconds. After 11,000 seconds, there are significant differences between experimental data and the MELCOR results for the fuel and cladding temperature. It was caused by thermocouple integrity issues in high-temperature conditions. Similarly, as shown in Figure 9, the cumulative mass of hydrogen generated from the oxidation reaction was observed close to the experimental data; after the core degradation phase was finished, the deviation from the measured value was less than 20%.

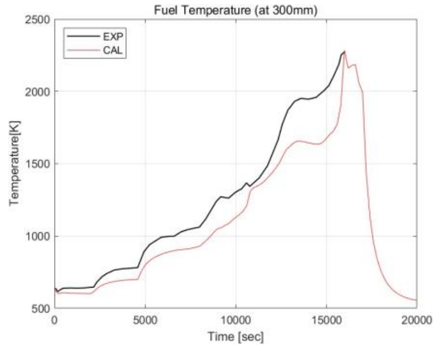

Figure 5. Fuel temperature at 300 mm

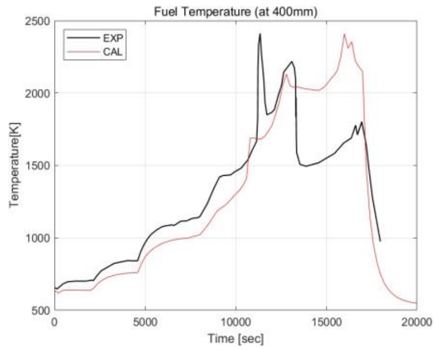

Figure 6. Fuel temperature at 400 mm

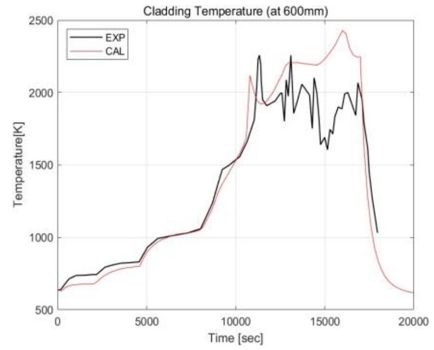

Figure 7. Cladding temperature at 600 mm

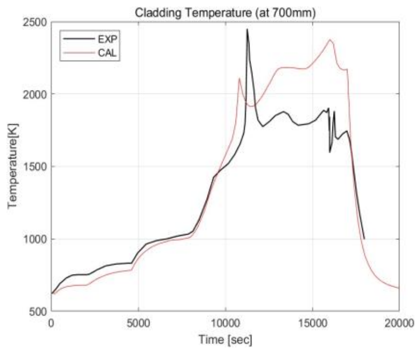

Figure 8. Cladding temperature at 700mm

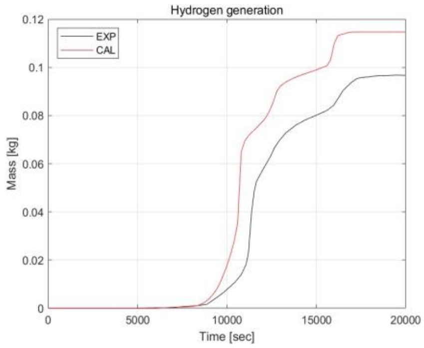

Figure 9. Hydrogen generation

We also analyzed the release fraction of fission product to the containment vessel. The analysis included the release fraction of xenon, tellurium, cesium, and iodine. Xenon and tellurium are representative of noble gases and Chalcogens, respectively. Cesium and iodine are fission products that have significant impacts on the environment and public health. Figure 10,11,12 and 13 show the release fraction of Xe, Te, Cs, and I to the containment, respectively. The experimental data was calculated using the mass ratio of fission products released to the containment relative to the initial inventory. As shown in Figure 10, 11, and 12, the deviation from the measured release fraction of xenon, tellurium and cesium was turned out to be less than 5%. Figure 13 shows the release fraction of iodine released in the form of I_2 and CsI. It shows a difference of about 20% with experimental data.

The calculation results revealed that the MELCOR code could predict the release of most source terms with the reasonable accuracy, provided the cladding temperature of fuels was reproduced well by the developed input model. Due to complicated iodine chemistry in the circuit and the containment, however, further analysis is needed to better predict the release and transport behavior of iodine.

