ENC 2010
European Nuclear Conference
TRANSACTIONS
30 May to 2 June 2010, Palau de Congressos de Catalunya, Barcelona, Spain

NEUTRONIC AND THERMAL-HYDRAULIC EXPERIMENTAL PROGRAM IN THE IPR-R1 TRIGA REACTOR AT CDTN

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ABSTRACT

The IPR-R1 TRIGA reactor, located at CDTN (Belo Horizonte/Brazil), is a typical 100 kW Mark I light-water reactor cooled by assisted natural convection with an annular graphite reflector. In order to study the safety aspects connected with the increase of the maximum steady state power of the IPR-R1 TRIGA reactor, experimental measures were taken. This paper summarizes the experimental program and some recent results and procedures of the neutronic and thermal-hydraulic experiments carried out in the IPR-R1 TRIGA reactor.

1. Introduction

The reactor power will be upgraded from 100 kW to 250 kW [1, 2]. This research project aims at the reactor operation security and reliability. It also has the objective of maintaining the team who has the expertise in theoretical and experimental neutronics and thermal-hydraulics. To implement this project it is important to bring up to date the TRIGA IPR-R1 reactor operation instrumentation in order to monitor and control the reactor variables. Some experiments must be periodically done in nuclear reactors.

The neutronic experiments which have bee performed in the reactor are to determine: control rods worth; core reactivity excess and shutdown margin; reactivity changes induced by a simulated void; reactivity power coefficient and power defect; temperature and isothermal reactivity coefficient; xenon poisoning; and neutrons flux. From the results, it is possible to balance all the determined reactivity losses with the reactivity excess available in the reactor, considering the present and the future power.

The core thermal power has been determined by calorimetric and heat exchanger balance methods. Some thermo-hydraulic parameters like the coolant velocity, mass flow rate and Reynolds's number have been monitored in the hot channel with the forced cooling system switched off and on. The process used was the monitoring of the hot channel inlet and outlet temperatures.

Nuclear reactor operators need to know the basic reactor behaviour in order to understand and safely operate a nuclear reactor. The natural circulation test was performed to confirm the cooling capability of the natural convection in the IPR-R1 TRIGA reactor. The IPR-R1 has capability of long term core cooling by natural circulation. Fuel, channel and pool temperatures depend on the reactor power. The water also depends on the environment temperature. All the operational parameters were collected, in real-time, monitored and recorded in a data acquisition system.

2. The IPR-R1 TRIGA reactor

The IPR-R1 TRIGA reactor core consists of a lattice of cylindrical fuel-moderator elements, in which the zirconium-hydride moderator is homogeneously combined with 20 %-enriched uranium, and graphite elements [3, 4, 5]. The core configuration has 63 fuel elements (59 Al-clad and 4 SS-clad elements) arranged in five concentric rings. The spaces between the rods are filled with water that acts as coolant and moderator. The power level of the reactor is controlled with three control rods: Regulating, Shim, and Safety. The reactor is equipped with three experimental and irradiation facilities: rotary specimen rack, pneumatic transfer tube, and central thimble. Figure 1 shows the geometrical configuration of the core.

