# UNCERTAINTY ANALYSIS ON THERMOHYDRAULIC PARAMETERS OF THE IPR-R1 TRIGA RESEARCH REACTOR

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## ABSTRACT

Experimental studies have been performed in the IPR-R1 TRIGA Mark 1 Research Nuclear Reactor of CDTN/CNEN at Belo Horizonte (Brazil) to find out the temperature distribution as a function of reactor power, under steady-state conditions. During these experiments the reactor was set in many different power levels. These experiments are part of the research program that has the main objective the commissioning of IPR-R1 Reactor for routine operation at 250 kW. This paper presents the uncertainty analysis in the results of the thermohydraulic experiments performed. The methodology used to evaluate the propagation of uncertainty in the results was done based on the pioneering article of Kline and McClintock, with the propagation of uncertainties based on the specification of uncertainties in various primary measurements.

## 1. INTRODUCTION

The IPR-R1 Reactor (Fig. 1) of Belo Horizonte (Brazil) is a nuclear research reactor, pool type with an open water surface and the core has a cylindrical configuration. The core power is 250 kWth, cooled by light water and with graphite reflectors. It contains 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements with 20 % enrichment and 8.5 wt % uranium. One of these steel-clad fuel elements is instrumented in the center with three thermocouples (Fig. 1). In standard operating conditions, thermal power is measured by four nuclear channels.

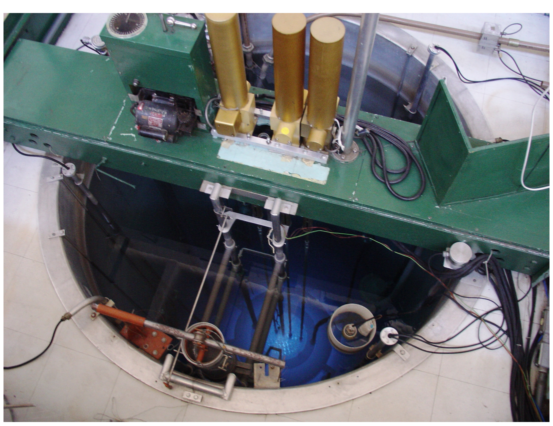


**Figure 1. The IPR-R1 TRIGA Reactor and the instrumented fuel element**

Experimental and analytical studies have been performed in the IPR-R1 Reactor [1] [2] to find out the core thermal power, the temperature distribution as a function of the reactor power under steady-state conditions, the flow distribution in the coolant channels, the heat transfer coefficient on the heated surface and a prediction of critical heat flux. This paper describes the methodology used to evaluate the propagation of uncertainty in experimental results.

## 2. EXPERIMENTAL METHODOLOGY

The thermal power calibration is made by the measurements of the coolant flow and temperature difference in the heat exchanger of the primary cooling loop [3], according to Equation 1,

$$q = \dot{m} \, . \, c_p \, . \, \Delta T \quad . \tag{1}$$

Where $\dot{m}$ is the flow rate of the coolant water in the primary loop, $c_p$ is the specific heat of the coolant, and $\Delta T$ is the difference between the temperatures at the inlet and the outlet of the primary loop. Table 1 presents the results and some calibration data.

**Table 1. The IPR-R1 TRIGA Reactor thermal power [3]**

| | |
|---|---|
| Calibration date | August 19, 2004 |
| Average flow rate | 32.7 $\pm$ 0.4 $m^3/h$ |
| Average inlet primary temperature | 41.7 $\pm$ 0.3 $^oC$ |
| Average outlet primary temperature | 34.8 $\pm$ 0.3 $^oC$ |
| Heat power transferred to the primary loop | 261 kW |
| Thermal losses from the reactor pool | 3.8 kW |
| **Reactor thermal power** | **265 kW** |
| Standard deviation of the measuring | 3.7 kW |
| Average power uncertainty | ±19 kW (±7.2%) |
| Heat power dissipated in the secondary loop | 248 kW |

The fuel temperature was measured with an instrumented fuel element which contains three chromel-alumel (type K) thermocouples [4]. This instrumented rod is located at position B1. Two thermocouples were inserted into the core through some holes on the top grid plate. These thermocouples were placed near position B1 and measured the inlet and outlet temperatures in the hot channel.

