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HIGH FIDELITY ANALYSIS OF BWR FUEL ASSEMBLY WITH COBRA-TF/PARCS AND TRACE CODES

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ABSTRACT

The growing importance of detailed reactor core and fuel assembly description for light water reactors (LWRs) as well as the sub-channel safety analysis requires high fidelity models and coupled neutronic/thermalhydraulic codes. Hand in hand with advances in the computer technology, the nuclear safety analysis is beginning to use a more detailed thermalhydraulics and neutronics.

Previously, a PWR core and a 16 by 16 fuel assembly models were developed to test and validate our COBRA-TF/PARCS v2.7 (CTF/PARCS) coupled code. In this work, a comparison of the modeling and simulation advantages and disadvantages of modern 10 by 10 BWR fuel assembly with CTF/PARCS and TRACE codes has been done.

The objective of the comparison is making known the main advantages of using the sub-channel codes to perform high resolution nuclear safety analysis. The sub-channel codes, like CTF, permits obtain accurate predictions, in two flow regime, of the thermalhydraulic parameters important to safety with high local resolution.

The modeled BWR fuel assembly has 91 fuel rods (81 full length and 10 partial length fuel rods) and a big square central water rod. This assembly has been modeled with high level of detail with CTF code and using the BWR modeling parameters provided by TRACE. The same neutronic PARCS's model has been used for the simulation with both codes. To compare the codes a coupled steady state has be performed.

1. INTRODUCTION.

The use of three-dimensional (3D) kinetics/thermalhydraulics coupled codes for best-estimate nuclear safety calculations is a common practice in nuclear industry and research centres. However, together with the software and computer technology advances, the evaluation at local pin level has been developed to operate nearest of the real safety margins.

This work presents a comparative study that will show the capabilities of system and subchannel coupled codes, such as TRACE and CTF/PARCS [1-4] respectively, to model and simulate the phenomenology expected in the newest design of BWR fuel assemblies. The selected fuel assembly is a modern 10 by 10 BWR type which has 91 fuel rods (81 full length and 10 partial length fuel rods) and a big square central water rod. The models developed try to represent in realistic way the design of the fuel assembly with the available capabilities of each code.

The paper is organized as follows: the section 2 presents the developed models for the both codes, TRACE and CTF. The section 3 shows the results for the steady state simulation and the discussion of them. Finally, in the section 4 the main conclusions of the work are presented.

2. DESCRIPTIONS AND MODELS.

As can be seen in the Figure 1, the analyzed fuel assembly is a 10 by 10 BWR type; it has 91 fuel rods (81 full length rods (FLRs) and 10 2/3 partial length fuel rods (PLRs)) and a big square central water rod. The Table 1 lists the main geometrical data used to develop the models.

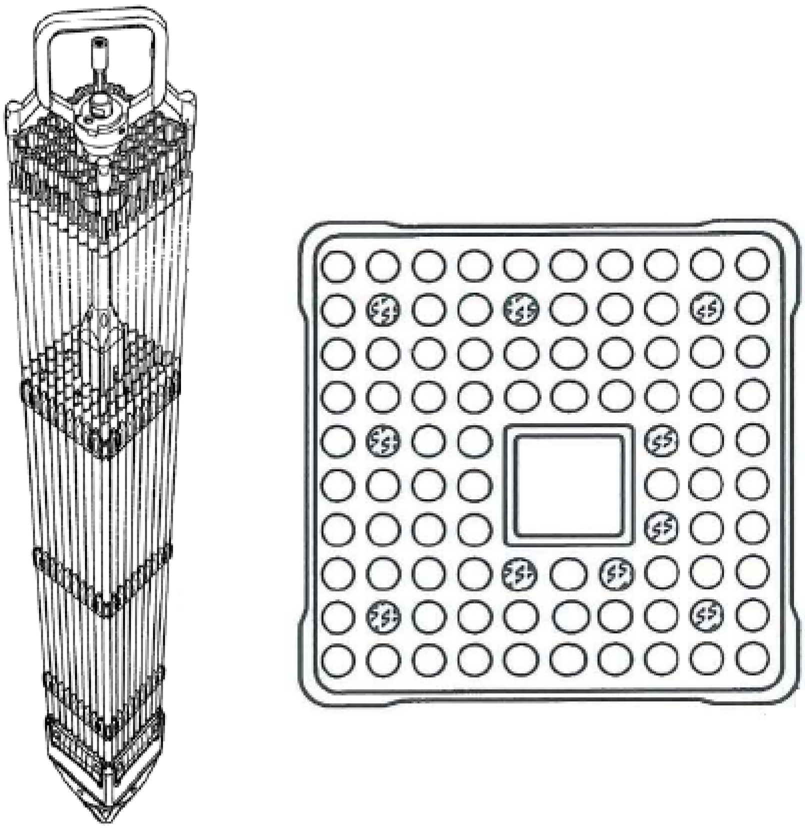

Figure 1: Scheme of the analyzed 10x10 BWR fuel assembly.

Table 1: Geometrical data of the BWR fuel assembly.

Fuel pitch (mm)	152.4
Pin pitch (mm)	12.95
Pellet diameter (mm)	8.87
Clad inside diameter (mm)	9.04
Clad outside diameter (mm)	10.28
Rod to channel gap (mm)	3.858
Water channel to rod gap (mm)	3.26
Channel wall thickness (mm)	1.70
Water channel outer width (mm)	35.0
Water channel wall thickness (mm)	0.80

A single detailed CTF model and two different TRACE models have been made using the available data. The two models for the latter code differ in its complexity; the simple one uses the normal channel option, while the advance uses the advance channel option of the last version of TRACE code (version 5, patch3).

