Steady State Natural Convection in Vertical Heated Rectangular Channel between Two Vertical Parallel MTR-Type Fuel Plates

Djalal Hamed

Abstract—The aim of this paper is to perform an analytic solution of steady state natural convection in a narrow rectangular channel between two vertical parallel MTR-type fuel plates, imposed under a cosine shape heat flux to determine the margin of the nuclear core power at which the natural convection cooling mode can ensure a safe core cooling, where the cladding temperature should not be reach the specific safety limits (90 °C). For this purpose, a simple computer program is developed to determine the principal parameter related to the nuclear core safety such as the temperature distribution in the fuel plate and in the coolant (light water) as a function of the reactor power. Our results are validated throughout a comparison against the results of another published work, which is considered like a reference of this study.

Keywords—Buoyancy force, friction force, friction factor, MTR-type fuel, natural convection, vertical heated rectangular channel.

I. INTRODUCTION

THE uses of the passive cooling mode like the free natural convection in the reactor technologies is increased in the last decade, which can evacuate the reactor decay heat after a normal or accident shutdown efficiently without the need of any power supply, which can evacuate the reactor decay heat after a normal or accident shutdown efficiently without the need of any power supply.

For the open pool nuclear research reactor, the upward natural convection induced by the density difference between the core and the pool should be investigated to ensure that the cladding temperature will not reach 90 °C, which is considered like a safety limit, to avoid any undesired effect to the integrity of the fuel plat.

In this last decade, many authors are attracted by the features of the natural convection cooling mode especially in the nuclear field, mainly by the passive residual heat removal from the nuclear core. Among the many published works, our whole interest goes toward the work done by Jo [1], which is considered like a reference of this study, for the application and the verification of our computer program. In his work [1], the author carried out by the both RELAP5/MOD3 and NATCON codes a numerical simulation of plate type research reactors during the natural convective cooling mode in a hot spot of a fuel assembly. Several convective heat transfer correlations are implemented into the simulations; then the coolant and surface temperatures and ONB temperature

margin, as a function of core power, are obtained from the simulations with a good agreement between the both used codes.

In the present study, a steady state thermal hydraulic analysis of an upward free natural convection of light water in a vertical channel between two parallel MTR-type fuel plates is investigated. To determine the margin of the nuclear core power for ensure a safe passive residual heat removal from the nuclear core, where the cladding temperature should not be reach a proposed safety limits. For this purpose, a computer program is developed to determine the distribution of the cladding and the coolant temperatures along the channel as a function of the core power.

II. STEADY STATE NATURAL CONVECTION GOVERNING EQUATIONS

For the case of one-dimensional monophasic steady state and free fluid flow, the momentum and the energy equations are respectively expressed by the two equations below [5].

$$\rho \, v \frac{\partial v}{\partial z} = (\rho_c - \rho_p)g - \frac{f}{2}\frac{(\rho_{in}v_{in})^2}{\rho D_h} \tag{1}$$

$$\dot{m}c_p \frac{\partial T}{\partial z} = p_h \, q_{max} \cdot cos\left(\frac{\pi . z}{2.l_p}\right) \tag{2}$$

where the first and the second terms of the right side of (1), are represent respectively the buoyancy and the friction forces with, ρ_c , ρ, ρ_{in} and ρ_p(kg/m^3) are respectively the mean, the local, the inlet and the pool coolant density. v and v_{in}(m/s) are respectively the local and the inlet coolant velocity. f is the friction factor; P_h(m) is the channel heated perimeter; $l_p(m)$ is the extrapolated length. $T(°C)$, C_p(J/k$_g$ s) and $q_{max}(W/m^2)$ are the coolant temperature, the coolant specific heat and the surface heat flux generated in the hot channel.