### 2.1 Heat Transfer Regimes of the Cladding to Coolant

As the IPR-R1 TRIGA reactor core power is increased, the heat transfer regime from the fuel cladding to the coolant changes from the single phase natural convection regime to subcooled nucleate boiling. Dittus-Boelter [5] proposed the following correlation to predict heat transfer coefficient ($h_{sp}$) for turbulent single-phase flow in long straight channels in the fully developed region:

$$h_{sp} = \frac{0.023k \, Re^{0.8} \, Pr^{0.4}}{D_w}, \quad \text{or:} \quad h_{sp} = 0.023 \frac{k}{D_w} \left( \frac{G D_w}{\mu} \right)^{0.8} \left( \frac{c_p \mu}{k} \right)^{0.4} , \tag{2}$$

where $Re$ is the *Reynolds number* and $Pr$ the *Prandtl number*, $D_w = 4A/P_w$ is the hydraulic diameter of the channel based on the wet perimeter, $A$ is the flow area in [m$^2$]; $P_w$ is the wet perimeter in [m]. $G$ is the mass flow in [kg/m$^2$s], $c_p$ is the isobaric specific heat in [J/kgK], $k$ is the thermal conductivity in [W/mK] and $\mu$ is the fluid dynamic viscosity in [kg/ms]. To the IPR-R1 TRIGA the fluid properties are calculated at the bulk water temperature on the sub-saturated at 1.5 bar. Direct measurement of the flow rate in a coolant channel is very difficult because of the bulky size and low accuracy of flowmeter. The mass flow rate in the channel is given by the mass flux divided by the channel area. The mass flux is given by the thermal balance in the channel. The Table 2 shows the coolant properties as function of power to the channel beside the position B1 of the core. In table, $G$ is the mass flux given by G $= \dot{m}$ */channel area;* $u$ is the velocity given by $u=G/\rho$, where $\rho$ is the water density (995 kg/m$^3$). The water thermodynamic properties to the IPR-R1 TRIGA are calculated at the bulk water temperature on the sub-saturated at 1.5 bar [6]. Table 2 shows, in last column, the heat transfer coefficient in the single-phase flow ($h_{sur}$) calculated by the Dittus-Boelter correlation.

**Table 2 – Coolant properties and the single-phase heat transfer coefficient [1]**

| q Core [kW] | q Channel [kW] | $\Delta T$ [°C] | $c_p$ [kJ/kgK] | $\dot{m}$ [kg/s] | $G$ [kg/m$^2$s] | $u$ [m/s] | $\mu$ [$10^{-3}$kg/ms] | $k$ [W/mK] | Re | Pr | $h_{sur}$ [kW/m$^2$K] |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 265 | 9,81 | 13.9 | 4.1809 | 0.169 | 205.40 | 0.21 | 0.549 | 0.639 | 6968 | 3.6 | **1.562** |
| 212 | 7.84 | 9.6 | 4.1800 | 0.195 | 237.98 | 0.24 | 0.575 | 0.638 | 7708 | 3.8 | **1.724** |
| 160 | 5.92 | 7.0 | 4.1795 | 0.202 | 246.35 | 0.25 | 0.596 | 0.636 | 7697 | 3.9 | **1.743** |
| 108 | 4.00 | 4.6 | 4.1793 | 0.208 | 253.05 | 0.25 | 0.620 | 0.634 | 7601 | 4.1 | **1.750** |
| 53 | 1.96 | 2.5 | 4.1789 | 0.188 | 228.52 | 0.23 | 0.638 | 0.632 | 6670 | 4.2 | **1.591** |
| 35 | 1.30 | 1.8 | 4.1780 | 0.172 | 209.64 | 0.21 | 0.642 | 0.630 | 6081 | 4.3 | **1.479** |

