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Abstract – *The thermal-hydraulic nuclear reactor core channel analysis is done thanks to the conservation equations of mass, momentum and energy, for an incompressible fluid. The set of equation is solved numerically using the finite volume method. This approach is applied for a 02 MW and for a 10 MW nuclear research reactor. The temperature profiles of the coolant and the clad along the channel are plotted. For the case of 02 MW and for an upward flow, the obtained results were compared to those given by Boudali and Salhi and to those given by the code TERMIC. For the case of 10 MW and a downward flow, the obtained results were compared to the results given by Lu et al. and also to those given by the code TERMIC. The obtained results are very close to those obtained by the cited authors and the calculated relative differences are minor. The results obtained thanks to this method, are more conservative than the results given by the presented comparative studies.*

Keywords: *Thermal-hydraulic, Nuclear Reactor Core, Nuclear Channel, Clad, Coolant*

I. Introduction

The thermal hydraulic analysis of a nuclear research reactor channel allows the prediction of the behavior of a set of physical quantities. These physical quantities are directly linked to the optimal margins required by nuclear safety in research nuclear reactors or nuclear power plants. Most of the established mathematical models, allow the thermal hydraulic study of a nuclear research reactor channel, in normal or accidental situation. One can cite several recent works established in this area, among them the work done by Housiadas (2000) [1] who used the code PARET to treat flow instabilities during loss of coolant transient in an MTR (Material Testing Reactor) type reactor. The case where the residual heat is removed in an MTR type reactor was studied by Kazeminejad (2006) [2]. He studied the free convection heat transfer numerically after a reactivity insertion accident. A Reactivity-Initiated Accident in an MTR type reactor was also studied by Khater (2007) [3]. The author developed a dynamic model to perform a thermal hydraulic analysis of the accident. The formulated model coupled the kinetic parameters with the thermal hydraulic ones. The model was validated using data from the generic 10 MW nuclear reactor of the TECDOC-643 (1980) [4], and then applied to the reactor ETRR-2. Another approach was applied by Kazeminejad (2008) [5]. The lumped parameters approach was used for coupling thermal hydraulic and neutron parameters. A numerical treatment of the flow reversal was applied in an MTR nuclear research reactor type of 10 MW during a loss of flow accident and under free convection conditions. The analysis of two types of accidents RIA and LOFA (Loss Of Flow Accident) with a coupled thermal and neutron EUREKA-2/RR code was done by Badrun et al. (2012) [6]. An application was made for a TRIGA 03 MW reactor. The results show that this code is able to be used for the analysis of these two kinds of accidents. Another work dealing with the analysis of thermal hydraulic parameters and local hot spot during the flow reversal was established by Al-Yahia et al. (2013) [7]. In this study, a coupling model was developed and the obtained results were compared to published works. In the case of steady state, a thermal hydraulic analysis was done for the nuclear reactor PARR-1 using the code RELAP (best estimate system code). This study, performed by Bokhari et al. (2007) [8], was dedicated to the analysis of the Onset of Flow Instabilities (OFI) and the critical heat flux for the Departure from Nucleate Boiling (DNB). Thus, a set of physical quantities has been calculated in the nuclear reactor core. In the same context, Salama et al. (2011) [9] used the CFD code FLUENT to perform a 3D simulation of a LOFA for a generic 10 MW nuclear reactor. Daeseong et al. (2012) [10] made a numerical treatment of the cooling capacities, by free convection, of an MTR nuclear research reactor. The work dealt with the determination of the optimal temperature margin compared to the conditions of Onset of nucleate Boiling (ONB). A thermal hydraulic analysis code THAC-PRR dedicated to MTR nuclear research reactors was developed by Lu et al. (2009) [11]. This thermal hydraulic code is based on the fundamental equations of conservation. The finite difference method was used to solve the obtained system of differential equations. This study will be used as a reference to validate the work done in this paper. Among other numerical methods, we note that used by Barati (2013) [12]. He presented a third order numerical model to simulate the kinetic and the dynamic behavior of the nuclear research reactor core. Computational Fluid Dynamices (CFD) have been also used by Salama et al. (2012) [13] to perform 2D simulations of partial and total flow blockage in the generic 10 MW nuclear reactor in TECDOC-643 (1980) [4].One notes also the proposals for improving the numerical methods used during the coupling of the parameters in nuclear research reactors proposed by Ragusa et al. (2009) [14]. Finally, one cites the work done by Sidi-Ali et al. (2012) [15], who studied the effect of the critical velocity on the thermal-hydraulic behavior of a nuclear research reactor core channel. The study was carried out for three critical velocities: that of Miller (1960) [16], that of Wambsganss (1967) [17] and that of Cekirge and Ural (1978) [18].

