COOLANT CHANNEL DISTRIBUTION IN THE CORE OF THE IPR-R1 TRIGA NUCLEAR REACTOR

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Abstract. *The 250 kW IPR-R1 TRIGA research reactor of the Nuclear Technology Development Center (CDTN) at Belo Horizonte is a pool type reactor cooled by light water under natural circulation. The core has an annular configuration of six rings. The core coolant channels extend from the bottom grid plate to the top grid plate. The cooling water flows through the holes in the bottom grid plate, passes through the lower unheated region of the element, flows upwards through the active region, passes through the upper unheated region, and finally leaves the channel through the differential area between a triangular spacer block on the top of the fuel element and a round hole in the grid. In the natural convection the driving force is supplied by the buoyancy of the heated water in the core channels. Direct measurement of the flow rate in a coolant channel is difficult because of the bulky size and low accuracy of flow meters. The flow rate through the channel may be determined indirectly from the heat balance across the channel using measurements of the water inlet and outlet temperatures. This paper presents the experiments performed in the IPR-R1 reactor to monitoring some thermohydraulic parameters in the core coolant channels, such as: the radial and axial temperature profile, temperature, velocity, mass flow rate, mass flux and Reynolds's number. Some results were compared with theoretical predictions, as it was expected the variables follow the power distribution (or neutron flux) in the core.*

Keywords*: Coolant flow rate, TRIGA nuclear research reactor, temperature, thermalhydraulic.*

1. INTRODUCTION

The IPR-R1 TRIGA reactor at CDTN (Fig. 1) has started up on November 11th, 1960 with a maximum thermal power of 30 kW, in the 70th the power was upgraded to 100 kW. Recently the power was upgraded again to 250 kW at steady state. TRIGA reactors, developed by General Atomics (GA), are the most widely used research reactor in the world. They are cooled by light water under natural convection and are characterized by being inherently safety. The IPR-1 was designed for research, training and radioisotope production and the core is placed at the bottom of an open tank of about 6m height and 2m diameter, able to assure an adequate radioactive shielding. Under full power conditions, the reactor coolant is constrained to flow in parallel to the fuel elements through the active zone of the reactor core. The gradient of fluid density produces a buoyancy force that drives the fluid upward through the reactor core. Countering this buoyancy force are the pressure losses due to the contraction and expansion at the entrance and exit of the core as well as the acceleration and friction pressure losses in the flow channels. Since each flow channel provides its own driving force, it is possible to consider flow channel independently. A forced heat removal system is provided for removing heat from the reactor pool water. The water is pumped through a heat exchanger, where heat is transferred from the primary to the secondary loop. The forced cooling system acts in opposition to the natural circulation, and its main purpose is to create a standing water volume at the pool top in order to improve the biological shield. This article presents the experiments performed in the IPR-R1 reactor for monitoring some thermohydraulic parameters such as: the radial and axial temperature profile, coolant velocity, mass flow rate and Reynolds's number at reactor core channels.

Two probes with type K thermocouples were used in the experiments to measure the bulk temperature of the coolant in the reactor channels, one at the channel exit and another at the channel entrance. Some results were compared with the calculations performed by theoretical analyses. The sensor signs were sent to an amplifier and multiplexing board of the data acquisition system (DAS). The DAS makes the temperature compensation for the thermocouples. All data were obtained as the average of 120 readings and were recorded together with their standard deviations. The system was developed to monitor and to register the operational parameters once a second in a historical database (Mesquita e Rezende, 2010).

