

EXPERIMENTAL INVESTIGATION OF THE TEMPERATURE DISTRIBUTION ON THE TRIGA-IPR-R1 RESEARCH NUCLEAR REACTOR AND DEVELOPMENT OF POWER MEASURING CHANNELS BY THERMAL PROCESSES

Amir Zacarias Mesquita

Centro de Desenvolvimento da Tecnologia Nuclear – CDTN/CNEN
amir@cdtn.br

Hugo Cesar Resende

Centro de Desenvolvimento da Tecnologia Nuclear – CDTN/CNEN
hcr@cdtn.br

Antônio Carlos Lopes da Costa

Centro de Desenvolvimento da Tecnologia Nuclear – CDTN/CNEN
aclc@cdtn.br

Abstract. *This work presents the experiments and theoretical analysis to determine the temperature distribution in the TRIGA-IPR-R1 Research Nuclear Reactor. It was also presented a methodology developed for calibration and monitoring of the reactor thermal power. This methodology allowed to add others power measuring channels to the reactor by using thermal processes. The experimental results were compared with theoretical calculations and with data obtained from technical literature. A data acquisition and processing system and a software were developed to help the investigation. This system allows on line monitoring and registration of all the reactor operational parameters.*

Keywords: *nuclear fuel, temperature, thermal power calibration, TRIGA research reactors.*

1. Introduction

The TRIGA IPR-R1 Research Nuclear Reactor (Fig. 1) of the *Nuclear Technology Development Center– CDTN*, in Belo Horizonte (Brazil), is a TRIGA Mark I type reactor. The core is cooled with demineralized light water and the reactor utilizes rods of uranium-zirconium-hydride homogeneously mixed as fuel elements. The uranium enrichment level is 20% in uranium 235. The reactor was designed for training, research and radioisotope production. TRIGA (Training, Research, Isotopes, General Atomic) reactors are inherently safe due to its large, prompt and negative temperature coefficient. It means that an increase in the reactor power, which produces a temperature rise in the fuel-moderator mixture, also produces a negative reactivity and reduces the power increasing rate what makes the power to stabilize in a safe level. Another safety characteristics of the TRIGA reactors are its passive heat removing system and its high retention of fission products in the fuel in case of cladding failure.

In November 2004 the TRIGA-IPR-R1 has completed 44 years in operation. Initially its nominal thermal power was 30 kW. In 1979 this power was increased to 100 kW by adding new fuel elements to the reactor. Recently some more fuel elements were added to the core increasing the power to 250 kW. Like others TRIGA it is a pool type reactor with a free circulation core cooling system. Although the large number of experiments carried out with the reactor, mainly on neutron activation analysis, there is not many data on its thermal-hydraulics processes whether experimental or theoretical. So, a number of experiments were carried out with temperature measurements inside the fuel element, in the reactor core and along the reactor pool. During these experiments the reactor was set in many different power levels. These experiments are part of the CDTN/CNEN research program, and have the main objective of commissioning the TRIGA-IPR-R1 reactor for routine operation at 250 kW.

Reliable measurements are essential to a correct prediction of the reactor thermal-hydraulic behavior. The most important parameter for this prediction is the thermal power produced and removed from the fuel elements. The first scope for this work was then the development of a methodology for the reactor power calibration. Other results of this project are three more channels for the reactor power measuring. The first channel measure the power removed through the primary cooling loop. The second one measure the power removed through the secondary cooling loop. Finally, the power obtained by measuring the temperature at the center of an instrumented fuel element, which was positioned in the reactor core. This last measuring channel can also be used as a safety device by automatically turning off (scram) the reactor if the fuel temperature rises over a safety limit.

An operational computer program and a data acquisition and signal processing system were developed for the reactor to allow on line monitoring the operational parameters, plotting graphics on the computer screen. Parameters like the thermal power in the cooling loops and the fuel temperature that were not used to be measured can now be seen in graphics or as numerical values.

