2009 International Nuclear Atlantic Conference - INAC 2009
Rio de Janeiro, RJ, Brazil, September 27 to October 2, 2009
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR - ABEN
ISBN: 978-85-99141-03-8

VERIFICATION AND VALIDATION OF ONE DIMENSIONAL MODELS USED IN SUBCOOLED FLOW BOILING ANALYSIS

Francisco A. Braz Filho[1], Alexandre D. Caldeira[1], Eduardo M. Borges[1] and Gaianê Sabundjian[2]

[1] Instituto de Estudos Avançados (IEAv)
Rodovia dos Tamoios, km 5,5
12228-001 São José dos Campos, SP
fbraz@ieav.cta.br

[2] Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP)
Av. Professor Lineu Prestes 2242
05508-000 São Paulo, SP

ABSTRACT

Subcooled flow boiling occurs in many industrial applications and it is characterized by large heat transfer coefficients. However, this efficient heat transfer mechanism is limited by the critical heat flux, where the heat transfer coefficient decreases leading to a fast heater temperature excursion, potentially leading to heater melting and destruction. Subcooled flow boiling is especially important in water-cooled nuclear power reactors, where the presence of vapor bubbles in the core influences the reactor system behavior at operating and accident conditions. With the aim of verifying the subcooled flow boiling calculation models of the most important nuclear reactor thermal-hydraulic computer codes, such as RELAP5, COBRA-EN and COTHA-2tp, the main purpose of this work is to compare experimental data with results from these codes in the pressure range between 15 and 45 bar. For the pressure of 45 bar the results are in good agreement, while for low pressures (15 and 30 bar) the results start to become conflicting. Besides, as a sub-product of this analysis, a comparison among the models is also presented.

1. INTRODUCTION

Subcooled flow boiling occurs in many industrial applications and is characterized by large heat transfer coefficients. However, this efficient heat transfer mechanism is limited by the critical heat flux where the heat transfer coefficient decreases leading to a fast heater temperature excursion potentially leading to heater melting and destruction.

Prediction of the location of the onset of significant voids and void fraction along the flow direction during subcooled flow boiling is very important for nuclear reactors. As the void fraction increases, the reactivity in the reactor core decreases and vice-versa. The vapor void fraction in water subcooled flow boiling is of significant importance in predicting the inception of two-phase instability and the onset of the critical heat flux condition in boiling and pressurized water reactors [1]. Furthermore, it was observed that the occurrence of subcooled nucleate boiling in the core region of pressurized water reactors causes the accumulation of boron compounds on fuel surfaces, leading to an unexpected deviation in the axial power distribution [2].

In the subcooled region of upward nucleate boiling flow in a vertical channel with a heated wall, the temperature near the wall and the bulk fluid temperature are, respectively, higher and lower than the saturation temperature. Subcooled boiling is thus characterized by a ‘‘higher-temperature’’ two-phase region near the heated surface and a ‘‘lower-temperature’’ single-phase liquid region away from the heated surface.

The evolution of the void fraction in subcooled boiling flow over a heated channel may be modeled using various approaches with different time and length scales. Recently, several CFD (Computational Fluid Dynamics) commercial computer codes [3], like CFX, are being used for multi-dimensional calculations in subcooled boiling flow, however these models depend on parameters that are not so easy to measure. This work is focused in one dimensional geometry models, which are implemented in the most important thermal-hydraulic and accident analysis computer codes utilized for nuclear reactor design calculations. The one dimensional models range from the simple homogeneous model, constituted of three conservation equations for the mixture liquid-steam, until the most complete model of six equations, i.e. three equations for each phase, and go through intermediate models of four equations. With the aim of verifying the subcooled flow boiling calculation of the most important thermal-hydraulic computer codes used in nuclear reactor simulations, the purpose of this work is a comparison of experimental data with results of these codes in the pressure range between 15 and 45 bar. There is a tendency that for high pressures ($\geq$ 45 bar) the obtained results are reasonable, while for lower pressures ($\leq$ 30 bar) the results begin to become conflicting. Besides, as a sub-product of this analysis, a comparison amongst the models is also presented.

