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

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ABSTRACT

The IPR-R1 is a typical TRIGA Mark I light-water and open pool type reactor. The core has an annular configuration of six rings and is cooled by natural circulation. 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. 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 thermo-hydraulic 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.

1. INTRODUCTION

Understanding the behavior of the operational parameters of nuclear reactors allow the development of improved analytical models to predict the fuel temperature, and contributing to their safety. The recent natural disaster that caused damage in four reactors at the Fukushima nuclear power plant shows the importance of studies and experiments on heat removal by natural convection to remove heat from the residual remaining after the shutdown. Experiments, developments and innovations used for research reactors can be later applied to larger power reactors. Their relatively low cost allows research reactors to provide an excellent testing ground for the reactors of tomorrow.

The IPR-R1 TRIGA (*Instituto de Pesquisas Radiativas* - Reactor 1, Training Research Isotope production, General Atomic) reactor is managed by Nuclear Technology Development Center (CDTN) a research institute of the Brazilian Nuclear Energy Commission (CNEN). The IPR-R1 reactor, shown in Figure 1, is located at the campus of Federal University of Minas Gerais. The IPR-R1 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.

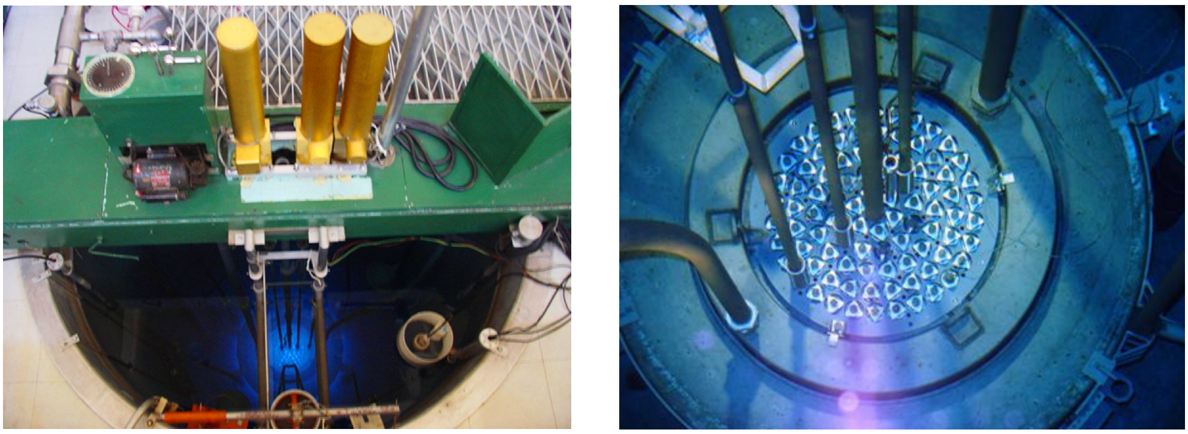

Figure. 1. IPR-R1 TRIGA reactor pool and core.

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 paper presents the experiments performed in the IPR-R1 reactor for monitoring some thermo-hydraulic parameters such as: the radial and axial temperature profile, coolant velocity, mass flow rate and Reynolds's number at reactor core channels.

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.

