*2007 International Nuclear Atlantic Conference - INAC 2007*
*Santos, SP, Brazil, September 30 to October 5, 2007*
*ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR - ABEN*
ISBN: 978-85-99141-02-1

# THE BRAZILIAN IAEA EXPERT MISSIONS IN THE RECOMMISSIONING OF THE IAN-R1 TRIGA REACTOR

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## ABSTRACT

This paper describes the works that were done by two Brazilian researchers of CDTN/CNEN, during the re-activation of the IAN-R1 Reactor in Bogotá (Colombia) in October 2005. The main aim of this mission was to participate in the Ad-hoc committee, established by Colombian Authorities, as International Atomic Energy Agency experts, to follow the recommissioning of the reactor IAN-R1, and assist the Colombian staff in the safe operation of the reactor. The work was carried out during two weeks and consisted in reviewing the operational procedures, results and records and providing lectures for the operating group. The reactor core was brought critical by adding two clusters (8 fuel elements); the three control rods were calibrated; the excess reactivity and shutdown margin were determined; and the thermal power evaluation was performed.

## 1. INTRODUCTION

Two researchers from the Nuclear Technology Development Center (CDTN), worked as experts of the International Atomic Energy Agency (IAEA), in the activities of hot commissioning of IAN-R1 Research Reactor [1] (Fig. 1), in Bogotá. The operation of this reactor, the only one in this country, started the recovery of the Colombian nuclear program.

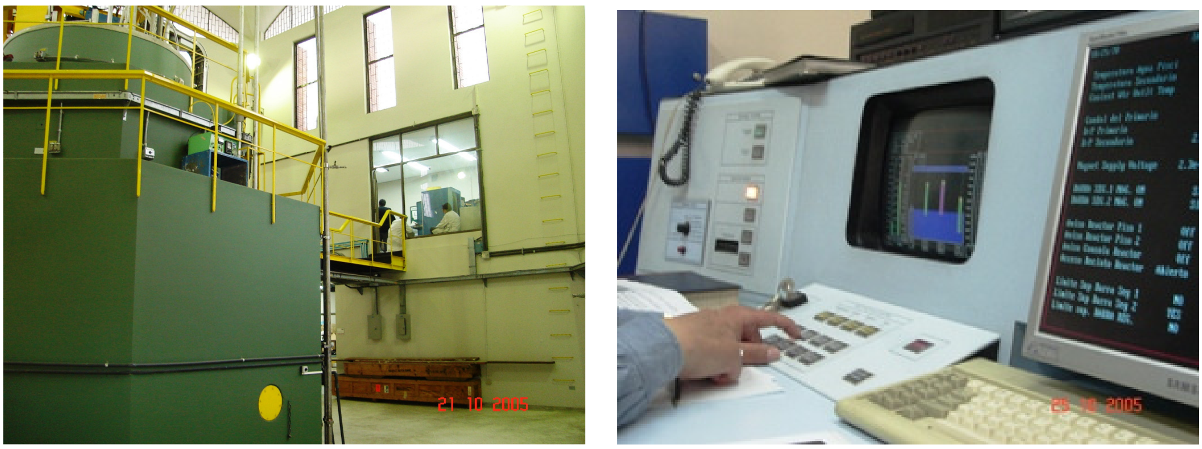

**Figure 1. The IAN-R1 TRIGA Reactor and console.**

The reactor was designed in 1965 as a small, 10 kW facility using aluminum plate-type, highly enriched fuel (MTR), under the United State's Atoms for Peace Program. General

Atomic has been involved in a gradual upgrading of the facility since 1988, and, in late 1994, a tripartite contract was signed with the IAEA, Colombian Authority and General Atomic to manufacture, install and commission the reactor with TRIGA-type, conversion to TRIGA low-enriched fuel [2, 3], and increasing the power of this reactor to 100 kW.

The nuclear activities in Colombia were interrupted in 1998 with the extinction of the Institute of Nuclear Affairs (IAN). In this year, eight fuels elements were removed from the reactor core to kept IAN-R1 TRIGA Reactor in a secure subcritical condition. It was brought into an extended shutdown condition in 1998 after the core conversion, which included the commissioning conducted by the supplier (General Atomic).[4]. The Colombian Authorities decided to reactivate the nuclear activities in the country with IAEA support [5]. In 2005, the authors of this paper were invited to participate in this work, due to their experience in working at experimental reactor physics and thermal hydraulic of the CDTN's IPR-R1 TRIGA Reactor.