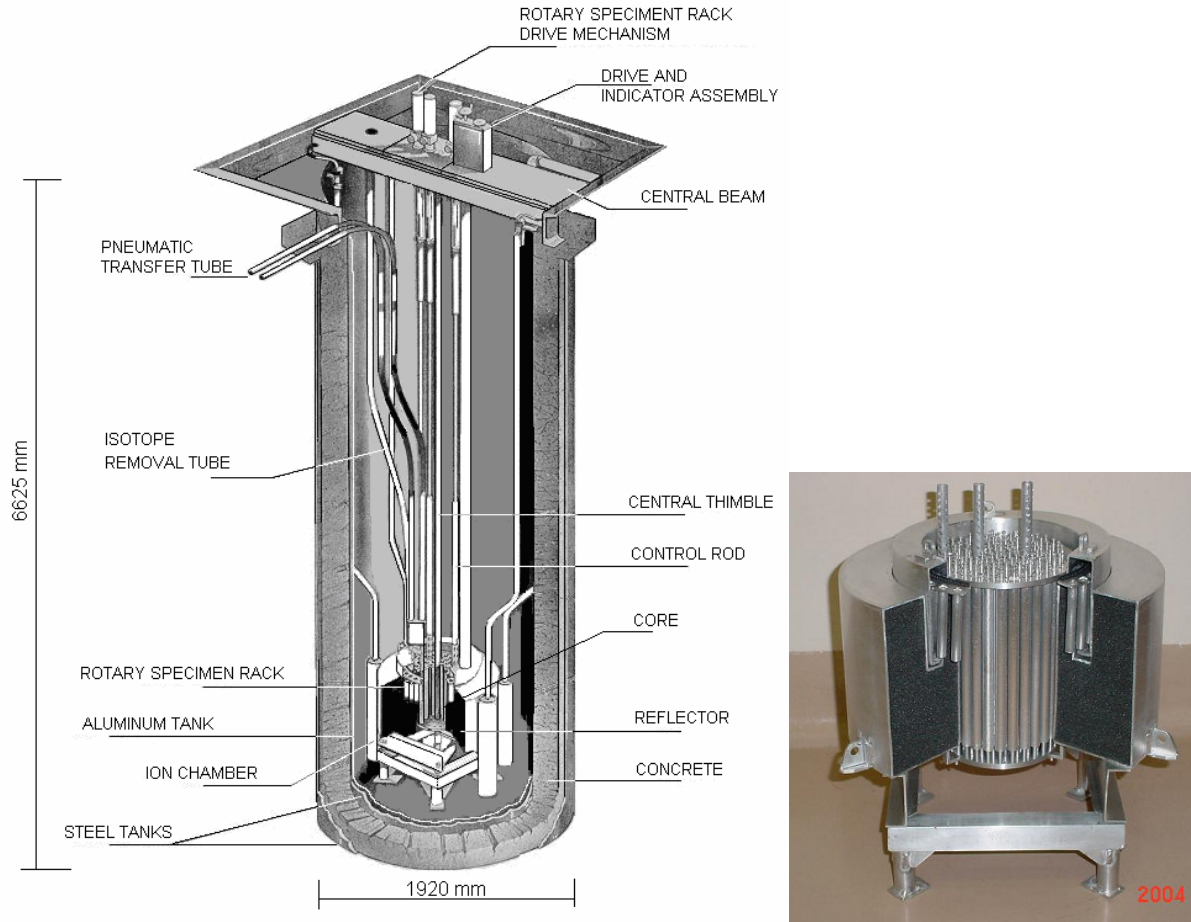


Figure 1. Cutaway drawing of the TRIGA IPR-R1 Nuclear Reactor and core model

2. Methodology

The original fuel element at the position B1 (Fig. 2) was removed from the reactor core and replaced by a new fuel element instrumented with three type k thermocouples positioned in the fuel center line (Fig. 3). The position B1 is the place with the greater power density in the core, according to the neutron calculations (Dalle, 2003). Two additional type k thermocouples were placed in the reactor core in two channels close to the position B1. Nine thermocouples and one platinum resistance thermometer were used to monitor the reactor pool temperature. The thermocouples were positioned in a vertical aluminum probe with the first thermocouple 143 mm above the core top grid plate. The reactor power was increased from 50 kW to 250 kW in 50 kW steps (measured by the neutron linear channel). The reactor thermal power calibration was carried out at 250 kW as described in the next section. After finding the power reference value, the instrumented fuel element was replaced to new positions in each one of the fuel rings from B to F. At the same way, the two flow channel thermocouples probes were replaced to channels close to the instrumented fuel element new positions. At each position the power was increased in steps of 50 kW from 50 kW to 250 kW. After these experiments the temperature measuring probes were taken out from the reactor and the instrumented fuel element returned to position B1. The original fuel element that was at this position was positioned in the fuel-element storage rack in the reactor pool.

2.1 Thermal Power Calibration

The reactor thermal power calibration was based on the energy balance in the primary cooling system under steady state conditions by measuring the values of the inlet and outlet temperatures of the water and its flow rate. In these conditions the energy produced in the reactor core must be equal to the heat transferred through the cooling loop added to the thermal losses from the pool to the environment. These losses represent a very small fraction of the total power. The power dissipated at the cooling loop will be closer to the reactor power the closer the water temperature at the

reactor pool were of the environment temperature. It means that the reactor pool temperature must be set the closer of the soil temperature around the pool, and that the air temperature at the reactor room set the closer of the pool temperature. Therefore, it is important to obtain these conditions and also a stability of the pool temperature over a long period of time. This can be obtained only after some hours of reactor operation, better at night when the changes of the outside air temperature are smaller. The power dissipated in the secondary loop was also measured.