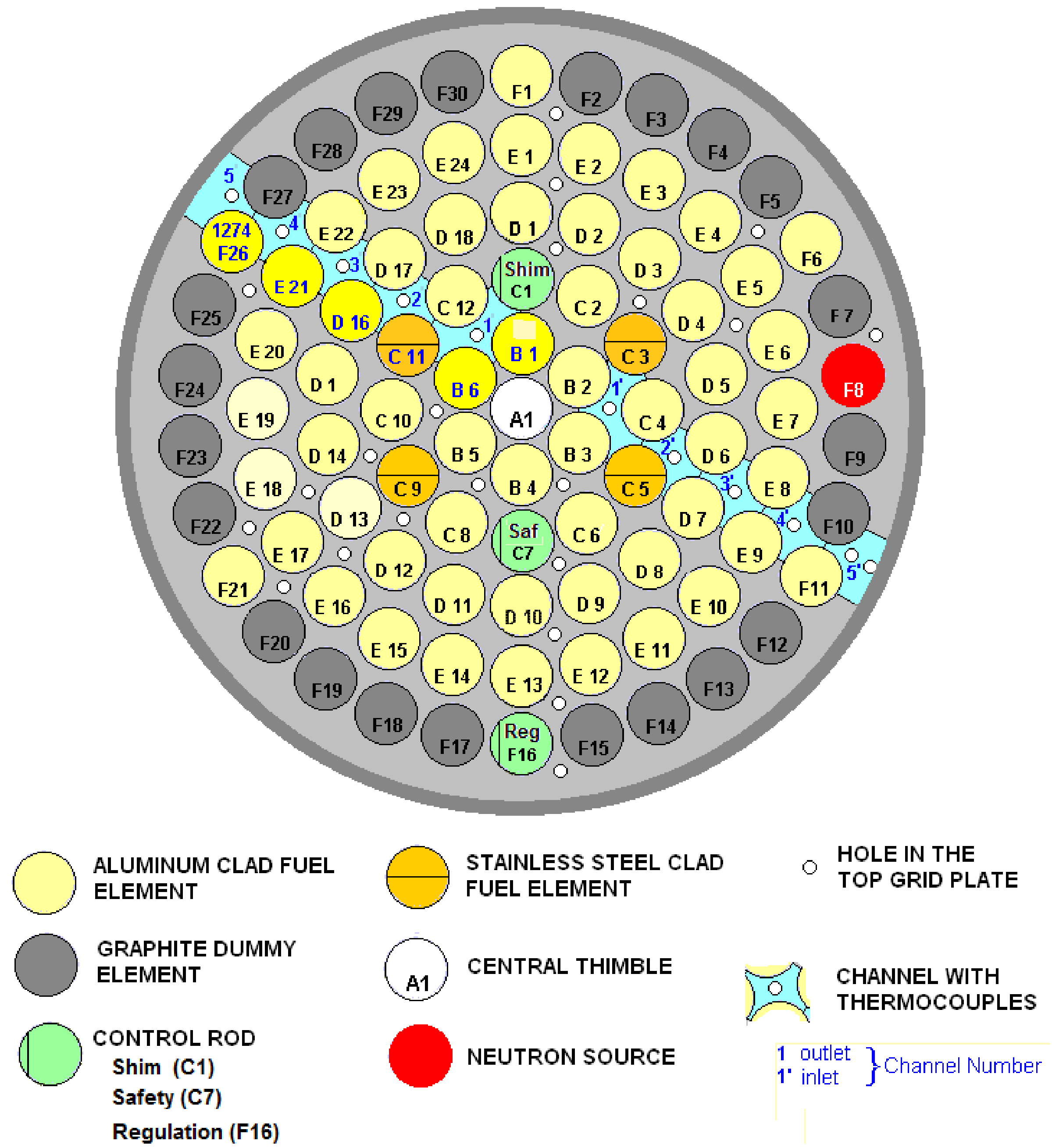

Figure 2. Core configuration

3. METHODOLOGY

Before starting the experiments the thermal power released by the core was calibrated, according with the methodology developed by Mesquita et al. [1]. It was found 265 kW while the linear neutronic channel indicated 250 kW. The calibration method used consisted of the steady-state energy balance of the primary cooling loop. For this balance, the inlet and outlet temperatures and the water flow in this primary cooling loop were measured. The heat transferred through the primary loop was added to the heat leakage from the reactor pool.

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 (Fig. 3). Some results were compared with the calculations performed by theoretical analyses. The temperature measuring lines were calibrated as a whole, including sensors, cables, data acquisition cards and computer. The uncertainties for the temperature measurement circuit were ±0.8 ^{o}C. The adjusted equations were added to the program of the data acquisition system (DAS). The sensor signs were sent to an amplifier and multiplexing board of the DAS, which also makes the temperature compensation for the thermocouples. The temperatures were monitored in real time on the DAS computer screen. 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 [2].

