POWER MEASURE CHANNELS OF THE IPR-R1 TRIGA RESEARCH NUCLEAR REACTOR BY THERMAL METHODS

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Abstract. *The IPR-R1 Research Nuclear Reactor at the Nuclear Technology Development Centre – CDTN, in Belo Horizonte, is a TRIGA Mark I type reactor. The IPR-R1 is a pool reactor, and the fuel elements at the core are cooled by water natural convection. The heat removal capacity of this process is great enough for safety reasons at the current maximum 250 kW power levels of the reactor. However, a 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. Power monitoring of nuclear reactors is always done by means of neutronics instruments (neutron flux measurement). In the IPR-R1 the power is measured by four nuclear channels. This work presents the results and methodology for the monitoring the power of this reactor by thermal processes. An improved method for thermal measuring channel is described using the fuel element and the water pool temperatures. Another thermal measuring channel consisted in the steady-state energy balance of the primary and secondary cooling loops of the reactor. For this balance, the inlet and outlet temperatures and the water flow of the cooling loops are measured.*

Keywords: *nuclear fuel element, nuclear reactor, thermal power, TRIGA reactor, instrumented fuel element*

1. INTRODUCTION

The 250 kW IPR-R1 TRIGA Reactor (Training, Research, Isotopes, General Atomic), of the Nuclear Technology Development Center (CDTN), is a pool type reactor cooled by natural circulation of light water. The reactor fuel is an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U. Figures 1, 2 and 3 show two photographs and two drawings of the reactor pool and core. The reactor core has 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented with three chromel/alumel thermocouples along its centerline. This instrumented fuel element was put in the reactor core in order to evaluate the thermal hydraulic performance of the IPR-R1 reactor under steady-state condition (Mesquita, 2005). Fuel temperatures were measured in various locations throughout the core with the use of the instrumented fuel element at different power levels. Three new processes for reactor power measurement by thermal ways were developed as a result of the experiments. These process make it possible on-line or off-line evaluation of the reactor power and the analysis of its behavior.

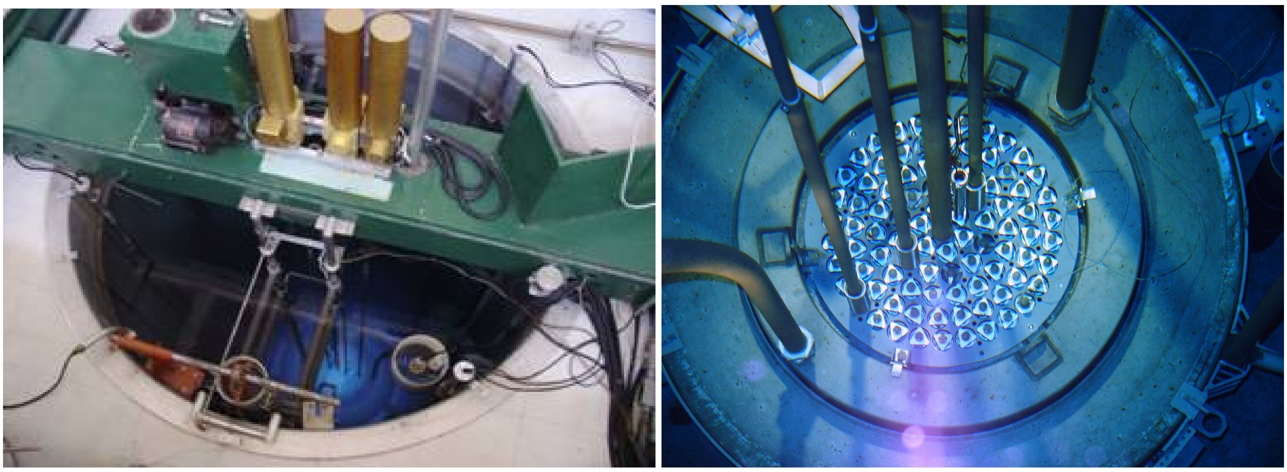


Figure 1. Top view of the pool and core of the IPR-R1 TRIGA Research Nuclear Reactor