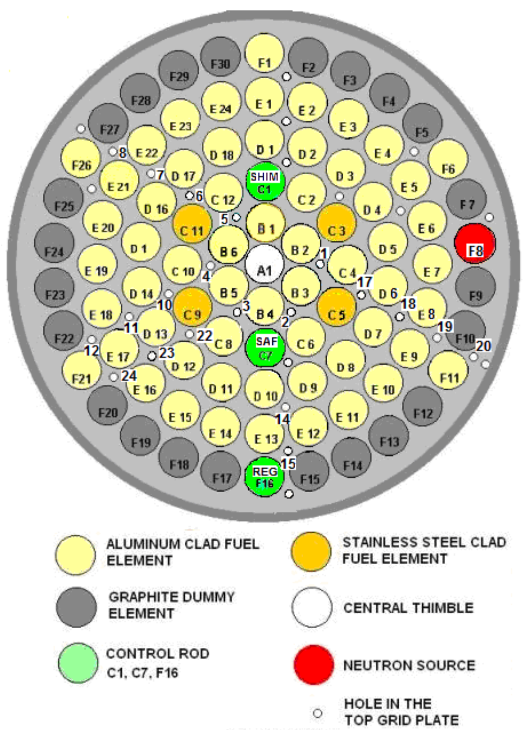

Fig. 1. IPR-R1 TRIGA reactor core

3. Neutronic experimental results

3.1 Control rod worth and excess of reactivity

The control rods were calibrated by the positive period method. The Control and Safety rods were intercalibrated. The reactivity measurements were performed at a power of less than 25 W so the temperature increase during the experiment was negligible. Integral worth of the Regulating and Shim rods as a function of their positions are shown in Fig. 2 and 3, respectively. The excess reactivity (ρ_{exc}) of the core was determined from the critical positions, at low power, of the control rods and the corresponding calibration curves. The average value obtained was 1.99 $. Table 1 shows the measured values of the control rods worth, the reactivity excess, and the shutdown margin, with the assumption that the highest worth control rod remains fully withdrawn (stuck rod condition) [6].

Tab. 1: Results of reactivity (β_{eff} of the IPR-R1 reactor is 0.0079)

	ρ ($)	ρ (pcm)
Regulating Worth	0.52	411
Shim Worth	2.63	2078
Safety Worth	2.34	1849
Reactivity Excess	1.99	1572
Shutdown Margin – Shim rod out	0.87	687

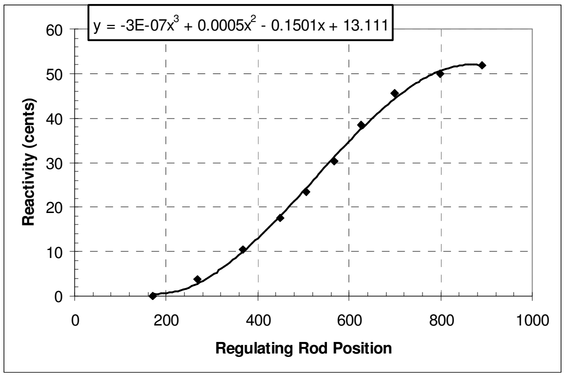

Fig. 2. Integral curve of the Regulating rod

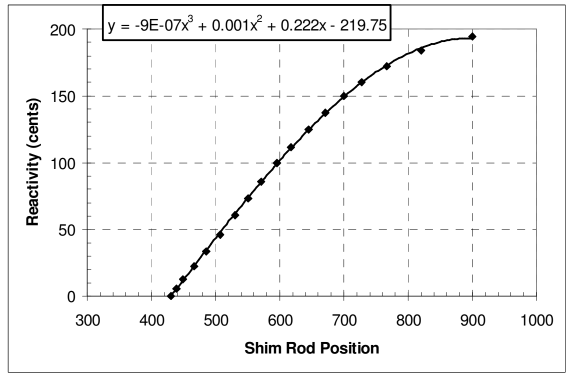

Fig. 3. Integral curve of the Shim rod

3.2 Loss of reactivity with power increase and xenon poisoning

Because of the prompt negative temperature coefficient, a significant amount of reactivity is needed to overcome temperature and allow the reactor to operate at high power levels. Figure 4 shows the relationship between reactor power level, raised in steps of 10 kW, and the associated reactivity loss to achieve a given power. The reactivity was determined from the calibrated control rod curves, considering each critical rod position. The reactivity needed to operate the IPR-R1 reactor at 100 kW, or the power defect, is 72 cents (569 pcm) [6].

Captures of neutrons in ^{135}Xe result in a negative reactivity effect in thermal reactors, because of the large capture cross section of this fission product [7]. The effect of the xenon poisoning during eight hours of operation at a steady-state power level of 100 kW was investigated. The reactivity loss versus the running time is plotted in Fig. 5. The estimated value of the negative reactivity introduced in the reactor was around 20 cents (158 pcm) [6].