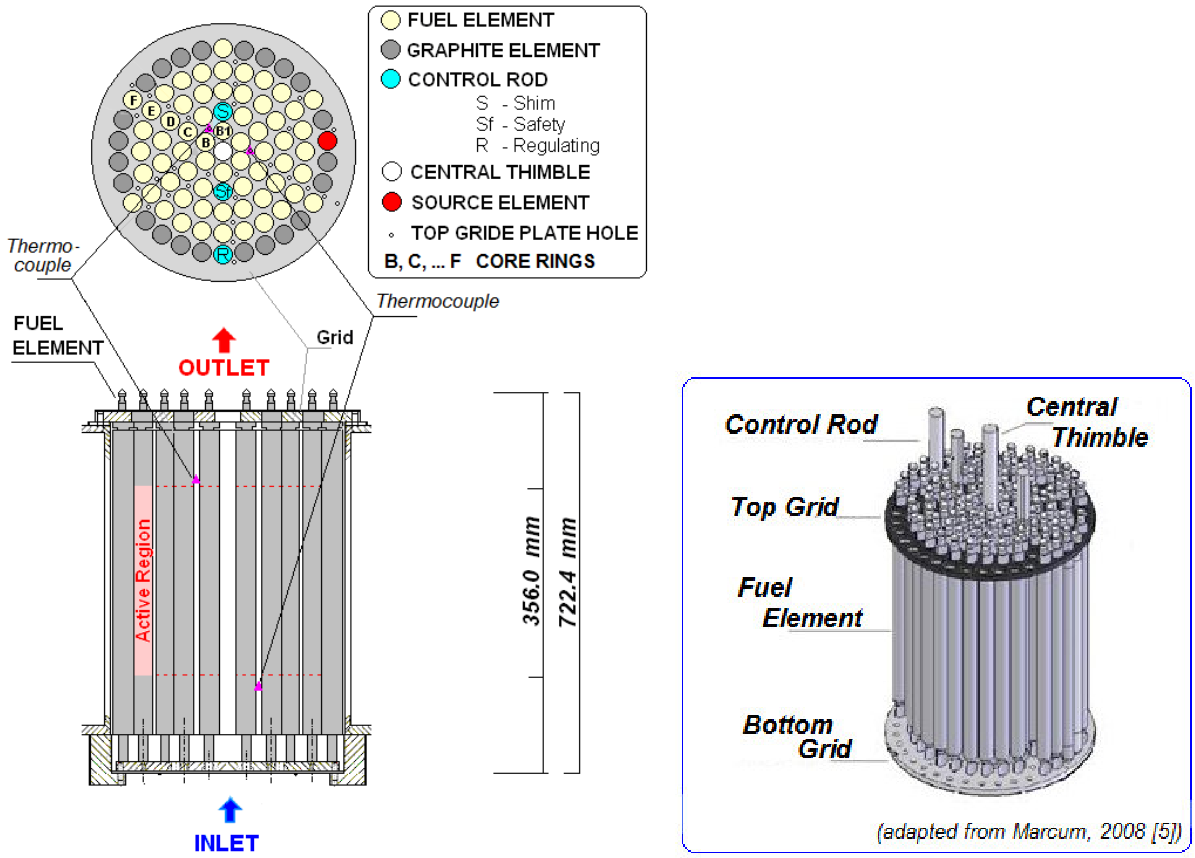

Figure 3. Temperature measures places in the reactor core

3.1. Coolant Channels Temperature

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 Fig. 3) and measured the inlet and outlet coolant channel temperatures. 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. This procedure was used also for the Channels 1' 2', 3', 4' and 5' (Fig. 2). 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.

To found the bulk coolant temperature axial profile at hot channel, with the reactor operating at 265 kW, the probe that measures the channel inlet temperature was raised in steps of 10 cm and the temperature was monitored. The same procedure was done with the reactor operating at 106 kW, but the probe was raised in steps of 5 cm.

3.2. Hydraulic Parameters of the Coolant

In the TRIGA type reactors the buoyancy force induced by the density differential across the core maintains the water circulation through the core. Direct measurement of the flow rate in a coolant channel is difficult because of the bulky size and low accuracy of flow meters. The mass flow rate through the channel may be 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 cross flow or mass transfer between adjacent channels was ignored. As said 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).

Figure 4 illustrates schematically the general natural convection process established by the fuel elements bounding one flow channel in the core. 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 leaving 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. As mentioned, in natural convection the driving force is supplied by the buoyancy of the heated water in the core channels.

In a typical TRIGA flow channel entire fuel element is cooled by single phase convection as long as the maximum wall temperature is kept below that required to initiate boiling. However, at higher power levels the inlet and outlet regions of the core, where the heat fluxes are the lowest, are cooled by single phase convection. In the central region, where the axial heat flux is highest, the mode of heat transfer is predominantly subcooled boiling [3] and [4].

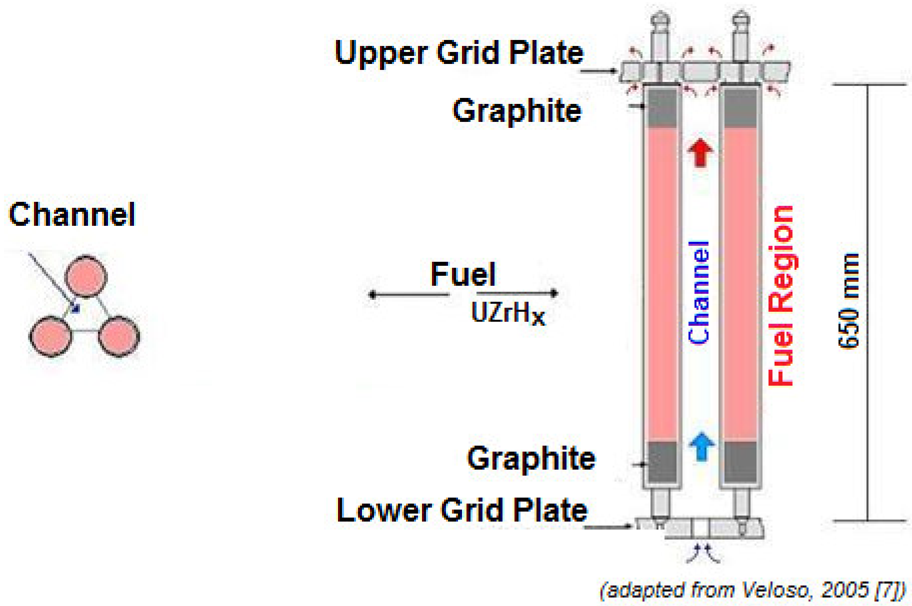

Figure 4. A schematic of one flow channel in the TRIGA core.

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 at 265 kW of core total power. The values are multiplied by the fuel element axial power distribution and core radial power distribution factors as shown in profiles of Fig. 5. The power axial distribution factor in the fuel is 1.25 [5]. Figure 6 shows in detail the coolant channels geometry. The core radial power distribution factors, shown in Fig. 6 were calculated using WIMS-D4 and CITATION codes [6]. The products are multiplied by the fractions of the perimeters of each fuel in contact with the coolant in each channel. The two hottest channels in the core are Channel 0 and Channel 1'. Channel 0 is located closer to the core centre, where density of neutron flux is larger, but there is no hole in the top grid plate in the direction of this channel. Table 1 gives the geometric data of the coolant channels and the percent contribution of each fuel element to the channels power [4] and [7].