2. EXPERIMENTAL APPARATUS

For comparison purposes, experimental data for a heated vertical tube were used [4]. For the heated channel, tubes with an inner diameter of 15.4 mm and length of 2000 mm, fabricated of stainless steel, were utilized. The fluid used in the experience was water, flowing in upward direction at the test section.

3. COMPUTER CODE MODELS

Two-phase flow involves a relative movement between the two phases, liquid and vapor, therefore, this problem can be formulated in terms of two fields of velocity. The complete formulation of this phenomenon in one-dimensional geometry is described by conservation equations for the two fluids, totaling six equations. Besides, a series of constitutive equations can be coupled to the problem, to specify the interface interaction between the two phases. There are a lot of difficulties in the modeling with two separate fluids, mainly in the elaboration of the constitutive equations. There are simpler models, which in certain conditions can simulate two-phase flow well, due to experimental validation. The basic concept of these models considers the liquid-vapor mixture as a single component, as for instance the homogeneous and drift flux models [5, 6]. In this work, RELAP5 [7], COBRA-EN [8] and COTHA-2tp [9] computational codes were used in the simulations. To proceed, a succinct description of these codes is presented.

3.1. The RELAP5 Code

The RELAP5 code has been developed for best-estimate simulations of transients in light water reactor (LWR) coolant systems. The code is based on a non-homogeneous and nonequilibrium two-phase model of six equations that is solved by a fast, partially implicit numerical scheme permitting fast calculation of system transients.

The subcooled boiling model used in the RELAP5 code is based on a mechanistic concept of evaporation and condensation. The net vapor generation rate is a result of the competition between evaporation and condensation rates. The prediction of steam contents in the boiling channel also depends on the relative velocity between vapor and bulk liquid flow as well as on the flow regime. The relevant sub-models in the RELAP5 code which affect the void fraction prediction in the subcooled boiling regime are described in Volume IV of the code manual and concisely in References 10 and 11. Regarding the simulation conditions, per RELAP5 code nodalization requirements, the channel was converted into a pipe having an equivalent cross-sectional flow area and hydraulic diameter. The nodalization scheme for the pipe test section used in the simulation is shown in Figure 1. The heated test section (PIPE component 3) was subdivided into forty control volumes. Time Dependent Volumes (TMDPVOL) and Time Dependent Junctions (TMDPJUN) were added to simulate the problem boundary conditions: system pressure, inlet temperature and inlet mass flow rate. To obtain the stationary state conditions, initially, temperatures and mass flow rates were supplied for all volumes and junctions and taken equal to the inlet temperature and the inlet mass flow rate, respectively. The code was executed until a steady state condition was reached.

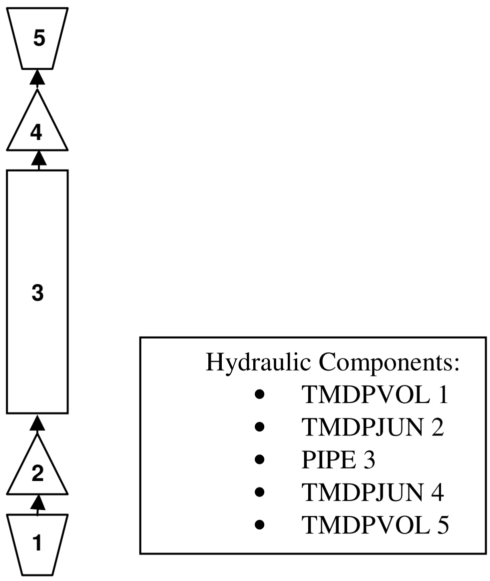

Figure 1. Nodalization scheme employed in the RELAP5 code.

3.2. The COBRA-EN Code

The COBRA-EN code is an upgraded version of the COBRA-3C/MIT code used for thermal-hydraulic transient analysis of reactor cores. Starting from a steady state condition in a LWR core or fuel element, the code allows to simulate the thermal-hydraulic transient response to user-supplied changes of the total power, of the outlet pressure and of the inlet enthalpy and mass flow rate.