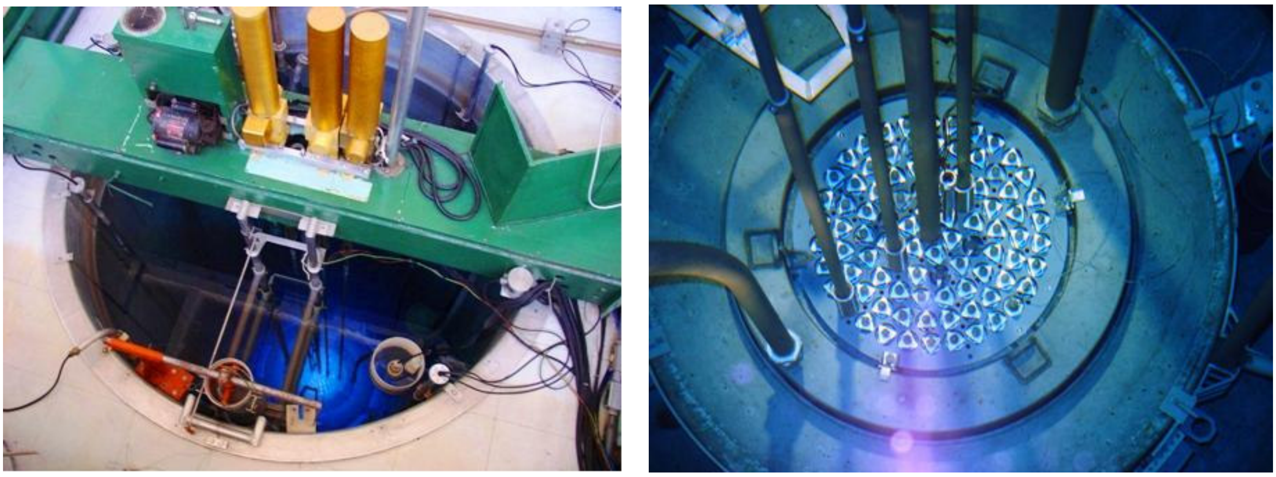

Figure. 1. IPR-R1 TRIGA reactor pool and core

2. CORE CONFIGURATION

The core has an annular graphite reflector and cylindrical configuration of six rings (A, B, C, D, E, F) with 90 positions able to host either fuel rods or other components like control rods, graphite dummies elements (mobile reflector), irradiating and measurement channels (e.g. central thimble or A ring). The core is surrounded by a graphite reflector and water. The fuel is an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U.

Figure 2 shows the IPR R1 TRIGA core configuration. As it is shown in the figure, there are small holes in the core upper grid plate. These holes were used to insert thermocouples to monitor the coolant channel temperatures.