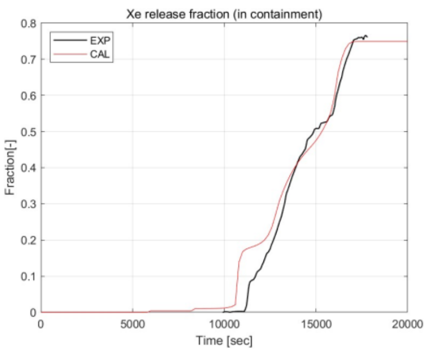

Figure 10. Xenon release fraction to containment

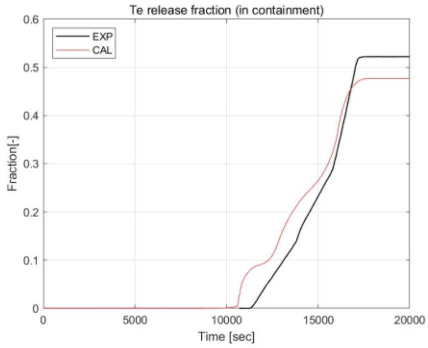

Figure 11. Tellurium release fraction to containment

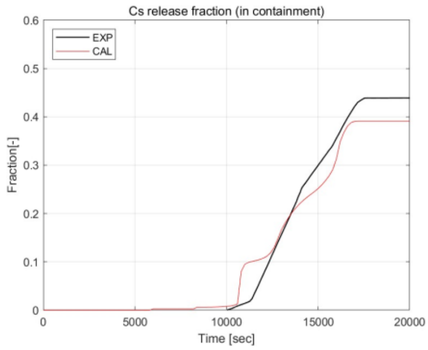

Figure 12. Cesium release fraction to containment

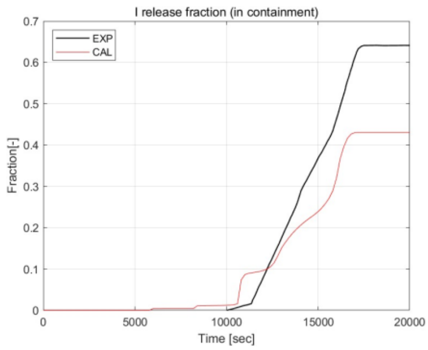

Figure 13. Iodine release fraction to containment

Furthermore, we conducted a sensitivity analysis of the core release model for the release fraction of cesium. The sensitivity analysis was conducted for the three models, such as the CORSOR-BOOTH model, the 1st revised CORSOR-BOOTH model, and the 2nd revised CORSOR-BOOTH model. Figure 14 shows the results of the sensitivity analysis of the core release model. It was found that the difference between the predicted release fractions by three models was less than 5%. Among the three models, the CORSOR-BOOTH model, which calculates total mass release rate as a combination of diffusion and gas-phase mass transport (evaporation) rates, predicted the experimental data most closely.

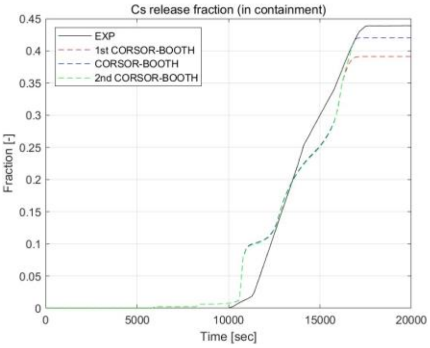

Figure 14. The release fraction of Cesium for the core release model

4. Conclusions

In this study, benchmarking of the PHEBUS FPT-1 experiment was performed using the MELCOR severe accident analysis code. Major variables related to source terms, such as the temperature of fuel and cladding, hydrogen generation, and the release fraction of fission products to containment, were compared with experimental data. The MELCOR predicted the temperature of fuel and cladding, hydrogen generation close to the experimental data. Furthermore, the MELCOR predicted the release fraction of fission products to containment, except for Iodine, with differences lower than 5% compared to experimental data. The effect of the choice of the core release model was not significant, but the prediction by the CORSOR-BOOTH model was closest to the test data of the release fraction of Cs. Uncertainty and sensitivity analyses of the MELCOR results on the source terms will be conducted in the future work.

Acknowledgements

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