COMPUTATION OF BWR TURBINE TRIP WITH CRONOS2 AND FLICA4

E. Royer
Commissariat à l'Energie Atomique
Direction de l'Energie Nucléaire
DM2S/SFME, Centre d'Etudes de Saclay
91 191 Gif sur Yvette Cedex, France
eric.royer@cea.fr

N. Todorova
Penn State University
Department of Mechanical and Nuclear Engineering
University Park, PA 16802, USA
nadejda@psu.edu

Abstract

This paper presents the computation of Exercise 2 of the OECD Boiling Water Reactor Turbine Trip benchmark with the CRONOS2 and FLICA4 codes. Exercise 2 is dedicated to the core analysis, using coupled thermal-hydraulic-neutronic codes and provided boundary conditions for the transient. Accurate thermalhydraulic model is required to analyze the Turbine Trip transient, because the reactivity insertion is mainly driven by the void fraction in the core.

The first section of the paper presents the studies carried out on the thermalhydraulic model, in order to assess the sensitivity of the transient power response. The second section is devoted to the optimization of the computation, by grouping similar assemblies into a single thermalhydraulic channel. A suitable model for the 764 assemblies of the BWR core is achieved when approximately one hundred channels are used.

1 INTRODUCTION

The OECD Boiling Water Reactor (BWR) Turbine Trip (TT) benchmark [1] was defined in order to analyze the coupled phenomena between core and system, due to the pressurization dynamics. Actual data from the Peach Bottom reactor are available for the benchmark, including experimental measurements during the transient.

CEA participation in the benchmark is aimed at validating its codes against a BWR transient. Particularly CRONOS2 [2] and FLICA4 [3] are used to compute Exercise 2, defined as a core boundary condition problem with coupled 3D kinetics and core thermalhydraulics. Exercise 1 was also computed with CATHARE [4], and Exercise 3 will combine system thermalhydraulics with core neutronics and thermalhydraulics.

2 THERMALHYDRAULIC MODEL

FLICA4 is a 3D thermalhydraulic code used for many reactor types and has a wide range of applications. The physical model is one-fluid with thermodynamic and velocity disequilibrium between liquid and vapor. Many closure laws are available, and a sensitivity study was carried out to assess the effect on the power response during the transient.

2.1 FLICA4 code

FLICA4 is the thermalhydraulic code of the SAPHYR system, which also includes CRONOS2 for 3D neutronic core calculations and APOLLO2 for 2D neutronic assembly calculations.

SAPHYR codes are based on a modular structure that allows a great flexibility of use. A special user oriented language, named GIBIANE, and a shared numerical toolbox have been developed to chain the various computation modules.

FLICA4 is a 3D two-phase compressible flow code, specially devoted to reactor core analysis. The fluid is modeled by a set of four equations : mass, momentum, and energy conservation for the two-phase mixture, and mass conservation for the vapor. The velocity disequilibrium is taken into account by a drift flux correlation. A 1D thermal module is used to solve the conduction in solids (fuel).

A complete set of closure laws is qualified for PWR, but due to its large flexibility, FLICA4 provides numerous correlations for wall friction, drift flux, heat transfer and critical heat flux, and many fluids can be calculated (liquids like lead or freons, gas like hydrogen or carbon dioxide).

FLICA4 includes an object-oriented pre-processor to define the geometry and the boundary conditions. Radial unstructured mesh is available, without any limitation on the number of cells. Zooming on a specific radial zone can be performed by a second calculation using a finer mesh (for instance a sub-channel calculation on the hot assembly).

The fully implicit numerical scheme is based on the finite volumes and the Roe solver. This kind of method is particularly accurate, with a low numerical diffusion.

2.2 Peach Bottom core model

Boundary conditions provided for Exercise 2 include core inlet pressure, temperature, mass flow, and outlet pressure. Inlet mass flow is actually defined for 33 channels, and is consistent with the core model used by the benchmark team [1]. For the CEA reference calculation, the core thermalhydraulic model is made of a 3D inlet area, in order to compute the flow distribution among the 764 assemblies, and a multi-1D area to model the closed assemblies. In addition, a separate channel represents the by-pass, and is used to correct the moderator density in the feedback process (calculation of neutronic cross-sections). Thus each assembly is modeled separately (by a single channel) and its flow rate is consistent with its operating conditions: power, pressure drop.