A. The Friction Factor Correlations

To calculate the friction factor in the rectangular heated channel more accurately, two corrections are introduced; the first one is given by the factor (ξ) as follows [3]:

$$f = \xi f_d \tag{3}$$

where for water $\xi = \left[\frac{\mu_w}{\mu_b}\right]^{0.85}$ and μ_w, μ_b are respectively the

Hamed. Djalal is with the Nuclear Research Center of Draria, Algiers, Algeria (e-mail: homjal@ yahoo.fr).

fluid dynamics viscosity for the temperature of the wall and the bulk temperature.

And the Darcy friction factor f_d is calculated according to the flow regime of the coolant where the three following cases are considered.

B. Laminar Fluid Flow

For laminar flow the correlation used is valid only for Reynolds number less than 2000, and K_R represents the Reynolds correction for the non-circular channel [3].

$$f_d = \frac{64}{Re K_R} \qquad (4)$$

where,

$$K_R = \frac{2}{3} + \frac{11}{24}\alpha(1 - \alpha)$$
$$\alpha = \frac{the\ channel\ width}{the\ channel\ length}$$

C. Transient Fluid Flow

For the case of transient fluid flow where the Reynolds number varies between 2000 and 5000, the friction factor without taking into account the Reynolds correction for the non-circular channel, is evaluated by a linear interpolation as follows [4].

$$f_d = fl + \left(\frac{Re-2000}{3000}\right)(f_t - f_l) \qquad (5)$$

fl is the friction factor for laminar flow for Reynolds number equal to 2000. f_t is the friction factor for turbulent flow for Reynolds number equal to 5000.

D. Turbulent Fluid Flow

For turbulent fluid flow in unheated channel and without taking into account the Reynolds correction for the non-circular channel, the correlation used is valid only for Reynolds number greater than 5000 [2].

$$\frac{1}{\sqrt{f_d}} = 0.8686\ ln\left(\frac{Re}{1.964\ ln(Re)-3.8215}\right) \qquad (6)$$
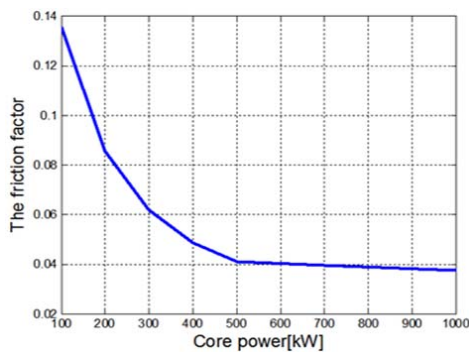

Fig. 1 The friction factor as a function of the core power

After the calculation of the friction factor as a function of the core power, which depends only on the coolant velocity, when the coolant velocity increases with the core power, the

friction factor decreases as shown in Fig. 1.

III. THE COOLANT VELOCITY

The coolant velocity along the channel is determined by integrating (1) analytically, then we got

$$v(z) = v_{in} + \frac{(\rho_c-\rho_p)}{\rho_c\ \bar{v}}gz - \frac{(\rho_{in}v_{in})^2}{2\ \bar{v}}\frac{\bar{f}}{\rho_c^2 D_h}z \qquad (7)$$

where, $\bar{v}(m/s)$, $D_h(m)$ and $\bar{f}$ are respectively the mean coolant velocity, the hydraulic diameter and the mean friction factor.

The previous equation is solved by an iterative process where, at each iteration, the inlet velocity (v_{in}) is increased with constant increment (Δv) until the difference between the buoyancy and the friction forces satisfies a convergence criterion as shown in Fig. 2.
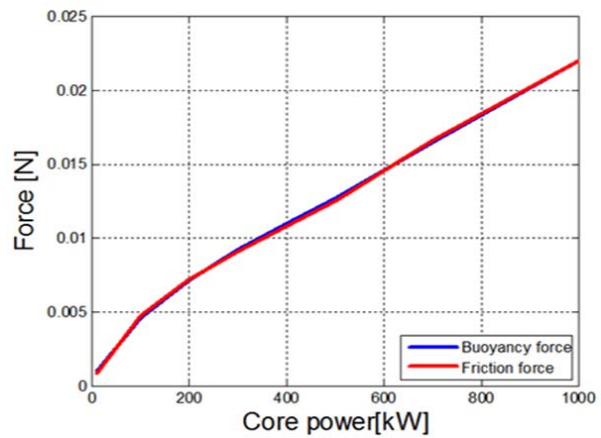

