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Analysis of the CASP2 Experiment in the Battelle Containment Model using the Code COCOSYS

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1. Introduction

Postulated accidents in nuclear reactors are considered admissible events for the purpose of analysis, allowing the establishment of safety conditions capable of preventing the accident or mitigate its potential consequences [1].

The comprehension of thermo-hydraulic phenomena involved in a postulated accident was developed using a large number of experiments as support, reproducing effects on local or global scales. Additionally, the experimental basis was crucial for the evolution of computational codes in the nuclear field, enabling the assessment of the response of reactor systems and components during accident scenarios, as well as the development of additional measures to prevent and mitigate the consequences of a possible release of radioactive material [2].

Among the variety of experiments carried out to support the safety analysis under accident scenarios, this study uses the data of the experiment Containment Standard Problem No. 2 (CASP2), conducted at the Battelle-Institut Containment model in Frankfurt, Germany [3]. To reproduce the results of this experiment, the boundary conditions were simulated using a thermos-hydraulic code [4] and the containment analysis employed the code COCOSYS v2.4 [5]. To verify the level of accuracy obtained between experimental data and simulated values, the Fast Fourier Transform Based Method (FFTBM) was also applied [6].

2. Methodology

The Battelle Containment Model (BCM) is a testing facility located in Germany, where experiments were carried out to investigate various phenomena that can occur inside the containment of a nuclear reactor (Figures 1 and 2). The experiments performed at this facility have provided a foundation for numerous user qualification studies, as well as for the development of numerical models for various codes.

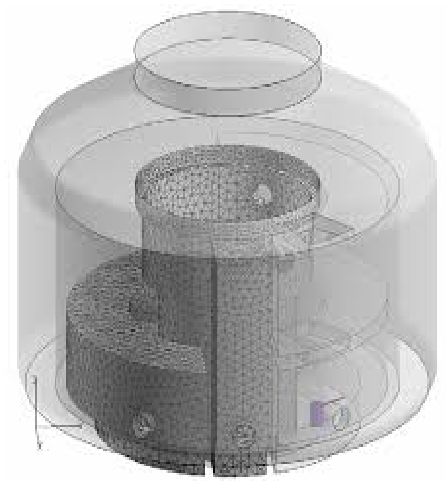

Figure 1: BCM facility[6]

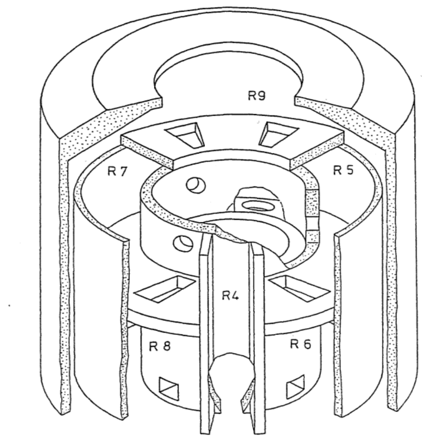

Figure 2: BCM Subcompartments [7]

The CASP2 experiment involved coupling the containment of the BCM to a pressure vessel filled with subcooled water at pressure of 141 bar and temperature of 289 °C, simulating operational conditions of Pressurized Water Reactors (PWR). The actuation of valves in the piping, which connects the containment to the pressure vessel, resulted in a depressurization in this experiment similar to the effects generated by the occurrence of postulated accident of Large Break Loss of Coolant (LBLOCA) in the containment system (Figure 3) [3].

The BCM containment is primarily composed of concrete and steel, with a total free volume of 641.90 m³, subdivided into 6 compartments: R4, R5, R6, R7, R8, and R9, which are interconnected by junctions that can be open or closed according to the experiment to be performed [3].

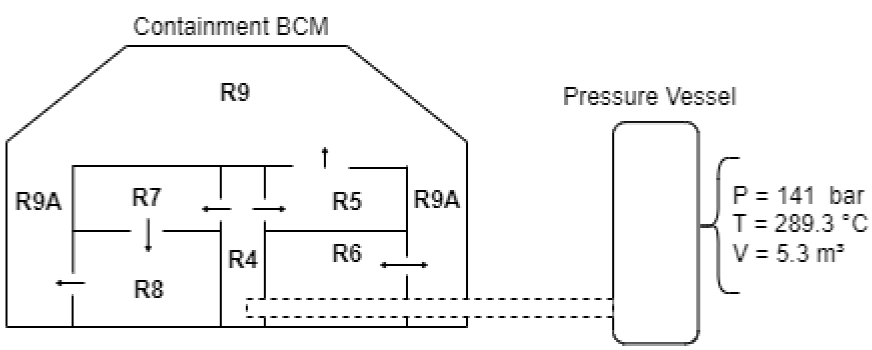

Figure 3: CASP2 Experiment [3]

The methodology employed to computationally reproduce the CASP2 experiment was based on the coupling a thermo-hydraulic code and the COCOSYS, applying the initial and boundary conditions of the experiment. Figure 4 represents the developed and coupled nodalization for simulating CASP2.