The thermal-hydraulic homogeneous model implemented in the COBRA-EN code is based on three partial differential equations which, using what is known as "subchannel approximation", describe the conservation of mass, energy and the momentum vector in axial and lateral directions for the water liquid/vapor mixture and the interaction of the two-phase coolant with the system structures. Optionally, a fourth equation can be added which tracks the vapor mass separately and which, along with the correlations for vapor generation and slip ratio, replaces the subcooled quality and quality/void fraction correlations, needed to extend the capabilities of the essentially homogeneous three-equation model.

If the computational domain is subdivided into a number of axial intervals, the control volume for mass, energy and axial momentum is a segment of the subchannel while the control volume for the lateral momentum is a segment of the somewhat arbitrary region which couples the two adjoining subchannels around a lateral gap. In each control volume, the flow equations as well as the one-dimensional heat conduction equations in the fuel rods are approximated by finite differences. The resulting equations for the hydrodynamic phenomena form a system of coupled nonlinear equations that are solved either by an implicit iterative scheme based on the calculation of the pressure gradients in the axial direction or by a Newton-Raphson iteration procedure. The heat conduction equations in the solid structures are treated implicitly. Moreover, the rod-to-coolant heat transfer model is described by a full boiling curve, comprising the basic heat-transfer regimes (forced convection, nucleate boiling, transition and film boiling), each represented by a set of optional correlations for the heat-transfer coefficient.

For this problem, the momentum equation in the lateral direction is not used, because just one channel of one-dimensional geometry in the axial direction is being analyzed. The channel is again divided in forty axial volumes.

3.3. The COTHA-2tp Code

The COTHA-2tp code, developed at the Institute of Advanced Studies (IEAv), uses the homogeneous model for two-phase flow analyses of light water reactor cores. This code is restricted to one-dimensional geometry and steady state calculations. The homogeneous model, at first, does not include the sub-cooled flow boiling regime. Two empiric models, those of Levy [12] and Lellouche (developed by EPRI [13]), were employed in the COTHA-2tp code to correct the homogeneous model for the occurrence of sub-cooled flow boiling.

4. RESULTS

To perform the comparisons, the code models were chosen as follows: for RELAP5, the six equations model; for COBRA-EN, the four equations model (three equations for the mixture plus one equation for the steam); for COTHA-2tp, the homogeneous model (three equations for the mixture), with Levy’s and Lellouche’s corrections for the thermodynamic quality calculation.

Two sample cases were studied: in the “A” case, the velocity mass flow is 900 kg/(m^2.s), the heat flux is 0.57 10^6 W/m^2 and the pressure is 45.0 bar and, in the “B” case, the velocity mass flow is 900 kg/m^2 s, the heat flux is 0.38 10^6 W/m^2 and the pressures are 15.0 bar, 30.0 bar and 45.0 bar.

4.1 The “A” Sample Case

This test is especially important because, besides the void fraction data, other important parameters, such as the fluid bulk temperature and the wall temperature data, are compared with the code results.

Figure 2 shows a graph of the void fraction versus fluid enthalpy along the channel. RELAP5 code results are in very good agreement with the experimental data, describing very well the beginning of the boiling regime, however overestimating the void fraction values when the regime of nucleate boiling is reached. With the COBRA-EN code a small delay in predicting the beginning of the boiling regime was observed, however the curve describes well the experimental data and stays close until the nucleate boiling regime is attained. The homogeneous model of the COTHA-2tp code presented the most conflicting results. Levy's formulation gives rise to a delay in the beginning of the boiling regime and overestimates the void fraction values when nucleate boiling occurs, while Lellouche's formulation produces better results.

Figure 3 shows a graph of the fluid bulk temperature versus fluid enthalpy along the channel. All code results show good agreement with the experimental data.

The results of the calculation of the wall temperature versus the fluid enthalpy along the channel are shown in Figure 4. For the experimental data the temperature arises quickly and stays practically constant along the channel, while for the results of the codes the temperature increases continuously in a slower way. This difference can be attributed to the heat transfer coefficient calculation, since the temperature of the fluid is in reasonable agreement with the experiment.