Table 1. Channel geometry and hydraulic parameters

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

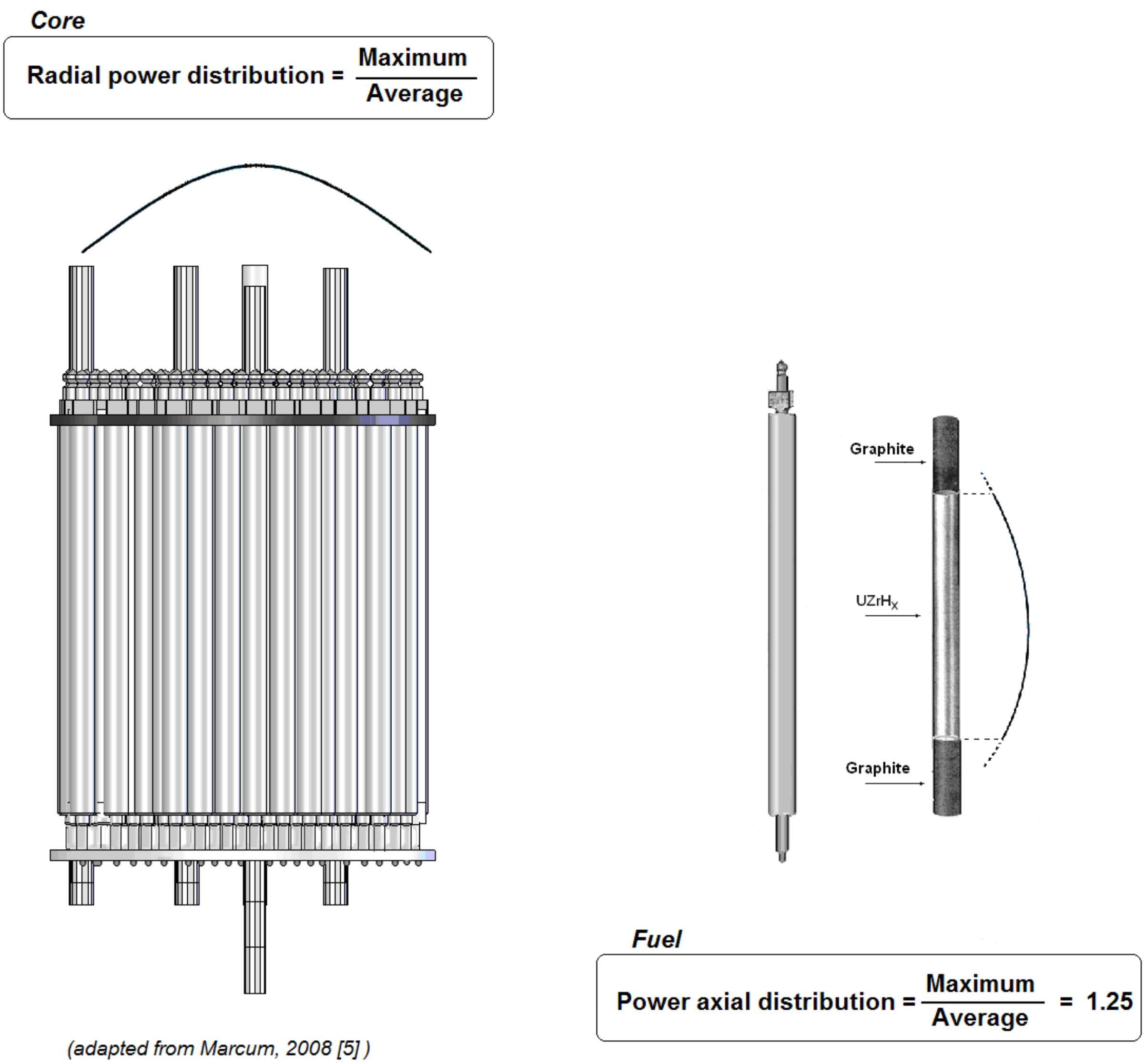

Figure. 5. Core radial and fuel element axial power profiles.

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 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 were obtained as function of the bulk water temperature at the channel for the pressure 1.5 bar [8]. Reynolds number (Re), used to characterize the flow regime, it is given by:

$$Re = \frac{GD_w}{\mu} \tag{2}$$

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].