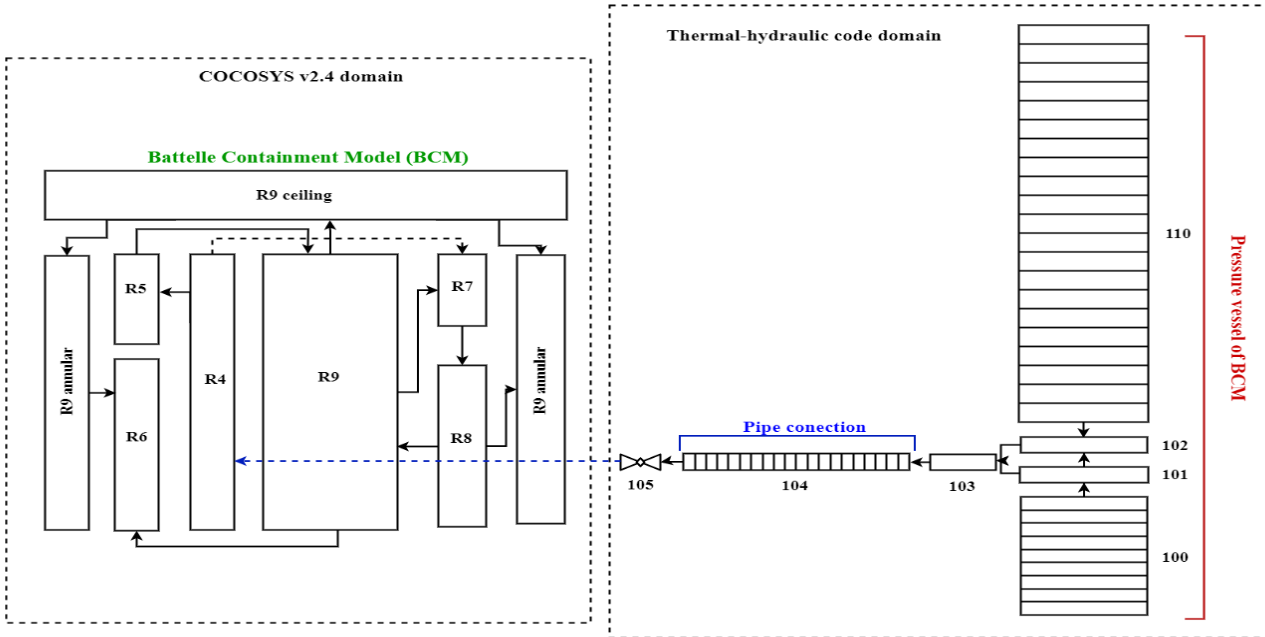

Figure 4: coupled nodalization model

3. Results and Discussion

The model developed for the pressure vessel reproduced the depressurization effect, providing the appropriate boundary condition for the COCOSYS simulation to assess the progression of the CASP2 experiment within the BCM containment environment. Furthermore, it is noteworthy that the obtained simulated values are within the experimental margin of error of 10% (Figure 5). The obtained results related to the containment region, simulated with COCOSYS, were divided into three time intervals. The first interval, from 0 to 2.5 seconds (Figure 6), aimed to demonstrate the effect generated by the rapid pressurization of compartment R4 and the subsequent pressure distribution. From 0 to 50 seconds (Figure 7), the purpose was to identify the peaks of pressure and temperature reached. Finally, the interval from 0 to 1000 seconds (Figure 8) provided a general overview of the pressure evolution behavior in the BCM containment.