The reactor IAN-R1 is a swimming pool type with concrete shield and two beamports. The fuel (U-$ZrH_{1.6}$) is contained in 4-rod clusters. The core configuration is a rectangular grid plate that holds a combination of 4-rod and 3-rod clusters. The 3-rod clusters provide a fourth cluster space to be used either for in-core irradiation or control rod locations. The core contains 50 fuel rods, 3 control rods and 3 in-core water filled experimental locations. The maximum core power level is 100 kW corresponding to a thermal neutron flux level varying from 1.9 x $10^{12}$ to 4.2 x $10^{12}$ n/cm$^2$.s, depending on the core locations [6]. The assembly is located inside an open tank full of light water which acts as biological shielding, partial neutron moderation and core coolant. The reactor core is cooled by natural circulation. The tank water is cooled by the primary and secondary systems.

## 2. DESCRIPTION OF THE ACTIVITIES

The two clusters (8 fuels rods) removed from the core in 1998 were replaced. The reactor was critical at 100 W during one hour. The Ad-Hoc committee was present during these experiments. When the reactor was critical, it could be observed that the reactivity values of each control rod are almost the same. The top view of the core without the two clusters and the core configuration after the fuel loading are shown in Fig.2.

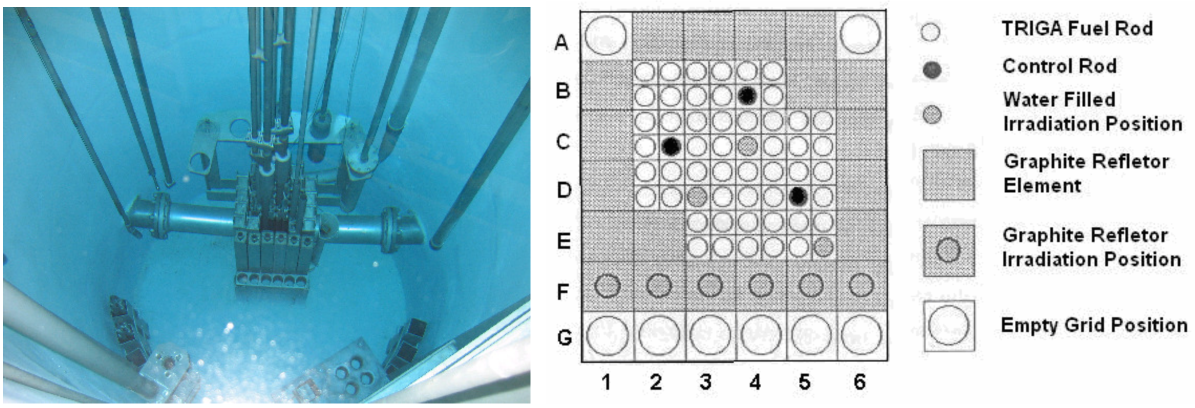


**Figure 2. IAN-R1 reactor core and the configuration after the fuel loading.**

## 2.1 Control Rod Calibration

All three-control rods were calibrated by the positive period method. The method consists of withdrawing the control rod from a known critical position through a small distance. Each successive step is compensated by lowering the other control rod just enough to reestablish criticality. In this process the control rod under calibration proceeds from the most inserted position (maintaining the reactor critical) to fully removed. The reactor period was obtained using the doubling time (DT), that is the time required for the power to increase by a factor of two.

The doubling time was measured with two digital chronometers, observing the power showed in the digital display in the console, 2 minutes after withdrawing the control rod under calibration, in order to finish the transition region. The reactivity associated with the measurement was gotten from the graphical form of the Inhour equation. It is important to note that for periods longer than one second, the curve is essentially independent of both $\ell$ and $\beta$. The reactivity measurements were performed at a low power so that the temperature increase during the experiment was negligible.