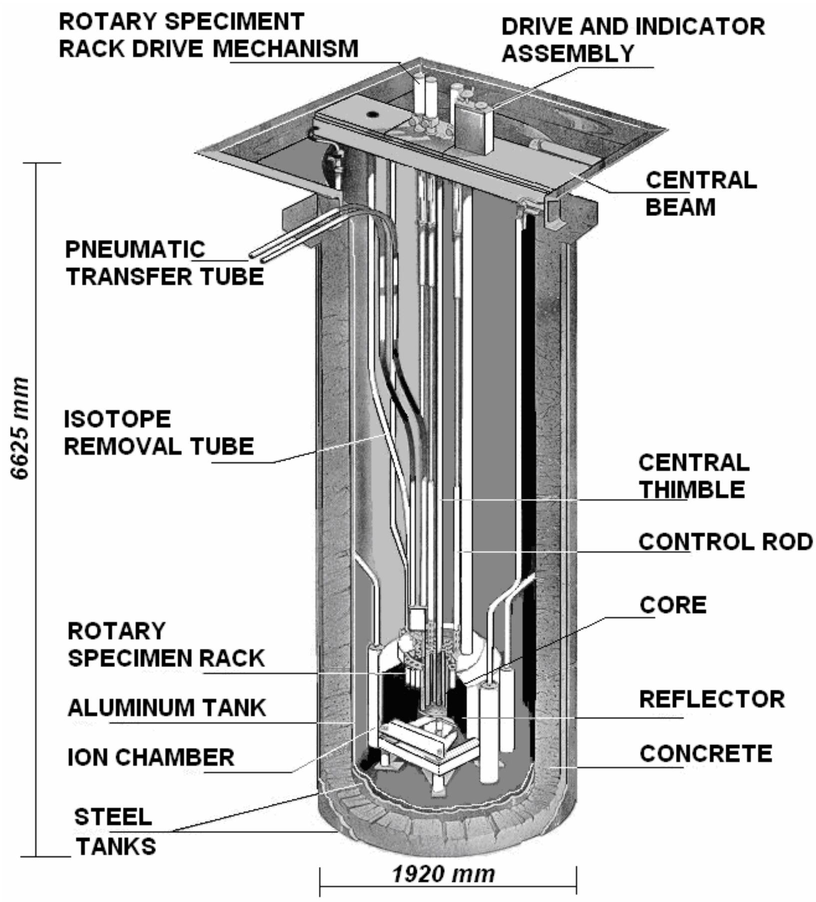


Figure 2: The pool of the IPR-R1 TRIGA Nuclear Reactor

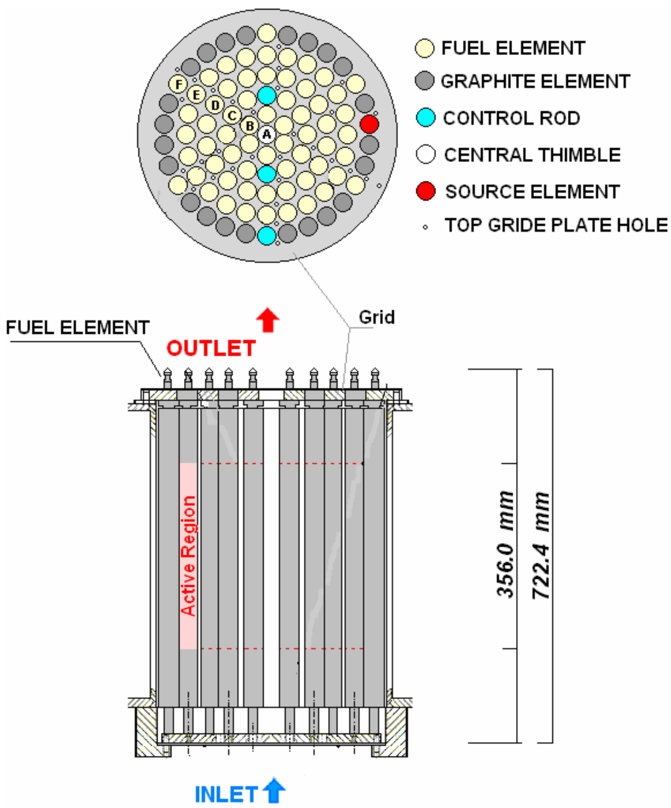


Figure 3: The core of the IPR-R1 TRIGA Nuclear Reactor

2. THE POWER MEASURING CHANNELS USING THE NEUTRONIC METHODS

Power monitoring of nuclear reactors is always done by means of nuclear detectors, which are calibrated by thermal methods. In the IPR-R1 Reactor four neutron-sensitive chambers are mounted around the reactor core for flux measurement. The type of chamber used and its position with respect to the core determine the range of neutron flux measured, as described bellow:

- The departure channel consists of a fission counter with a pulse amplifier that feeds a logarithmic count rate circuit and gives useful power indication from the neutron source level to a few watts.
- The logarithmic channel consists of a compensated ion chamber feeding a logarithmic (log n) amplifier and recorder and a period amplifier, which gives a logarithmic power indication on a recorder from less then 0.1 W to full power.
- The linear channel consists of a compensated ion chamber feeding a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder.
- The percent channel consists of an uncompensated ion chamber feeding a power level monitor circuit and meter, which is calibrated in percentage of full power.

The nuclear instrumentation is used to detect neutrons when sub-critical multiplication occurs during the reactor starp-up, and after achieving the criticality the variation of neutron flux, to obtain the automatic control of reactivity for maintaining a stable power level.

Unfortunately, the ionization chamber neutron detector measures the flux of neutrons thermalized in the vicinity of the detector. This signal is not always proportional to the integral neutron flux in the core and consequently to the core power. Besides the response of a single nuclear detector is sensitive to the changes in the core configuration, particularly to the control rod position. This is important in the TRIGA reactor, which do not have distributed absorbers for reactivity control and maintaining criticality is by insertion of control rods (Zagar *et al*., 1999).

3. THE POWER MEASURING CHANNEL USING THERMAL PROCESS

3.1. Power measuring by the thermal balance

The reactor core is cooled by natural convection flow of demineralized water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere by primary cooling loop, secondary cooling and a cooling tower (Fig. 4). Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total dissipated power is determined by making the thermal balance of the inlet and outlet cooling water that flows through in the primary and secondary loops and the calculation of the heat losses. These losses represent a very small fraction of the total power (about 1.5% of total) (Mesquita *et al*., 2005).