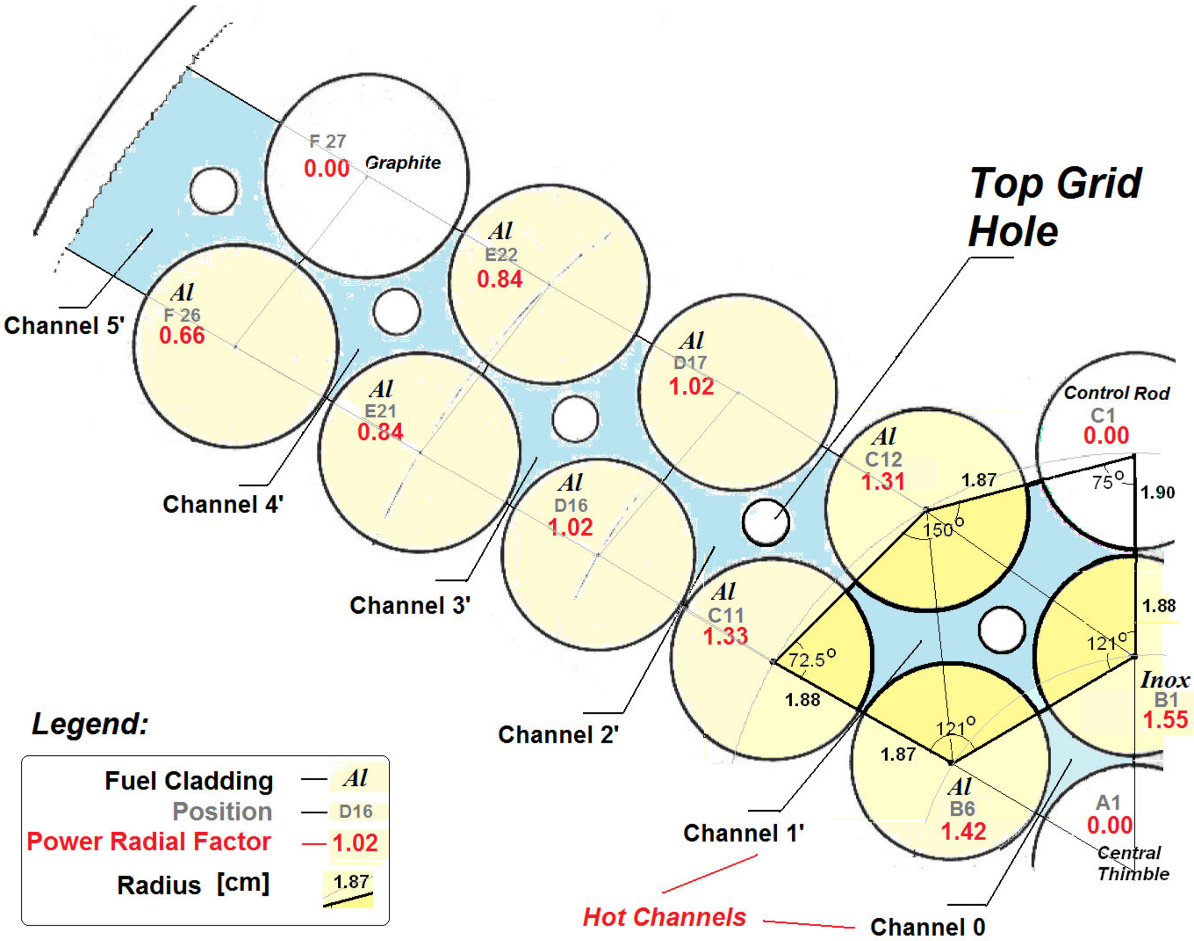

Figure 6. Core coolant channels geometry and radial power distribution.

4. RESULTS

4.1 Outlet Coolant Temperature as Function of the Thermal Power

The experimental coolant exit temperature for each core ring is shown in Fig 7 as a function of the reactor power. The aluminum probe with thermocouple was inserted in each hole at top grid plate, and the coolant inlet temperature was about 38 ^{o}C in all measurements.

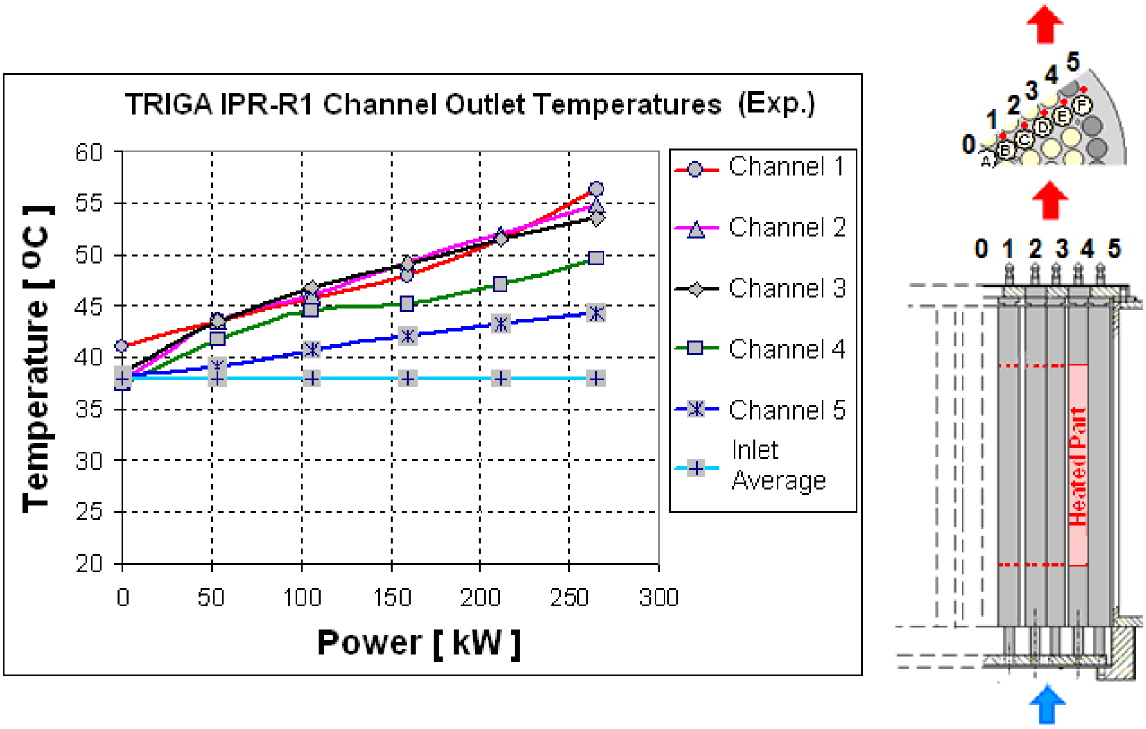

Figure 7. Outlet coolant temperature as function of the thermal power.

4.2 Radial temperature profile along the core coolant channels

Figure 8 shows the radial core coolant temperature profiles (inlet/outlet channel temperatures) at 265 kW. Theoretical results using the PANTERA code are also shown in the figure [7].