In the present work, the main objective is to calculate, at steady state, two physical quantities which are the temperature of the coolant and the temperature of the clad along a nuclear research reactor core channel by solving a set of equations describing the thermal system.

II. Problem analysis

Let us consider a channel of a nuclear research reactor core as presented in figure (1). The channel, in the plane (x,z), is defined as the area between two fuel plates. The cross section is constant. The flow is upward in the z direction with a mass flow rate G. The coolant, in contact with the clad, is heated by convection and the maximum power density in the channel i is given by:

$$q_c^{'}(i) = F(i)\, q_a^{'} \tag{1}$$

where $q_a^{'}$ is the average power density base which is the base power divided by the volume of fuel and $F(i)$ is the factor of nuclear power given by $F(i) = F_r\, F_a$ where F_r and F_a are respectively the radial and the axial factor of power.

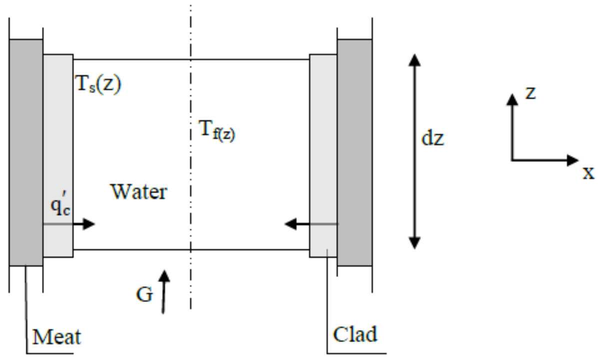

Figure 1: Channel configuration

III. Mathematical model

The mathematical model describing the thermal hydraulics in the core channel was used by Lu et al (2009) [11], Kazeminedjad (2012) [19] and by Zhao et al (2013) [20]. The model is based on the three equations of conservation presented in (2-4). The assumptions made are as follows: the study is conducted at steady state; the flow is incompressible; turbulent and fully developed (thermally and hydrodynamically). The channel is smooth and has a hydraulic diameter D_h.

The conservation equations of mass, momentum and energy, for an incompressible fluid, are written according to Zhao et al. (2013) [20]:

$$\frac{1}{A}\frac{dG}{dz} = 0 \tag{2}$$

$$\frac{d}{dz}\left(\frac{G^2}{\rho_f\, A}\right) = -A\frac{dP}{dz} - \rho_f g A - \int \tau_f dl \tag{3}$$

$$\frac{dh}{dz}\left(\frac{G}{A}\right) = \frac{q^{'}_c \pi_h}{A} \tag{4}$$

where A is the flow cross section, G the coolant mass flow rate, h the enthalpy, π_h the heated perimeter, ρ_f the coolant density, g the gravitational acceleration and dl is the element length.

In equation (3) $\frac{dP}{dz}$ is the axial pressure gradient and the last term of the same equation is evaluated by:

$$\int \tau_f dl = f\ \frac{1}{D_h}\frac{\rho_f V^2}{2}\frac{1}{A} = \frac{f G^2}{2\,\rho_f D_h A} \tag{5}$$

where f is the friction factor, D_h the hydraulic diameter and V the velocity of the cooling fluid.

The equation of momentum becomes, after injection of (5) in (3) and dividing by A,

$$\frac{d}{dz}\left(\frac{G^2}{\rho_f\, A^2}\right) = -\frac{dP}{dz} - \frac{f\,G^2}{2\,\rho_f\, D_h A^2} \mp \rho_f g \qquad (6)$$

In the nuclear reactor core channel, for upward flow one writes $-\rho_f g$ and for downward flow + $\rho_f g$ in equation (6).