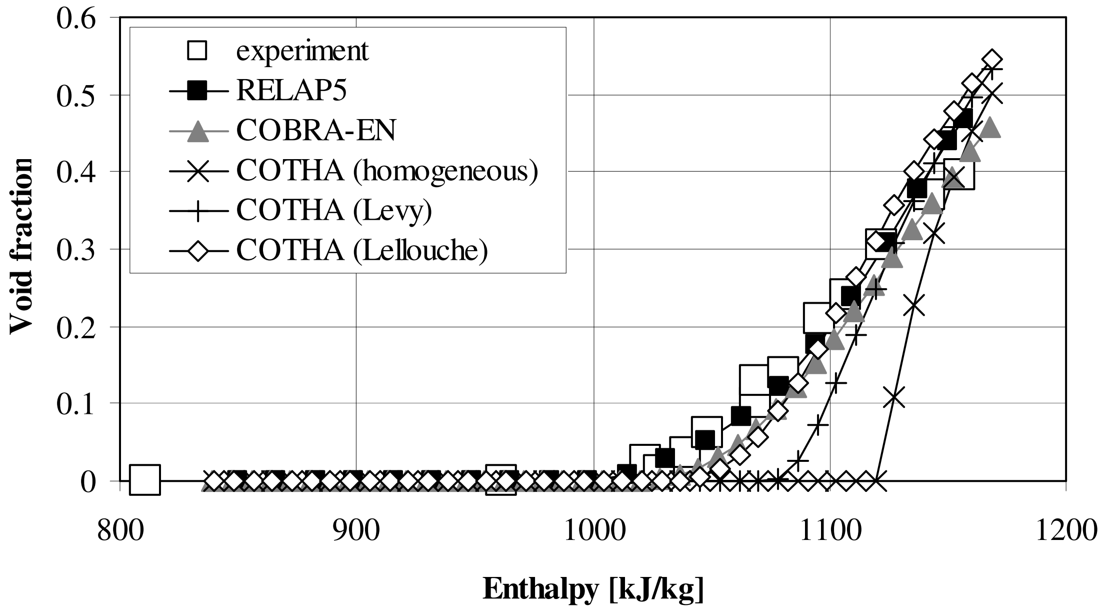

Figure 2. Void fraction versus enthalpy along the channel for “A” sample case.

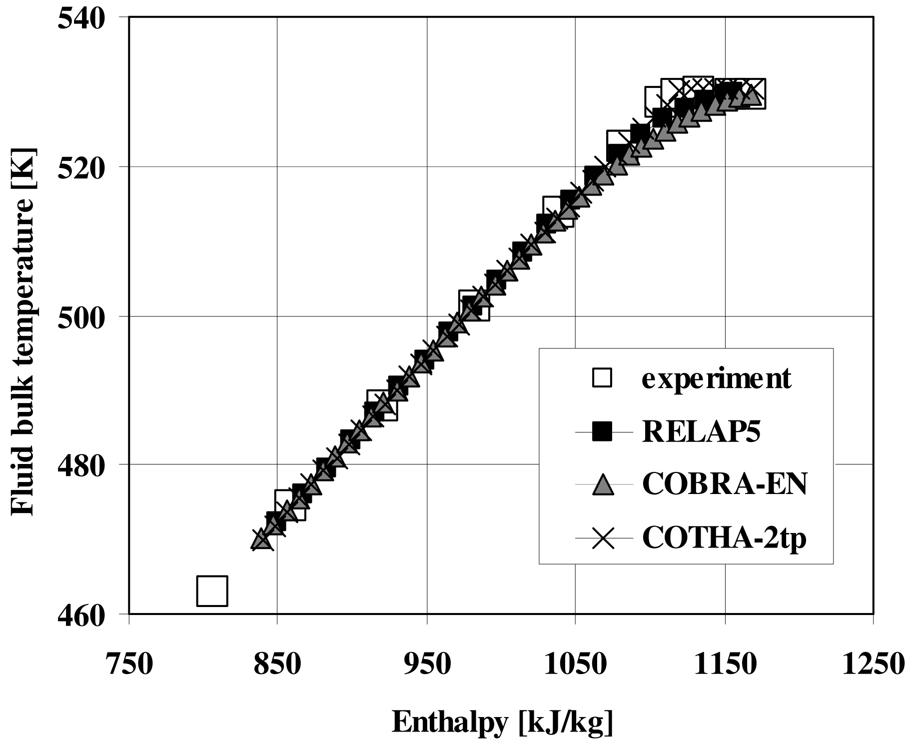

Figure 3. Fluid bulk temperature versus enthalpy along the channel for "A" sample case.