The inlet and outlet temperatures are measured by four platinum resistance thermometers (PT-100) positioned at the inlet and at the outlet pipes of the primary and secondary cooling loops. The flow-rate in the primary is measured by the differential pressure on a orifice plate and a differential pressure transmitter, in the secondary the flow is measured by a flowmeter. The pressure transmitter was calibrated and an adjusted equation was obtained and added to the data acquisition system. The temperature measuring lines were calibrated as a whole, including thermometers, cables, data acquisition cards and computer. The adjusted equations were also added to the data acquisition system (Mesquita and Rezende, 2004).

The power dissipated at the cooling loop will be closer to the reactor power the closer the water temperature in the reactor pool is to the environment temperature. The steady state is reached after some hours of reactor operation, then the power dissipated in the cooling system added with the losses is equal the core power. The uncertainty in the power measurement considered all the uncertainty propagation from primary parameters, according to the methodology described by Coleman and Steele (1999). The uncertainty is calculated only for the power of the primary loop because it is now the standard power measuring system for the IPR-R1 Reactor (CDTN/CNEN, 2007).

The thermal power dissipated in the primary and secondary loops were obtained through a thermal balance given by the following equation:

$$q_{cool} = \dot{m} \, . \, c_p \, . \, \Delta T \quad . \qquad (1)$$

Where q_{cool} is the thermal power dissipated in each loop [kW], $\dot{m}$ is the flow rate of the coolant water in the loop [kg/s] , c_p is the specific heat of the coolant [kJ/kg^{o}C], and ΔT is the difference between the temperatures at the inlet and the outlet of the loop [^{o}C].

The data acquisition computer program calculates the power dissipated in the cooling loop with the collected data being used in Equation (1), and with the $\dot{m}$ and c_p values corrected as function of the temperature of the coolant (Miller, 1989).