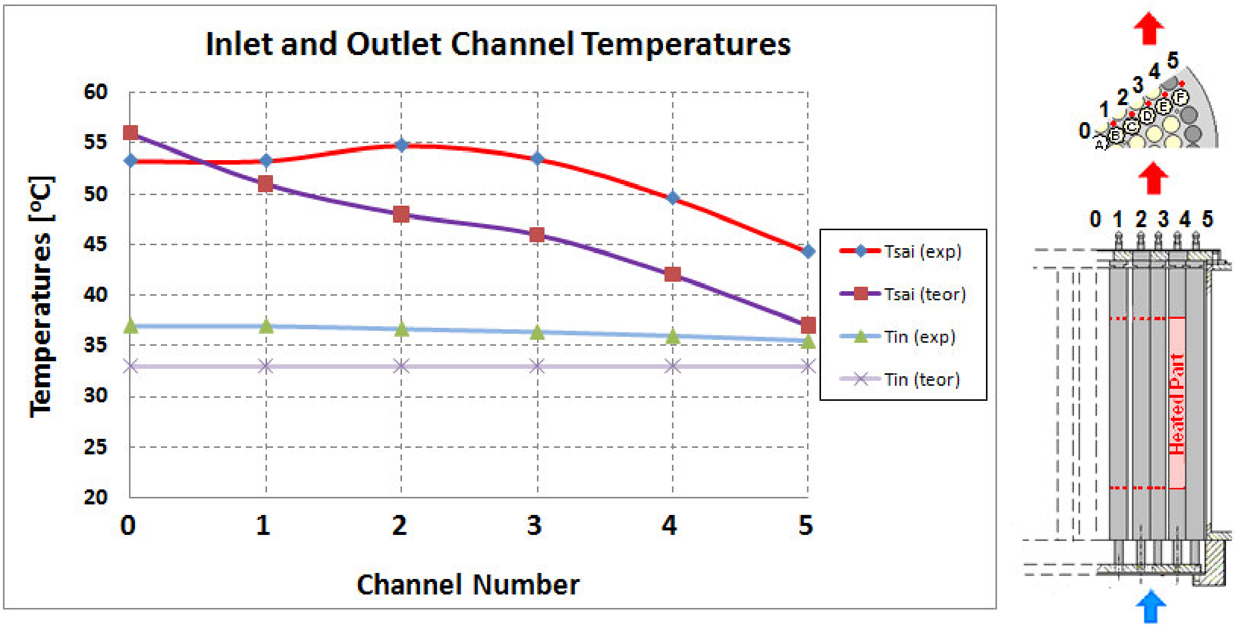

Figure 8. Radial temperature profile in the core coolant channels at 265 kW.

4.3 Axial temperature profile in the hot channel

The experimental bulk coolant temperatures profile in Channel 1 is shown in Fig. 9 as a function of the axial position, for the powers of 265 kW and 106 kW. Figure 9 shows also the curve predicted from the theoretical model using the PANTERA code at 265 kW [7]. Experimental results for other TRIGA reactors are also shown in the graphics [9], [10] and [11].

The experimental temperature profile along the coolant is different from that predicted from the theoretical model. 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. 9. Although Channel 1 is located beside the control rod, the axial temperature profile was not influenced by a possible deformation of the neutron flux caused by this rod, because it was in its upper position, i.e. outside the core. The actual coolant flow is quite different probably, because of the inflow of water from the core sides (colder than its center).

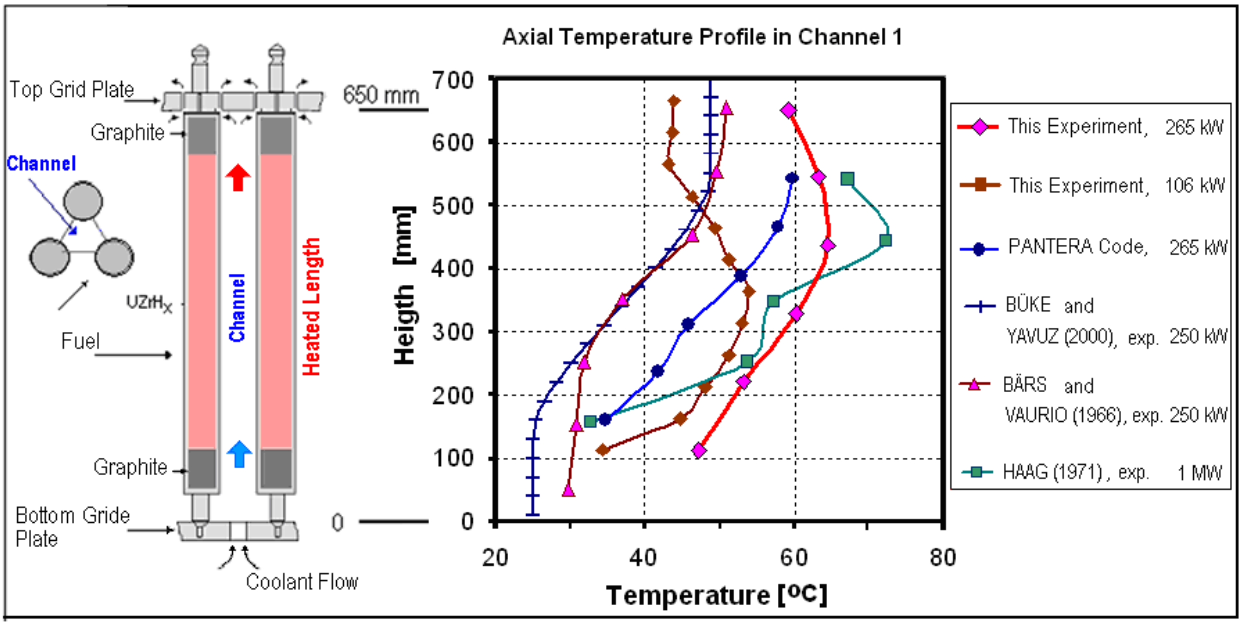

Figure 9. Axial bulk coolant temperature profile along the Channel 1.

