THERMAL-HYDRAULIC ANALYSIS CODE FOR PLATE-TYPE FUEL NUCLEAR REACTORS

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Abstract. *Plate-type fuel assemblies is mostly associated with research and material test reactors but currently such fuels have also been considered for research and naval propulsion reactors. This research work presents a computer code model written in FORTRAN language to perform thermal-hydraulic analysis of nuclear plate-type fuel elements, named COTENP. The code solves the conservation equation for mass, momentum and energy for a sub-channel based on geometric and thermal-hydraulic conditions. It calculates minimum DNBR for the hottest channel. The code uses the chain or cascade method for two stages in order to facilitate the whole analysis. In the first stage, we divide the core into channels with size equivalent to a fuel assembly. In the second stage, we divide the hottest fuel assembly into sub-channels with size equivalent to one actual coolant channel. For the code validation, we considered two different problems. The first was the CARR research reactor with low pressure and temperature conditions, and the second was the LABGENE reactor, a small PWR prototype with high temperature and pressure conditions. The code yields detailed information such as static pressure in the channel, mass flow rate distribution among the channels, coolant temperature axial distribution, quality and local and critical fluxes. The COTENP code reproduced well the CARR reactor results, but presented important discrepancy regarding the temperature axial distribution for the LABGENE reactor results. The DNBR estimation for both problems were accurate.*

Keywords*: Thermal-hydraulic, reactor, research, plate-type fuel, naval propulsion*

1. NOMENCLATURE

FA	Fuel Assembly	MNDBR	Minimum Departure from Nuclear Boiling Ratio
MTR	Material Test Reactor		
PWR	Pressure Water Reactor	CARR	China Advance Research Reactor
CHF	Critical Heat Flux	LABGENE	Laboratory for Nuclear Power Generation
DNBR	Departure from Nuclear Boiling Ratio	FMD	Flow Mass Distribution
HEM	Homogeneous Equilibrium Model		

2. INTRODUCTION

Different types of fuel geometry are used for fuel assemblies of nuclear reactors depending on the application, and plate fuels have been used in research, material test and naval propulsion reactors (Andrezejewski, 2005). When they are used in material test or research reactors, the fuel assemblies (FA) are placed in the core of pool-type reactors which operate at lower pressure and temperature conditions, less than 0.85 MPa and 150 °C, respectively (Umbehaun, 2000; Andrezejewski, 2005; Lei and Zhijian, 2010). When used in naval propulsion reactors, which are typical small PWRs, the FAs are placed in a pressurized vessel, as the core of pressurized water reactors (PWR) operating in pressure and temperature conditions higher than research reactors, 15 MPa and 300 °C, respectively. This geometry provides compact cores with high power density because of better coolant conditions, and greater resistance to external dynamic loads in comparison with fuel rod FAs (Thomas, 1990; Tong and Weisman, 1996; Andrezejewski, 2005). Good fuel thermal-hydraulic assembly design is associated with a good reactor performance regarding heat removal during operational, transient, and accident conditions. The thermal-hydraulic analysis allows determining what cooling conditions are appropriate to avoid that fuel limits are reached. For steady-state flow and power regime, the analysis can be performed using the sub-channel methods considering nominal and conservative condition, i.e., considering average and highest core heat generation distribution, respectively. In both analyses one must assure that no critical heat flux will occur in the core (Cuervo, 2001; Umbehaun, 2000; Andrezejewski, 2005).

In the context of PWRs, computer codes are used to study the thermal-hydraulic core behavior at different operating conditions. These computer codes allow for a detailed analysis of the mass, energy and momentum transfer equations in

the fuel assembly in different core conditions For example, the COBRA code is usually used for design purposes in steady-state conditions, and the THINC code, for the analysis of the operational and safety reactor conditions (Carajilescov and Bastos, 2000; Cuervo, 2007). In the first computer codes the coolant was not allowed to achieve the saturation temperature and thus no bulk phase change was considered (Tong and Weisman, 1996; Cuervo, 2007). In fact, steam generation could occur near the fuel clad where the local coolant enthalpy can be slightly greater than that of saturation but the bulk of the coolant could still be considered in the liquid phase and the whole system could be treated as a monophasic system (Cuervo, 2007). For PWR power reactors, with fuel rod assemblies, most of thermal-hydraulic codes employ open channel arrangements in which coolant traverse from one coolant sub-channel to another in its vicinity. Plate-type fuel assemblies have closed channels, which imply a simpler physical model since no mass transfer occur from one channel to another in its vicinity (Lei and Zhijian, 2010). Many researchers developed thermal-hydraulics codes for plate-type geometry mainly for research and material test reactors operating at low pressure and temperature conditions, for example, the COBRA-3C/RERTR, PARET, COOLOD-N and MERSAT (Kaminaga, 1990; Umbehaun, 2000; Hainoun et al., 2010). In this work, we develop a thermal-hydraulic analysis code for plate-type channels named "COTENP" that allow analyzing both research and small PWR power reactors used in naval propulsion.

3. DEVELOPMENT OF COTENP

3.1. Thermal-hydraulic Model

The thermal analysis process for nuclear power conversion involves the solution of mass, momentum and energy equations taking into account the system conditions (Lu and Qui, 2009). In the COTENP code we adopt the sub-channel method for two sub channel sizes (Tong and Weisman, 1996; Cuervo, 2007). The first is equivalent to a fuel assembly and the second equivalent to one channel that comprises it.