2.1. COBRA-TF code model.

The CTF high fidelity model represents, with the higher accuracy allowed by the code, the design features of the fuel assembly. The model includes the representation of PLRs, water rod, fuel channel and the axial changes on the geometrical parameters (flow area, hydraulic diameter and gaps between rods) that take place through the assembly into a pin scale (subchannel approximation).

The scheme presented in the Figure 2 shows the CTF model that contains 92 subchannels including the water rod. Each flow subchannel is divided into 28 axial nodes and connected with its corresponding heat structure. Furthermore the water rod is connected to 9 heat structures representing the water channel cladding and also the fuel box is represented by passive heat structures that induces heat convection with the fluid of surrounding subchannels.

In the cross section of the left side of Figure 2 has been highlighted in blue the positions of the subchannels that contains PLRs as heat structures, whereas the water rod is represented in yellow. The right side shows a simplified scheme of axial nodalization for the CTF model and the connexion of the heat structures with the nodes.

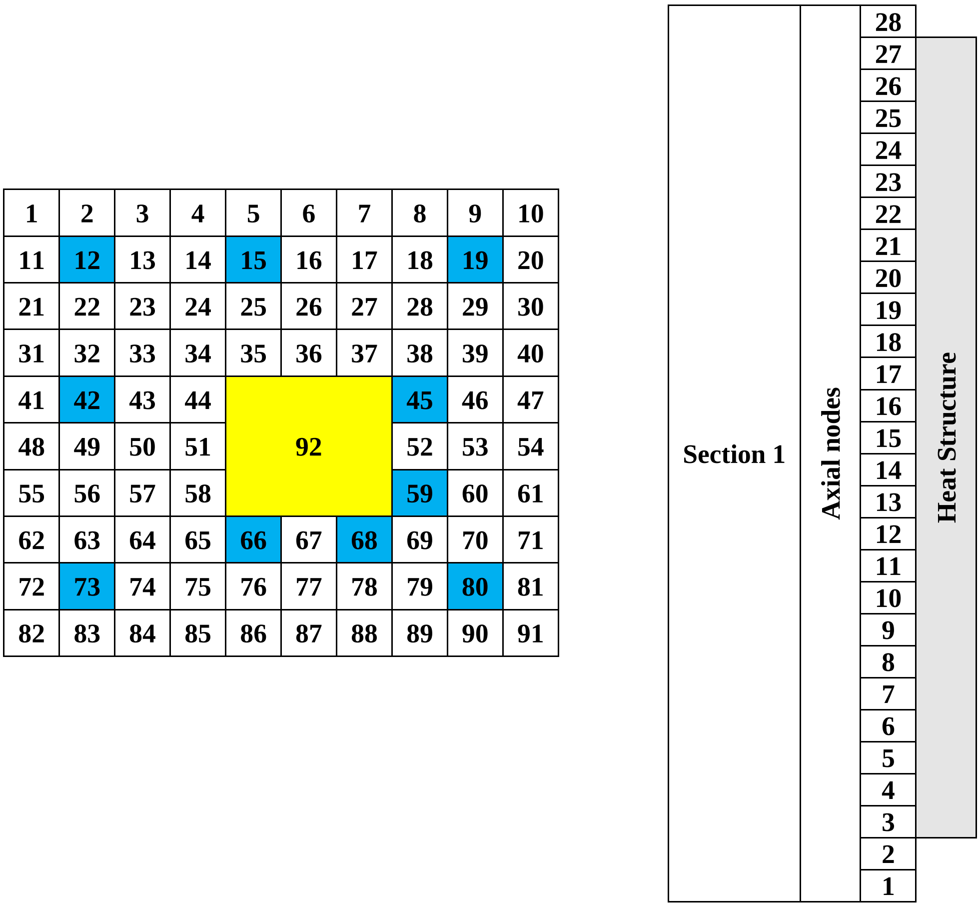

Figure 2: Nodalization scheme of the COBRA-TF model.

2.2. TRACE code models.

The Figures 3 and 4 represent, respectively, the scheme of the simple and advanced models developed for the TRACE code. In the simple model, the fuel assembly is represented by a single flow channel connected with a heat structure that represents the 91 fuel rods (including the PLRs), while in the advance model has been modelled the water rod (WR) and the PLRs using the TRACE's advanced channel feature. Axially, both models have 28 nodes like CTF model. This means that the fuel assembly is represented by two flow channel (one for fuelled flow path and other for the WR) and two heat structures for FLRs and PLRs respectively. Basically both models have different level of detail in the representation of PLRs and WRs.