2.3 Sensitivity study

Boiling Water Reactors are charcterized by a significant feedback effect due to the wide range of the fluid density. The fluid density is actually calculated from the void fraction, which mainly depends on the following three models: wall heat exchange, particularly the sub-cooled boiling, interface heat exchange model, particularly the condensation rate, and the drift flux model.

During the Turbine Trip transient, the core pressure increases, which affects the fluid density, and results in a reactivity insertion. This transient requires an accurate thermalhydraulic model to compute the void fraction history. The effect of the void fraction model is studied to assess the sensitivity of the reactivity insertion or the related power peak. Actually, measured data used for comparison are the steady-state axial power profile, and the history of power and reactivity during the transient.

Three standard drift flux models are compared: homogeneous model (no velocity difference between liquid and vapor), Ishii correlation, Bestion correlation [5]. The homogeneous model predicts higher void fraction (cf. Figure 1), which affects the power distribution (cf. Figure 2). The axial-offset for the three models ranges from -0.67 % for the homogeneous model to 11.2 % for the Ishii correlation. Both Bestion and Ishii correlations reasonably agree with the measured data (labelled as *P1 edit*).

Nevertheless, transient computations always over-estimate the power peak (cf. Figure 3). The timing is also slighty delayed, and the power excursion is stopped by the scram, occurring at the specified time of 0.75 *s*. Power is actually a very sensitive parameter in the transient, compared to reactivity (cf. Figure 4), for which the maximum varies with less than 0.05 $ for the different models.

2.4 Conclusion

The reference FLICA4 model uses a single thermalhydraulic channel for each assembly, and computes the mass flow rate for each assembly depending on its own operating conditions. A sensitivity study was carried out on the drift flux model. It appears that a small modification of the steady-state conditions due to the void fraction in the core results in a large variation of the power peak during the transient. In any case, the reactivity and the power peak are always over-estimated

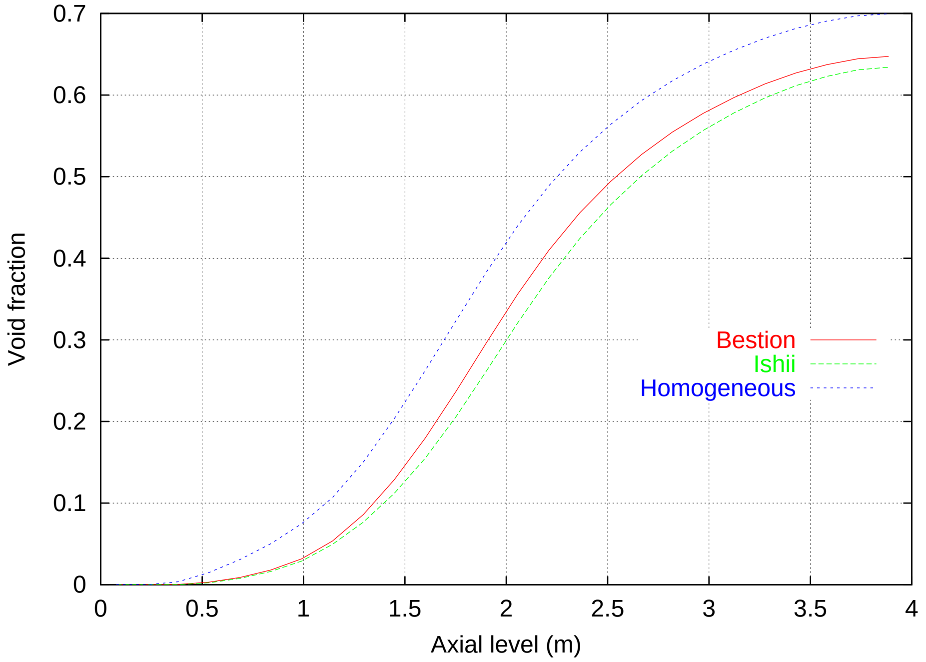

Figure 1: Initial steady-state void fraction for different drift flux models.