The sub-channel is divided into control volumes distributed axially along the length. The use of the sub-channel method, based on the integral transport equations, facilitates the calculation of pressure drop, heat transfer coefficient, coolant speed variation, and average temperature in the sub-channel. Figure 1 shows the variables pertaining to the three balance equations applied to the sub-channel analysis. The control volumes comprising the sub-channel allow the simultaneous solution of the conservation equations for mass, energy and momentum. The assumptions are steady-state flow and power regime, inlet and outlet pressures of all sub-channels are uniform, and no cross flow between the sub-channels.

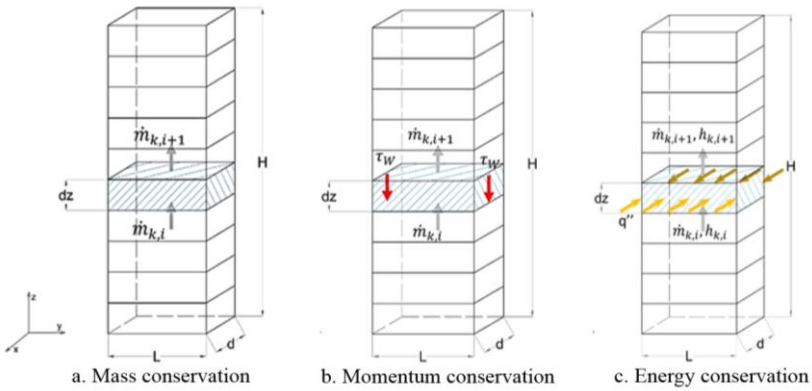


Figure 1. Variables from the balance equations in a control volume (node) of a sub-channel, where H is the length, L is the side length, and d is the distance between plates.

The mass balance equation in the k_{th} sub-channel and i_{th} control volume or node, as showed in Fig. (1a) and assuming steady-state and no cross flow between the sub-channels, is simply

$$\dot{m}_{k,i} - \dot{m}_{k,i+1} = 0\,, \tag{1}$$

where $\dot{m}_{k,i} = \int_A \rho_{k,i}\bar{u}_{k,i}dA = \rho_{k,i}\bar{u}_{k,i}A_K$, $\dot{m}$ is the mass flow rate (kg/s), ρ is the specific mass of the coolant (kg/m^3), u is the coolant speed, and A is the cross section area of the sub-channel (m^2).

The momentum balance equation due to the coolant flow in the control volume showed in Fig. (1b) considers friction pressure drop, form pressure drop, elevation pressure drop, and acceleration pressure drop and is given by for the k_{th} sub-channel and i_{th} node

$$\Delta P_{k,i} = f_{k,i}\frac{(z_{k,i+1}-z_{k,i})\rho_{k,i}(\bar{u}_{k,i})^2}{D_h} + k_{k,i}\frac{\rho_{k,i}\bar{u}_{k,i}^2}{2} + \rho_{k,i}g(z_{k,i+1}-z_{k,i}) + \frac{\dot{m}_k^2}{2\rho_{k,i}}\left(\frac{1}{A_{i+1}^2}-\frac{1}{A_i^2}\right), \quad (2)$$

where, ΔP is the total pressure drop in the volume control, f is the friction coefficient, z is the node elevation in the sub-channel, D_h is the hydraulic diameter of the sub-channel and, $k_{k,i}$ is the localized form loss coefficient. The friction coefficient is specified by the Hangen-Posiseulle correlation for laminar flow, Eq. (3), and by the Blasius correlation for turbulent flows, Eq. (4) (Incropera & Dewitt, 1999).

$$f_{k,i} = \frac{64}{\mathrm{Re}_{k,i}}, \quad (3)$$

$$f_{k,i} = 0{,}36\,\mathrm{Re}_{k,i}^{-0,25}. \quad (4)$$

The steady-state energy balance equation for the k_{th} sub-channel and i_{th} node takes into account the heat transferred from the fuel plate to the coolant, and is given by

$$\dot{m}_k(\bar{h}_{k,i+1}-\bar{h}_{k,i}) = q''_x p_a dz, \quad (5)$$

where $\bar{h}$ is the coolant enthalpy (J/kg), q'' is the heat flux across the surface area of the fuel plate (W/m^2), and P_a is the heated perimeter of the sub-channel (m).

The heat flux is obtained from reactor physics calculations, which yield the power density distribution in the whole reactor core. The power density distribution in different sub-channels can be represented by an axial distribution and a radial factor to account for their different radial position in the core (Todreas and Kazimi, 1990). The heat flux in the k_{th} sub-channel and i_{th} node is given by

$$q''_{x\_k,i} = q''_{\max} F_{z,i}^N F_{R,k}^N, \quad (6)$$

where, $F_{z,i}^N$ is the axial power distribution factor in the volume control; $F_{R,k}^N$ is the radial power distribution factor in the sub-channel.

The coolant temperature and the thermodynamic quality are given by Eqs. 7 and 8, respectively, for the k_{th} sub-channel and i_{th} node,

$$T_{k,i+1} = T_{k,i} + \frac{\Delta\bar{h}_{k,i}}{C_{k,i}}, \quad (7)$$

$$X_{T\_k,i} = \frac{\bar{h}_{vc}-\bar{h}_l}{\bar{h}_v-\bar{h}_l}, \quad (8)$$

where, T is the coolant temperature, $\Delta\bar{h}_{k,i} = \bar{h}_{k+1,i} - \bar{h}_{k,i}$ is the enthalpy gain in the k_{th} node, C is the heat capacity of the coolant, X_T is the thermodynamic quality, $\bar{h}_{vc}$, $\bar{h}_l$, $\bar{h}_v$ are the average node enthalpy and the corresponding liquid and vapor enthalpies with respect to the average node pressure and temperature.