4.4 Thermal hydraulic parameters of coolant channels

The pertinent parameters required for the analysis of coolant channels are tabulated in Table 2. Figure 10 shows the power dissipate and the temperature increase in each channel at 265 kW reactor total power. This power was the results of the thermal power calibration [1]. The profile of the mass flow rate and velocity in the core is shown in the graphs of Figure 11. Figure 12 compares experimental and theoretical profile of mass flux *G* in the core coolant channels. The theoretical values were calculated using PANTERA code [7]. As it can see by the Reynolds number the flow regime is turbulent in channels near the core center.

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

Channel	Channel Power q [kW]	T_{out} - T_{in} ΔT [°C]	Flow Rate $\dot{m}$ [kg/s]	Area [cm²]	Mass Flux G [kg/m²s]	Velocity u [m/s]	Reynolds Number Re -
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

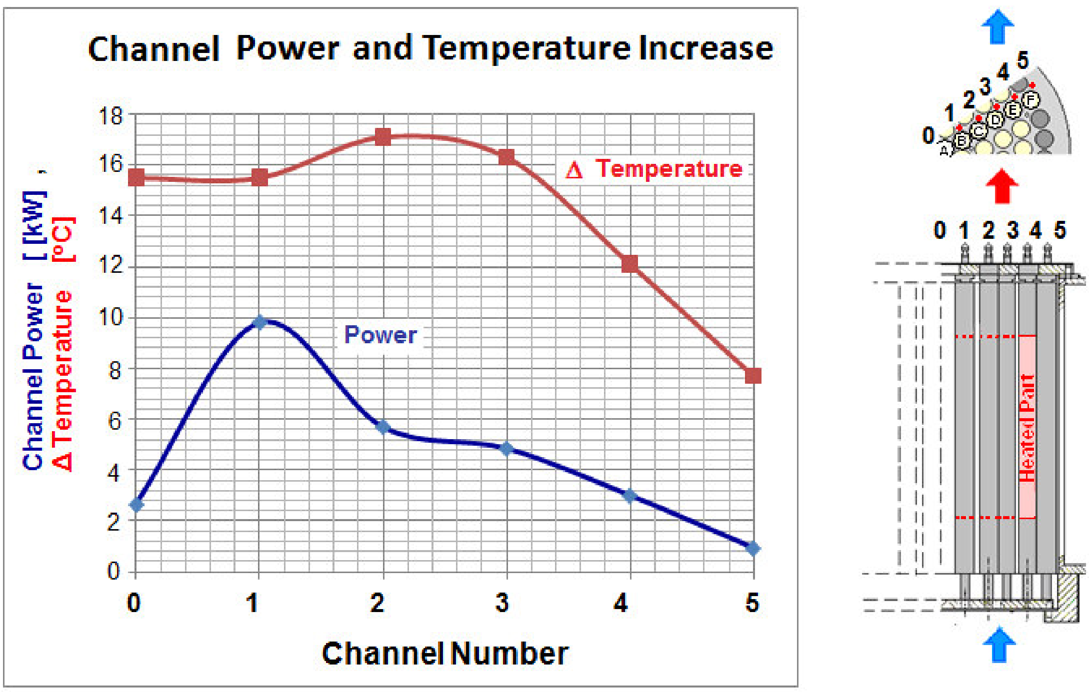

Figure 10. Power and temperature increase in coolant channels at 265 kW.

As can be seen in Figures 11 and 12 the velocity and mass flux in each channel are proportional to power dissipated in the channel.

[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 °C.