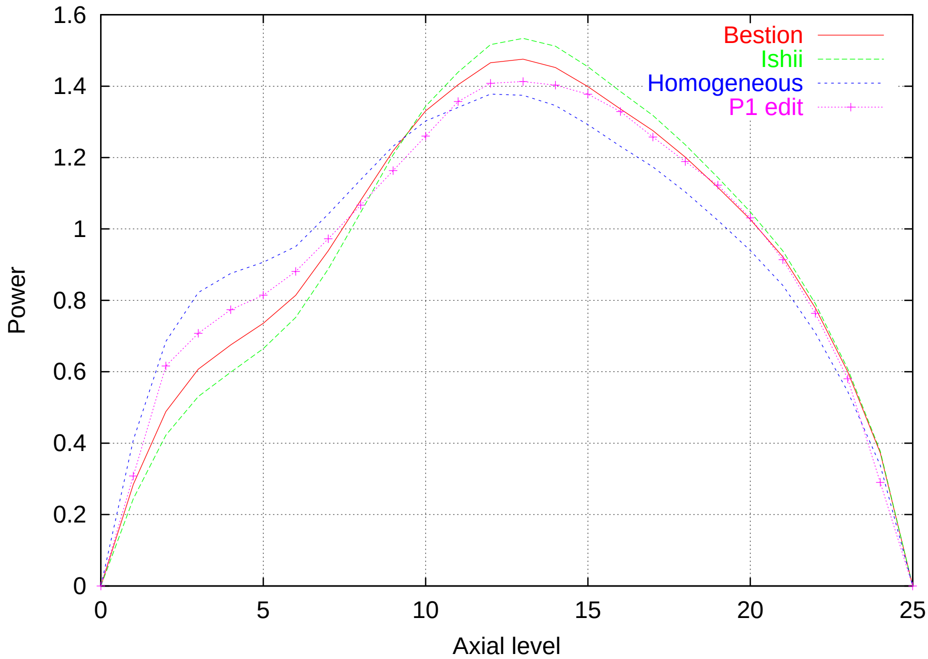

Figure 2: Initial steady-state axial power for different drift flux models.

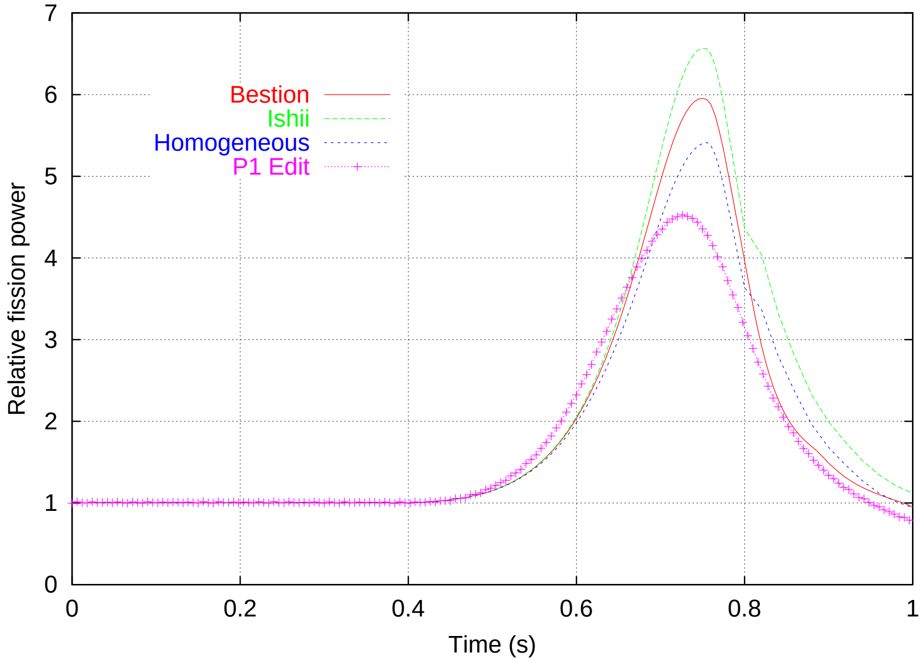

Figure 3: Power history for different drift flux models.