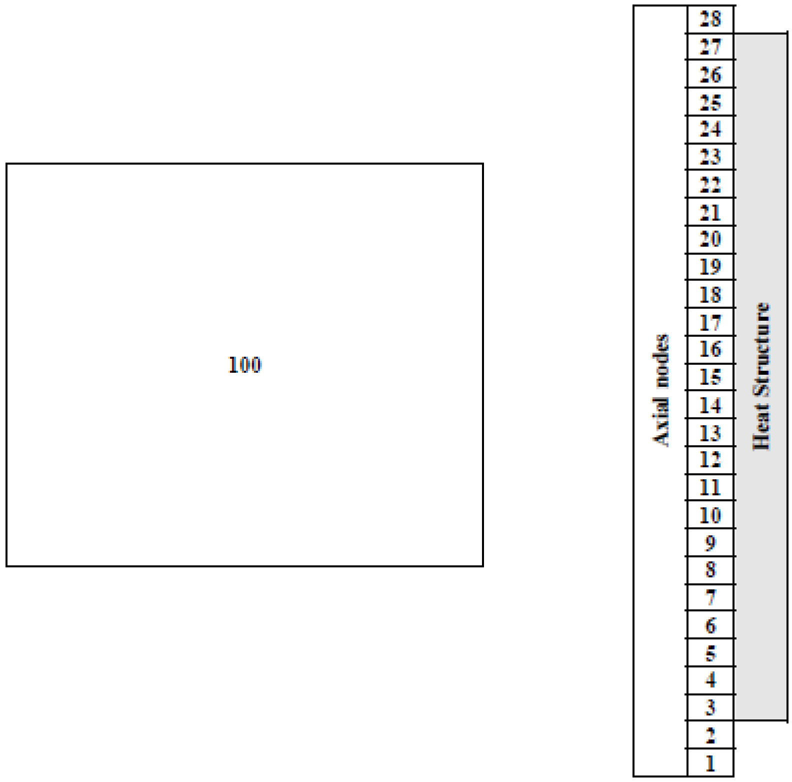

Figure 3: Nodalization scheme of the SIMPLE TRACE model.

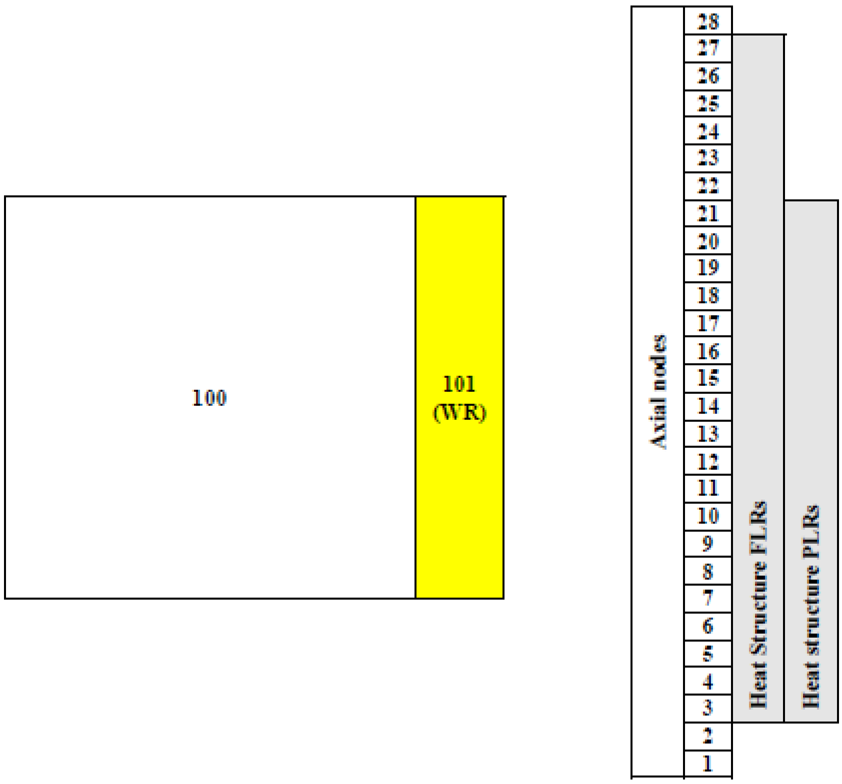

Figure 4: Nodalization scheme of the ADVANCE TRACE model.

PARCS model is the same for coupled CTF/PARCS and TRACE codes. Each axial node of the thermalhydraulic model is connected to a neutronic node through the coupling scheme that provides the corresponding feedback of variables. The two lower axial nodes in the thermalhydraulic models have been grouped in a reflector node for neutronic model as well the upper node. Also it should be noted that PARCS uses symmetry as a boundary condition.

3. RESULTS AND DISCUSSION.

The results of the simulations are presented in this section and the differences observed between the simulations are commented and discussed. In the Figures 5-12 are shown a comparison of the results obtained by the simulation of steady state with the three models previously presented and the coupled codes CTF/PARCS and TRACE.

As can be observed in the Figure 5, the axial power profile is quite different for all the simulations, the relative difference in the peaking factors obtained in the three simulations is about the 5% and the root mean square (RMS) error between TRACE and CTF/PARCS simulations is 3.68%. The CTF/PARCS and the advanced model of TRACE show higher power peak factors at the node 16, which corresponds to the end of PLRs. The results could be partially explained by the level of detail of the different models and codes for representing the partial rods and the consideration of the water rod in the advance TRACE and CTF models.