Because of the non linearity of the momentum equation (6), the set of equations (2, 6 and 4) will be solved using the finite volume numerical method. Equation (2) is integrated over the control volume VC:$\int_{VC} \frac{1}{A}\frac{dG}{dz} dV_c = 0$. One obtains according to Patankar (1980) [21] the value of G at the east and west faces of the control volume (Figure 2-a):

$$G_e - G_w = 0 \qquad (7)$$

The values of G at the faces are the average values at the two nodal points east and west: $G_e = (G_E + G_P)/2\ et\ G_w = (G_P + G_W)/2$. Substituting these equations into equation (7), one obtains:

$$G_E = G_W \qquad (8)$$

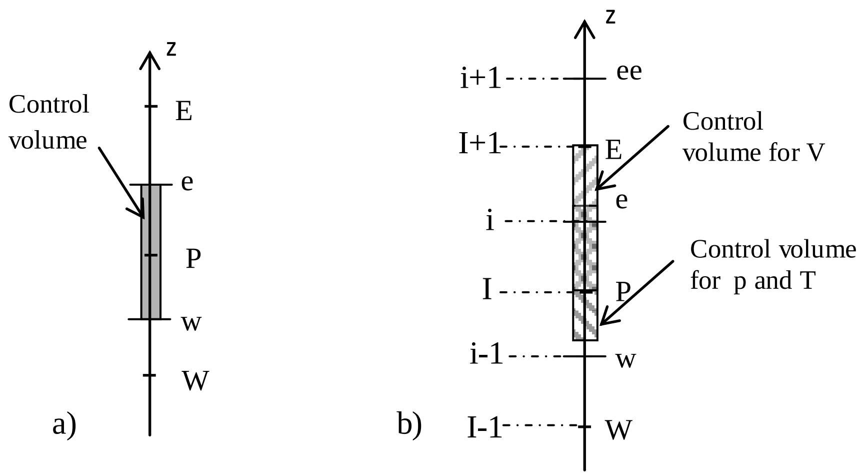

Figure (2): a- Control volume 1D, b- Control volume for velocity component.

The conservation equation of momentum for the G component, on the control volume for node 'e', shifted forward, shown in Figure (2-b) is written as:

$$\int_{VC} \frac{d}{dz}\left(\frac{G^2}{\rho_f\, A^2}\right) dV_c = -\int_{VC} \frac{dP}{dz} - \int_{VC} \frac{f\,G^2}{2\,\rho_f D_h A^2}\, dV_c - \int_{VC} \rho_f\ g\ dV_c \qquad (9)$$

Integrating equation (9) on the control volume gives:

$$\left(\frac{G\,G}{\rho_f\ A}\right)_P^E = -P_P^E - \left(\frac{f\ G\ G}{2\,\rho_f D_h A}\right)_P^E \Delta z - \rho_f\ g\ A\ \Delta z \qquad (10)$$

Developing (10), one gets:

$$\left(\frac{G}{\rho_f\ A}\right)_E G_E - \left(\frac{G}{\rho_f\ A}\right)_P G_P = (P_P - P_E)A - \rho_f\ g\ A\ \Delta z + \left(\frac{f\ G}{2\,\rho_f D_h A}\Delta z\right)_P G_P - \left(\frac{f\ G}{2\,\rho_f D_h A}\Delta z\right)_E G_E \qquad (11)$$

To evaluate $G_E$and G_P a centered differentiation scheme between neighboring nodes is used, one obtains: $G_E = \frac{G_e+G_{ee}}{2}$ and, $G_P = \frac{G_w+G_e}{2}$. As $G_E = V_E \rho_f A$, with V_E the fluid velocity in the channel, one can write G_E and G_E according to the velocities of V_E and V_P as follows $G_E = \rho_f A\left(\frac{V_e+V_{ee}}{2}\right)$ and $G_P = \rho_f A\left(\frac{V_w+V_e}{2}\right)$ and the velocities are $V_E = \frac{V_e+V_{ee}}{2}$ and $V_P = \frac{V_w+V_e}{2}$.

The discretized equation, in a general form, for the velocity components at the faces of the control volume is given by:

$$a_e V_e = \sum a_{nb} V_{nb} + b + (P_P - P_E)A \qquad (12)$$

Where P_p is the pressure at the center of the control volume and P_E is the pressure value at the east node.