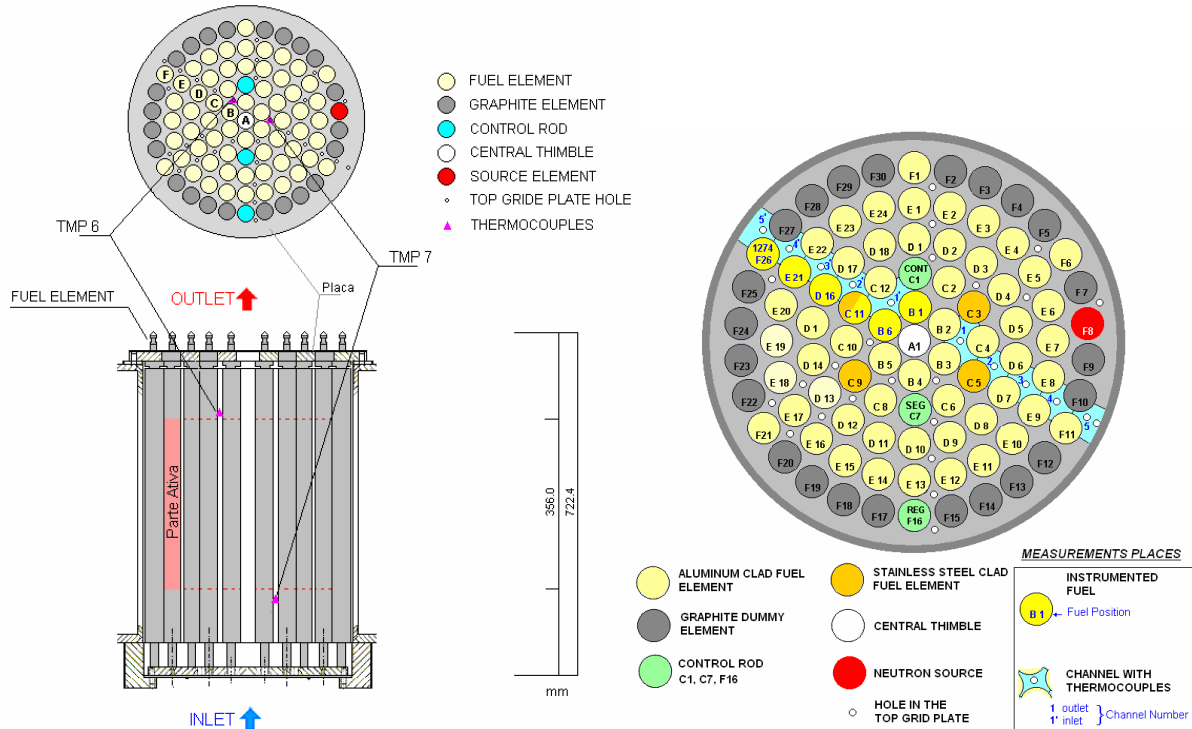


Figure 2. Thermocouple positions in the reactor core

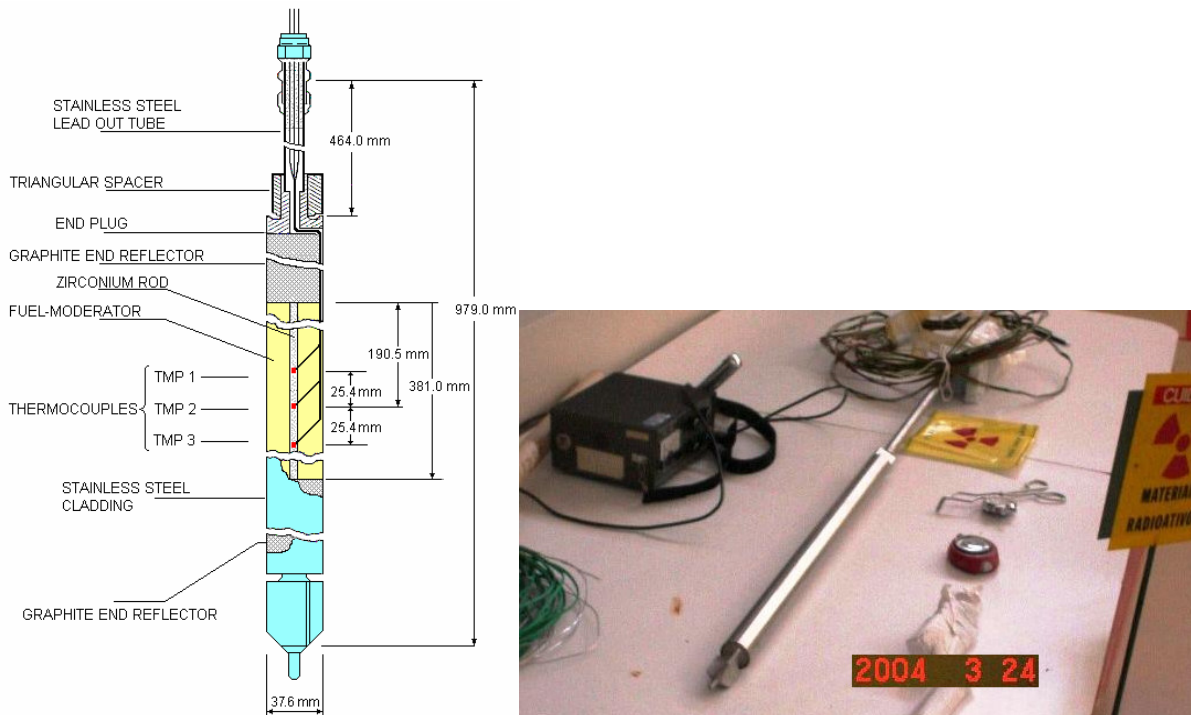


Figure 3. Instrumented fuel-moderator element

3. Results

The thermal power calibration is the first result presented, because all the following analysis were referred to it. The Microsoft Excel 2003 was used to process and present the experimental parameters and results.

3.1. Reactor Thermal Power Calibration

The experiment for the reactor thermal power calibration began at 15:30 hours and finished at about 23:30 hours. The reactor stayed critical for about eight hours with the neutron linear channel measuring 250 kW. The power transferred through the primary cooling loop was monitored during the full test and steady state occurs from 22:30h to 23:30h. Figure 4 shows the power evolution in the primary and secondary loops during the test. Table 1 presents the results and some calibration data. The uncertainty in the power measurement considered all the uncertainty propagation from primary parameters, according to the methodology described by Mesquita (2005).

3.2. Temperature Distribution in the Reactor Pool