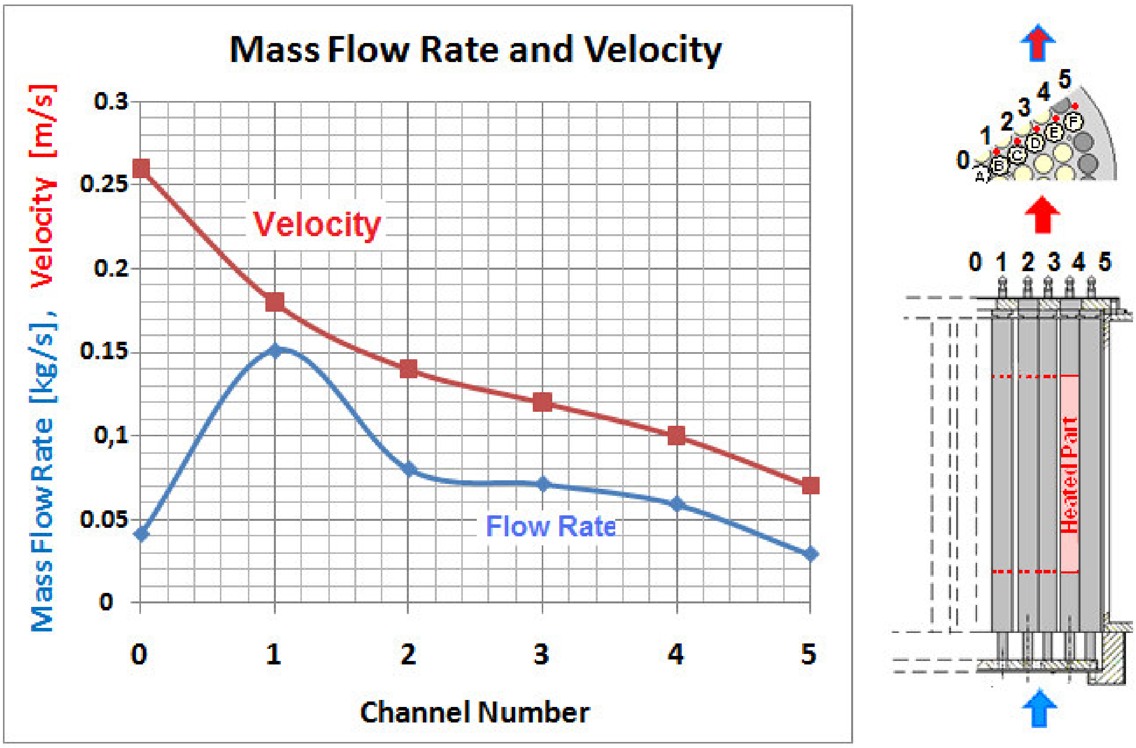

Figure 11. Mass flow rate and velocity in coolant channels at 265 kW.

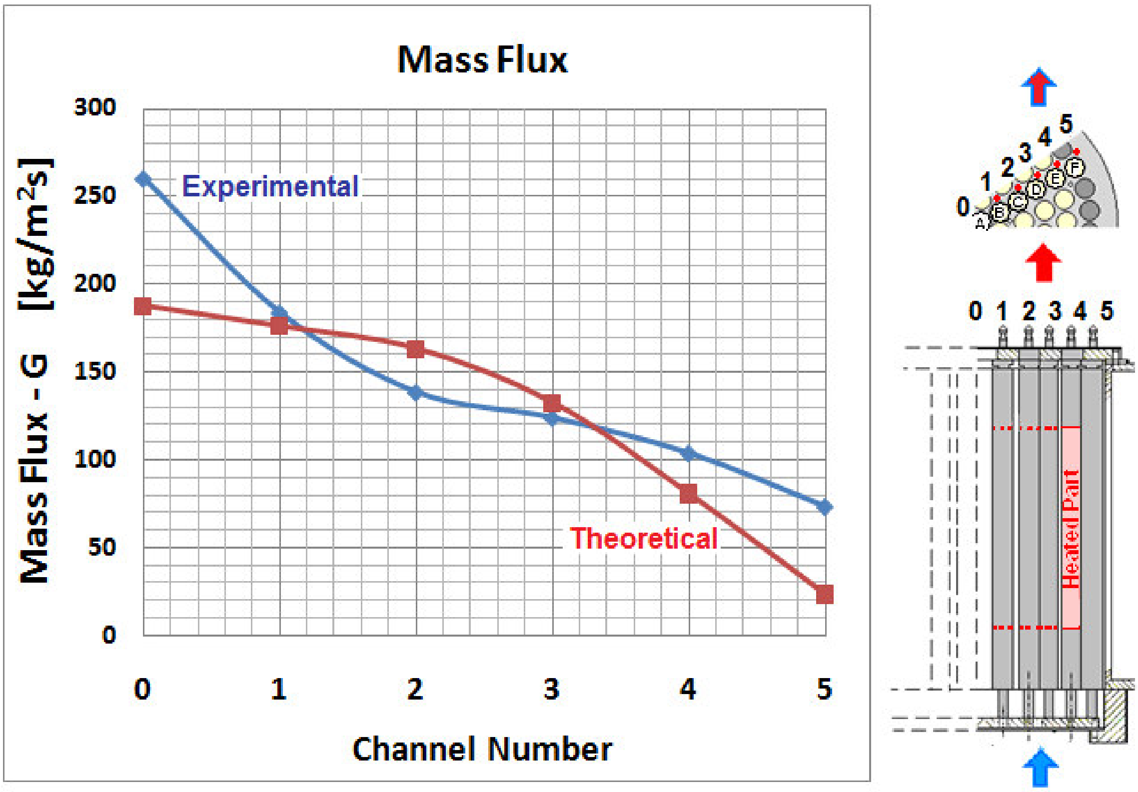

Figure 12. Mass flux in coolant channels at 265 kW.

5. CONCLUSIONS

Experiments to understand the behavior of the nuclear reactors operational parameters allow improve model predictions, contributing to their safety. Developments and innovations used for research reactors can be later applied to larger power reactors. Their relatively low cost allows research reactors to provide an excellent testing ground for the reactors of tomorrow. The experiments described here confirm the efficiency of 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 results can be considered as typical of pool-type research reactor.

The experimental temperature profile along the coolant channel 1 (Fig. 9) is different from that predicted from the theoretical model using the PANTERA Code [7]. 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 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.

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. The experimental data also provides information, which allows the computation of other parameters, such as the fuel cladding heat transfer coefficient [12]. 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 [13].

ACKNOWLEDGMENTS

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