Expressing equation (12) according to the new coordinate system numbering nodes, as shown in Figure (2-b), one gets:

$$a_i V_i = \sum a_{nb} V_{nb} + b + (P_I - P_{I+1})A \qquad (13)$$

In the new numbering system, the neighboring points involved in the sum $\sum a_{nb}V_{nb}$, shown in Figure (2-b) are the nodes (i-1) and (i +1).

The integration of the conservation equation of energy on the control volume is written:

$$\int_{VC} \frac{dH}{dz}\left(\frac{G}{A}\right) dV_c = \int_{VC} \frac{q^{'}U_h}{A} dV_c \qquad (14)$$

Integrating (14) gives:

$$[GH]_w^e = q^{'}U_h\Delta z \qquad (15)$$

Where H is the enthalpy of the fluid and is given by $H = Cp_f T_f$, Cp_f is the heat capacity at constant pressure of the coolant, T_f is the temperature of the coolant and is assumed to be constant.

Using centered differentiation scheme for east and west faces, one obtains:

$$G\left(\frac{T_{f\,E} + T_{f\,P}}{2}\right) - G\left(\frac{T_{fP} + T_{fW}}{2}\right) = \frac{q^{'}\pi_h}{Cp_f}\Delta z \qquad (16)$$

Finally, by grouping terms, one obtains the discretized equation for temperature:

$$T_{fE} = \left(a_W T_{fW} + b\right)/a_E \qquad (17)$$

If equation (17) is expressed according to the new coordinate system numbering nodes as shown in Figure (2-b), one obtains:

$$T_{f\,I+1} = \left(a_{I-1}T_{f\,I-1} + b\right)/a_{I+1} \qquad (18)$$

where:

$$b = \frac{q^{'}\pi_h}{C_{pf}}\Delta z\,, \quad a_E = a_W = \frac{G}{2}$$

Equation (12) can be solved if the pressure p is known or estimated. If the correct pressure is known, the velocity field obtained after solving the linear algebraic system will satisfy the continuity equation. The SIMPLE scheme is used for pressure evaluation.

Once the temperature of the coolant known, one can now evaluate the temperature in the clad T_c. The equation given by El wakil (1971) [22] is used:

$$T_c(z) = T_f(z) + \frac{q_c^{'}t_m}{2h} + cos\frac{\pi z}{l_e} \qquad (19)$$

Where $t_m$is the meat thickness, l_e the extrapolated length, and h the convection heat transfer coefficient given by $h = \frac{Nu_{Dh}\lambda}{D_h}$ where λ is the thermal conductivity of the fluid, D_h the hydraulic diameter and Nu_{Dh} the Nusselt number calculated thanks to the Colburn expression which is used for the turbulent and fully developed flow and given by:

$$N_u = 0.023\,{Re_{Dh}}^{0.8}Pr^{0.33} \qquad (20)$$

The Reynolds number is given by $Re_{Dh} = \frac{V\,D_h}{\nu_f}$ where ν_f is the kinematic viscosity of the fluid. The Prandtl number appearing in equation (20) is given by $Pr = \frac{\mu_f\,Cp_f}{\lambda}$ where μ_f is the dynamic viscosity of the fluid.

The boundary conditions of the system are as follows: the temperature of the coolant at the entry of the channel and the flow rate are those given in table 1. For pressure, a difference of 1 bar between the inlet and outlet pressure (given in table 1) is taken to ensure the fluid flow in the channel.

IV. Nuclear reactors data and validation

To verify the method developed to solve the set of equations describing the thermal hydraulic behavior in the studied channel, one chooses to do it for two different nuclear research reactors using water as coolant and plate type fuel. The first nuclear research reactor is about 02 MW referenced by IAEA as TECDOC-233 (1980)[23] and the second of about 10 MW and also referenced by AIEA as TECDOC-643 (1980)[4].