The reactor operated during a period of about eight hours at a thermal power of 250 kW measured with the linear channel, and then steady state was obtained. Figure 5 shows the water temperatures evolution at the reactor pool and the flow channel inlet and outlet temperature, until the steady state had been obtained. The flowing channel used, for temperature measurements, was the one close to the fuel element position B1 (channel 1).

Figure 5 also shows that the thermocouples positioned 143 mm over the top grid plate (Inf 7) measure a temperature higher than all the another thermocouples positioned over the reactor core. It means that the chimney effect is not much high, less than 400 mm above the reactor core, in agreement with the experiments of Rao et al. (1988). Finally, the primary loop suction point at the pool bottom has the lowest temperature in the reactor pool.

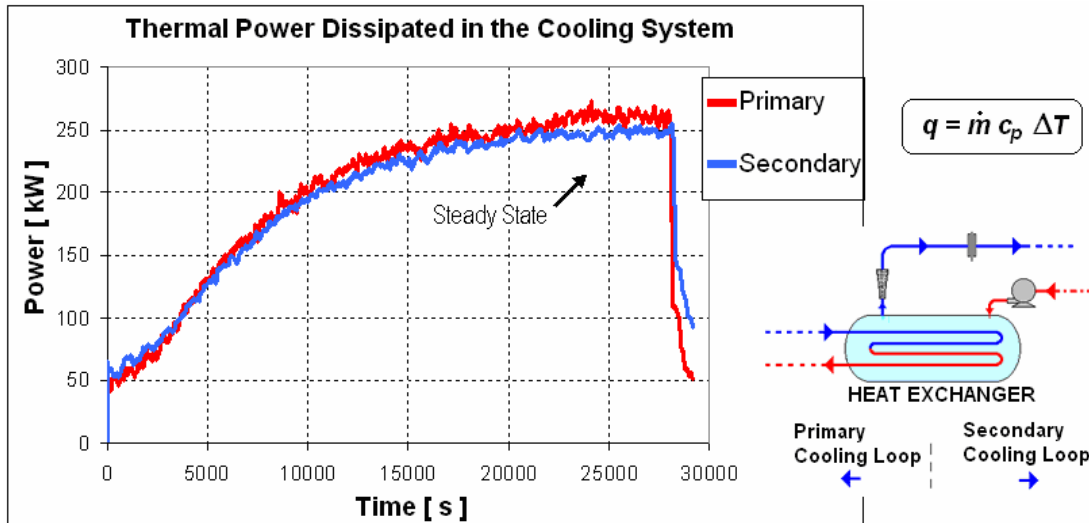


Figure 4. Thermal power evolution in the cooling system

Table 1. TRIGA IPR-R1 Reactor thermal power

Calibration date	August 19, 2004
Average inlet pressure	$32.7 \pm 0.4 \text{ m}^3/\text{h}$
Average inlet primary loop temperature	$41.7 \pm 0.3 \text{ }^\circ\text{C}$
Average outlet primary loop temperature	$34.8 \pm 0.3 \text{ }^\circ\text{C}$
Thermal power transferred through the primary loop	261 kW
Thermal losses from the reactor pool	3.8 kW
Reactor thermal power	265 kW
Standard deviation	3.7 kW
Average power uncertainty	$\pm 19 \text{ kW } (\pm 7.2\%)$
Heat power transferred through the secondary loop	248 kW