Fig. 2 The buoyancy and the friction forces after they satisfy the convergence criterion

The buoyancy and the friction forces are expressed respectively as [1]:

$$F_B = (\rho_c - \rho_p)\ A\ L\ g \qquad (8)$$

$$F_F = \frac{(\rho_{in}v_{in})^2}{2}A\left[\frac{1}{2\rho_{in}} + \sum_{i=1}^{n}f\frac{\Delta z_i}{\rho_i D_h} + \frac{1}{\rho_{out}}\right] \qquad (9)$$

with, ρ_{in}, ρ_{out} and $\rho_i(K_g\ m^{-3})$ are respectively the inlet, the outlet and the local coolant densities. $L(m)$ and $A(m^2)$ are the channel length and the channel cross section.

The first and third terms in (9) are the entrance and exit losses, while the middle term is the friction pressure loss over the channel.

IV. THE COOLANT PRESSURE

The pressure distribution along the channel is calculated after a static pressure analysis by:

$$P(z) = P_{atm} + \rho g(H - z) - \frac{1}{2}\rho\ \vartheta^2(z) \qquad (10)$$

V. THE TEMPERATURE IN THE FUEL PLATE

A. The Coolant Temperature

After integrating (2), we can express the variation of the coolant temperature as follow.

$$T(z) = Tin + \frac{2\,P_h\,l_p}{\dot{m}cp\pi} q_{max} \left[sin\left(\frac{\pi.z}{2.l_p}\right) + sin\left(\frac{\pi.l}{4.l_p}\right) \right] \quad (11)$$

where, Tin (°C) is the inlet coolant temperature.

B. The Cladding Temperature

The outer surface clad temperature is calculated by using the Newton law, which is expressed by the simple equation below

$$T_{cl}(z) = \frac{q_{max}}{h} cos\left(\frac{\pi z}{2\,l_p}\right) + T(z) \quad (12)$$

where, $h\,(W/m^2°C)$ is the convective heat transfer coefficient. To calculate this coefficient, three different correlations of Nusselt number are analyzed:

- *Elnabass correlation* [2]:

$$Nu = \frac{1}{24}\frac{R_h}{L}(G_r P_r)\left(1 - e^{-2.4\left[\frac{0.5\,L}{R_h G_r P_r}\right]^{0.75}}\right)$$

$$G_r = \left[\frac{\rho^2 g\beta\,L^3(T_{cl}-T)}{\mu^2}\right]^{0.75}$$

$$P_r = \frac{C_p\mu}{K}$$

- *Kreith & Bohn correlation* [3]:

$$Nu = \frac{R_a}{24}\left(1 - e^{-\frac{35}{R_a}}\right)^{0.75}$$

$$R_a = \frac{\rho g\beta C_p(T_{cl}-T)D_h^3}{K\nu}$$

- *McAdams correlation* [6]:

$$\begin{cases} N_u = 0.59\,R_a^{0.25} & 10^4 < R_a < 10^9 \\ N_u = 0.129\,R_a^{0.33} & 10^9 < R_a \end{cases}$$

After we have evaluated the convective heat transfer coefficient by the three correlations as a function of the core power, the resultsare presented in the figure below:

It is evident from Fig. 3 that the three correlations have almost the same values of the convective heat transfer coefficient. But, we must choose one of them. In our study, the McAdams correlation is chosen since it is the simplest and also it is used in some codes.