The properties of the two nuclear research reactors are given in table (1):

Table 1: MTR type reactors Data, TECDOC-233 (1980)[24] and TECDOC-643 (1980)[4]

Properties	**2 MW**	**10 MW**	**Properties**	**2 MW**	**10 MW**
Standard fuel elements number	19	23	Standart fuel elements plates number	19	23
Control fuel elements number	4	5	Control fuel elements plates number	15	17
Channel width (cm)	6.64	6.65	Plate total lenght (cm)	62.5	60
Fuel width	6.3	5.1	Channel thickness (cm)	0.2916	0.223
Heating length (cm)	60	60	Clad thickness (cm)	0.0381	0.038
Inlet temperature (°C)	38	38	Fuel thickness (cm)	0.051	
Outlet pressure (bar)	1.961		Radial factor of power	2	1.4
Clad thermal Conductivity (w/cm.K)	0.180	0.180	Axial factor of power	1.58	1.5
Fuel Thermal Conductivity (w/cm.K)	0.536	0.5	Total flow rate (m^3/h)	300	1000
			Flow velocity in the channel (m/s)	0.94	2.97

The results obtained from the calculations in the present work will be compared to some works done and to a thermal hydraulic code. The works are those of Lu et al. (2009) [11] and Boudali and Salhi (2011)[24]. The first authors, Lu et al.(2009)[11], developed a thermal-hydraulic analysis code for research reactors using plate-type fuel. The analysis code is based on the fundamental laws of conservation and appropriate constitutive correlations. A simple and improved lumped-differential method has been adopted to analyze the conjugate heat transfer between the fuel plate and the coolant. An application for a nuclear research reactor of 10 MW was done. The results obtained for the temperature of the clad and the temperature of the coolant at steady condition are used as a reference for the validation of this work for the case 10 MW. The second authors, Boudali and Salhi (2011)[24], developed a thermal hydraulic study of a nuclear research reactor core channel using El Wakil equation set-up for the thermal part of their work. They study this thermal hydraulic behavior for three critical velocities. They determine the profiles of the temperature in the fuel meat, in the clad and in the coolant for the 02 MW nuclear research reactor given in TECDOC-233(1980)[23], These profiles will be, also, a reference for the validation of this work for the case of 02 MW. The thermal hydraulic code involved in this work is TERMIC (1995) [25], which is a program that can perform thermal-hydraulic calculations of nuclear reactor cores in pressure and temperature ranges typical of MTR type reactors. TERMIC (1995) [25] is intended to calculate fuel and coolant temperature along the fuel channels, critical heat flux (CHF), departure from nucleate boiling ratio (DNBR) and also maximum allowable powers and heat fluxes using selectable limiting criteria of onset of nucleate boiling (ONB), CHF and onset of flow instability (OFI) as a function of the coolant velocity. This code will be run for the two cases of 02 MW and 10 MW and the obtained results will be also compared to this work.

V. Obtained results

The first application deals with a core channel of a nuclear research reactor of 02 MW, the flow is upward and the coolant temperature at the entry of the channel is taken equal to 38 °C. The obtained profiles of temperature for the coolant and the clad are presented in figure (3).