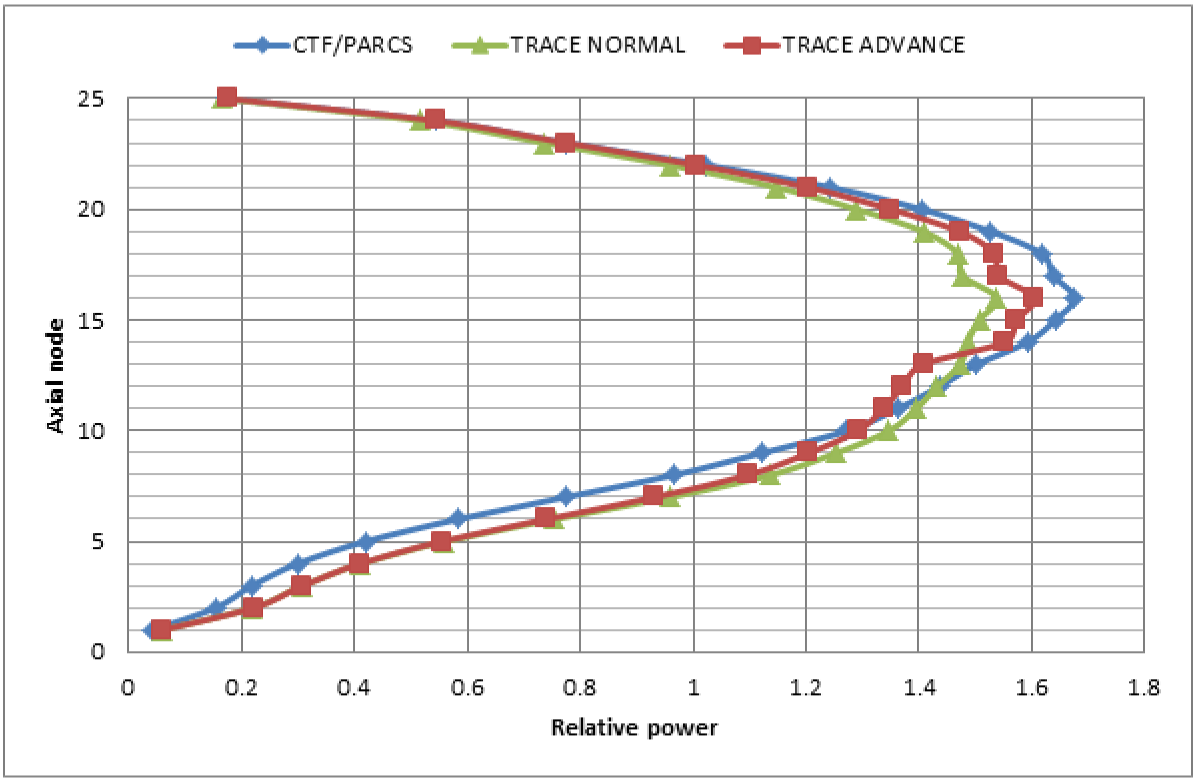

Figure 5: Comparison of the axial power profile.

The Figure 6 shows a considerable discrepancy between the pressure drops calculated by the two codes. CTF predicts a higher pressure drop, of about 0.14 MPa, while the two TRACE models predicts about 0.1 MPa. Since there are not abrupt differences in the pressure profile due to local effects, the explanation is the different models and correlations used by the codes to calculate the wall and interfacial friction.

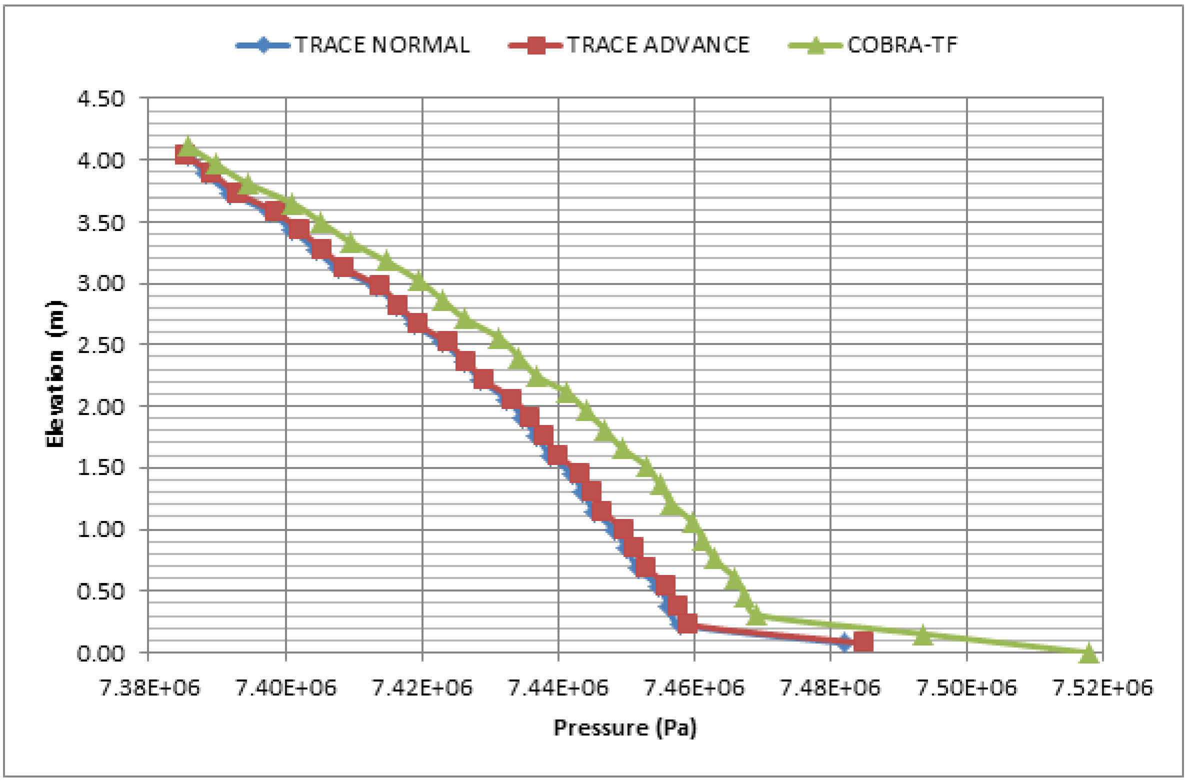

Figure 6: Comparison of pressure drop axial profile.