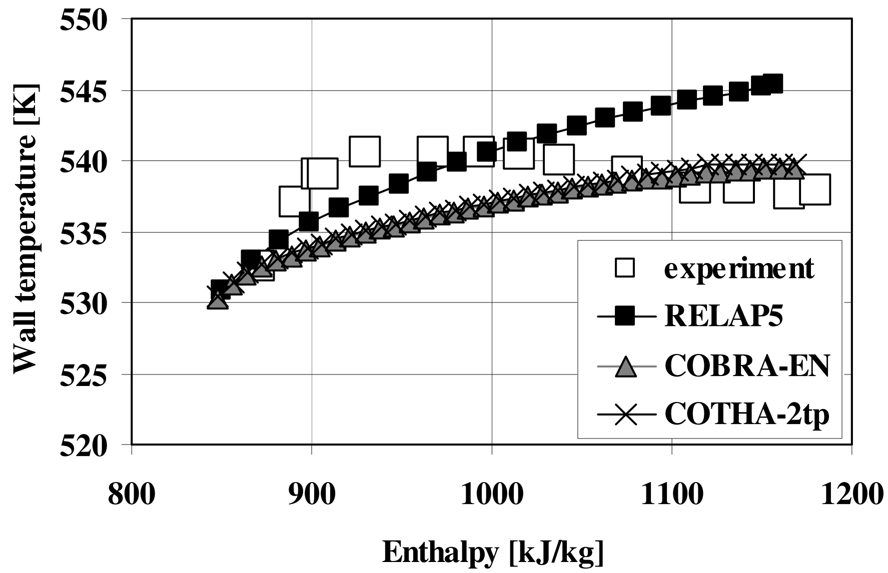

Figure 4. Wall temperature versus enthalpy along the channel for "A" sample case.

4.2 The "B" Sample Case

Figures 5 and 6 show the COTHA-2tp code results for the void fraction calculation versus the thermodynamic quality along the channel, using the Levy's and Lellouche's formulations, respectively. Levy's formula produced bad results, so much in the prediction of the beginning of boiling as in the underestimation of the void fraction values in the sub-cooled flow boiling regime. The results are more conflicting the lower the pressure is. The results of Lellouche obtained better agreement with experimental data, when compared with the results of Levy.

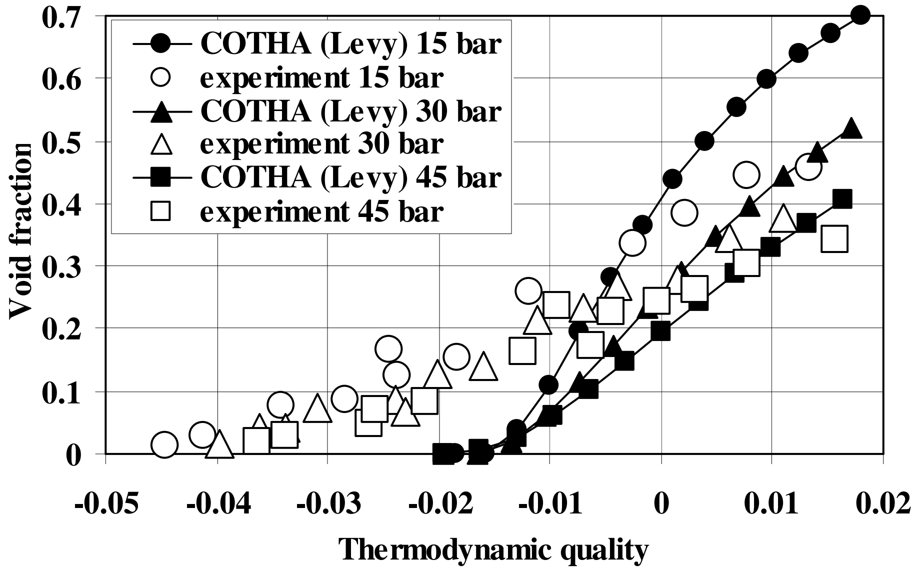

Figure 5. Void fraction versus thermodynamic quality along the channel for "B" sample case (COTHA).