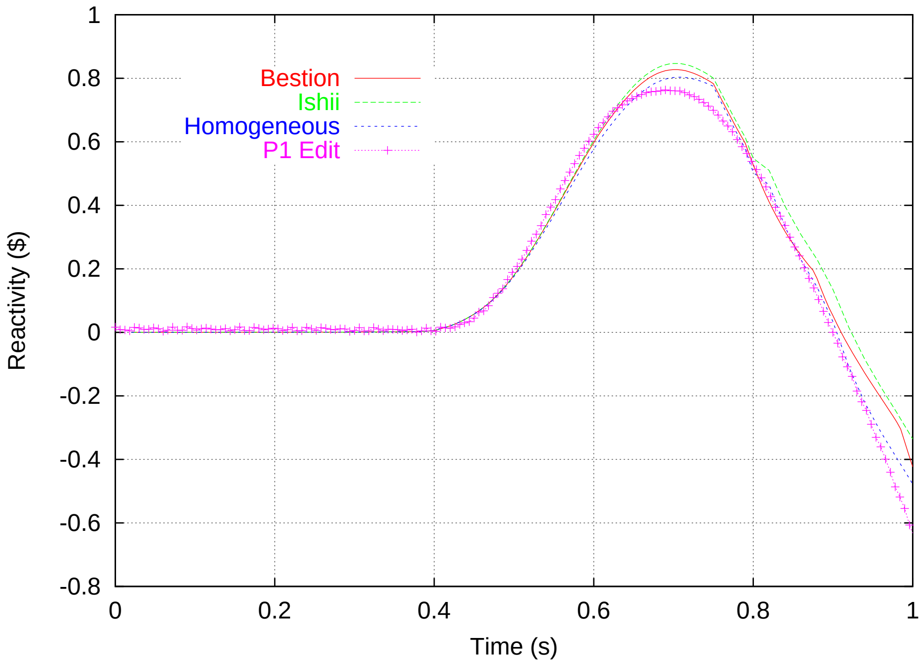

Figure 4: Reactivity history for different drift flux models.

compared to the measurements.

3 NEUTRONIC-THERMALHYDRAULIC COUPLING

From thermalhydraulic point of view, the fuel assemblies of a BWR core are considered as isolated 1D channels, and they are only related by the total mass flow rate through the core. This feature enables to reduce the computation cost by grouping similar assemblies into a same channel. The number of thermaldraulic channels was studied in order to define an optimized coupled model to compute the Peach Bottom reactor, compared to the reference model with 764 channels.

3.1 Analysis of the specified 33 channel model

The benchmark specifications [1] propose a 33 channel model for the core, and define boundary conditions for these 33 channels. Each thermalhydraulic channel is mapped to several assemblies in neutronic calculation. Actually the number of assemblies mapped to a channel varies between 4 and 76.

Comparison of the radial power distribution with the reference calculation for each assembly shows large discrepancies: the maximum relative error reaches 18.1 % (cf. Table 1). This can be explained when comparing the relative axial power of different assemblies mapped to the same thermalhydraulic channel. For instance channel 30 is mapped to 36 assemblies, but it appears that there are in fact 4 main clusters of assemblies (cf. Figure 5).

3.2 Optimized number of channels

Since the specified model with 33 channels is too coarse to keep an accurate power distribution in the core, it was decided to refine this model by splitting channels where there are too many assemblies. The normalized radial power of the assemblies mapped to the same channel was used as the relevant parameter to derive new radial mapping between neutronics and thermalhydraulics.

Four new thermalhydraulic models were defined, using 55, 87, 119 and 147 channels. Table 1 shows that the maximum error on the assembly power is significantly reduced when increasing the number of channels. 119 channels seems enough for an accurate computation of the initial steady-state of the reactor.

During the TT transient, the radial power distribution is only slightly modified (all rods are inserted at scram and boundary conditions are uniform). Then, the optimized model derived on steady-state analysis is expected to be worth for the transient. The effect of the number of channels on the integral power along the transient is actually very small (cf. Figure 6), but has to be considered on the radial power peak (cf. Figure 7). As for the steady-state, the evolution of the radial power peak exhibits different trends with the 33 channel model compared to the reference model (764 channels) and the optimized model (119 channels).