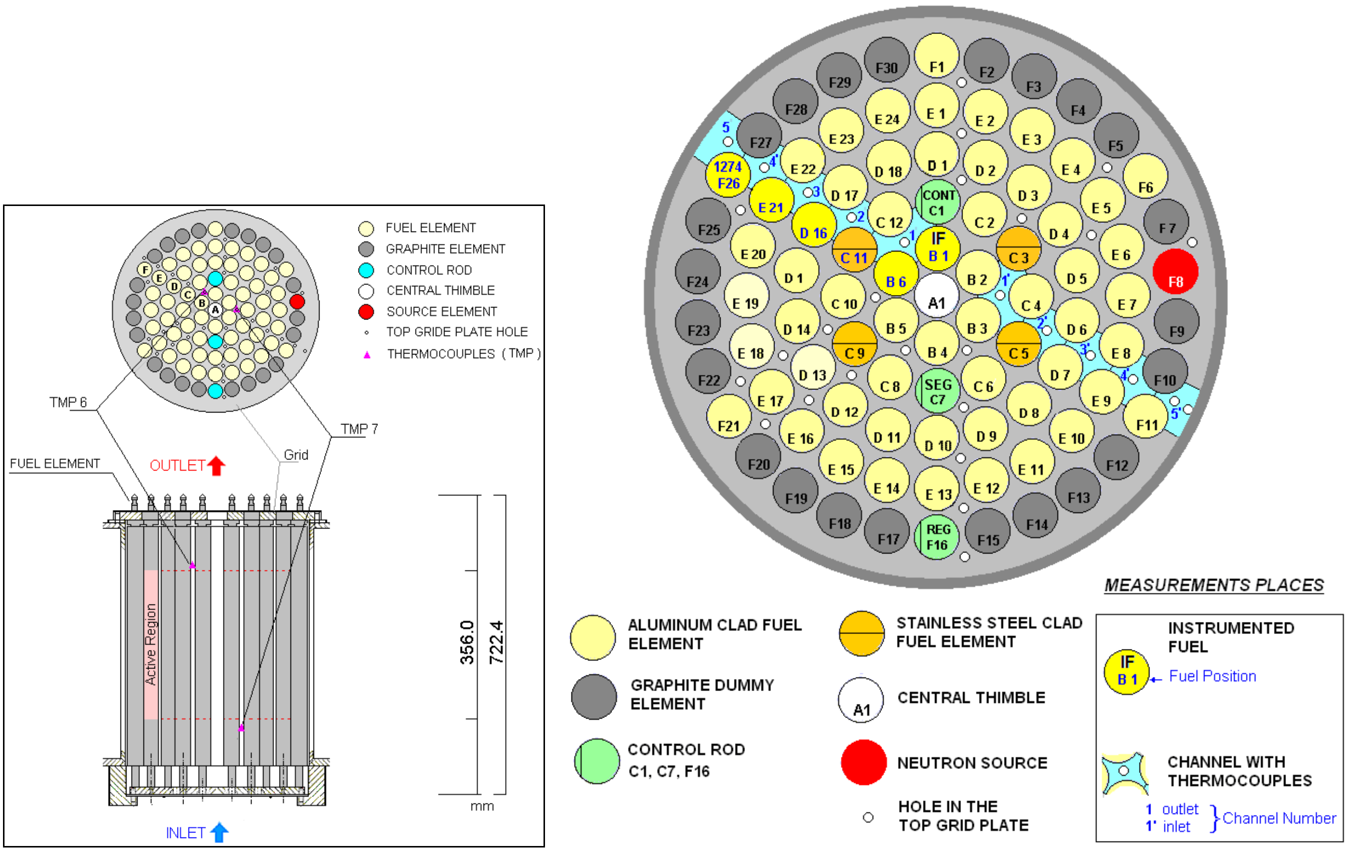

Figure 2. Temperature measures places in the reactor core

3. MONITORING OF THERMALHYDRAULIC PARAMETERS ALONG COOLANT CHANNELS

3.1. Radial temperature profile along the core coolant channels

Position B1 is the hottest location in the core, according to the neutronic calculation (Dalle, 2005). Two type K (chromel–alumel) thermocouples fixed in two rigid aluminum probes (7.9 mm of diameter), were inserted into the core in two channels close to position B1 (Channel 1 and 1' in Fig. 2) and measured the inlet and outlet coolant temperatures in the hot channel. The probes penetrated axially the channels through small holes in the core upper grid plate. The probes were positioned in diametrically opposite channels, so that when a probe measured the channel entrance temperature, the other one registered the channel exit temperature. In a subsequent run, the probe positions were inverted (Fig. 2). This procedure was used also for the Channels 2, 3, 4 and 5. There is no hole in the top grid plate in the direction of the Channel 0; so it was not possible to measure its temperature. The inlet and outlet temperatures in Channel 0 were considered as being the same of Channel 1. For the other channels there are holes in the top grid plate it was possible to insert the temperature probes. The thermocouples were calibrated to obtain measurements within the experimental resolution of ± 0.5 °C. The temperatures were monitored in real time on the data acquisition system computer screen. The reactor thermal power in all measures was 265 kW, according the reactor thermal calibration performed by Mesquita et al. (2007). Figure 3 shows the radial core coolant temperature profiles (inlet/outlet channel temperatures). Theoretical (PANTERA code) results are also shown in the figure (Veloso, 2005).

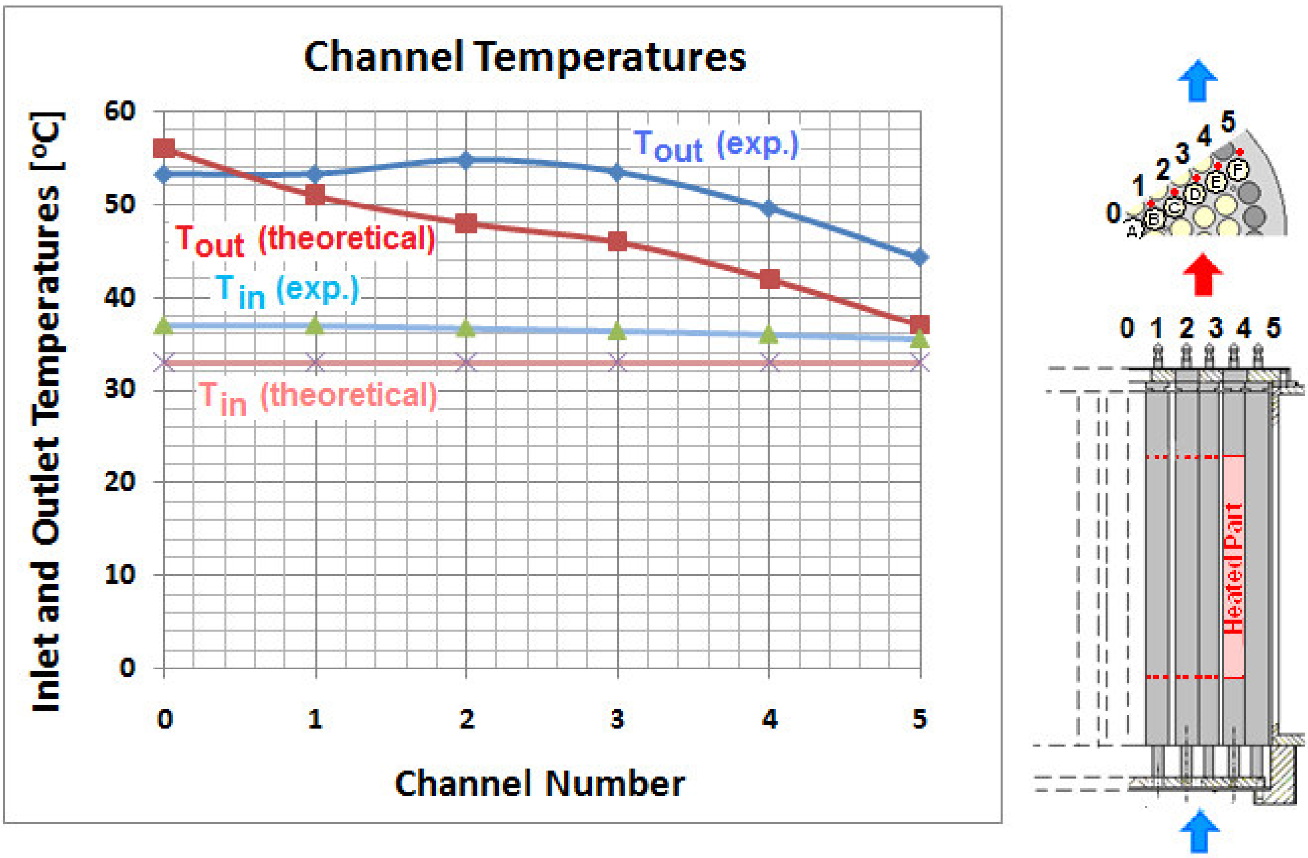

Figure 3. Radial temperature profile in the core coolant channels at 265 kW thermal power