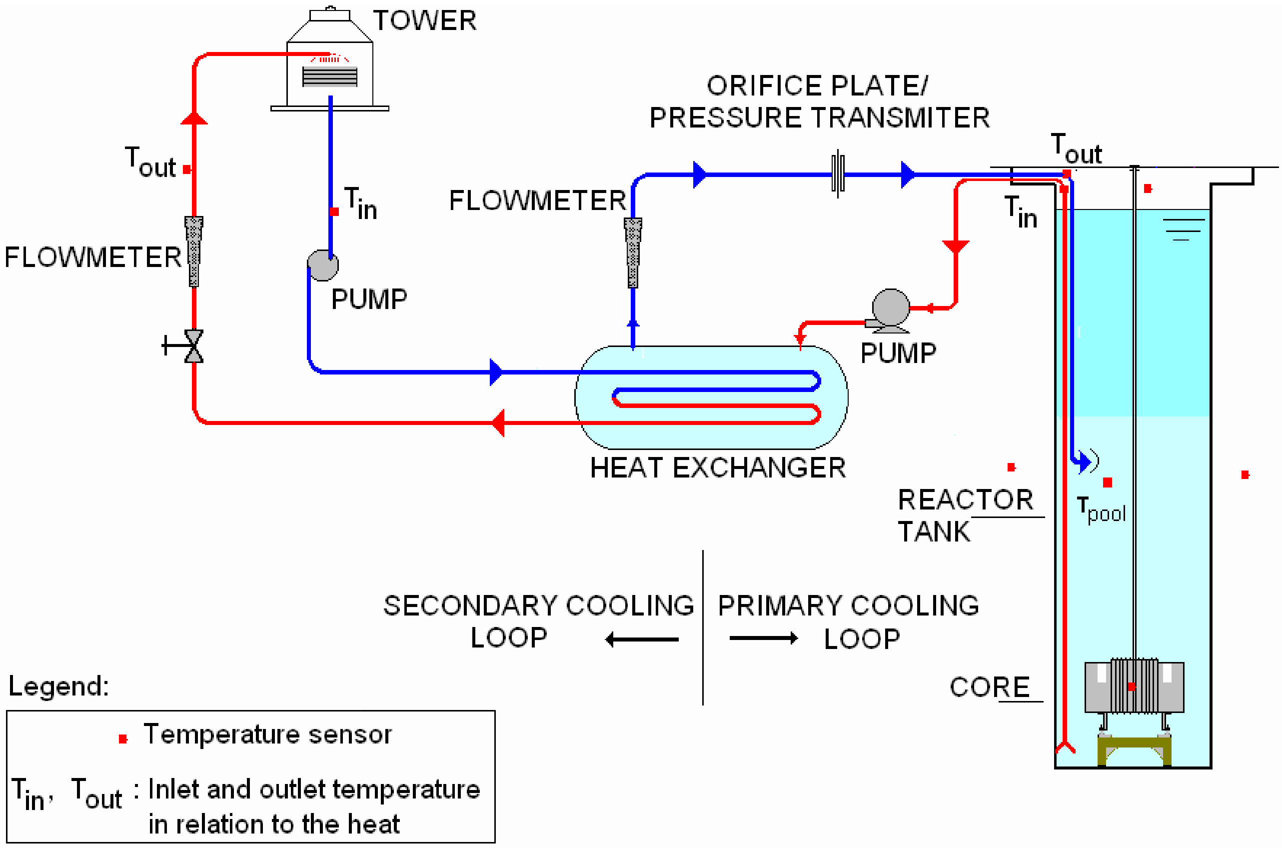


Figure 4. The reactor cooling system

To calculated the heat losses, one resistance thermometer (PT-100) were positioned inside the pool to measure the water pool temperature. A type K thermocouple was placed just above the pool surface to measure the air temperature at the reactor room. Two type K thermocouples were distributed around the pool, in holes in the reactor room floor, to measure the soil temperature. The core of the TRIGA Mark I IPR-R1 Nuclear Reactor is placed below the room floor, in the bottom of a cylindrical pool, 6.625 m deep and 1.92 m in diameter, whose upper surface is 25 cm below the level of the floor. The reactor pool transfers heat to the environment by conduction to the soil, through the lateral walls and through the bottom of the pool, and by convection and evaporation to the air at the reactor room, through the upper surface. All this losses is calculated by the data acquisition system as described by Mesquita *et al.* (2005).

Figure 5 shows the power evolution in the primary and secondary loops during one reactor operation. Table 1 presents the results of the thermal balance in this operation and some experimental data.

Table 1. IPR-R1 TRIGA Reactor thermal balance.

Average primary loop coolant flow rate	32.7 ± 0.4 m^3/h	
Average primary loop coolant inlet temperature	41.7 ± 0.3 ^{o}C	
Average primary loop coolant outlet temperature	34.8 ± 0.3 ^{o}C	
Power dissipated in the primary loop	261 kW	
Thermal losses from the reactor pool	3.8 kW	
Reactor thermal power	**265 kW**	
Standard deviation	3.7 kW	
Uncertainty in the measure of the reactor thermal power	±19 kW	(±7.2%)
Power dissipated in the secondary loop	248 kW	

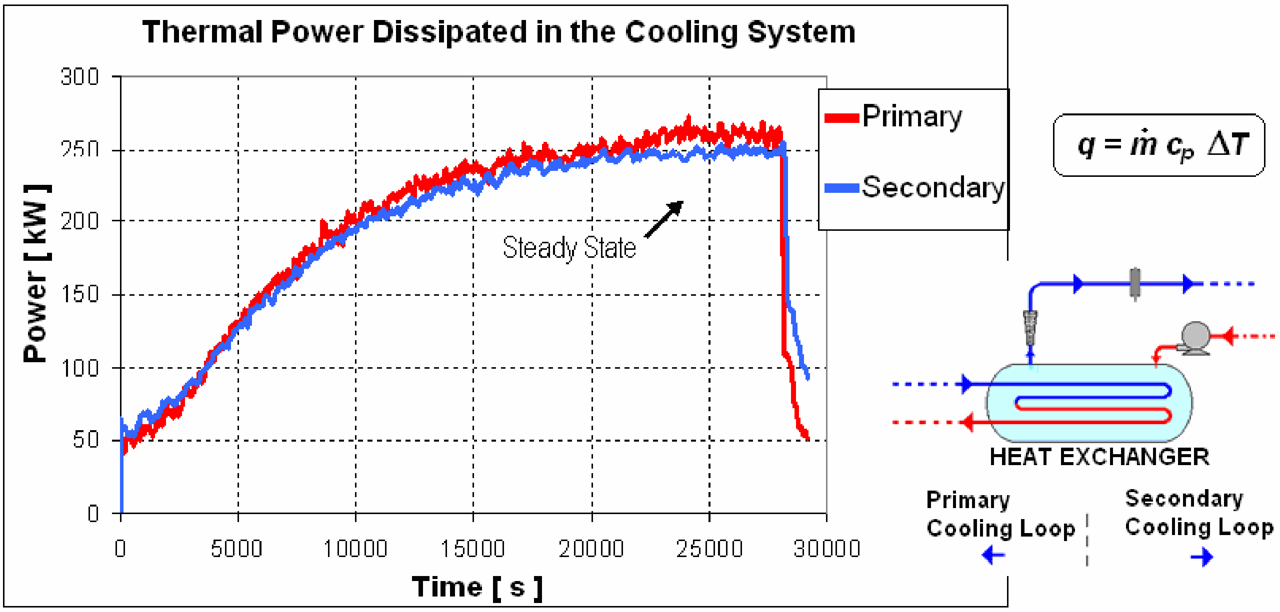
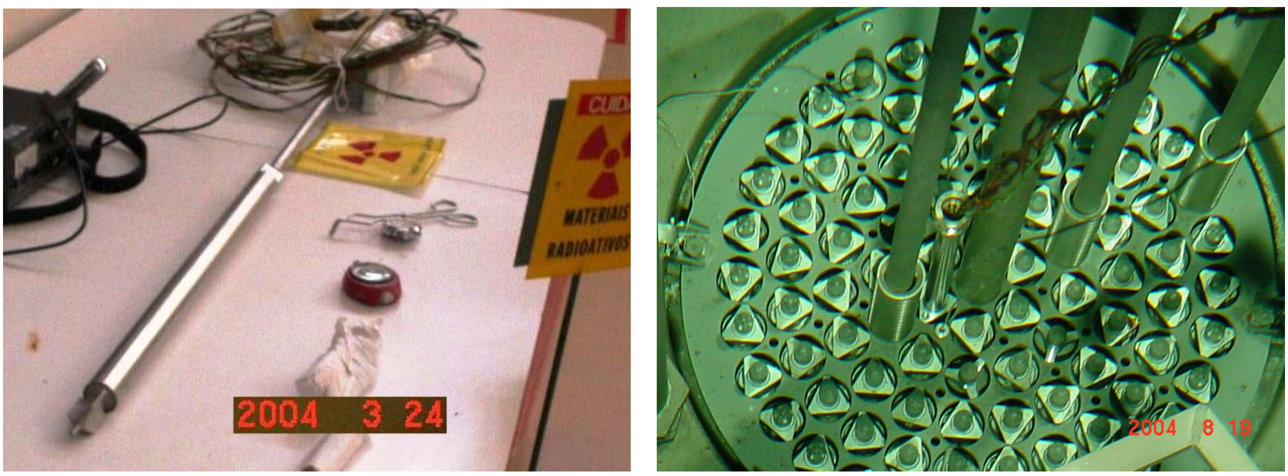