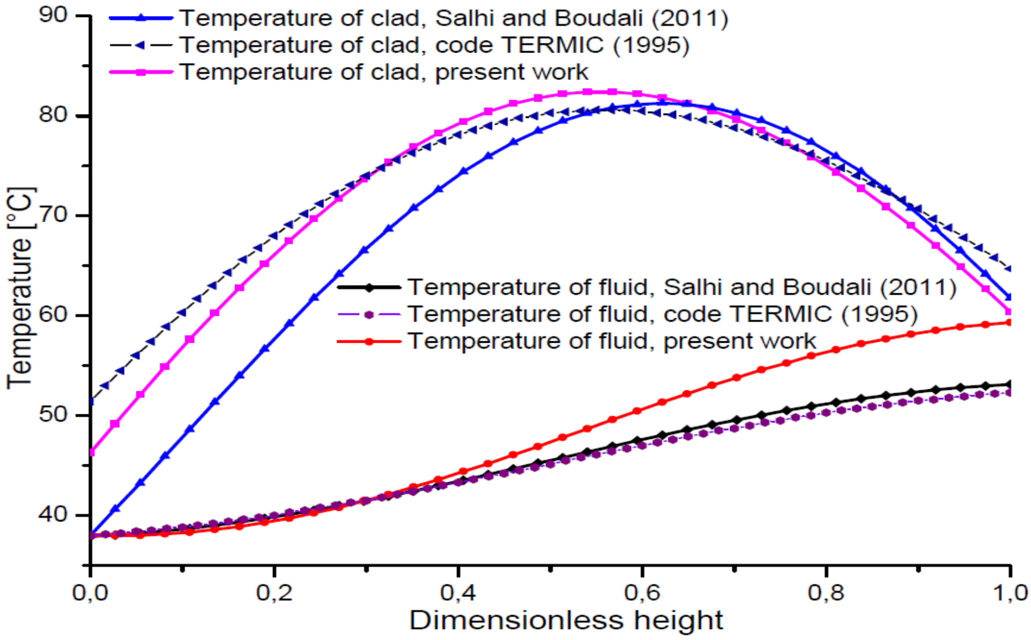

Figure 3: Temperature profiles of the coolant and the clad in the 02 MW nuclear research reactor core channel

In figure (3) are presented six profiles. In the upper part, the temperature profiles of the clad and in the lower part the temperature profiles of the coolant. The profiles obtained by the present work are compared to those of Boudali and Salhi (2011)[25] and to those given by the code TERMIC (1995) [25].

The second application deals with a core channel of a nuclear research reactor of 10 MW, the flow is downward and the coolant temperature at the entry of the channel is taken equal to 38 °C. The obtained profiles of temperature for the coolant and the clad are presented in figure (4). In this figure, are presented six profiles. In the upper part, the temperature profiles of the clad and in the lower part the temperature profiles of the coolant. The profiles obtained by the present work are compared to those of Lu et al. (2009)[11] and to those given by the code TERMIC (1995) [25].

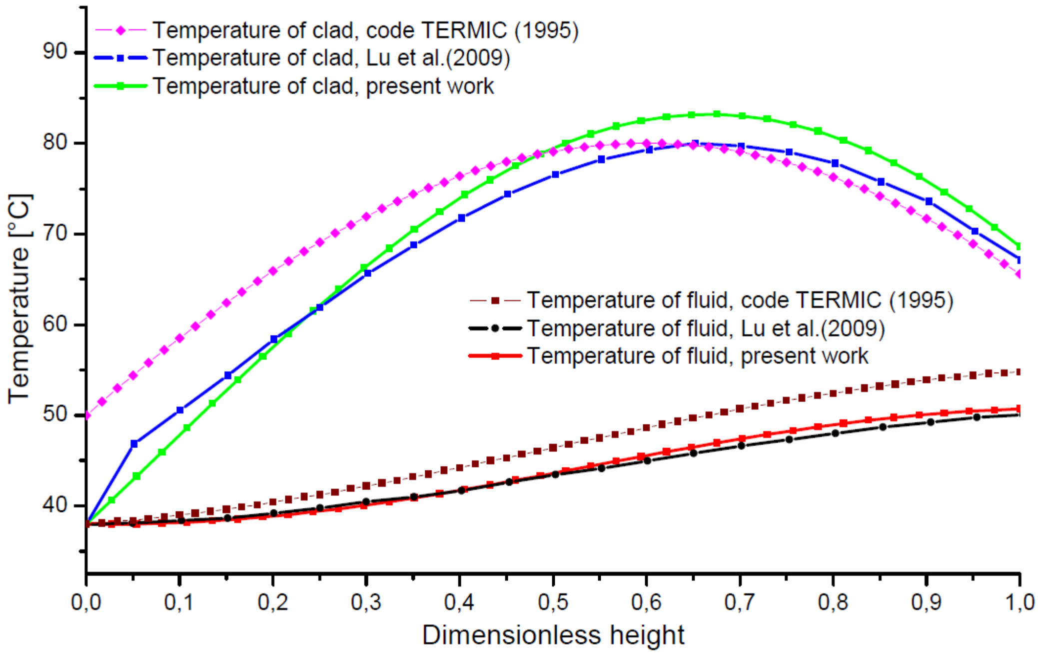

Figure 4: Temperature profiles of the coolant and the clad in the 10 MW nuclear research reactor core channel