The treatment for two-phase flow in nuclear systems has been an area of interest especially in thermal-hydraulic analysis. The main difficult appears in the description of the many different internal settings that must be taking into account for a more rigorous treatment of the problem (Todreas and Kazimi, 1990). The COTENP code uses the theory of Homogeneous Equilibrium Model (HEM) for the two-phase flow phenomena in the reactor core (Todreas and Kazimi, 1990). The thermodynamic quality is analyzed along the sub-channel length to determine the location where the two-phase flow starts and must be considered. The fictitious flow obeys the conservation equation presented above,

but it requires average properties which must be obtained from average two-phase flow parameters. The average coolant specific mass, average coolant viscosity, and average coolant speed are given by Eqs. 9 to 11, respectively,

$$\bar{\rho}_{k,i} = \frac{1}{\dfrac{X_{T\_k,i}}{(\bar{\rho}_{k,i})_v} + \dfrac{1 - X_{T\_k,i}}{(\bar{\rho}_{k,i})_l}}, \tag{9}$$

$$\mu_{k,i} = X_{T\_k,i}(\mu_{k,i})_v + (X_{T\_k,i} - 1)(\mu_{k,i})_l, \tag{10}$$

$$\bar{u}_{k,i} = \frac{\bar{u}_{k,i}\bar{\rho}_{k,i}A_{k,i}}{\bar{\rho}_{k,i}A_{k,i}}. \tag{11}$$

The main thermal-hydraulic limit imposed to the reactor core is the critical heat flux (CHF) which may cause the fuel rods to have localized burnouts and, therefore, leaks of radioactive fission products to the reactor coolant (Todreas and Kazimi, 1990). To avoid that to occur we monitor the heat transfer condition of nucleate boiling in the core, which occurs before the critical heat flux crisis. The safety heat transfer condition is given by the departure from nucleate boiling ratio (DNBR) defined as

$$DNBR = \frac{q''_{DNB}}{q''_{local}}, \tag{12}$$

where q''_{DNB} is the critical heat flux or the end of nucleate boiling heat transfer condition, and q''_{local} is the local heat flux. For safe operation, we demand that this ratio be greater than a prescribed safety margin, usually greater than 10 %, depending on how accurately both parameters in Eq. 12 are determined (Todreas and Kazimi, 1990; Tong and Weisman, 1996). The local heat flux is obtained from Eq. 6 and the critical heat flux is usually obtained from a correlation based on experiments that reproduce the reactor core operation conditions. We adopt EPRI correlation mainly because it fits well typical PWR conditions and is compatible with the thermal-hydraulic calculation scheme implemented in the COTENP code (EPRI, 1983; Tong and Weisman, 1996).

3.2. Computational Model

The computational method adopted to solve the equations presented in Sect. 3.1 considers radial symmetry and the chain or cascade method with two stages as shown in Fig. 2 (Chelemer et al., 1972; Tong and Weisman, 1996; Todreas and Kazimi, 2001). The chain or cascade method considers two stages: in the first stage, the core is divided into sub-channels with size equivalent to a fuel assembly. From this analysis, the sub-channel with the largest enthalpy is identified as the hot assembly. In the second stage, the hot fuel assembly is divided into sub-channels with size equivalent to an actual coolant channel. As in the previous stage, the sub-channel with largest final enthalpy is identified as the hottest sub-channel. For this sub-channel a detailed analysis including critical heat flux calculations and DNBR is undertaken.

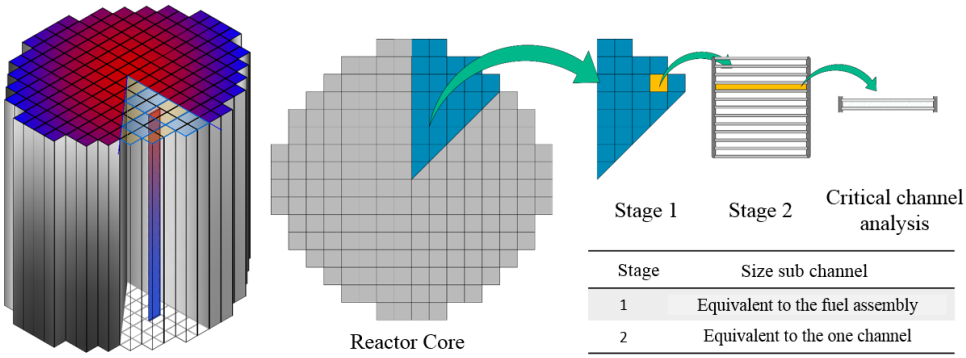
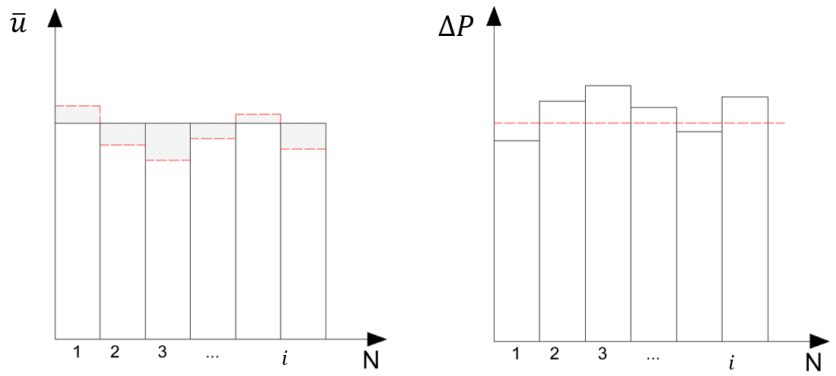


Figure 2. The chain method for two stages adopted in the COTENP code.

The code can analyze a symmetric region up to 1/8 of reactor core allowing judiciously selection of different sub-channels regarding the peak of their power density distribution. Thus, the approach allows studying a smaller number of sub-channels and reduces the computational effort to conduct complete reactor core thermal-hydraulic analyses.