3.2. Axial temperature profile in the hot channel

With the reactor operating at 265 kW, the probe that measures the channel inlet temperature raised in steps of 10 cm and the temperature was measured. The same procedure was done with the reactor operating at 106 kW, but the probe was raised in steps of 5 cm. The experimental temperature profile of the coolant water in Channel 1 is shown in Fig. 4 as a function of the axial position, for the powers of 265 kW and 106 kW. Figure 4 shows also the curve predicted from the theoretical model using the PANTERA Code at 265 kW (Veloso, 2005). The figure shows also the experimental results for other TRIGA reactors (Bärs and Vaurio, 1966), (Haag, 1971) and (Büke and Yavuz, 2000).

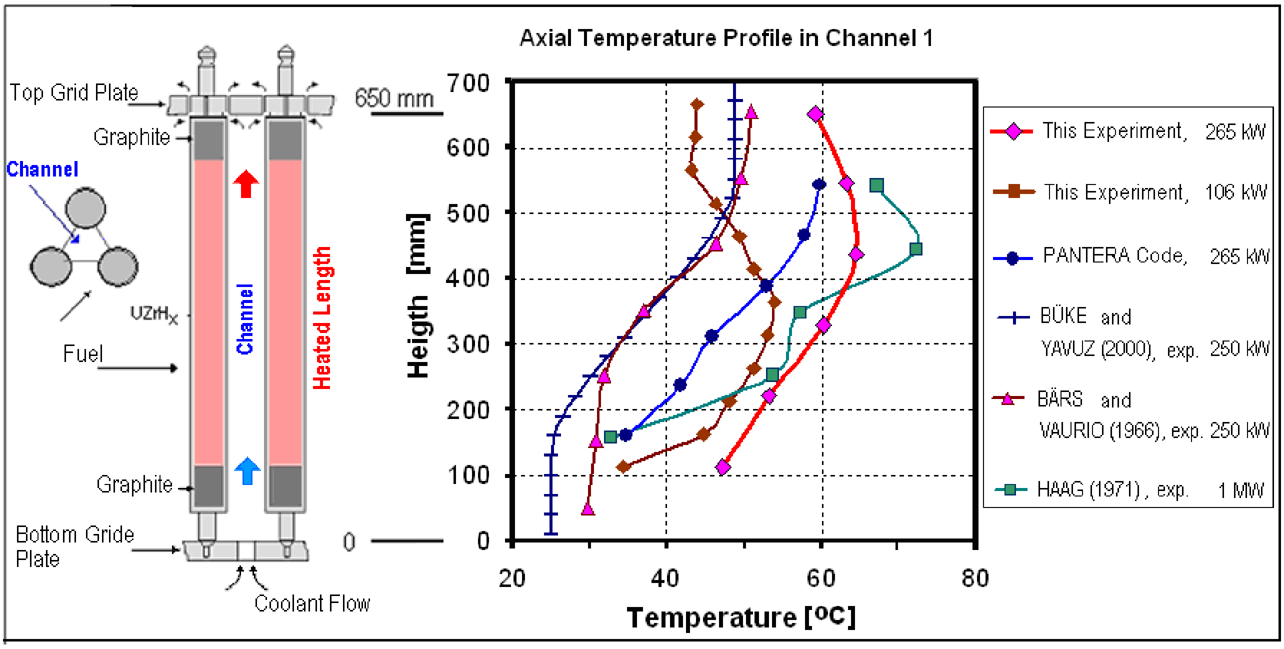

Figure 4. Axial temperature profile along the Channel 1

3.3 Hydraulic parameters of the coolant

The mass flow rate through the core coolant channels was determined indirectly from the heat balance across each channel using measurements of the water entrance and exit temperatures. Although the channels are laterally open, in this work not considered the mass transfer between adjacent channels. Inlet and outlet coolant temperatures in channels were measured with two rigid aluminum probes with thermocouples. They were inserted in the upper grid plate holes (Fig. 2). The reactor thermal power in all measures was 265 kW (already corrected by the thermal power calibration result). Figure 5 shows in detail the coolant channels geometry. The two hottest channels in the core are Channel 0 and Channel 1 (Fig. 2 and Fig. 5). Channel 0 is located closer to the core centre, where density of neutron flux is larger. Table 1 gives the geometric data of the coolant channels and the percent contribution of each fuel element to the channels power.

The channel heating process is the result of the thermal fraction contributions of the perimeter of each fuel around the channel. So there was an average power of 4.518 kW dissipated in each stainless steel cladding fuel element and 4.176 kW dissipated in each aluminum cladding fuel element. The values are multiplied by the core radial power distribution factors calculated by Dalle (2005) using WIMS-D4 and CITATION codes (Fig. 5), and by the power axial distribution factor in the fuel (1.25) as also shown in Fig. 5 (Mesquita, 2005). The products are multiplied by the fractions of the perimeters of each fuel in contact with the coolant in each channel.