Table 2: Obtained and comparative results

02 MW	Tcp Boudali & Salhi (2011)	Tcp code TERMIC (1995)	Tcp Present Work	Zp Boudali & Salhi (2011)	Zp code TERMIC (1995)	Zp Present Work	Tcout Boudali & Salhi (2011)	Tcout code TERMIC (1995)	Tcout Present Work	Tfin for present work	Tfout Boudali & Salhi (2011)	Tfout code TERMIC (1995)	Tfout Present Work
	81.2946	**80.6**	**82.397**	**0.6216**	**0.5667**	**0.5676**	**61.7709**	**64.7**	**60.373**	**38**	**53.112**	**52.3**	**59.3075**
10 MW	Tcp Lu et al. (2009)	Tcp code TERMIC (1995)	Tcp Work	Zp Lu et al. (2009)	Zp code TERMIC (1995)	Zp Present Work	Tcout Lu et al. (2009)	Tcout code TERMIC (1995)	Tcout Present Work	Tfin for presnt work	Tfout Lu et al. (2009)	Tcp code TERMIC (1995)	Tfout Present Work
	80	**80.6**	**83.2026**	**0.6524**	**0.5833**	**0.6757**	**61.1332**	**68.63**	**64.7**	**38**	**50.0457**	**55.27**	**50.7113**

The differences between the obtained results, for the clad and coolant temperatures, and those obtained by various authors were done thanks to the following formulation: (Value of the present work – Value from other author)/ Value of the present work.

Tcp : Clad peak temperature
Zp : Peak location
Tcout : Clad outlet temperature
Tfin : Fluid inlet temperature
Tfout: Fluid outlet temperature

The analysis, of the obtained results shows that most of the calculated relative differences are less than 10 %. The numerical method used in this work gives results close to the results obtained by Boudali and Salhi (2011)[24] and also those obtained by Lu et al. (2009)[11]. The results given by the code TERMIC (1995) [25] for the two power cases are also close to those obtained by this work. Moreover, the values of the clad temperature at the peaks obtained by this work are larger than those given in the three references cited before. So, the results obtained, using this numerical approach, are the more conservative. However, three calculated relative differences are slightly higher than 10 %. One value, for the 02 MW case, is the outlet temperature of the fluid given by the code TERMIC (1995) [25]. The two other relative differences are for the 10 MW case: the peak position of the clad temperature calculated with the code TERMIC (1995) [25], and finally the clad temperature at the exit of the channel given also by the code TERMIC (1995) [25].

One of the most interesting results is the maximal temperature at the peak for the two cases 02 MW and 10 MW. The four relative differences are less than 3.8% which is a very good result. As for the values expressing the peak positions, the relative differences are slightly larger but still very acceptable. For the first it is 13.6% for the peak position which has no effect on the maximal temperature at the peak. For the 02 MW case, the calculated outlet temperature of the fluid compared to that given by the code TERMIC (1995) [25] has a relative difference of 11%. The calculated temperature, of the fluid, increases to reach the clad temperature at the exit. This is due to the under evaluation of the heat transfer. The last value is the relative difference of 10.9% for the outlet calculated temperature of the clad and that given by the code TERMIC (1995) [25]. This difference is simply due to the big difference in the inlet clad temperature which is more than 12 degrees.

However, in a point of view of nuclear safety, the intrinsic values of the peak temperatures are more interesting than the position of these peaks and the results presented here show that the calculated temperatures are acceptable.

VI. Conclusions

In the present work, a system of nonlinear differential equations for thermal-hydraulic analysis of a core channel of a nuclear research reactor using plate type fuel was solved using the finite volume method. The system of differential equations was set up thanks to the conservative equations of mass, momentum and energy. The resolution method was applied to two nuclear research reactors, for an upward flow for the 02 MW case and a downward flow for the 10 MW case.

To enhance this work, relative difference calculations were done involving the obtained results and some comparative works. The obtained results for each case were compared to results given by a thermal-hydraulic code and also to validated results. This double comparison was necessary to know the precision of the results and their ability to predict correctly the temperatures of the clad and the coolant. These two physical quantities are of great importance in the calculation of the critical flux.

The analysis of the relative differences shows that the obtained profiles of the clad and coolant temperatures are very close to the comparative results presented in this work and the thermal-hydraulic behavior of the nuclear reactor core channel is physically correct. And finally, one can also say that the obtained results are the more conservative.

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