C. The Fuel Temperature

There are many methods to calculate the center fuel temperature. The simplest method is using the electrical analogy (Ohm law), so we can write:

$$T_f(z) = T_{cl} + q_{max}.cos\left(\frac{\pi.z}{2.l_p}\right)\left(\frac{e_f}{k_f} + \frac{e_{cl}}{k_{cl}}\right) \quad (13)$$

where, e_f(m) and e_{cl}(m) are respectively the half fuel and the cladding thicknesses, while $k_f(w/m°C)$ and $k_{cl}(w/m°C)$ are respectively the nuclear fuel and cladding thermal conductivities.
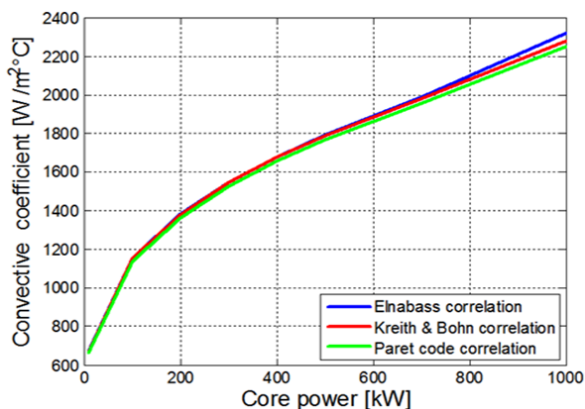

Fig. 3 The convective heat transfer coefficient for different correlations as a function of the core power

VI. RESULTS AND DISCUSSION

We applied our computer program to the same nuclear reactor core of the reference work, where the main nuclear core and fuel element characteristics data are presented in Table I [1]. And all the water properties are calculated as a function of pressure and temperature by the polynomial correlations of the work [7].

TABLE I
THE MAIN CORE GEOMETRIC DATA

Pool depth	9m
number of assembly	18
number of fuel plates	21
number of channels	22
Plate thickness	1.27mm
Meat thickness	0.52mm
Meat width	62.1mm
Clad thickness	0.38mm
Channel thickness	2.35mm
Channel width	66.6mm
Plate length	660mm
Heated length	640mm
Unheated length	10mm

In Figs. 4-6, we display the typical for 200 kW core power.

For the verification of the results that we can obtain by mean of our computer program. In Fig. 7, we show a comparison of the coolant and cladding temperatures in the hot channel at 400 kW core power for the both results obtained by our computer program and those published in the reference work. The both results are relatively close, where the difference is less than 5 °C for the two temperatures over the all channel length.
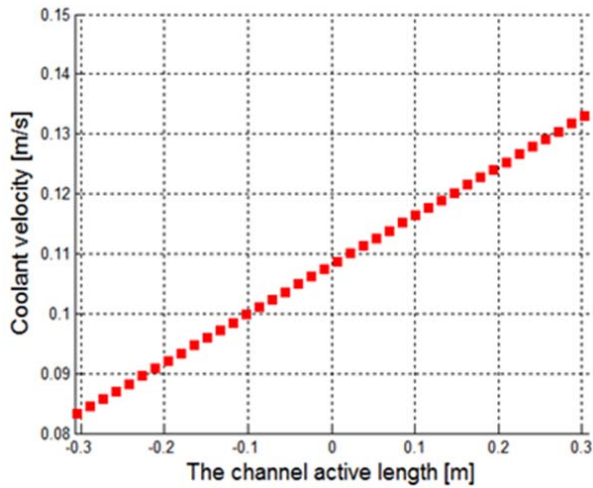
Fig. 4 The coolant velocity in the hot channel for 200 kW core power
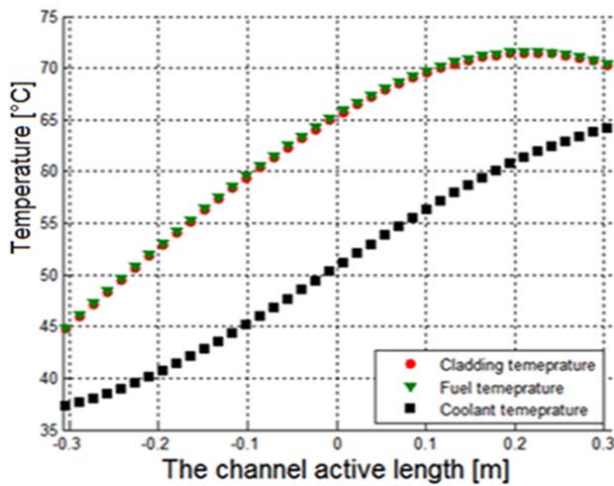