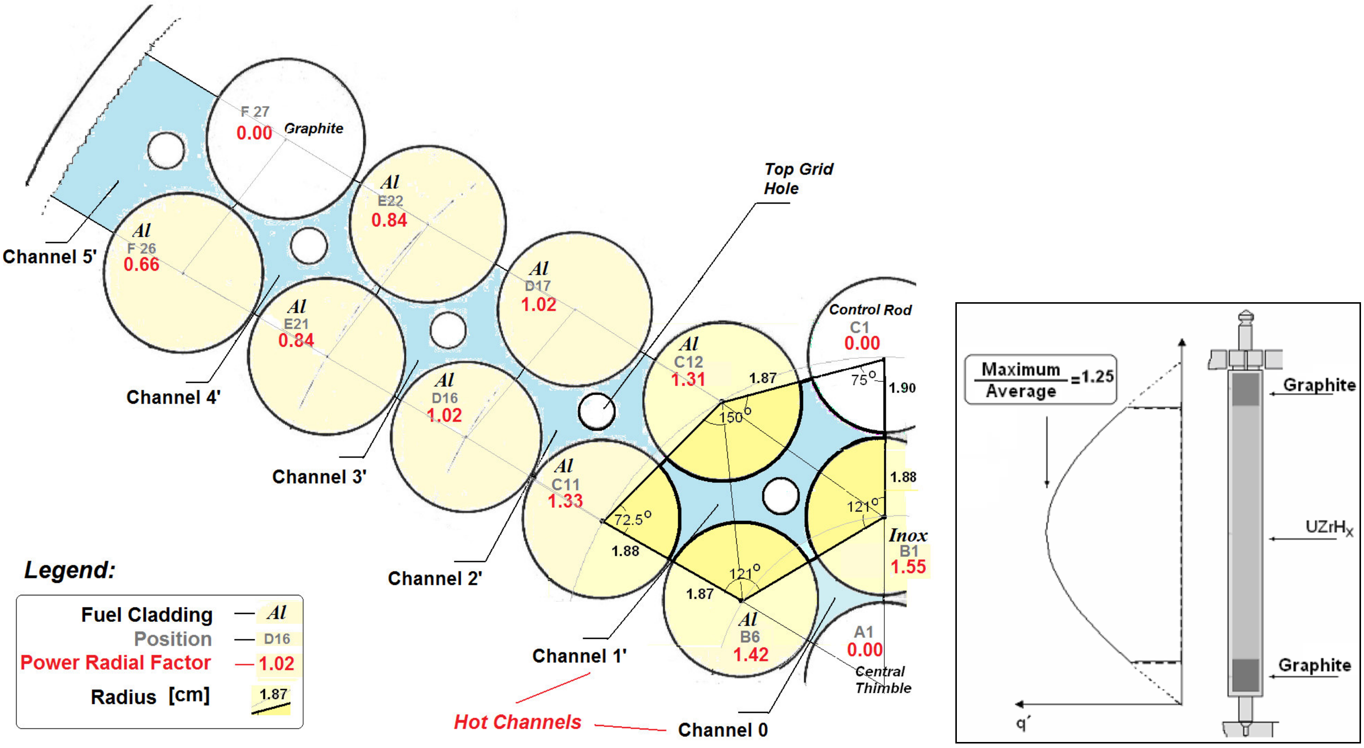

Figure 5. Coolant channels geometry in the core and axial power distribution within the fuel rod

Table 1. Channel geometry and hydraulic parameters (Veloso, 2005) (Mesquita, 2005)

Channel Number	Area [cm^2]	Wetted Perimeter [cm]	Heated Perimeter [cm]	Hydraulic Diameter [cm]	Channel Power [%]
0	1.5740	5.9010	3.9060	1.0669	1.00
1’	8.2139	17.6427	15.1556	1.8623	3.70
2’	5.7786	11.7456	11.7456	1.9679	2.15
3’	5.7354	11.7181	11.7181	1.9578	1.83
4’	5.6938	11.7181	8.6005	1.9436	1.13
5’	3.9693	10.8678	3.1248	1.4609	0.35

The mass flow rate in the hydraulic channel ($\dot{m}$) in [kg/s]; is given indirectly from the thermal balance along the channel using measurements of the water inlet and outlet temperatures:

$$\dot{m} = \frac{q_c}{c_p \Delta T} \quad . \tag{1}$$

Where q_c is the power supplied to the channel [kW], c_p is the isobaric specific heat of the water [J/kgK] and ΔT is the temperature difference along the channel [^{o}C]. The values of the water thermodynamic properties are obtained as function of the bulk water temperature at the channel for the pressure 1.5 bar.