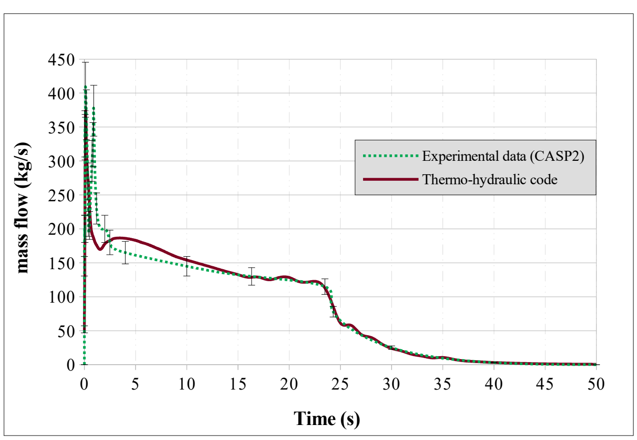

Figure 5: flow mass from pressure vessel

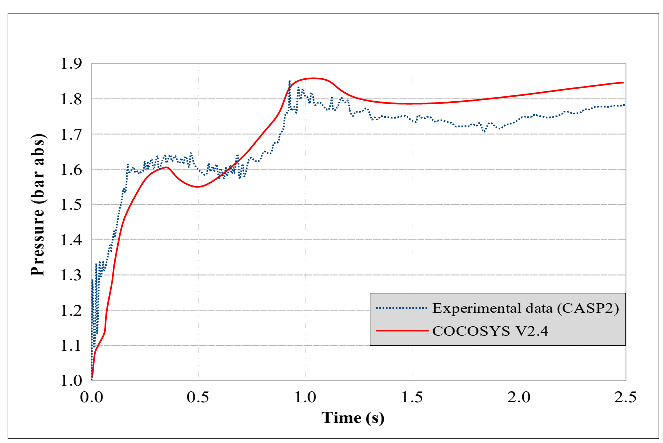

Figure 6: Pressure curve in R4 compartment

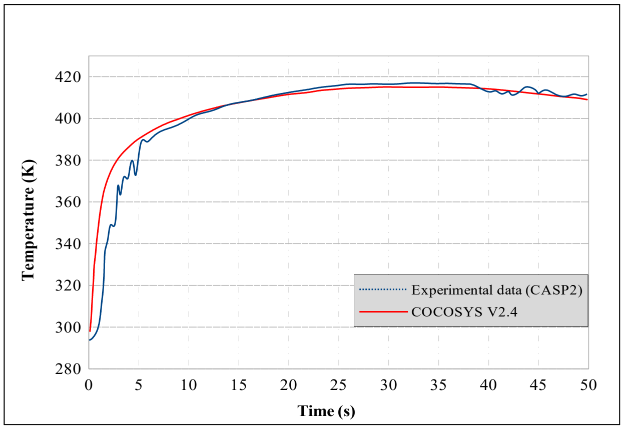

Figure 7: Temperature in R8 compartment

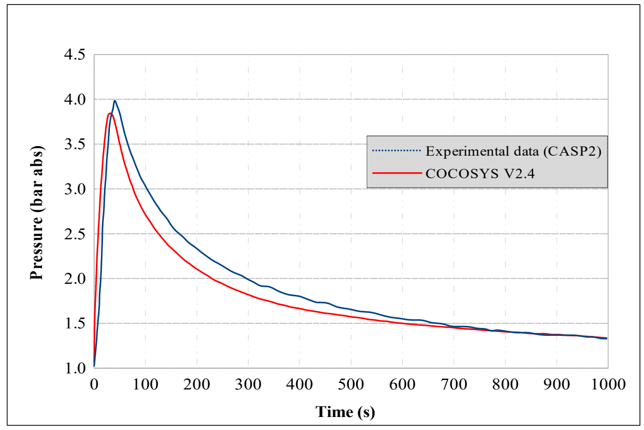

Figure 8: Global pressure

The Fast Fourier Transform Based Method (FFTBM) was applied to the obtained results for accuracy quantification. The FFTBM shows the measurement–prediction discrepancies in the frequency domain and quantifies the discrepancy magnitude. The FFTBM considers the maximum value equal to 0.4 of total amplitude for an acceptable adherence of the obtained results [6]. Above this value, the nodalization model must be refined to better interpret the phenomenology of the considered experiment.

Table 1: FFTBM results

Time interval (s)	0 a 2.5	0 a 50	0 a 1000
AAtot - Total accuracy	0.189	0.267	0.214

4. Conclusions

The results for accuracy quantification obtained applying the FFTBM show that the simulation model based in coupled nodalization with thermo-hydraulic and COCOSYS codes for CASP2 experiment reproduced with a good agreement the experimental data, as can be verified in Table 1 with all acceptability factors lower than 0.4. The results show that the simulation of this kind of experiment is an important tool to be applied for user qualification, enabling to improve the confidence degree of the results obtained in safety analysis particularly for new nuclear reactors designs.

References

[1] Comissão Nacional de Energia Nuclear (CNEN), Glossário de Segurança Nuclear, (2015).

[2] International Atomic Energy Agency (IAEA). Status and Evaluation of Severe Accident Simulation codes for Water-Cooled Reactors. Vienna, Austria. IAEA-TECDOC-1872 (2019).

[3] Nguyen, D.L. and Winkler, W. CSN report no. 65. Comparison Report On OECD-CSNI Containment Standard Problem N° 2 "Water Line Rupture in a Branched Compartment Chain II, Munich, Germany (1982).

[4] General Atomics. SCDAP/RELAP5/MOD 3.2 Code Manual. User's Guide and Input Manual. (1998).

[5] Gesellshaft Für Anlagen-Und Reaktorsicherheit (GRS) mbh. COCOSYS V2.4 User's Manual. Revision 1. (2016).

[6] International Atomic Energy Agency (IAEA). Safety Report Series N°. 52 – Best Estimate safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, Vienna, Austria (2008).

[7] Matthias Heitsch, Daniele Baraldi, Heinz Wilkening, Validation of CFD for Containment Jet Flows including Condensation. OECD/NEA and IAEA, p. 1-14. JRC45594 (2008).