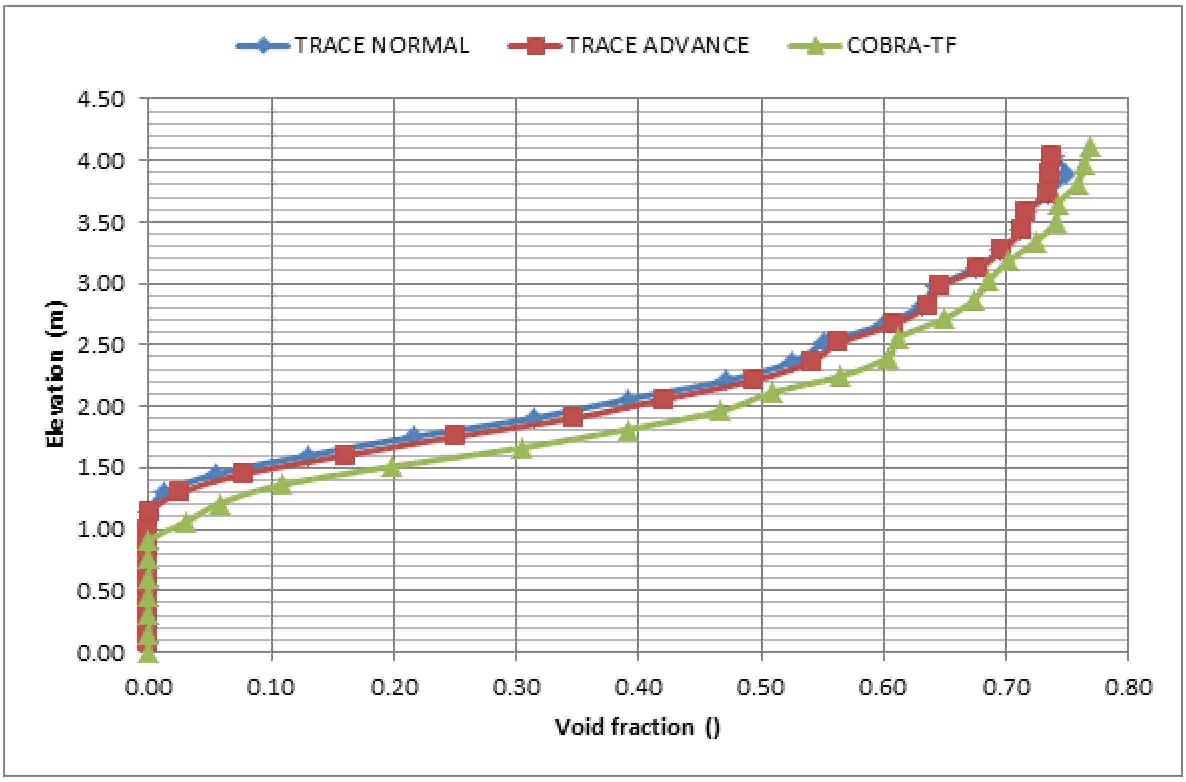

Figure 7: Comparison of the void fraction axial profile.

The differences in the void fraction prediction plotted in the Figure 7 could be explained by the previously commented pressure drops and axial power profiles. CTF code predicts higher void fractions, mainly in the middle and lower part of the bundle. Moreover the Figures 8 and 9 represent the liquid temperature and density profiles that presents slight gaps into the values calculated with the analyzed models and codes.