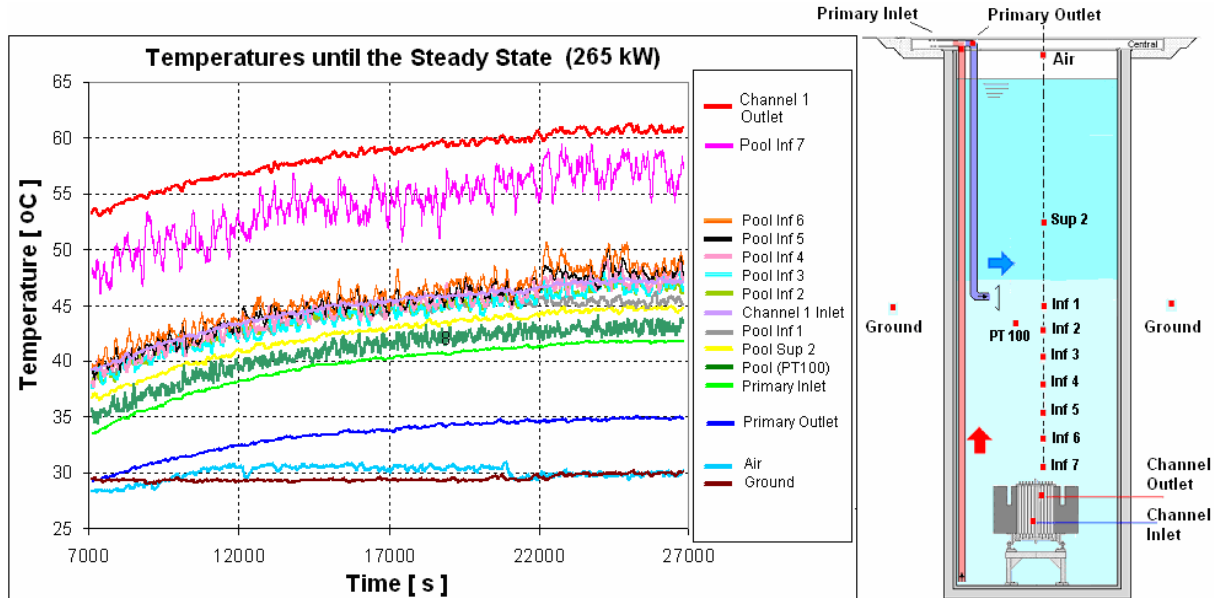


Figure 5. Temperatures evolution at 265 kW thermal power

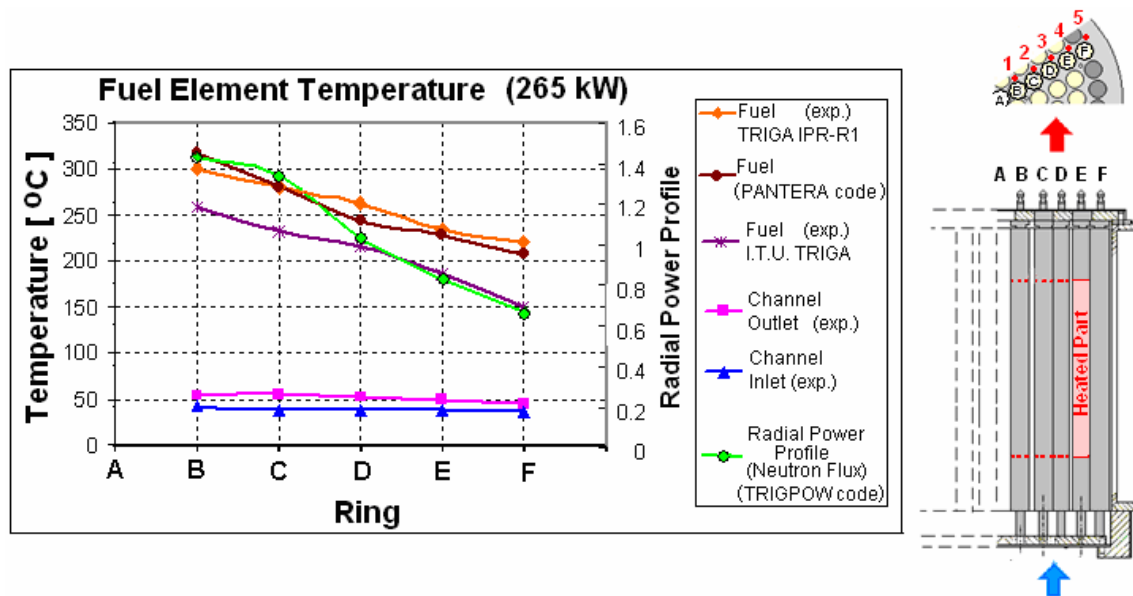


Figure 6. Fuel temperature radial profile at 265 kW thermal power

3.3. Temperature Distribution in the Core Rings

The instrumented fuel element position was changed during the experiments in such way that it was placed in all of the fuel rings at the reactor core, as shown in Fig. 2. The first position was the most central one at the ring B, the second at ring C until ring F. Experiments were carried out with the power changing from 50 kW to 250 kW in 50 kW steps for each position of the instrumented fuel element. Power was measured in the neutron linear channel, what means that measured 250 kW represent 265 kW of thermal power, in agreement with the calibration. Figure 6 gives the values obtained in each core ring for the average fuel