The Shim 1 and Shim 2 rods were intercalibrated. The idea was to measure one control rod in presence of another rod, used for compensating the reactivity introduced by step withdrawal of the measure rod. The Regulating, Shim 1 and Shim 2 rods worth were 3.25 \$, 2.89 \$ and 3.39 \$, respectively. The three control rods have sufficient reactivity worth to shutdown the reactor, independently.

During these calibrations one power measuring channel could not be used because the fission counter associated to it was not functioning (water had entered into the detector). We decided to continue the tests since the reactor was operating at low power, and the other two power measuring channels were still working. The initial operation checklist was accomplished and it was observed that all protection devices were available. Besides the manual scram button on the operator's control panel and close to the reactor pool, there are many automatic shutdown circuits (scram circuits).

## 2.2 Core Excess Reactivity and Shutdown Margin

The excess reactivity ($\rho_{exc}$) of the core was determined from different control rods critical positions, at low power, and the correspondent calibration curves. The average excess reactivity value obtained was (2.18 ± 0.08) \$.

The total reactivity worth of the control system was 9.53 \$. With a core excess reactivity of 2.18 \$, the shutdown margin with the most reactive rod (Shim 2) stuck out of the core was 3.96 \$. The shutdown margin is that margin the reactor can be shutdown from a critical condition, and is given by the difference between the reactivity worth of the considered control rods (the most worthy rod is assumed fully withdrawn) and the core excess reactivity. Table 1 presents the values of the control rods worth, the core excess reactivity, and the shutdown margin for IAN-R1 core configuration.

**Table 1. Results of reactivity**

| Parameter | ρ ($) |
|---|---|
| REGULATING Worth | 3.25 |
| SHIM 1 Worth | 2.89 |
| SHIM 2 Worth | 3.39 |
| Excess Reactivity | 2.18 |
| Shutdown Margin – SHIM 2 Out | 3.96 |

## 2.3 Thermal Power Estimate by Calorimetric Procedure

Before starting this test the fission detector had already been repaired, and it was operating properly. It was recommended that for routine operations the two fission counters and the ion chamber should be available. The thermal power calibration was performed using the calorimetric method [7]. Some thermocouples were put along the pool, and the top of the pool was thermally isolated. The reactor stayed critical at a constant power of 8 kW, indicated in the console, during about 3 hours, with manual power corrections because the automatic control system failed. The control rod initial positions were: Shim 1 (655), Shim 2 (661) and Regulating (674). The primary cooling system was switched off, and the rate of temperature rise was determined. With the specific heat of the system and water volume of the pool, the core power was then determined from the measured rate of temperature rise from operation of the reactor. During the experiment, all the pool temperatures were collected in intervals of 30 minutes. Fig. 3 shows the positions of the thermocouples in the pool, and the average water temperature versus the running time, during the thermal power estimation.