3.2.1. Flow Adjustment in the Core

The accuracy of the thermal-hydraulic analysis depends on a good estimate for the flow distribution. Similar to the COBRA and FLOW codes (Umbehaun, 2000; Cuervo, 2007), the COTENP method considers steady-state flow in the axial direction and equal pressure drop across all sub-channels. The main assumption of this approach is to equate the pressure drop of all sub-channels and determine the coolant speed through the mass and momentum balance equations. The inlet coolant speed for each sub-channel is adjusted so that the pressure drops of all sub-channels are the same. Figure 3 shows the adjustment scheme for the coolant speed and pressure drop in which we start with similar speeds which yield different pressure drop for each sub-channel (grey lines) and arrive at the end with different coolant speed and the same pressure drop (red lines).



(a) (b)

Figure 3. Flow adjustment approach in the core for the COTENP code. (a) Speed adjustment, (b) Pressure adjustment. The gray lines depict the initial speed and pressure values prior to the adjustment, and the red lines depict the final adjusted results.

The flow rate adjustment is calculated through Eqs. 13 and 14 representing the speed and pressure drop relation between sub-channels. In this approach the speed change in the i_{th} sub-channel depends on the other sub-channel speeds. The initial coolant speed distribution is obtained from the total flow rate divided by the total number of sub-channels. The change in the coolant speed for the jth sub-channel is calculated from

$$\Delta\bar{u}_j = \frac{\bar{u}_{j,0}}{2}\left(\frac{\Delta P_{i,0}}{\Delta P_{j,0}} - 1\right) + \left(\frac{\Delta P_{i,0}}{\Delta P_{j,0}}\right)\left(\frac{\bar{u}_{j,0}}{\bar{u}_{i,0}}\right)\Delta\bar{u}_i \quad ; \quad \Delta P_{i,0} \neq \Delta P_{j,0}\,, \tag{13}$$

$$\Delta\bar{u}_i = -\frac{1}{\rho_i \alpha_i}\left(\sum_{\substack{j=1 \\ j\neq i}}^{N} \rho_i \alpha_i \Delta\bar{u}_j\right). \tag{14}$$

The iterative procedure for *N* different sub-channels is stopped when a convergence criterion is met,

$$\max_i\left(\frac{\Delta\bar{u}_i}{\bar{u}_i}\right) << \in \,, \tag{15}$$

where for the i_{th} sub-channel $\Delta\bar{u}_i$ is the coolant speed change, $\bar{u}_i$ is the coolant speed, $\bar{u}_{i,0}$ is the initial coolant speed; $\Delta P_{i,0}$ is the initial pressure drop, ρ_i is the specific mass of the coolant, α_i is the area relation A_i/A_T, A_i is the cross

section area of the i_{th} sub-channel, A_T is the total cross section area of all sub-channels, and ϵ is the convergence criterion.

3.2.2. General Solution Procedure

The COTENP code was develop using FORTRAN language enabling modular programming techniques and easy modification of implemented routines. Figure 4 shows the flow chart of the solution procedure. The code structure has one main routine and five operation subroutines.