temperature, the inlet and outlet temperatures at the flow channel closest to the instrumented fuel, and the experimental results obtained by Özkul and Durmayaz (2000) in the I.T.U. TRIGA Mark II Reactor at the Istanbul University. The radial power profile found with reactor physics calculations (Dalle, 2003) were plotted on the same graphic.

Figure 7 shows the measured radial core inlet and outlet temperature profiles and compares it with theoretical calculations obtained with PANTERA code. Figure 8 shows the temperatures measured with the instrumented fuel

element in each core ring as a function of the reactor thermal power. Figure 9 shows the temperatures measured in the core flow channels outlet for each core ring as a function of the reactor power.

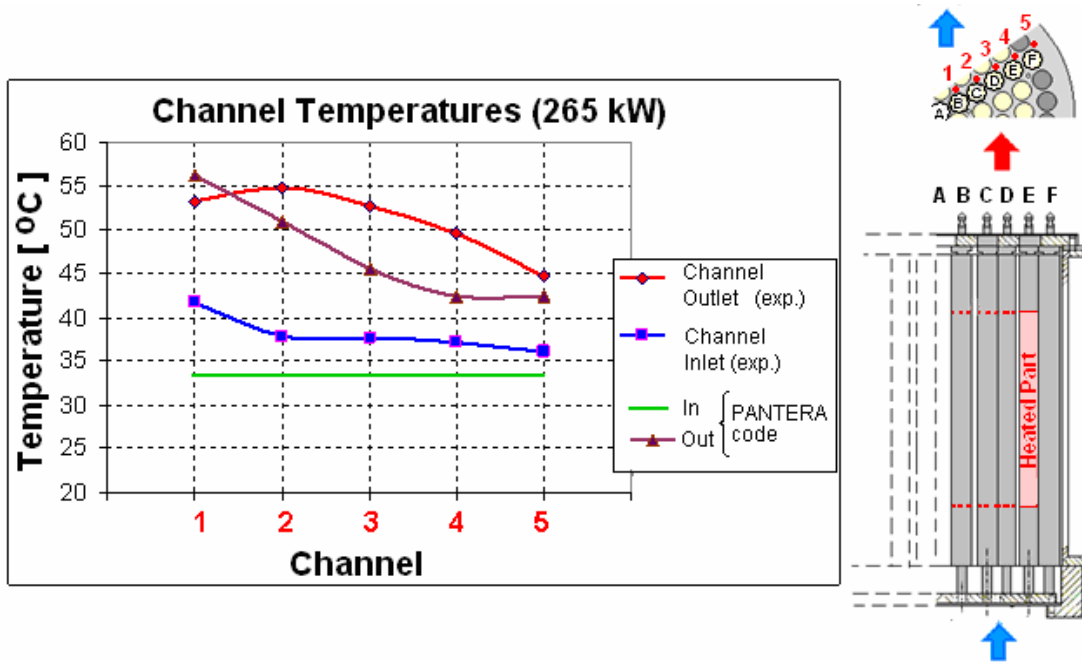


Figure 7. Temperature radial profile in the core channels at 265 kW thermal power