Figure 5. Thermal power evolution in the cooling system

3.2 Power measuring by fuel and pool temperature

One instrumented fuel element was put in the core for the experiments to evaluate the thermal hydraulic performance of the IPR-R1 Reactor (Mesquita, 2005). The instrumented fuel is identical to standard fuel elements but it is equipped with three chromel-alumel thermocouples, embedded in the zirconium centerline pin. The sensitive tips of the thermocouples are located one at the center of the fuel section and the other two 25.4 mm above, and 25.4 mm below the center. Figure 6 shows: a) The instrumented fuel element before it is put in the core; and, b) the core upper view with the instrumented fuel element in ring B. Table 2 presents some information about the instrumented fuel element (Gulf General Atomic, 1972). Figure 7 shows the diagram and design of this fuel element.



a) b)

Figure 6. a) The instrumented fuel element. b) core upper view with the instrumented fuel element in ring B

Table 2. Instrumented fuel element data.

Parameter	Value
Heated length	38.1 cm
External diameter	3.76 cm
External fuel element active area	450.05 cm^2
External fuel area (U-$ZrH_{1.6}$)	434.49 cm^2
Fuel element active volume	423.05 cm^3
Fuel volume (U-$ZrH_{1.6}$)	394.30 cm^3
Power (total in the core = 265 kW)	4.518 kW