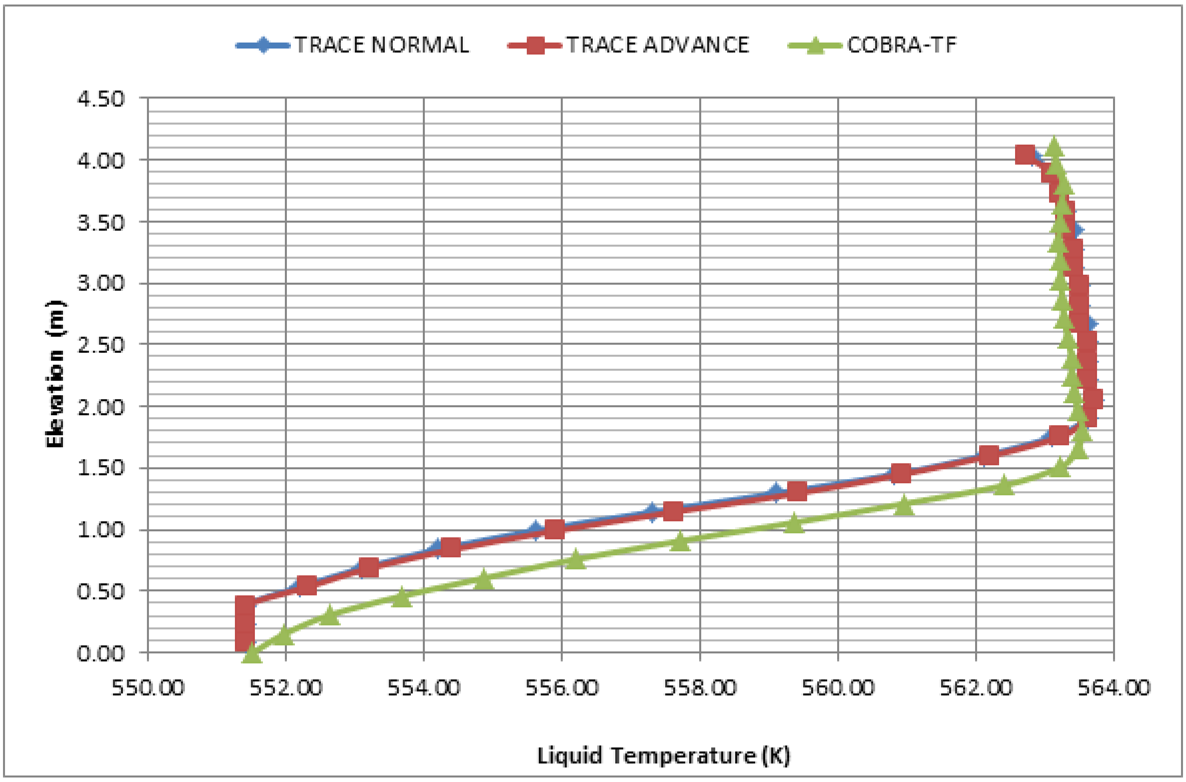

Figure 8: Comparison of the axial profile of liquid temperature.

Another aspect is the difference in the velocity profile calculated by CTF and TRACE codes. As shown in Figures 10 and 11, CTF predicts higher velocities, for both the liquid and the vapor phases, from the height where the two-phase flow regime is established. This behavior may be due to many factors, such as the differences in the formulation of the equations of subchannel code, which takes into account cross flows and 3D effects in the transport of momentum, the differences in the friction models and correlations, and also by the prediction of the fluid regimes that depends on the computed void fraction.

The Figure 12 presents the liquid temperature profile computed with the advanced model of TRACE and the CTF model in the water rod. As it can be seen, the temperature acquired by the coolant that pass through the water rod is similar in both cases.

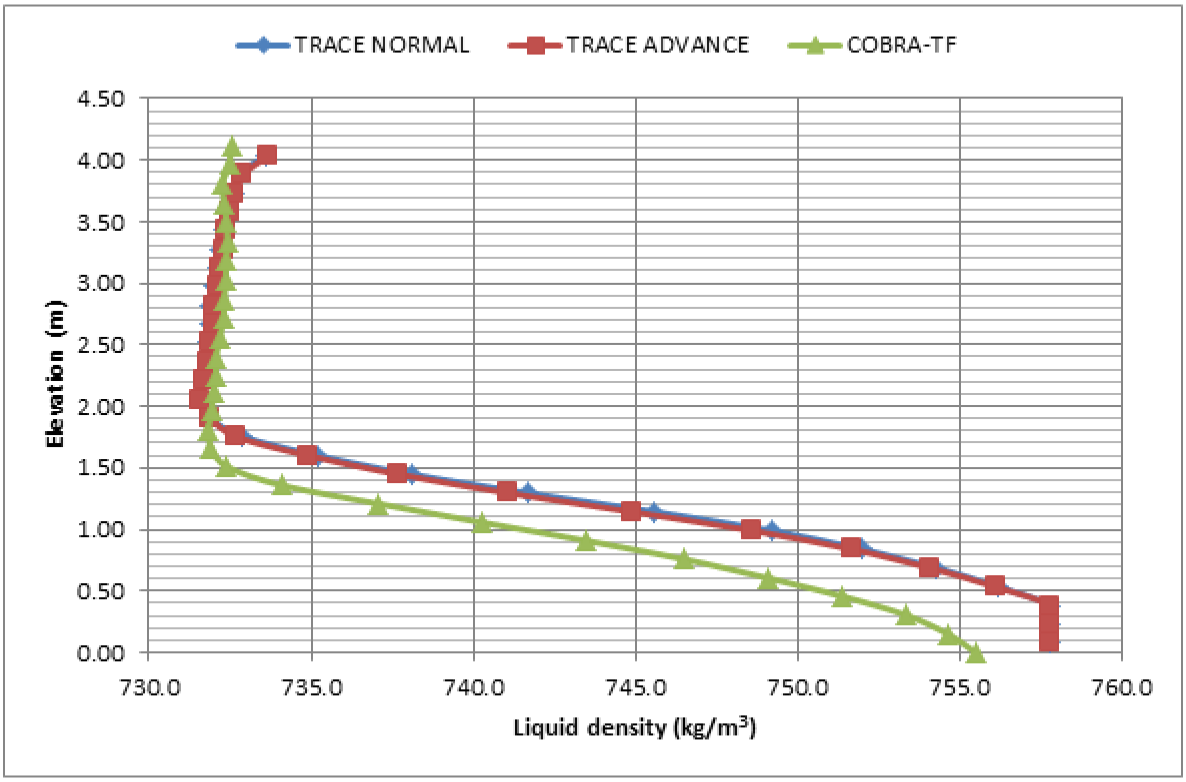

Figure 9: Comparison of the axial profile of liquid density.