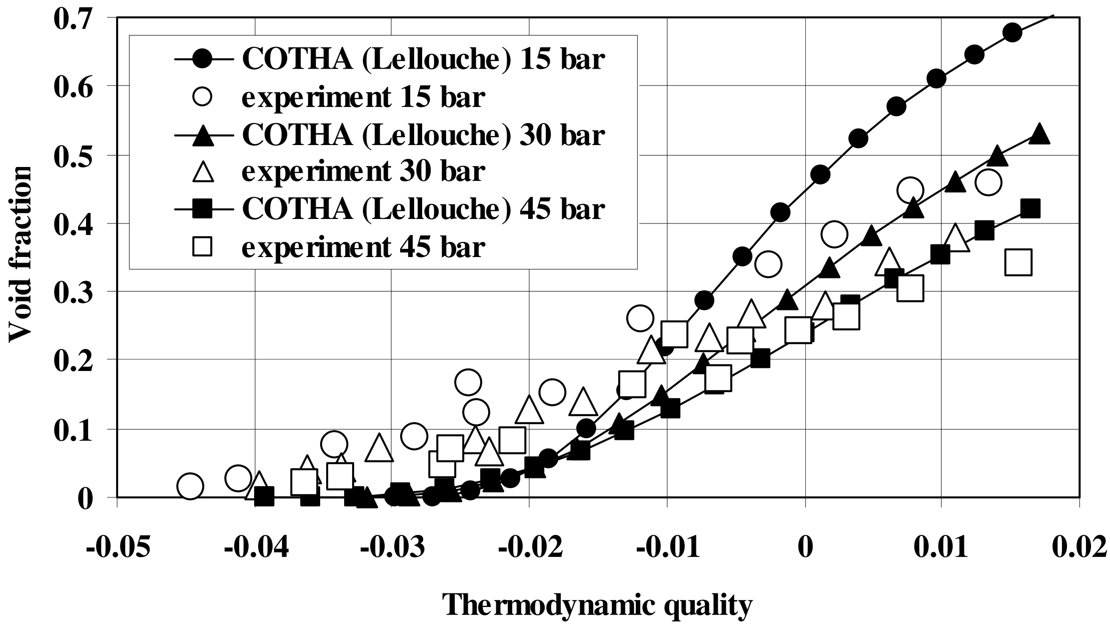

Figure 6. Void fraction versus thermodynamic quality along the channel for "B" sample case (COTHA).

Figure 7 shows the COBRA-EN code results for the void fraction calculation versus the thermodynamic quality along the channel. The results are in good agreement with experimental data for the pressure of 45 bar. The differences between the results and the experimental data increase as the system pressure is reduced.