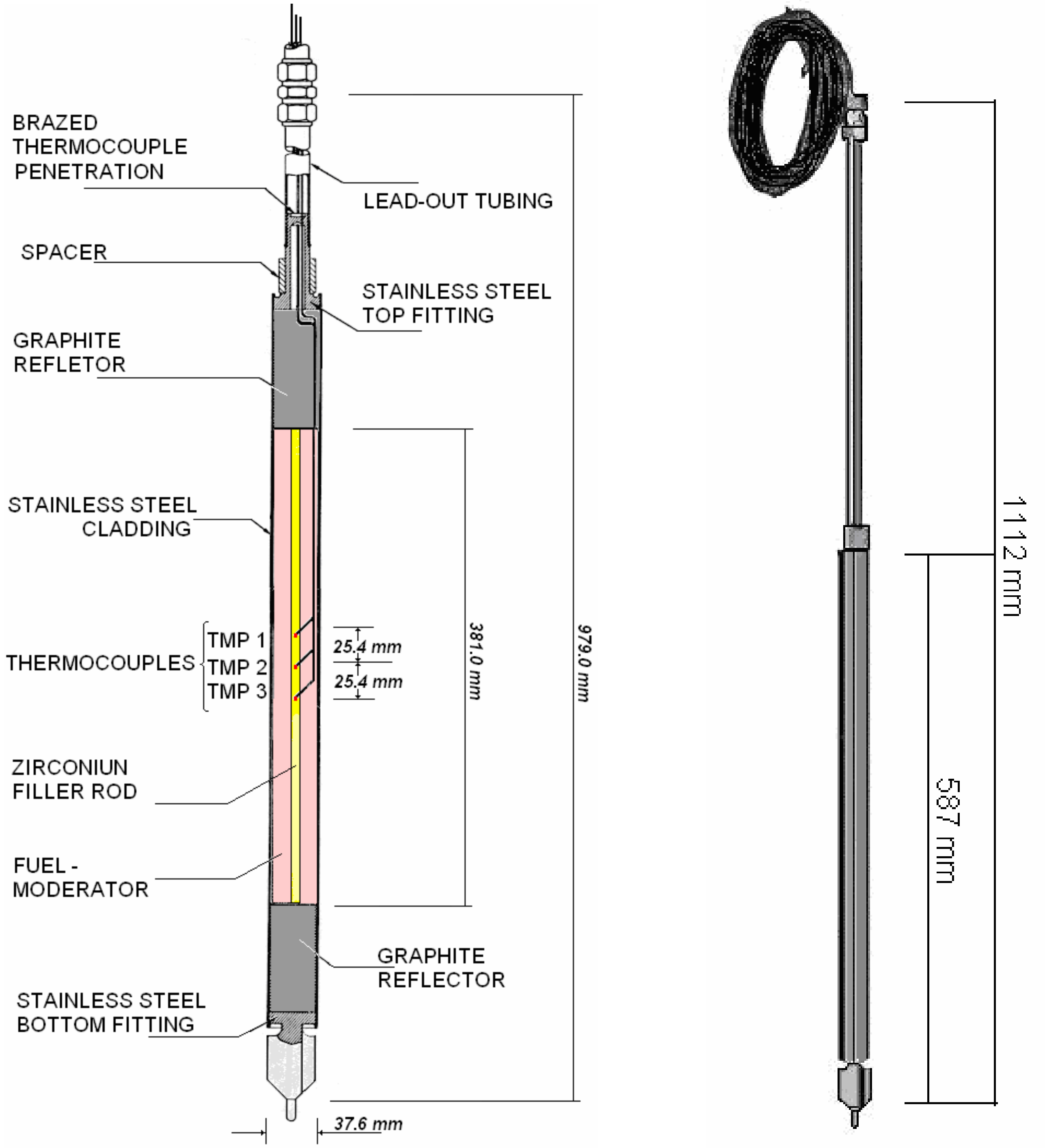


Figure 7. Diagram of the instrumented fuel element

During the experiments it was observed that the temperature difference between fuel element and the pool water bellow the reactor core (primary loop inlet temperature) do not change for the same power value as can been seen on Figure 8. With the instrumented fuel element in position B1 of the core (hottest fuel element), the power measured in linear channel (with the values corrected by the calibration results) was plotted as a function of the temperature difference between the fuel and the primary loop inlet temperature. The following polynomial expression was obtained that relates the two values:

$$q = 2 \,.\, 10^{-5} (\Delta T)^3 - 0.0045(\Delta T)^2 + 0.7666\,\Delta T - 2.4475 \quad . \qquad (2)$$