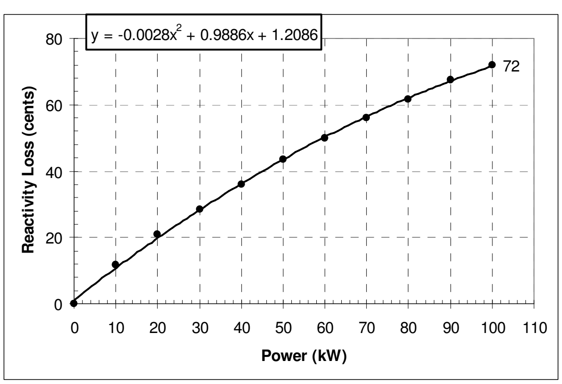

Fig. 4. Reactivity loss versus power level

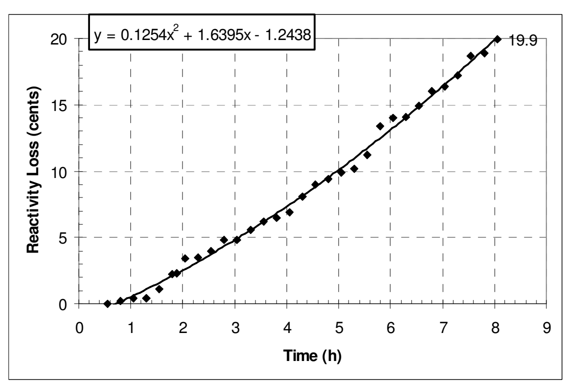

Fig. 5. Reactivity loss due to the ^{135}Xe buildup versus time

3.3 Loss of reactivity due to void

The presence of voids in a reactor has a significant effect on the reactivity, thus being of primary importance for the reactor safety and controllability. The voids can be produced either inherently, such as in cases of subcooled boiling, or they may be introduced on purpose, as for material irradiation. The reactivity changes induced by the void were measured by introducing in a critical core a simulated void (a closed aluminium tube which contained atmospheric air). Thus, the reactor was brought back to the critical state with the help of a calibrated control rod, and the change in the reactivity was determined by the change in the position of the control rod. The insertion of the void in the central thimble (higher flux) induces the largest reactivity loss, 22.0 cents (173.8 pcm) [6].

3.4 Temperature and isothermal reactivity coefficient

The prompt negative temperature reactivity coefficient of the reactor is defined as $\alpha_c=\Delta\rho/\Delta T$. The α_c negative means that an increase in temperature will cause a decrease in ρ, hence, a decrease in the reactor power and in the temperature which tends to stabilize the reactor power. The experiment was performed by increasing the power, and, consequently, the fuel temperature by withdrawing the Regulating rod in a number of steps at once. The heat of the core, and then ΔT, was estimated from the power versus time curve. The reactivity change $\Delta\rho$ was determined from the Regulating calibration curve considering its positions. The obtained average value of the temperature reactivity coefficient was (-1.1 ± 0.1) ¢ /ºC [6].

The isothermal temperature coefficient was measured by observing the reactivity change when the core temperature is raised by other means while the reactor is operating at a very low, almost zero power level [7]. When the reactor is at zero power there is no sensible heat being released in the fuel, and the entire reactor core can be characterized by a single temperature. The isothermal reactivity coefficient, α_{ISO} computed was -0.5 ¢/°C [8]. The results obtained demonstrated that the fuel temperature coefficient is the main contributor to the reactivity power coefficient of the TRIGA reactor.

3.5 Thermal neutron flux

To determine the neutron fluxes, bare and cadmium-covered gold and cobalt foil detectors were irradiated in the rotary specimen rack and in the central thimble, respectively, with the reactor operating at 100 kW. Their gamma activities were measured using Ge spectrometer. The average thermal neutron flux determined at the rotary specimen rack was (6.4 ± 0.4) $x10^{11}$ $n.cm^{-2}.s^{-1}$ and at the central thimble was (4.1 ± 0.3) $x10^{12}$ $n.cm^{-2}.s^{-1}$ [9].

4. Thermal-hydraulic experimental results

4.1 Thermal power calibration

It was used two procedures to the thermal power calibration of the IPR-R1 reactor: the calorimetric and heat balance methods. The calorimetric procedure was done with the reactor operating at a constant power, with primary cooling system switched off. The rate of temperature rise of the water was recorded. The reactor power is calculated as a function of the temperature-rise rate and the system heat capacity constant. The heat balance procedure consists in the steady-state energy balance of the primary cooling loop of the reactor. For this balance, the inlet and outlet temperatures and the water flow in the primary cooling loop were measured. The heat transferred through the primary loop was added to the heat leakage from the reactor pool.