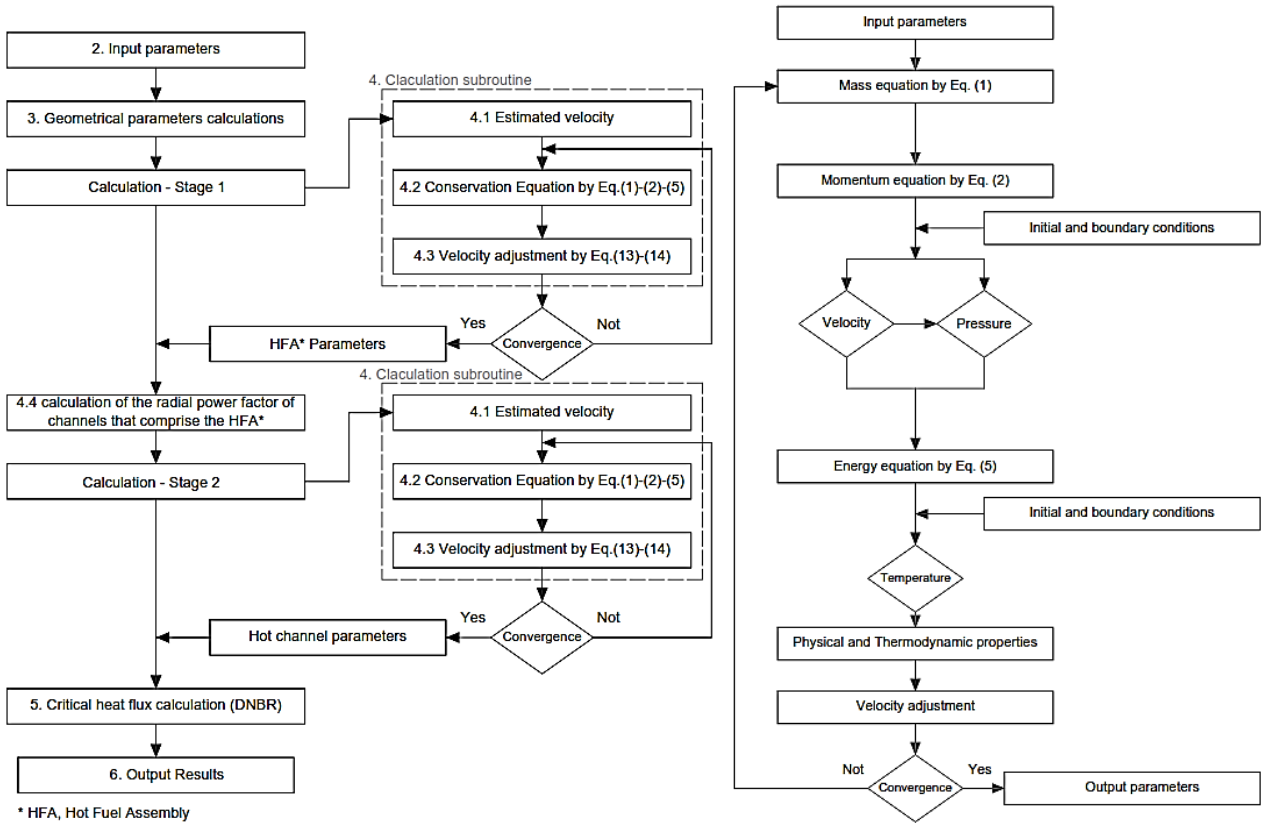


Figure 4. Flow chart for COTENP code. (a) Flow chart procedure of main subroutine, (b) Conservation equation solution.

The main program controls the five subroutines. The input data is done by subroutine 2 in two groups: general data (number of FA, types of FA, number of control volumes, geometrical data of FA), and operational data (radial and axial power factors, total flux, inlet temperature and inlet pressure in the core and power density in the core) are required to the code. Subroutine 3 holds the main geometry data calculation, and subroutine 4, through other three subroutines, holds the calculations to estimate coolant speed, for the conservation equations and for the coolant speed adjustment. Subroutine PROPAGUA, a homemade code (Carajlescov, 2014) which furnishes the state properties of water and vapor used throughout the code.

The code conducts the first and second stage calculations according to the size of the sub-channels under study. Before starting the second stage, the code estimates the radial power factor for each channel that comprises the hottest fuel assembly. The solution is carried out using the Thomas algorithm (Chapra and Canale, 2006).

Subroutine 5 holds the CHF calculations for the hottest channel. Subroutine 6 outputs the calculated results, namely mass flow rates, coolant speed distribution, temperature distribution, enthalpy rise and thermodynamic quality along each sub-channel for the two stages. The core pressure drop and the location of the hottest channel with regard to the CHF are also calculated.

4. SIMULATION RESULTS AND DISCUSSIONS

To verify accuracy of the COTENP code we selected reported results from two small power reactors namely the China Advanced Research Reactor (CARR) (Lei & Zhijian, 2010), and the Laboratory for Nuclear Power Generation (LABGENE) (Andrezejewski, 2005). The COTENP results for these two reactors were compared with the thermal-hydraulic parameters calculated by Lei and Zhijian (2010) and Andrezejewski (2005).

4.1. Analysis of research reactor CARR

The CARR is a multipurpose research reactor for providing high neutron flux for different applications. The reactor core contains 21 plate-type fuel assemblies with rectangular coolant channels. There are two types of fuel assemblies: 17 standard assemblies and 4 open assemblies as described in Table 1 and depicted Fig. 5a. Table 1 also presents the main thermal-hydraulic parameters and geometric data for the CARR reactor.

Table 1. Main thermal-hydraulic parameters and geometric data of CARR.

Parameter	Value	Unit
Thermal power reactor	56.4	*MW*
Flow rate	2400	m^3/h
Inlet pressure	0.62	*MPa*
Inlet temperature	318	*K*
Average power density in the core	568	W/cm^3
Height of the active area	0.85	*m*

Parameter	Geometrical data Standard FA*	Open FA*	Units
Number plates	20	2	
Plate length	0.85	0.85	*m*
Plate width	0.0772	0.073	*m*
Plate thickness	0.00289	0.00289	*m*
Distance between plates	0.00289	0.073	*m*

*FA, Fuel assembly

Figure 5 shows the details of the thermal-hydraulic data utilized to analyze the CARR reactor. Fig. 5a shows the arrangement of the fuel assemblies in the reactor core and the radial power factor for each fuel assembly. Fig. 5b presents the axial power distribution for both types of fuel assemblies. Since the reactor is small, the thermal-hydraulic analysis carried out with the COTENP code encompasses the whole core. We considered 17 control volumes in axial direction, i.e, 17 axial nodes in each sub-channel analysis.

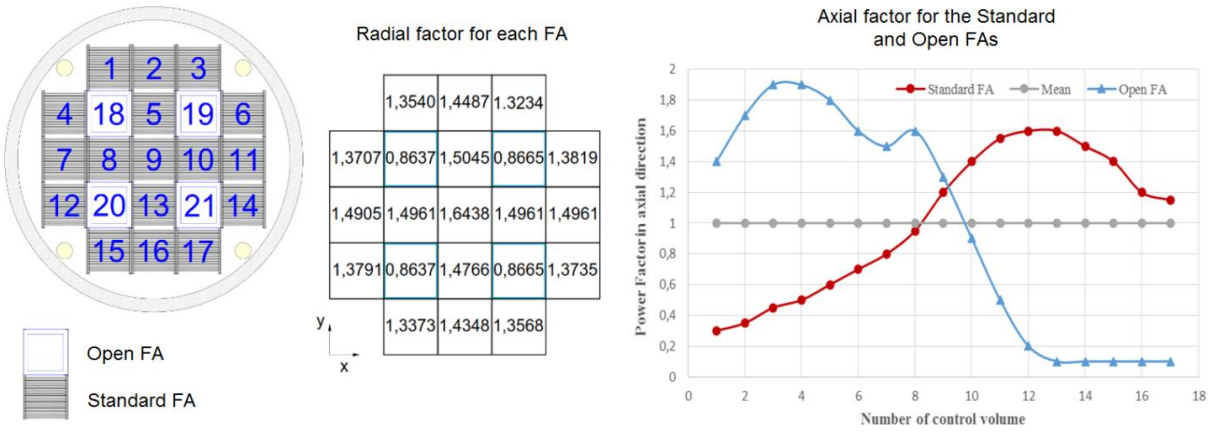


Figure 5. Thermal-hydraulic data utilized for the CARR reactor. a. Identification of the fuel assemblies in the core (1 to 17 are the Standard FAs and 18 to 21 are the Open FAs). (a)Location and radial power factors of FA. (b) Axial power factor of standard and open FA.