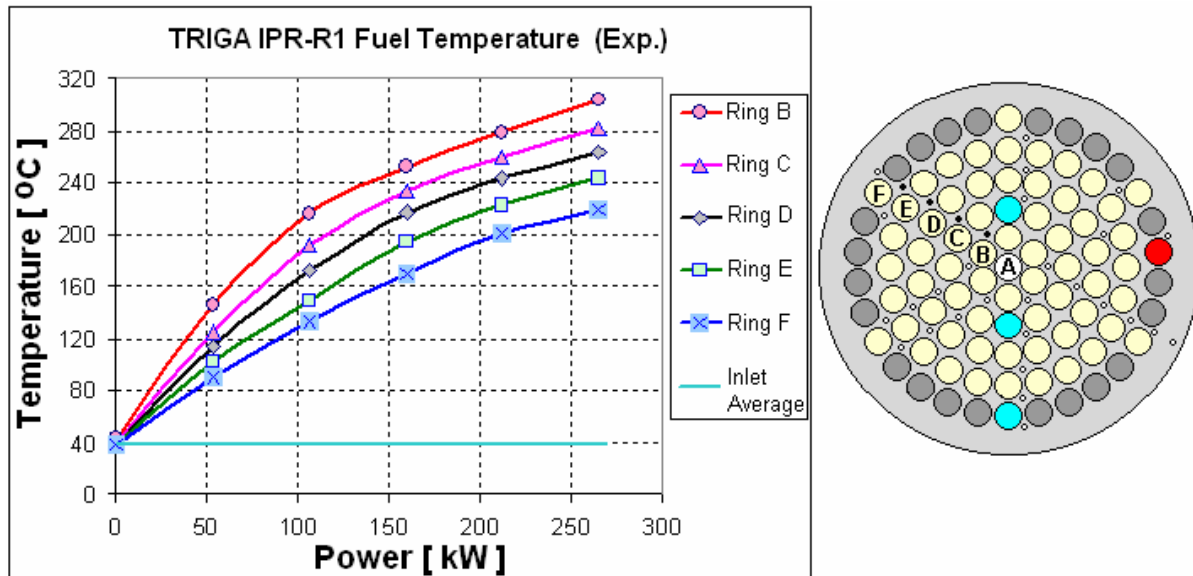


Figure 8. Instrumented fuel element temperature in all core rings

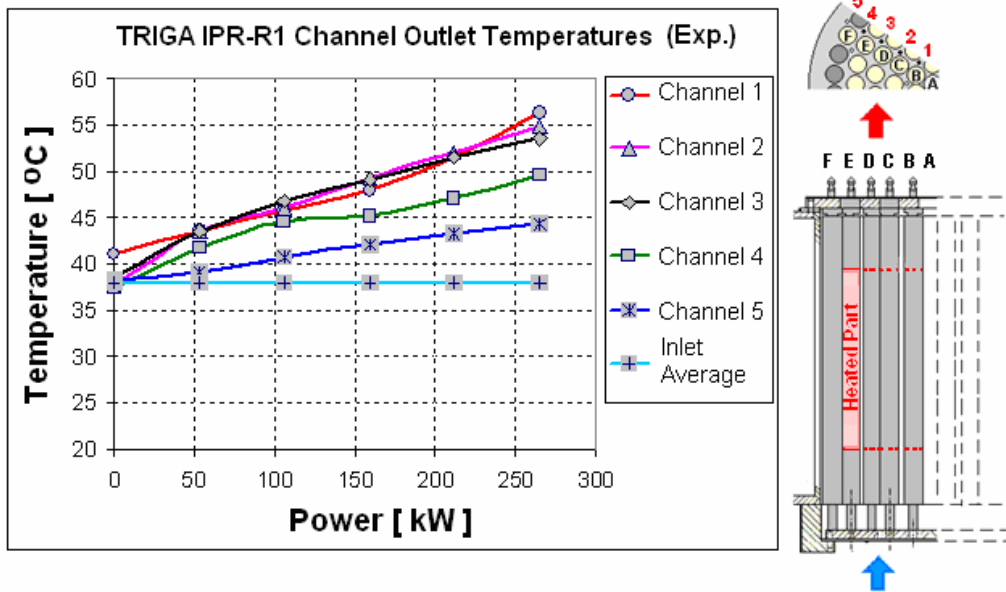


Figure 9. Outlet channels temperatures as function of the thermal power

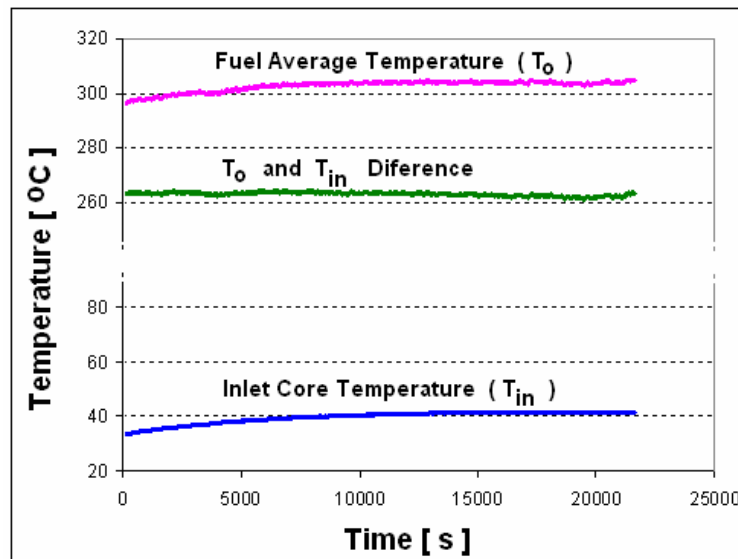


Figure 10. Fuel and inlet core water temperature evolution

3.4. Power Measuring Channel using Thermal Process

During the experiments it was observed that the temperature difference between fuel element and the pool water below the reactor core (primary loop inlet temperature) stay unchanged for the same power, as can be seen on Figure 10. With the instrumented fuel element at position B1, the power measured in the 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 these two values:

$$q = 2 \times 10^{-5} (\Delta T)^3 - 0.0045 (\Delta T)^2 + 0.7666 \Delta T - 2.4475 \quad , \quad (1)$$

where q is the 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 for this equation was one ($R^2 = 1$).

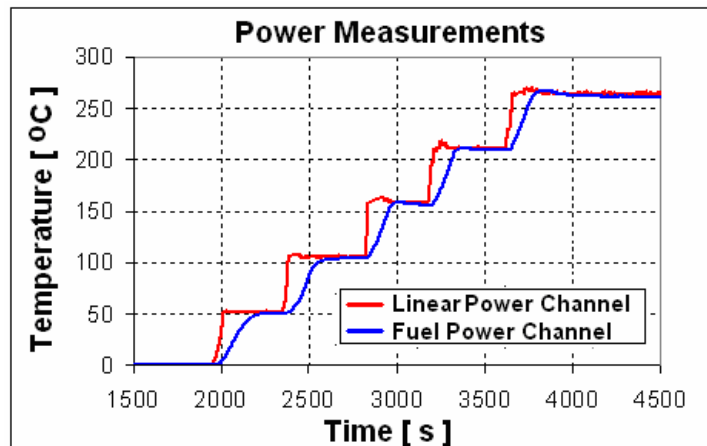


Figure 11. Reactor power measured by neutron channel and by fuel element temperature

Equation 1 was included in the data acquisition system and this new power measurement channel is available for the TRIGA IPR-R1 Reactor. After the experiments the instrumented fuel element was kept in position B1 (Figure 2) to monitor the reactor power and core temperature. Figure 11 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 TRIGA IPR-R1 Reactor Accidents Report (CDTN/CNEN, 2000) during steady state operation is 550 °C. A power operation limit over 1 MW was found based in this temperature and using Equation 1.

4. Conclusion