Figure 6 shows the evolution of the measured temperatures during the calorimetric calibration and the heat balance calibration. Table 2 sumarizes the results. The thermal power obtained by the calorimetric method was 102 kW (±21 %), and by the heat balance calibration was 112 kW (±5.9 %). The calorimetric method calibration presented a large uncertainty. The main source of error was the determination of the heat content of the system, due to a large uncertainty in the volume of the water in the system and a lack of homogenization of the water temperature. The heat balance calibration in the primary loop is the standard procedure for calibrating the power of the IPR-R1 TRIGA reactor.

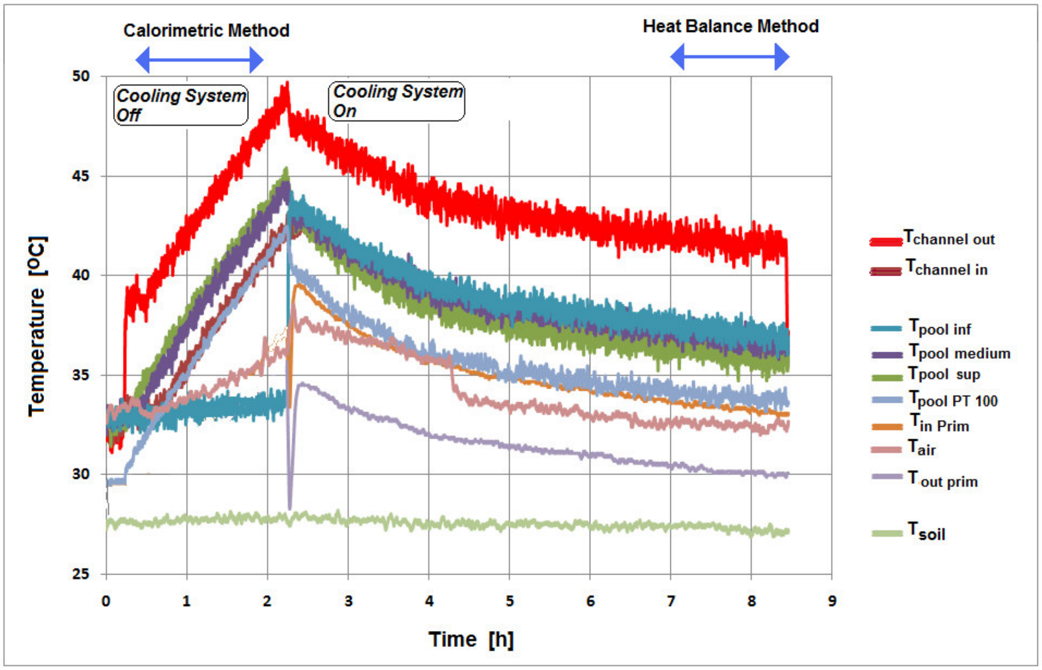

Fig. 6. Temperature evolution during the heat balance power calibration procedure

Tab. 2: Results of the power calibration by calorimetric and heat balance methods

Parameter	Calorimetric Method	Heat Balance Method
Temperature-rise rate ($\Delta T/\Delta t$)	4.84 °C/h	--
Average water temperature rise	34 °C to 41 °C	--
Water volume rise	17.36 m^3 to 17.86 m^3	--
Average water volume	17.7 m^3	17.7 m^3
Power dissipated	99 kW	--
Thermal losses from the reactor pool	3 kW	1.4 kW
Uncertainty	± 21 kW (± 21 %)	±6.6 kW(± 5.9 %)
Average primary loop coolant flow rate	--	30.09 ± 0.02 m^3/h
Average primary loop inlet temperature	--	33.4 ± 0.2 °C
Average primary loop outlet temperature	--	30.2 ± 0.2 °C
Power dissipated in the primary loop	--	111 kW
Standard deviation of the readings	--	± 4.0 kW
Power dissipated in the secondary loop(*)	--	85 kW
Total reactor power	**102 kW**	**112 kW**

(*) Not considered for thermal power calibration

4.2 Coolant mass flow rate and velocity in the hot channel

The mass flow through the core hot channel was determined indirectly from the heat balance across the core using measurements of the water entrance and exit temperatures. The channel heating process is the result of the thermal fraction contributions of the perimeter of each fuel around the channel. The Reynolds number found shows that the coolant flow is turbulent, in agreement with the experiments carried out by Mesquita [10], which showed that the heat transfer is subcooled nucleate boiling for operations over 85 kW. Results of the experiments indicated an increase in mass flow rate and velocity in the hot channel when forced cooling is turned on, as shown in Fig.7.