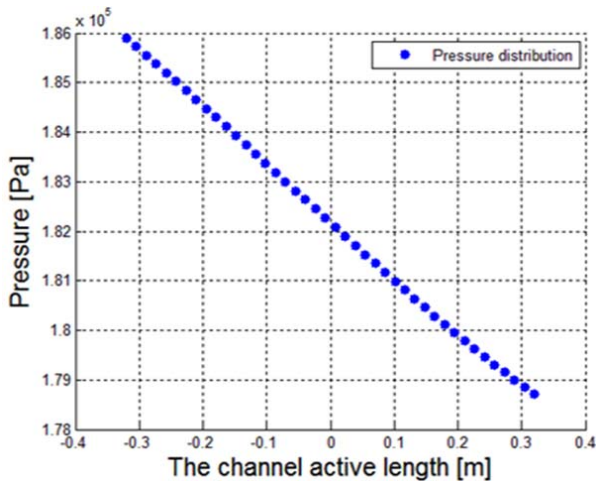Fig. 5 The temperature distribution in the hot channel for 200 kW core power

Fig. 6 The pressure distribution along the Hot Channel for 200 kW core power
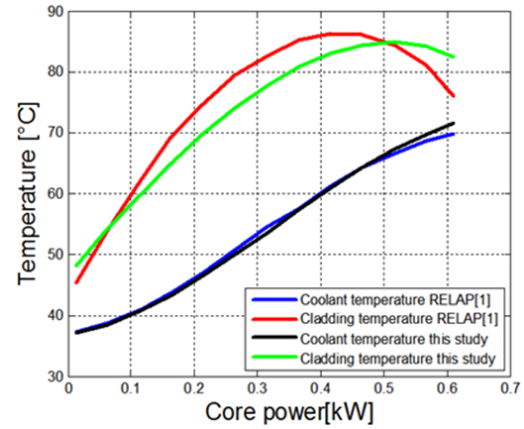

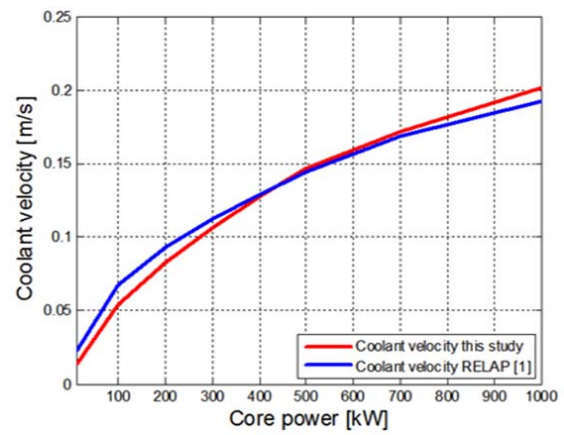Fig. 7 The temperature distribution in the hot channel for 400kW core power