The reactor power calibration methodology presented in this report is now the standard methodology used for the TRIGA IPR-R1 Reactor power calibration (CDTN/CNEN, 2001). The uncertainty value obtained does not differ significantly from another thermal calibration processes described in technical literature. The experiments also show the efficiency of the free circulation in removing the heat produced in the reactor core by nuclear fission. The instrumented fuel temperature stayed almost constant at about 300 °C at 265 kW thermal power and with the primary cooling loop circulation off.

5. Acknowledgements

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6. References

- CDTN/CNEN, 2000, Relatório de Análise de Segurança do Reator TRIGA IPR-R1, Belo Horizonte, Brazil, 321p.
- CDTN/CNEN, 2001, "Manual de Operação do Reator TRIGA IPR-R1", Belo Horizonte, Brazil, 242p.
- Dalle, H.M., 2003, "Avaliação Neutrônica do Reator TRIGA IPR-R1-R1 com Configuração de 63 Elementos Combustíveis e Barra de Regulação em F16", CDTN/CNEN. (NI-EC3-01/03), Belo Horizonte, Brazil, 18 p
- Mesquita, A. Z., 2005, "Investigação Experimental da Distribuição de Temperaturas no Reator Nuclear de Pesquisa TRIGA IPR-R1", Campinas, Faculdade de Engenharia Química, UNICAMP, Brazil, Thesis, 183 p.
- Özkul, E.H.; Durmayaz, A., 2000, "A Parametric Thermal-hydraulic Analysis of ITU TRIGA Mark II Reactor", Proceedings of the 16th European Triga Conference, Institute for Nuclear Research, Pitesti, Romania., p3-23 - p3-42.
- Rao, D.V. et al., 1988, "Thermal Hydraulics for Sandia's Annular Core Research Reactor", Proceedings of Eleventh Biennial U.S. Triga Users' Conference, 1988, Washington, General Atomics, p. 4-89, 4-113.
- Veloso, M.A., 1999, "Análise Termo-hidráulica do Reator TRIGA IPR-R1 a 250 kW", CNEN/CDTN (NI-CT4-03/99), Belo Horizonte, Brazil, 141 p

7. Responsibility notice

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