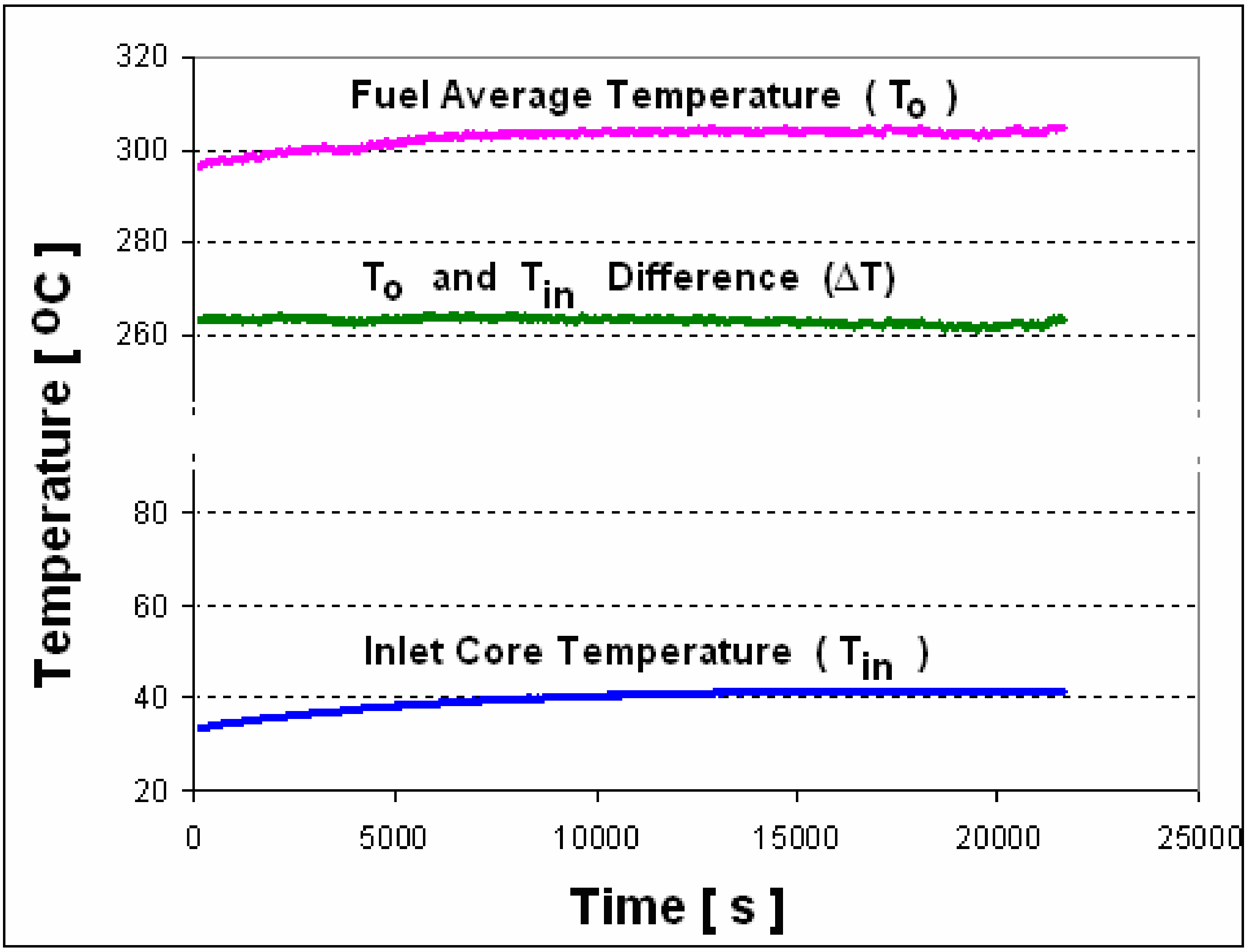


Figure 8. Fuel and inlet core water temperatures evolution

Where q is the calibrated reactor thermal power, in [kW] and ΔT is the difference between the average fuel temperature and the primary loop inlet temperature, in [°C].

The determination coefficient obtained was one (R^2 = 1). The Equation (2) was included in the data acquisition system and this new power measurement channel is available for the IPR-R1 TRIGA Reactor. After the experiments the instrumented fuel element was maintained in position B1 of the core to monitor the reactor power and core temperature in all reactor operation. Figure 9 shows reactor power measuring results using the linear neutron channel and the temperature difference channel method. It can be seen a delay in the second channel response due to the system thermal inertia.

The fuel temperature limit defined in the IPR-R1 TRIGA Reactor Accidents Report (CDTN/CNEN, 2007) during steady state operation is 550 °C. A power operation limit over 1 MW was found based in this temperature and using Equation (2).

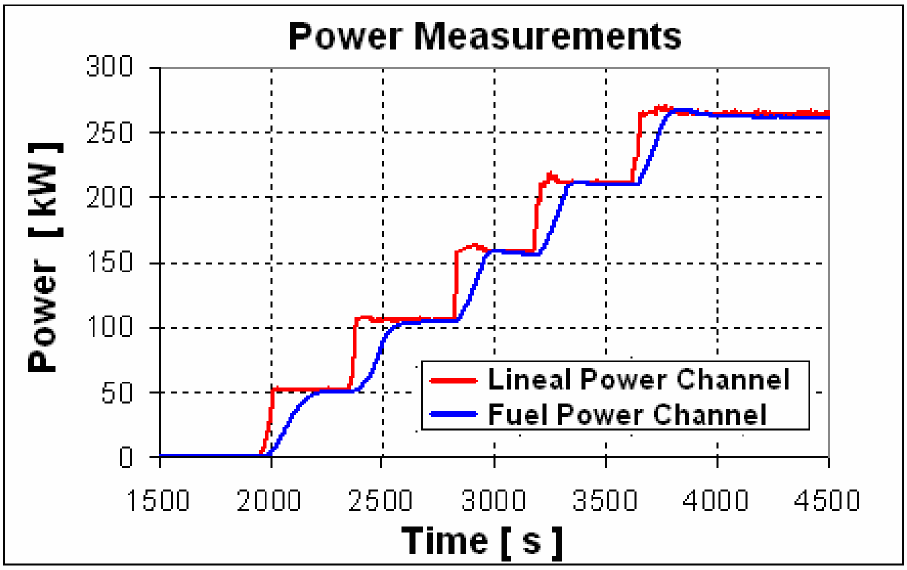


Figure 9. Reactor power measured by neutron channel and by fuel element temperature

Figure 10 shows one of the video-screens displays of the digital monitoring system computer that consolidates information, in real time, of the reactor power status. This screen monitors the power measured by the neutronics channels and by the three new thermal channels (Mesquita and Rezende, 2004).