The results for the steady-state conditions for the CARR reactor are shown in Figures 6 to 10. In the first stage of calculation, the code estimated the mass flow distribution among the several sub-channels. Fig. 6 shows the comparison between COTENP and the results from Lei and Zhijian (2010) for the mass flow rate distribution. We observe that the two curves have a similar behavior and coincident results for fuel assemblies 18, 19, 20 and 21 (Open FAs). Note that the axial power distribution for the Open FA is very different from that of the Standard FA. This fact may have influenced the mass flow rate results for these FAs, but the differences are very small. Both calculations present channel 9 as the one with largest mass flow rate. As also shown in Fig. 6, the maximum discrepancy for channel mass flow rate between the COTENP code and Lei and Zhang (2010) occurs in channel 9 and it is 0.74 %.

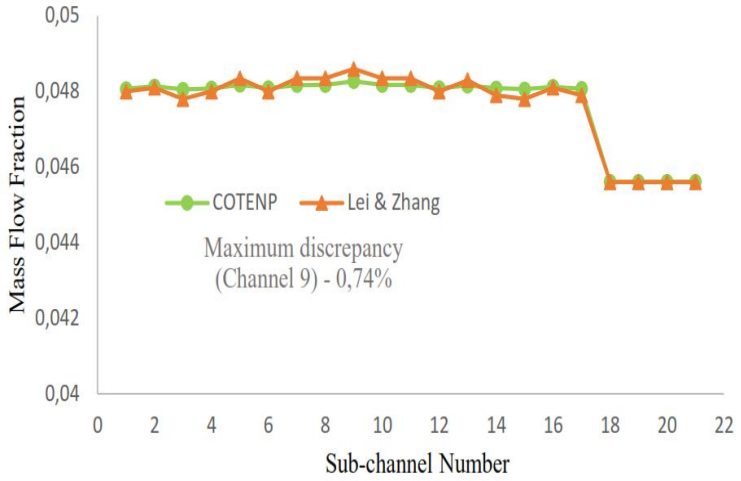


Figure 6. Mass flow rate distribution for the CARR reactor.

The hottest FA is identified through the enthalpy rise. Fig. 7 shows that the largest enthalpy rise occurs in the FA number 9. For the second stage, 19 different sub-channels were modeled as shown in Fig. 8. The hottest sub-channel is the one we have numbered as 19. Figure 9 shows the coolant temperature axial distribution in the hottest sub-channel. It varies along the sub-channel from 318 K to 345 K and is always being below the corresponding saturation temperature. Figure 10 shows the critical heat flux analysis in the hottest sub-channel. The results present similar trend with the COTENP code presenting lower DNBR for colder sub-channels but with similar results for the hotter ones. Lei and Zhijian present minimum DNBR of 11 while the COTENP code presents a minimum DNBR of 13. We should note that Lei and Zhijian used the W3 correlation in their analysis while se have used the EPRI correlation.

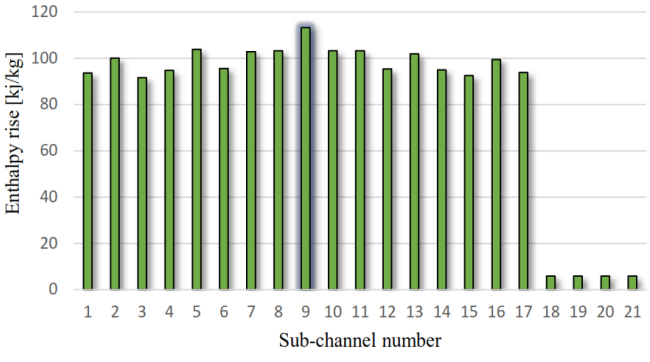


Figure 7. Enthalpy rise for the first stage (fuel elements).

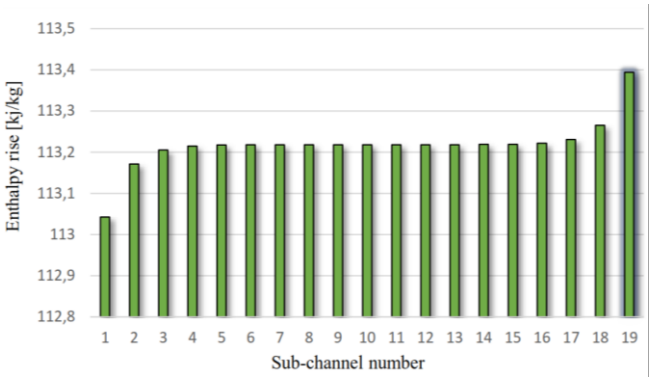


Figure 8.Enthalpy rise for the second stage (sub-channels from FA 9).

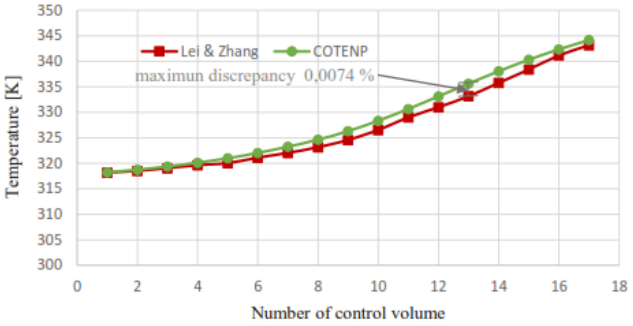


Figure 9. Coolant temperature in the hottest sub-channel.