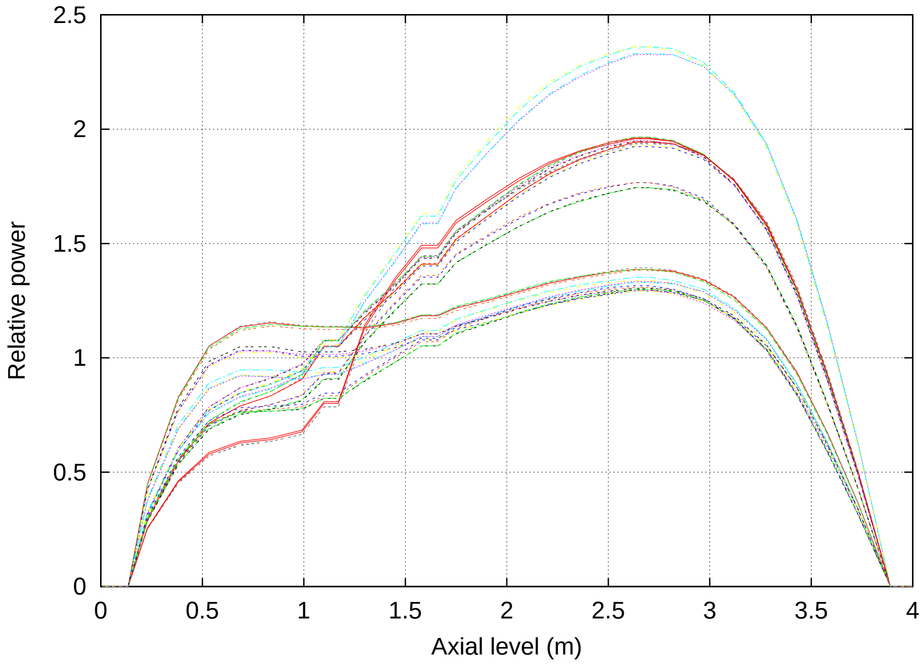

Figure 5: Relative power for assemblies mapped to channel 30.

Table 1: Error on radial power for different number of channels.

Number of channels	Average error (%)	Maximum error (%)
33	4.50	18.1
55	3.16	10.3
87	3.12	8.13
119	3.04	6.61
147	3.07	6.38

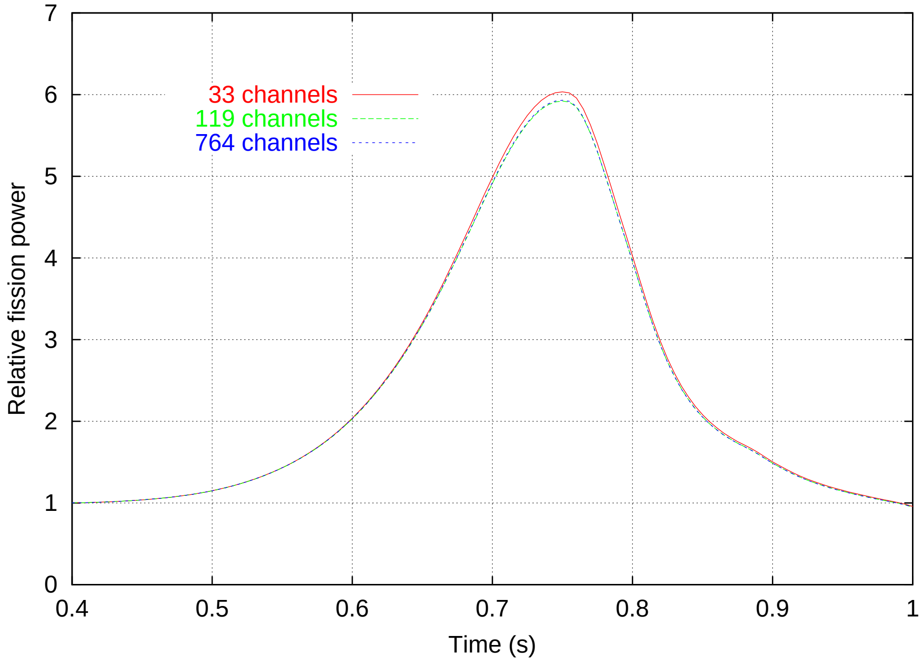

Figure 6: Power history for different number of channels.