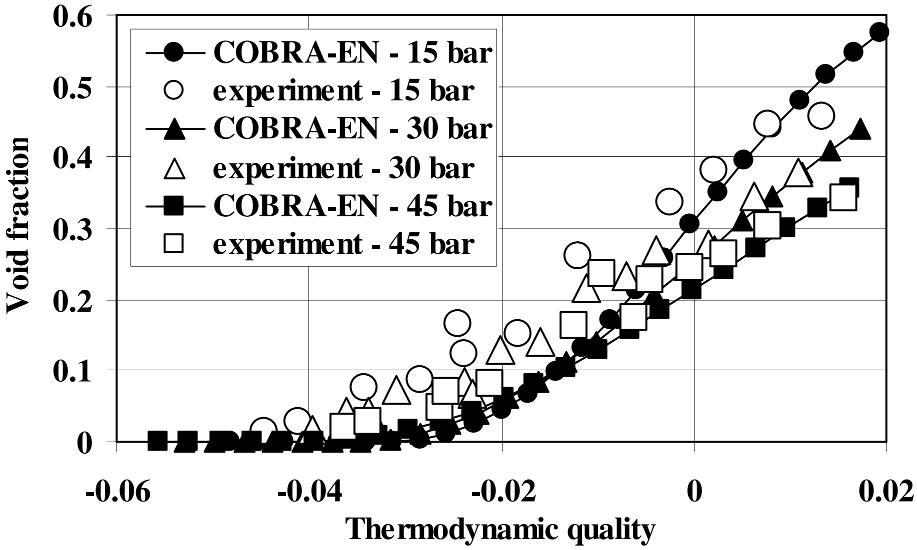

Figure 7. Void fraction versus thermodynamic quality along the channel for "B" sample case (COBRA).

Figure 8 shows the RELAP5 code results for the void fraction calculation versus the thermodynamic quality along the channel. These results are the best ones, but, as observed for the other codes, the differences with respect to the experimental data increase as the system pressure is reduced.

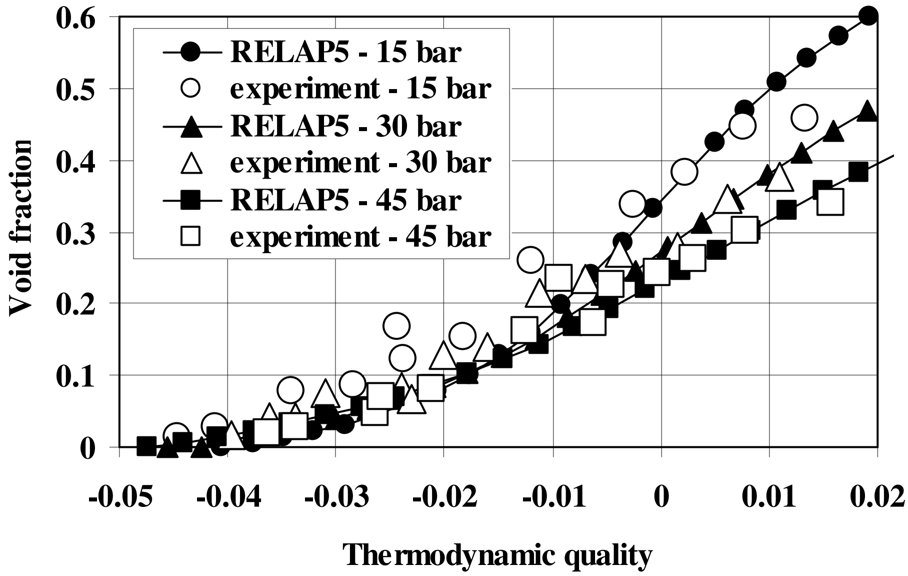

Figure 8. Void fraction versus thermodynamic quality along the channel for "B" sample case (RELAP5).

5. FINAL COMMENTS

Subcooled flow boiling is especially important in water-cooled nuclear power reactors, where the presence of vapor bubbles in the core influences the reactor system behavior at operating and accident conditions.

Reactors operate in a wide range of pressure, from one to hundreds of atmospheres. One-dimensional models with various degrees of empiricism may predict fairly well the void fraction, averaged over the channel cross-section. These models are used for transient simulations in nuclear reactors at high-pressure conditions. The interest in the simulation of subcooled flow boiling at low-pressures (1–2 bars) has increased over the past few years, driven mainly by the need of performing safety analyses of research nuclear reactors operating near atmospheric pressure [10] and investigation of new concepts for Advanced Light Water Reactors. At low pressures, the saturation temperature is lower and the difference between liquid and vapor densities is much higher than at high pressure, leading to an increase in the size of generated bubbles and a decrease in nucleation site density and bubble nucleation frequency [14]. As a consequence of changed bubble dynamics and different thermodynamic properties of water, the void fraction behavior is significantly different at low pressures, so that the extrapolation of high-pressure data to low-pressure conditions usually leads to erroneous results. The "B" sample case results show clearly that the results for the pressure of 45 bar are better than those for the pressures of 30 and 15 bar, in all cases.