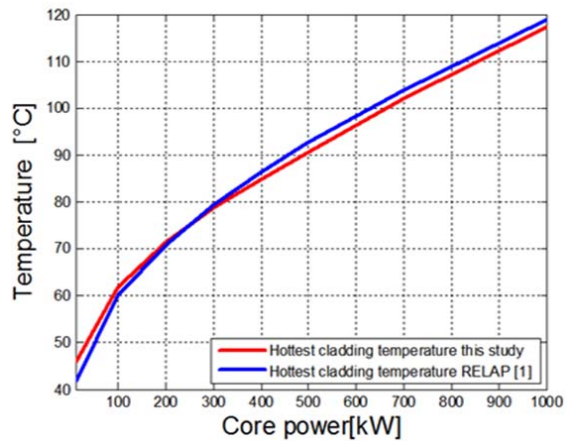Fig. 8 The Coolant Velocity as a Function of The Core Power

Fig. 9 The hottest cladding temperature as a function of the core power

In Figs. 8-10, we present respectively the coolant velocity, the hottest cladding and outlet coolant temperatures which are always compared with the same reference work, in the hot channel as a function of the core power. It is evident from the three figures that there is not a significant difference between

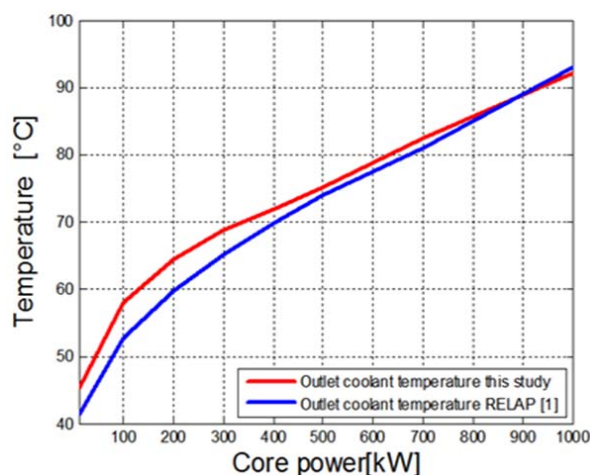the results of the two works. Finally, in Fig. 11, we present the outlet coolant pressure.

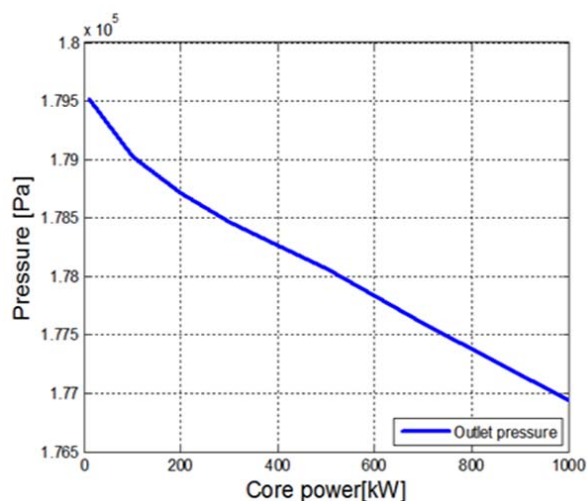Fig. 10 The outlet coolant temperature as a function of the core power

Fig. 11 The outlet coolant pressure as a function of the core power

VII. CONCLUSION

In this study, we performed a thermal hydraulic analysis by solving analytically a steady state natural convection in a heated rectangular channel between two vertical parallel MTR-type fuel plates of the same core reactor of the reference work.

The main goal of this study is to determine the core power margin to employ safely the natural convection cooling mode without reaching a critical state, where the cladding temperature must stay below a specific safety limit (90 °C).

For this purpose, a computer program is developed to calculate and determine the coolant and cladding temperatures distributions in the hot channel of nuclear fuel element, as a function of the core power. Then, our computer calculation program results are validated throughout a comparison against other published study.

The interest conclusions are summarized as follows:

1) A good agreement between the results of our simple and fast computer calculation program and those of the reference work which are carried out by using two validate code for nuclear use.

2) The core power should not reach the 400 kW, to ensure a safe, passive residual heat removal from the nuclear core by the upward natural convection cooling mode.

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