The pertinent parameters required for the analysis of coolant channels are tabulated in Table 2. In Table 2 the mass flux G is given by: $G = \dot{m}$ / *channel area*. The velocity u is given by $u = G / \rho$, where ρ is the water density (995 kg/m^3). The values of the water thermodynamic properties at a pressure of 1.5 bar, as a function of the fluid average temperature of in the channels were found from the table provided by Wagner and Kruse (1988). Reynolds number (Re) is given in the last column of Table 2, given by:

$$\mathrm{Re} = \frac{G D_w}{\mu} \tag{2}$$

Where G is the mass flux in [kg/m^{2}s], D_w is the hydraulic diameter in [m] and μ is the dynamic viscosity [kg/ms].

Figure 6 shows the power dissipate and the temperature increase in each channel. The profile of the mass flow rate and velocity in the core is shown in the graphs of Figure 7. Figure 8 compares experimental and theoretical profile of mass flux G in the core coolant channels. The theoretical values were calculated using PANTERA code (Veloso, 2005).

Table 2. Properties of the coolant channel at the power of 265 kW[1]

Channel	Channel Power q	T_{out} - T_{in} Δ T	Flow Rate $\dot{m}$	Area	Mass Flux G	Velocity u	Reynolds Number Re
	[kW]	[^{o}C]	[kg/s]	[cm^2]	[kg/m^{2}s]	[m/s]	-
0	2.65	15.5	0.041	1.574	260.48	0.26	3228
1	9.81	15.5	0.151	8.214	183.83	0.18	5285
2	5.70	17.1	0.080	5.779	138.44	0.14	5181
3	4.85	16.3	0.071	5.735	123.79	0.12	4184
4	3.00	12.1	0.059	5.694	103.62	0.10	2525
5	0.93	7.7	0.029	3.969	73.06	0.07	549

[1]. Specific heat (c_p) = 4.1809 [kJ/kgK], water density (ρ) 995 kg/m3 and dynamic viscosity(μ) = 0.620 10^{-3} kg/ms at 45 ^{o}C.