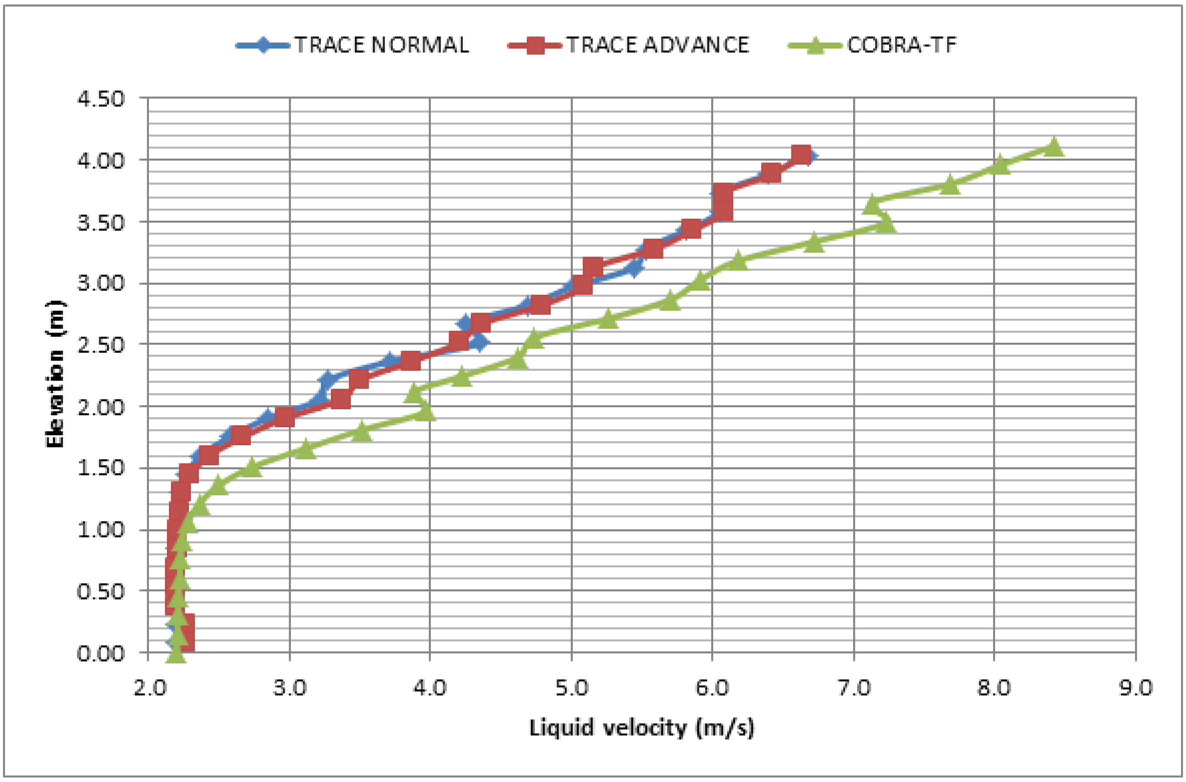

Figure 10: Comparison of the axial profile of liquid velocity.

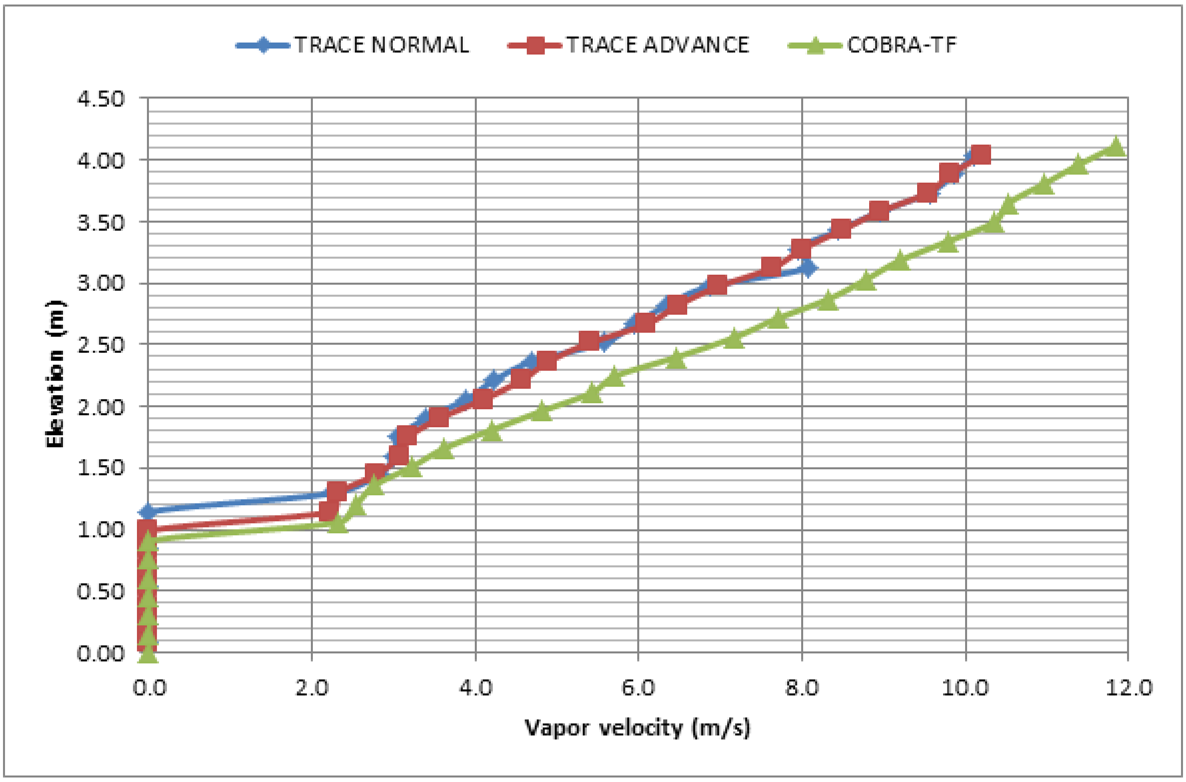

Figure 11: Comparison of the axial profile of vapor velocity.