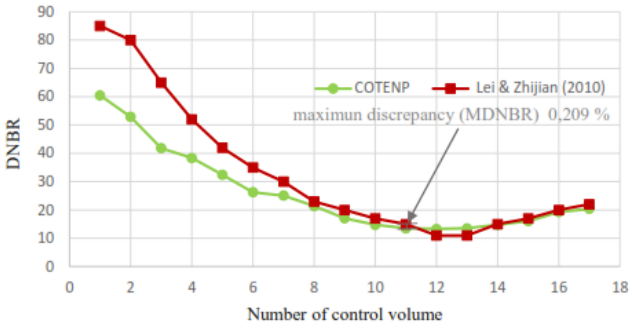


Figure 10. DNBR in the hottest sub-channel.

4.2. Analysis for the Laboratory for Nuclear Power Generation (LABGENE)

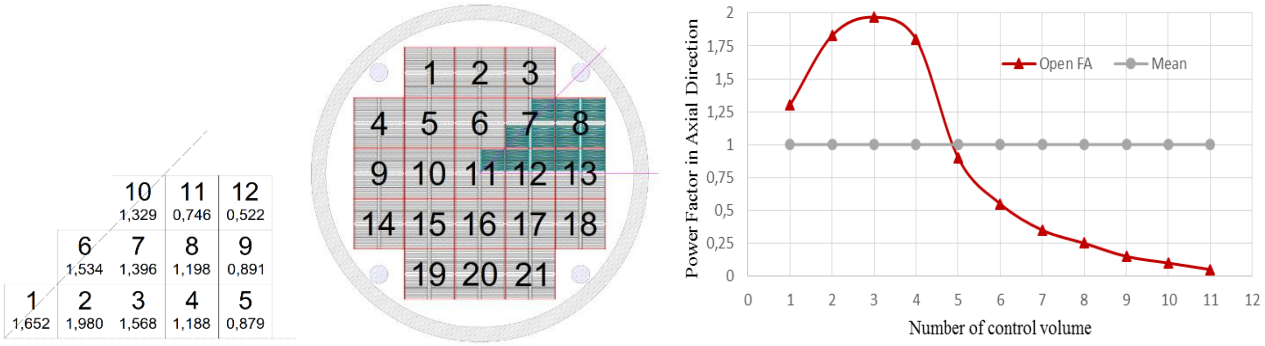
The LABGENE is a laboratory for nuclear power generation aiming at small reactor technology and the naval propulsion developed by Brazilian Navy, which will be used to validate the design conditions and test the possible operating conditions in a naval propulsion plant (CTMSP, 2015). This prototype is planned to operate in typical PWR operation conditions, i.e., high pressure and temperature. Table 2 shows the main thermal-hydraulic parameters and geometric data for the LABGENE reactor.

Table 2. Main thermal-hydraulic parameters and geometric data for the LABGENE reactor.

Operational data			Geometrical data		
Parameter	Value	Unit	Parameter	Standard FA*	Unit
Thermal power reactor	58	*MW*	Number plates	18	
Flow rate	240	m^3/h	Plate length	1.105	*m*
Inlet pressure	13	*MPa*	Plate width	0.09435	*m*
Inlet temperature	518	*K*	Plate thickness	0.00243	*m*
Power density in the core	150	W/cm^3	Distance between plates	0.00300	*m*
Height of the active area	1.105	*m*			

*FA, Fuel assembly

The reactor contains 21 plate-type cruciform FA with 4 fuel sub-assemblies with 18 fuel plates. The total number of fuel plates in each FA is 72. Figure 11 shows a schematic of the LABGENE reactor core. Making use of radial symmetry, we have considered in the calculations only 1/8 of the core cross section as shown in Fig. 11, i.e., fuel assemblies 7, 8, 11, 12 and 13. Figure 11a shows the radial power factors for the subassemblies, which was adapted from (Tong and Weisman, 1996) for a typical PWR reactor. We divided the axial channel length into 11 control volumes. Figure 11b shows the axial power density distribution for the average fuel assembly (Andrezejewski, 2005).



(a) (b)

Figure 11. Thermal-hydraulic data utilized for the LABGENE reactor. a. Identification of the fuel assemblies in the core. (b) Axial power density distribution for the average fuel assembly.

For the LABGENE analysis, the results of steady-state are shown in Figs.12 to 16. Figure 12 shows mass flow distribution among the several sub-assemblies defined in Fig. 11a. We observe that the sub-assembly with larger radial power factor presents the largest mass flow rate fraction (number 2).

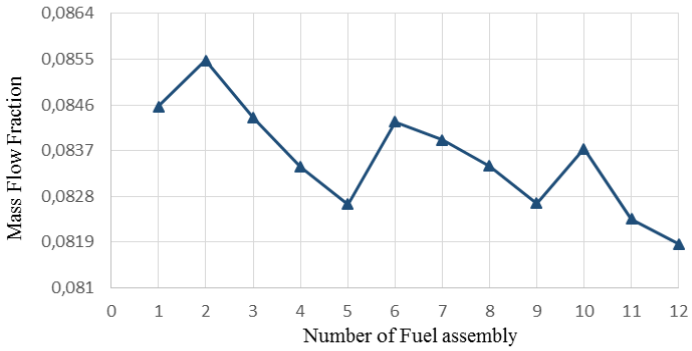


Figure 12. Mass flow rate mass rate distribution for the LABGENE core.