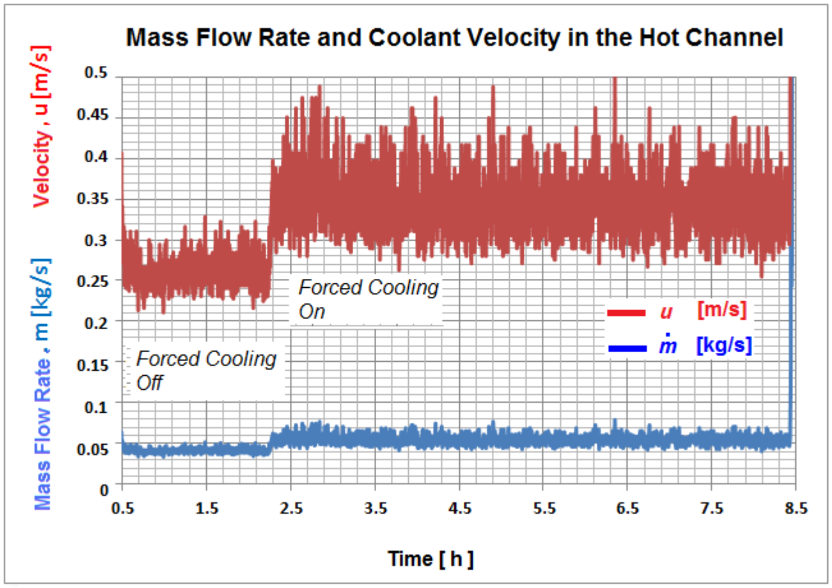

Fig. 7. Evolution of the coolant flow rate and velocity in the hot channel of IPR-R1 core

5. Conclusion

To evaluate the reactor operation security and reliability, and to maintain the team who has the expertise in neutronics and thermal-hydraulics some experiments are realized periodically in the reactor.

The experiments were performed in the IPR-R1 TRIGA reactor with 63 fuel elements in the core. The excess reactivity obtained for the core was 1.99 \$, and the shutdown margin with the most reactive rod stuck out of the core was 687 pcm, greater than the minimum safety limit required (200 pcm). The reactivity worth of the control rods is adequate to allow complete control of the reactor during operation from a shutdown condition to full power. The condition which at least two control rods should have sufficient reactivity worth to shutdown the reactor, independently, is satisfied. The isothermal coefficient of -0.5 ¢/kW was determined. The inherent safety of the reactor arises from the prompt negative temperature reactivity coefficient, whose measured value was (-1.1 ± 0.1) ¢/ºC, which effectively limits the power when excess reactivity is suddenly inserted. The reactivity needed to overcome the temperature and allow the IPR-R1 reactor to operate at 100 kW was 0.72 \$. Most of the negative reactivity change due to the power increase must be attributed to the change in the fuel temperature (prompt coefficient). The negative reactivity introduced by the xenon was around 0.20 \$, after eight hours of operation at 100 kW. The insertion of a void in the central thimble induced a loss of reactivity of 22.0 cents. The thermal neutron flux determined at the reactor rotary specimen rack was 6.4×10^{11} $n.cm^{-2}.s^{-1}$, and at the central thimble was 4.1×10^{12} $n.cm^{-2}.s^{-1}$.

For continuous monitoring of the thermal reactor power level, the instrumentation to measure the temperature and the power was incorporated in the data acquisition system. The evolution of these parameters, and also of some neutronic parameters, is displayed, in real time, and recorded on a digital monitoring system computer developed for the IPR-R1 reactor [11]. The values of the thermal power obtained by the calorimetric method and by the heat balance calibration were 112 kW and 102 kW, respectively.

The IPR-R1 TRIGA core design has sufficient natural convective flow to maintain continuous flow of water throughout the core. By this means significant vapour formation is avoided, and steam bubbles, in the vicinity of the fuel element surfaces, are restricted. The spacing between adjoining fuel elements, and hence the water fraction in the core, was selected not only from neutronic considerations but also from thermo hydrodynamic considerations.

6. Acknowledgments

The authors express their thanks to the *Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG)* and to the *Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq)* for the financial support.

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