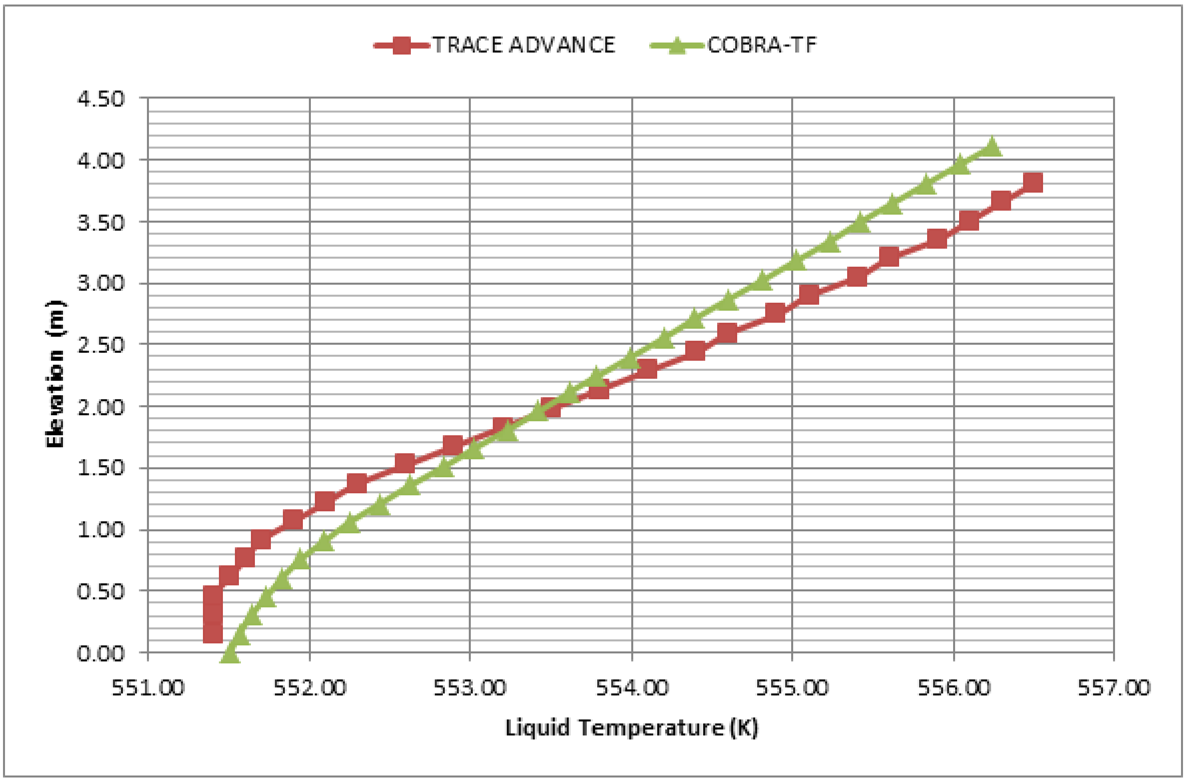

Figure 12: Comparison of the axial profile of liquid temperature in the water rod.

4. CONCLUSIONS.

A comparison between the CTF/PARCS and TRACE codes has been done using a modern 10 by 10 BWR fuel assembly. The results obtained with both models of TRACE (simple and advance channel) show similar trends. In the other hand, the subchannel code CTF shows slightly different results in the steady state. The results of CTF shows higher pressure drop and void fractions. These differences between the codes could be explained by the high fidelity of CTF code, which computes cross flow around the length of fuel bundle, and geometrical changes in a subchannel scale, but also due to the different models and correlations used by the codes.

To understand better the different behavior of both codes (TRACE and CTF) for simulate high fidelity models, as modern BWR fuel assemblies, would desirable to make further studios of models and correlations used for the codes, and carry out the comparison of the results with experimental data obtained with this type of nuclear fuel assemblies. Hence, this study could be considered a first attempt to show the different capabilities of the system and subchannels codes to represent with high fidelity the advance BWR fuel assemblies, and the different results obtained with each approximation.

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REFERENCES

1. T. Downar, D. Lee, Y. Xu, T. Kozlowski, J. Staundenmier, “PARCS v2.7 US NRC Core Neutronics Simulator” (2006).
2. COBRA/TRAC – “A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems. Equations and Constitutive Models”. **Vol. 1**. *NUREG/CR-3046 PNL-4385* (1983).
3. COBRA/TRAC – “A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems. Users’ Manual”. *Vol. 3*. *NUREG/CR-3046 PNL-4385* (1983).
4. A. Abarca, T. Barrachina, R. Miró, G. Verdú, “A Procedure for Coupled Thermal-hydraulic Subchannel and Neutronic Codes Using COBRA-TF and PARCS”. *INAC* (2011).