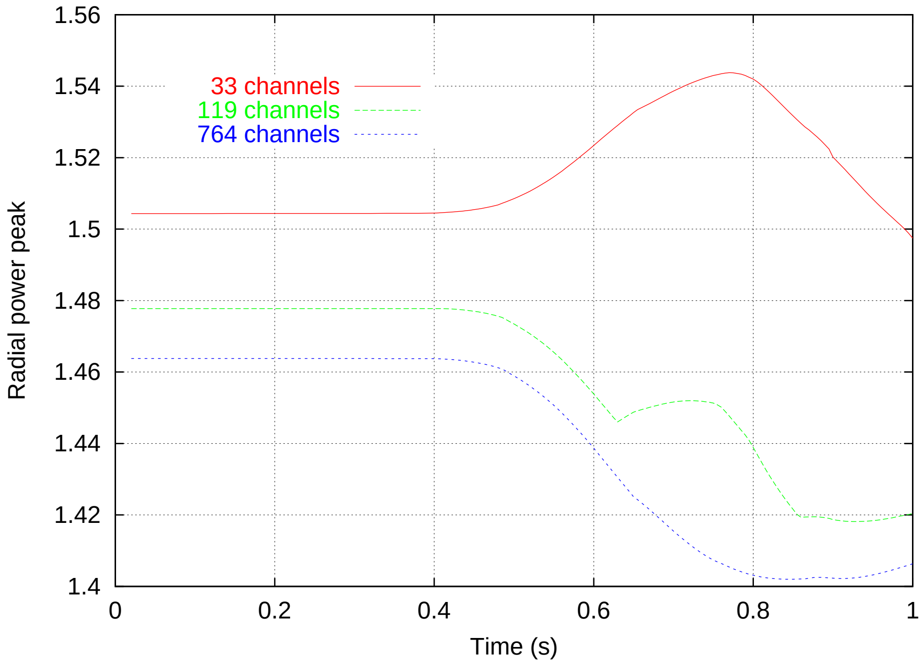

Figure 7: Radial power peak history for different number of channels.

3.3 Conclusion

In order to provide an optimized model for coupled thermalhydraulic-neutronic analysis of the Peach Bottom reactor, a study of the number of thermalhydraulic channels was performed. The 33 channel model proposed by the benchmark specifications appears to be too coarse for an accurate computation of the power distribution. After a detailed investigation of the mapping between assemblies and channels, an improved model was derived with 119 channels. This model can be used for both the steady-state and the TT transient analysis.

4 CONCLUSIONS

A suitable model was defined to model the Peach Bottom Boiling Water Reactor with the FLICA4 and CRONOS2 coupled codes. The sensitivity of the thermalhydraulic closure relationships was first studied by comparison against experimental data. The accuracy on the void fraction calculation appears to be critical for correctly predicting the initial steady state of the reactor and the reactivity insertion during the Turbine Trip transient.

In order to reduce the computation time of a BWR model, an appropriate methodology has been used to collapse assemblies into average thermalhydraulic channel without affecting significantly the results. An optimal number of channels to model the Peach Bottom reactor core is found to be around one hundred.

Experience gained in computing Exercise 2 will further be applied for Exercise 3, when coupling core (CRONOS2 and FLICA4) and system (CATHARE code).

References

[1] J. Solis, K.N. Ivanov, B. Sarikaya, A.M. Olson, K.W. Hunt, "Boiling Water Reactor Turbine Trip (TT) Benchmark, Volume I: Final Specifications", *Nuclear Energy Agency, NEA/NSC/DOC(2001)1*, June 2001

[2] J.J. Lautard, D. Schneider, A.M. Baudron, "Mixed dual methods for neutronic reactor core calculations in the CRONOS system", *proceedings of M&C*, Madrid, September 27-30 1999

[3] I. Toumi, D. Gallo, A. Bergeron, E. Royer, D. Caruge, "FLICA4 : a three dimensional two-phase flow computer code with advanced numerical methods for nuclear applications", *Nuclear Engineering and Design*, **vol 200**, pp 139-155, 2000

[4] B. Rameau, G. Mignot "BWR Turbine Trip calculations with the CATHARE code", *proceedings of Physor 2002*, Seoul, October 7-10 2002

[5] P. Coddington, R. Macian, "A study of the performance of void fraction correlations used in the context of drift-flux two-phase flow models", *Nuclear Engineering and Design*, **vol 215**, pp 199-216, 2002