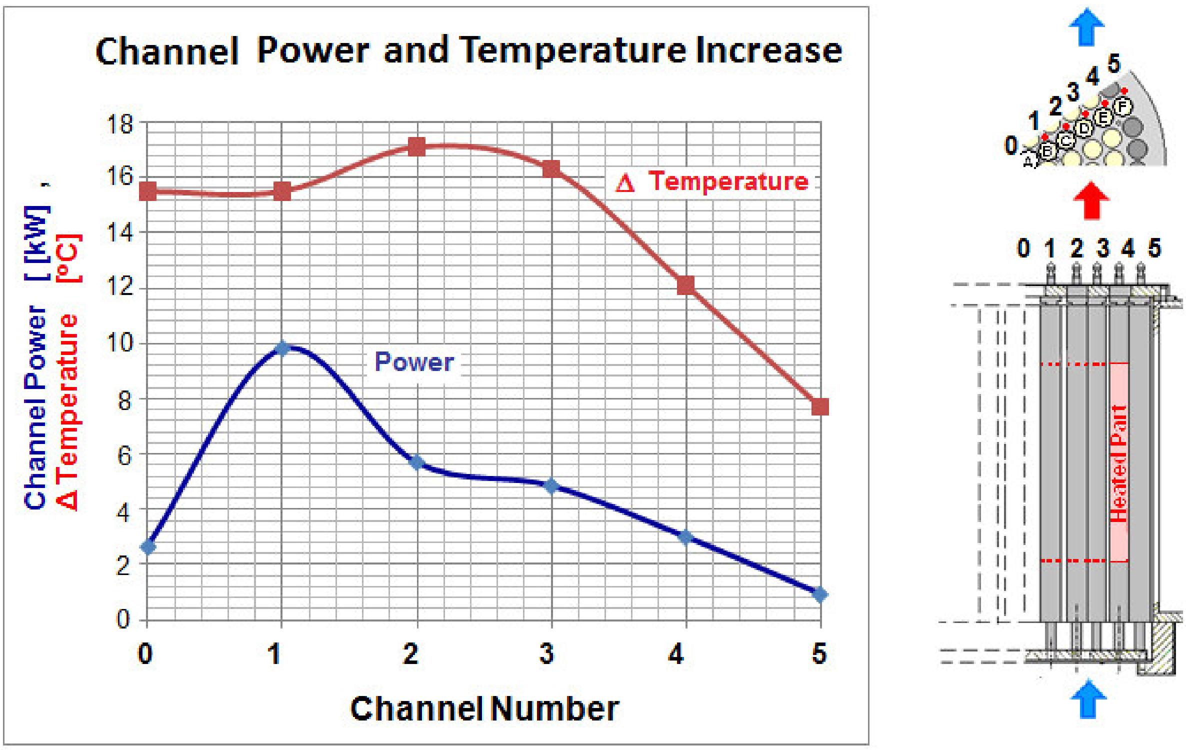

Figure 6. Power and temperature increase in coolant channels

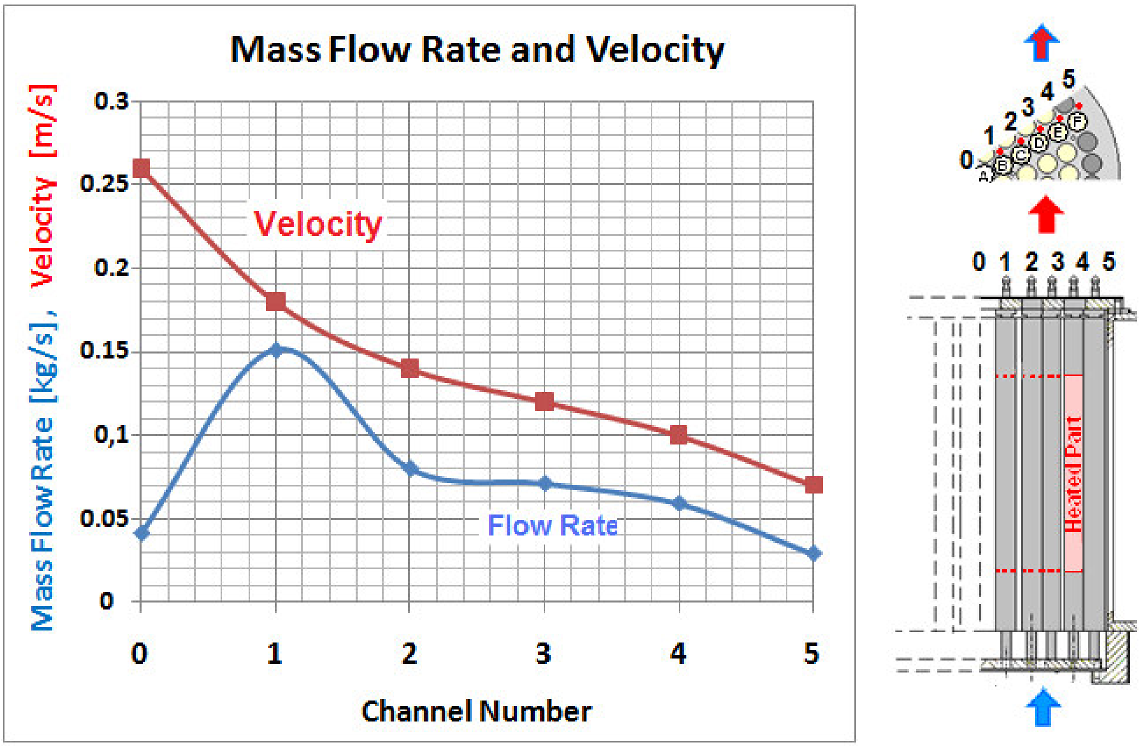

Figure 7. Mass flow rate and velocity in coolant channels

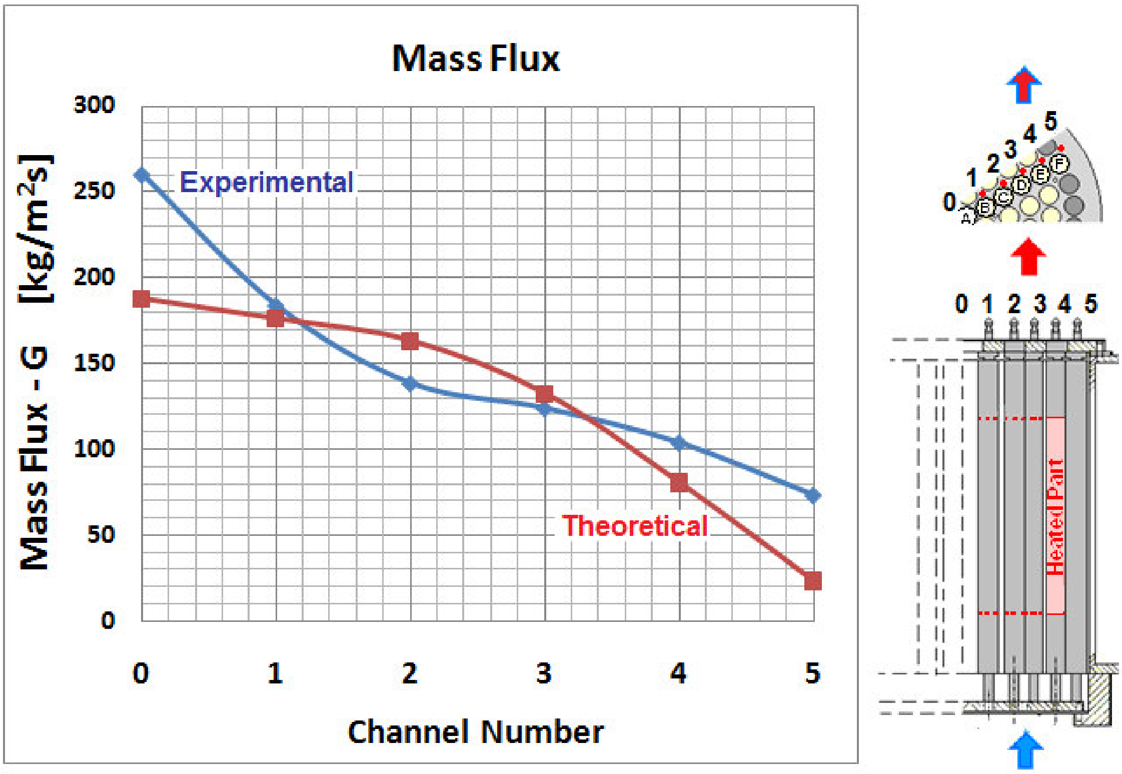

Figure 8. Mass flux in coolant channels

4. CONCLUSIONS

The experiments confirmed the efficiency of the natural convection in removing the heat produced in the reactor core by nuclear fission. The data taken during the experiments provides an excellent picture of the thermal performance of the IPR-R1 reactor core.

The experimental temperature profile along the coolant channel 1 (Fig. 4) is different from that predicted from the theoretical model using the PANTERA Code (Veloso, 2005). Ideally, the coolant temperature would increase along the entire length of the channel, because heat is being added to the water by all fuel regions in the channel. Experimentally, the water temperature reaches a maximum near the middle length and then decreases along the remaining channel. The shape of the experimental curves is similar to the axial power distribution within the fuel rod as shown in Fig. 5. The theoretical temperatures and mass flux were determined under ideal conditions. The actual coolant flow is quite different because of the inflow of water from the core sides (colder than its center). There is a considerable coolant crossflow throughout the channels. Note that the natural convection flow is turbulent in all channels near the center. As can be seen in Figures 6 and 7, the mass flow in each channel is directly proportional to power dissipated in the channel.

The IPR-R1 TRIGA core design accommodates sufficient natural convective flow to maintain continuous flow of water throughout the core, which thereby avoids significant bubbles formation and restricts possible steam bubbles to the vicinity of the fuel element surface. The spacing between adjoining fuel elements was selected not only from neutronic considerations but also from thermohydrodynamic considerations.

It is suggested to repeat the experiments reported here, by placing a hollow cylinder over the core, with the same diameter of it, to verify the improvement of the mass flow rate by the chimney effect. These experiments can help the designers of the Brazilian research Multipurpose Reactor (RBM), which will be a pool reactor equipped with a chimney to improve the heat removal of from the core (CDTN/CNEN, 2009).