Comparing the models, the worst results were obtained with the homogeneous model corrected by the Levy's and Lellouche's formulations. It is important to note that these models are used in one of the main thermo-hydraulic analysis codes, COBRA-IV-I [15].

Although the focus of this article is the subcooled flow boiling regime, all the cases include the beginning of the nucleate boiling regime. It is observed that all the models yield underestimated values for void fraction subcooled boiling and overestimated values for nucleate boiling in comparison with experimental data.

As a future work, computer code results for low (1–2 bars) and high (150–200 bars) pressures will be compared with experimental data, to complete this study.

ACKNOWLEDGMENTS

The authors wish to thank R. D. M. Garcia for his many useful comments and suggestions to the manuscript of this work.

REFERENCES

1. T. Hamidouche, A. Bousbia-salah and F. D'Auria, "RELAP5/Mod 3.2 Code Assessment Study Under Low Pressure Onset of Flow Instabilities", *International Conference in Nuclear Energy for New Europe 2004*, Slovenia, pp. 122.1-122.9 (2004).

2. V. K. Dhir, "Complete Numerical Simulation of Subcooled Flow Boiling in the Presence of Thermal and Chemical Interactions", NERI Project 99-0134, Report: DE-FG03-99SF-21930, Los Angeles, (2003).

3. E. Krepper, B. Konkar and Y. Egorov, "CFD Modelling of Subcooled Boiling – Concept, Validation and Application to Fuel Assembly Design", *Nuclear Engineering and Design*, **v. 237**, pp 716-731 (2007).

4. G. G. Bartolomei and V. M. Chanturiya, "Experimental Study of True Void Fraction When Boiling Subcooled Water in Vertical Tubes", *Thermal Engineering*, **v. 14**, pp. 123-128 (1967).

5. P. Coddington and 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*, **v. 215**, pp. 199-216 (2002).

6. B. Xu and P. K. Bansal, "Non-adiabatic Capillary Tube Flow: a Homogeneous Model and Process Description", *Applied Thermal Engineering*, **v. 22 (16)**, pp. 1801-1819 (2002).

7. RELAP5 Code Development Team, "RELAP5/MOD3 Code Manual", NUREG/CR-5535, Scientech, Inc., Idaho Falls (1999).

8. D. Basile, M.Beghi, R. Chierici and E. Salina, "COBRA-EN: an Upgraded Version of the COBRA-3C/MIT Code for Thermal-hydraulic Transient Analysis of Light Water Reactor Fuel Assemblies and Cores", Report 1010/1 (1999).

9. F. A. Braz Filho, E. M. Borges and A. D. Caldeira, "COTHA-2tp: A Code for Two-phase Flow Analysis of PWR Cores Using the Homogeneous Model", *International Nuclear Atlantic Conference – INAC 2007*, Santos (2007).

10. B. Konkar and B. Mavko, "Modelling of Low-pressure Subcooled Flow Boiling Using the RELAP5 Code", *Nuclear Engineering and Design*, **v. 220**, pp. 255-273 (2003).

11. S. Petelin and B. Konkar, "Prediction of Void Fraction in Subcooled Flow Boiling", *International Conference in Nuclear Energy in Central Europe 1998*, Slovenia, pp. 195-222 (1998).

12. S. Levy, "Forced Convection Subcooled Boiling Prediction of Vapor Volumetric Fraction", *International Journal Heat Mass Transfer*, **v. 10(7)**, pp. 951-965 (1967).

13. International Atomic Energy Agency, "Thermal-hydraulic Relationships for Advanced Water Cooled Reactors", IAEA-TECDOC-1203, IAEA, Vienna, Austria (2001).

14. V. Prodanovic, D. Fraser and M. Salcudean, "Bubble Behavior in Subcooled Flow Boiling of Water at Low Pressures and Low Flow Rates", *International Journal of Multiphase Flow*, **v. 28**, pp. 1-19 (2002).

15. Wheeler C. L., C. W. Stewart, R. J. Cena, D.S. Rowe and A. M. Sutey, "COBRA-IV-I: An Interim Version of COBRA for Thermal-hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores", BNWL-1962, Battelle-Northwest, Richland, WA (1976).