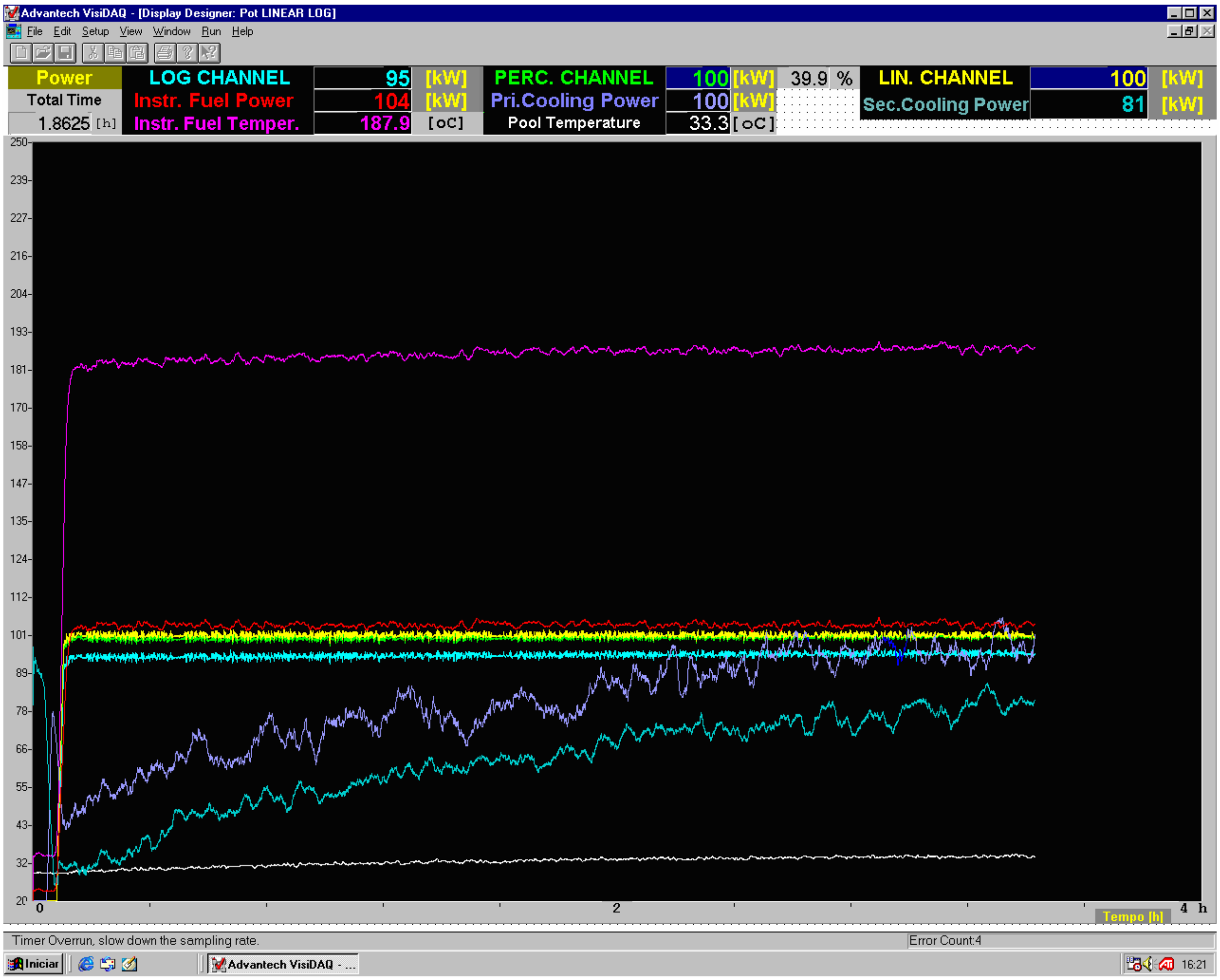


Figure 10. Power monitoring on the screen of the data acquisition system

4. CONCLUSION

The knowledge of the reactor thermal power is very important for precise neutron flux and fuel element burnup calculations. The burnup is linearly dependent on the reactor thermal power and its accuracy is important to the determination of the mass of burned ^{235}U, fission products, fuel element activity, decay heat power generation and radiotoxicity. The thermal balance method presented in this report is now the standard methodology used for the IPR-R1 TRIGA Reactor power calibration (CDTN/CNEN, 2007). The uncertainty value obtained does not differ significantly from another thermal calibration processes described in technical literature (Zagar *et al*., 1999).

The heat balance and fuel temperature methods are accurate, but impractical methods for monitoring the instantaneous reactor power level, particularly during transients. For transients the power is monitored by the nuclear detectors, which are calibrated by the thermal balance method. On the other hand, the response of one nuclear detector is sensitive to the changes in the core configuration, mainly to the control rod position. This is important in research reactors, which do not have distributed absorbers for reactivity control and the normal mode of maintaining criticality is by insertion of control rods. The heating of the thermocouple due the gamma ray is negligible because the small mass and good thermal radiation equilibrium with the surrounding fuel.

5. ACKNOWLEDGEMENTS

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6. REFERENCES

CDTN/CNEN, 2007, “Relatório de Análise de Segurança do Reator TRIGA IPR-R1”, Belo Horizonte, Brazil, 321p.

Coleman, H.C.; Steele, W.G., 1999. “Experimentation and Uncertainty Analysis for Engineers”. 2nd. Ed. John Wiley & Sons, Inc. New York, USA. 275p.

Gulf General Atomic, 1972. “15” SST Fuel Element Assembly Instrumented Core”. San Diego, CA, USA. Drawing Number TOS210J220.

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

Mesquita, A.Z.; Rezende, H.C., 2004, “Data Acquisition System for TRIGA Mark I Nuclear Research Reactor of CDTN”. Proceedings of the America Nuclear Energy Symposium (ANES 2004), Miami Beach, Flórida, USA. America Nuclear Energy.

Mesquita, A.Z.; Rezende, H.C.; Tambourgi, E.B., 2005. “Power Calibration of the IPR-R1 TRIGA Reactor”. Revista Iberoamericana de Ingeniería Mecánica. Madrid, Spain. Vol. 7 N.º 1, March.

Miller, R.W., 1989. “Flow Measurement Engineering Handbook”, 2nd. Ed. New York, McGraw-Hill Publishing Company. New York, USA. p. E19-E21.

Zagar, T; Ravnik, M; Persic, 1999. A. “Analysis of Reactor Thermal Power Calibration Method”. Proceedings of the International Conference Nuclear Energy in Central Europe’99, Portoroz, Slovenia.. p 91-p 98.

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