5. ACKNOWLEDGEMENTS

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6. REFERENCES

Bärs, B. and Vaurio, J., 1966. “Power Increasing Experiments on a TRIGA Reactor”. Technical University of Helsinki, Department of Technical Physics. Otaniemi Filand. Report No. 445, 19 p.

Büke, T. and Yavuz, H., 2000. “Thermal-hydraulic Analysis of the ITU TRIGA Mark-II Reactor”. Proceeding of 1st Eurasia Conference On Nuclear Science and its Application. Izmir, Turquia. 23-27 Oct. p. 333-347.

CDTN/CNEN - Nuclear Technology Development Center/Brazilian Nuclear Energy Commission, 2009. “Brazilian Multipurpose Reactor (RMB), Preliminary Report of Reactor Engineering Group, General Characteristics and Reactors Reference”. (in Portuguese).

Dalle, H.M., 2005. “TRIGA IPR-R1 Reactor Simulation Using Monte Carlo Transport Methods”. ScD thesis , Universidade Estadual de Campinas, São Paulo, (in Portuguese).

Haag, J.A., 1971. “Thermal Analysis of the Pennsylvania State TRIGA Reactor”. Pennsylvania: The Graduate School, Department of Nuclear Engineering, Dissertation (M. Sc.). 96 p.

Mesquita, A.Z., 2005. “Experimental Investigation on Temperatures Distribuitions in a Research Nuclear Reactor TRIGA IPR-R1”, ScD thesis, Universidade Estadual de Campinas, São Paulo, (in Portuguese).

Mesquita, A.Z. and Rezende, H.C., 2010. “Thermal Methods for On-Line Power Monitoring of the IPR-R1 TRIGA Reactor”. Progress in Nuclear Energy (New series), v. 52, p. 268-272, April 2010. Elsevier Ltd. Oxford UK.. doi:10.1016/j.pnucene.2009.07.013.

Mesquita, A.Z.; Rezende, H.C. and Tambourgi, E.B., 2007. “Power Calibration of the TRIGA Mark I Nuclear Research Reactor”. Journal of the Brazilian Society of Mechanical Sciences and Engineering, v. XXIX, p. 240-245. Rio de Janeiro. doi: 10.1590/S1678-58782007000300002.

Veloso, M.A., 2005. “Thermal–hydraulic Analyses of the IPR-R1 TRIGA Reactor on 250 kW”, CDTN/CNEN, NI-EC3-05/05, Belo Horizonte, (in Portuguese).

Wagner, W and Kruse, A., 1998, “Properties of Water and Steam – The industrial standard IAPWS-IF97 for the thermodynamics properties”. Springer, Berlin.

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