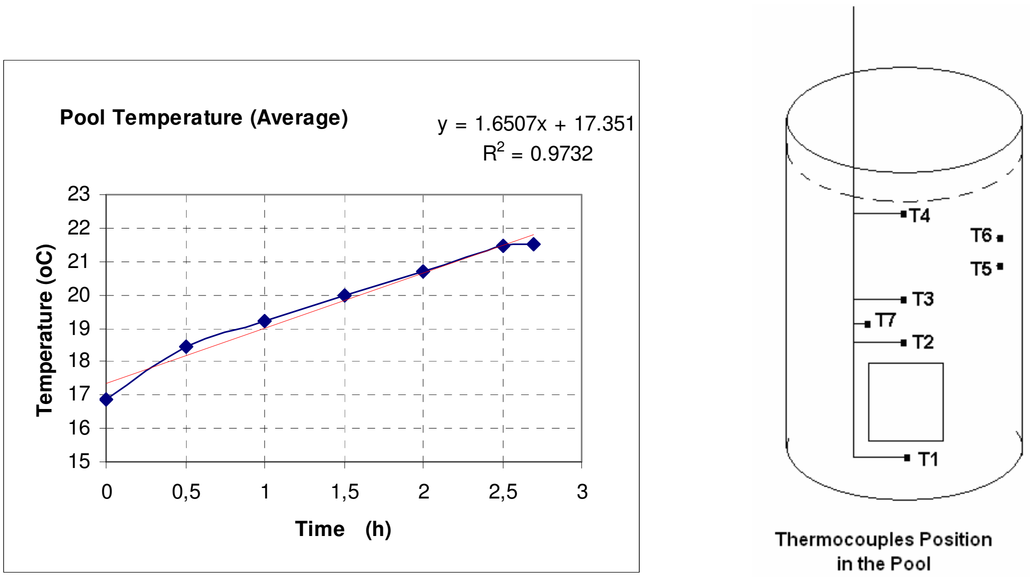


**Figure 3. Pool water temperature increase during the thermal power estimation.**

At 8 kW in the control console the radiation level was approximately of the same order of that one obtained during General Atomics tests [4]. The Cerenkov radiation could already be visualized in the reactor core (Fig. 4). There was a scram for high radiation, indicating that the actual power was larger than the console indication. The power obtained by the calorimetric method was 30 kW with an uncertainty of 20 %.

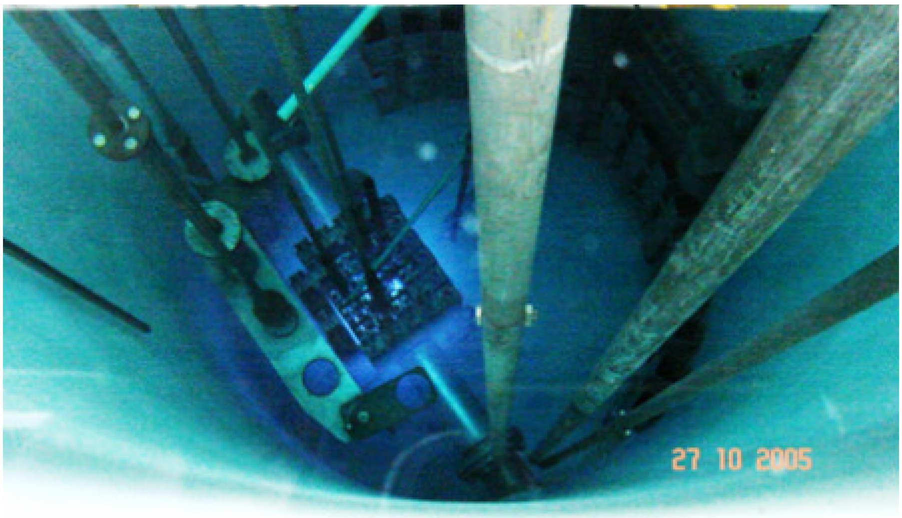


**Figure 4. The Cerenkov radiation during the thermal experiment.**

In the next day, the reactor was turned on in critical conditions with the control console indicating 8 kW. The positions of the ion chamber, and the two fission counters were adjusted until the console indications became 30 kW.

## 3. CONCLUSIONS

The neutronic and thermal-hydraulic parameters obtained in the recommissioning program were close to those found by General Atomics in the tests conducted in 1997 [4].

The reactor core was brought critical by adding two clusters (8 fuel elements). The control rods were calibrated by the positive period method, and the Regulating, Shim 1 and Shim 2 control rods worth were 3.25 $, 2.89 $ and 3.39 $, respectively. The three control rods worth were almost the same, and they have sufficient reactivity to shutdown the reactor, independently. The excess reactivity obtained for the proposed core was 2.18 $, and the shutdown margin, with the most reactive rod stuck out of the core, was 3.96 $, hence greater than the minimum safety limit required. The thermal power calibration was performed using the calorimetric method. It was determined that the real reactor power was 30 kW with an uncertainty of 20 %.

The IAN-R1 Reactor installation has good computer equipment and electronic instrumentation, a data acquisition system has already been implemented [8], as well the communication and physical protection systems [9]. The control bar graph display of the console screen is very friendly. There is a No-Break that feeds the control console and the control rods electromagnets. So, in the event of electrical energy failure the operator has time to make decisions.

The recommissioning of the IAN-R1 TRIGA research reactor was successfully completed, in safety conditions. The technological interchange and cooperation among the American Latin countries were a very positive fact of this mission.


## ACKNOWLEDGMENTS

The authors would like to thank the operation staff of the IAN-R1 Research Reactor, and to compliment IAEA, and, specially, Dr. Heriberto José Boado Magan for giving incentive to the cooperation among Latin American countries in the solution of their nuclear problems.


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