Figure 12 and 13 show enthalpy rise for the first and second stage calculations. The hottest sub-channel is located in the fuel sub-assembly number 2 located inside the fuel assembly the number 12. The hottest sub-channel is identified in Fig. 14 as the number 1. Fig. 15 shows the coolant temperature axial distributions for the hottest sub-channel calculated with the COTENP code and provided by Andrezejewski (2005). The temperature is below the saturation temperature for the corresponding temperature and pressure in the sub-channel (603.9 K to 130 bar). The temperature discrepancy between the two calculations is large. Fig. 16 shows the DNBR along the axial direction for the hottest sub-channel. The critical heat flux correlation used in the COTENP code is the EPRI correlation while in the Andrezejewski (2005) calculation was the one from Jens and Lottes. Despite the important temperature discrepancies and different CHF correlation, the minimum DNBR from both calculations were quite close, 2.79 from Andrezejewski (2005) and 2.56 from the COTENP code.

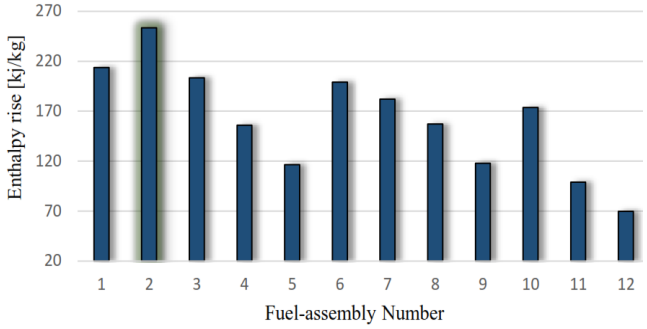


Figure 13. Enthalpy rise for the first stage (fuel sub-assemblies)

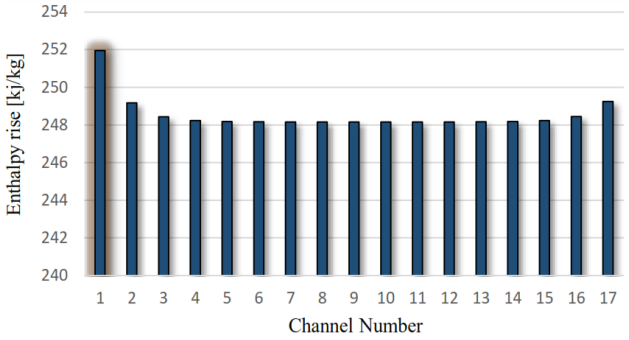


Figure 14. Enthalpy rise for the second stage (sub-channels).

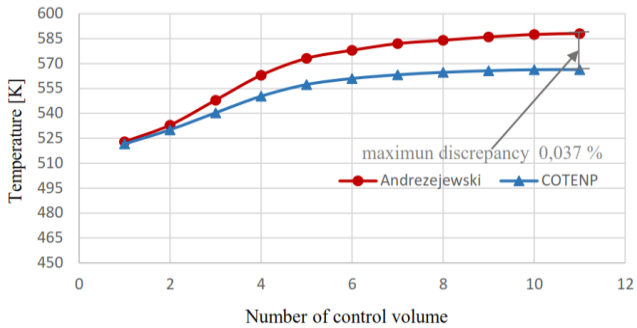


Figure 15. Coolant temperature in the hottest sub-channel.

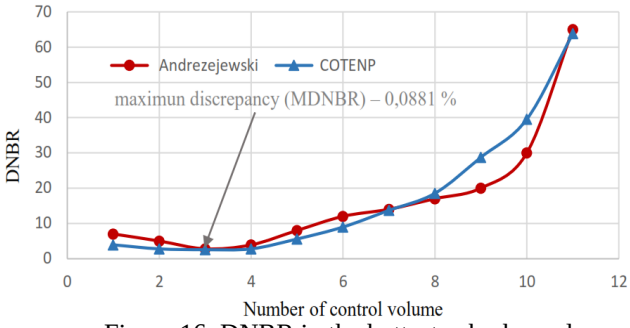


Figure 16. DNBR in the hottest sub-channel.

The thermal-hydraulic analysis from the CONTENP code produced information regarding the two-phase flow regime. Although the analysis shows that the coolant temperature was always below the corresponding saturation, the hottest channel achieved the two-phase state in its final stretch of channel with the thermodynamic quality equal to $0{,}35 \times 10^{-2}$.

5. CONCLUSIONS

The physical and mathematical models adopted in this research work allowed the development of the COTENP code, a thermal-hydraulic analysis code for plate-type fuel assembly nuclear reactors. The COTENP code can be considered a tool for analyzing the thermal-hydraulic limits for the average and hot channels of nuclear reactors in steady-state conditions. The sub-channel and the chain methods showed suitable to represent the coolant behavior. The model allows representation of radial and axial direction through normalized power density distributions. The COTENP code produces information for the hottest channels such as coolant temperature axial distribution, pressure drop in the core and critical heat flux analysis. The critical heat flux is verified using the EPRI correlation. The COTENP code results were compared with similar results provide in the literature for the CARR and LABGENE. The COTENP code reproduced quite well the CARR results but presented very different temperature distribution in the hot channel when compared with the LABGENE results. More comparisons should be carried out in order to obtain a better estimation of uncertainties of the models employed in the COTENP code, especially to analyze plate-type fuels in small nuclear reactors.

The CONTENP code reproduced well the CARR reactor results, but presented important discrepancy regarding the temperature axial distribution for the LABGENE reactor results. The DNBR estimation for both problems were accurate.

6. RESPONSIBILITY NOTICE

The authors are the only responsible for the printed material included in this paper.

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