For local boiling the Newton Equation of cooling is modified to the form:

$$h_b = \frac{q''}{T_{sur} - T_f}, \qquad (3)$$

where $h_b$ is the coefficient for nucleate boiling heat transfer; $q''$ is the heat transfer rate per unit of surface area [W/m$^2$]; $T_f$ is the bulk fluid temperature [°C]; $T_{sur}$ is the surface temperature [°C], given by:

$$T_{sur} = T_{sat} + \Delta T_{sat} \quad . \qquad (4)$$

The surface superheat was calculated by the McAdams correlation [5];

$$\Delta T_{sat} = 0.81(q'')^{0.259}, \qquad (5)$$

with $q''$ in [W/m$^2$] and $\Delta T_{sat}$ in [°C]. This correlation reproduces experimental data for subcooled water from 11 to 83 °C, pressure of 2 to 6 bar, velocity from 0.3 to 11 m/s and hydraulic diameter of 0.43 cm to 1.22 cm. The heat flux for fully developed subcooled nucleate boiling is given by the equation [7]:

$$h_{sur} = q'' / \Delta T_{sat} \quad , \qquad (6)$$

where $h_{sur}$ is the heat transfer coefficient for local pool boiling between the cladding surface and the coolant [kW/m$^2$K], $q''$ is the heat flux in fuel surface [kW/m$^2$] and $\Delta T_{sat}$ is the wall superheat [$^o$C]. The $h_{sur}$ as function of the power, with the instrumented fuel element positioned in the position B1 are shown in the last column of the Table 3.

**Table 3 – Thermal parameters of the fuel element in subcooled boiling regime [1]**

| $q_{core}$ [kW] | $q_{B1}$ [W] | $T_o$ [$^o$C] | $q'$ [W/m] | $q''$ [W/m$^2$] | $q'''$ MW/m$^3$ | $\Delta T_{sat}$ [$^o$C] | $T_{sur}$ [$^o$C] | $k_g$ **[W/mK]** | $h_{sur}$ **[kW/m$^2$K]** |
|---|---|---|---|---|---|---|---|---|---|
| 265 | 8759 | 300.6 | 22988 | 194613 | 20.70 | 19.0 | 130.4 | **10.75** | **10.25** |
| 212 | 7007 | 278 | 18391 | 155690 | 16.56 | 17.9 | 129.3 | **9.84** | **8.69** |
| 160 | 5288 | 251.6 | 13880 | 117502 | 12.50 | 16.7 | 128.0 | **8.94** | **7.05** |
| 108 | 3570 | 216.1 | 9369 | 79314 | 8.44 | 15.0 | 126.4 | **8.31** | **5.27** |

## 3. ANALISYS OF UNCERTAINTIES

This item presents the uncertainties associated with values of the experimental measurement and the expressions deduced to calculate propagation of uncertainties in thermal power and heat-transfer coefficients, always taking into account the law-physical equations used in theoretical calculations [8]. In the found expressions, the contributions of the uncertainties associated with the geometry of the fuel element are negligible due to the rigorous tolerances specified in the maker's drawings [4]. The uncertainties associated with the physical properties of the water are also negligible, because they are insignificant when compared with the uncertainties of the variables measured during the experiments. The thermocouples, the resistance temperature detectors and the flowmeter were all calibrated and they had their respective uncertainties determined, considering the uncertainties of the circuit, the uncertainties of the other components of the data acquisition system, the statistical uncertainties of the calibration process and the standard error associated with the regression analysis for the respective calibration curve. The uncertainties ($U$) for the temperature measurement circuit were $U = \pm 0.4$ $^o$C for resistance temperature detectors, and $U = \pm 1.0$ $^o$C for thermocouples.

The uncertainty in the thermal power of the reactor is determined, mainly, by the uncertainty in the measure of the flow rate of the coolant loop and by the uncertainty in the value of its temperature in the inlet and outlet of the coolant loop. The flow rate of the primary circuit is measured through a group formed by an orifice plate and a differential pressure transmitter, with digital indication in the data acquisition system. The uncertainty consolidated with the measurement of the flow rate, from 28 m$^3$/h to 33 m$^3$/h, was evaluated in $U = \pm 0.41$ m$^3$/h (±1.1%).

The method adopted to calculate the propagation of uncertainty was proposed by Kline and McClintock [9]. Suppose a set of measurements is made and the uncertainty in each measurement is estimated. Then, these measurements are used to calculate some desired result for the experiments. We wish to estimate the uncertainty in the calculated result on the basis of the uncertainties in the primary measurements. The result R is a given function of the independent variables $x_1, x_2, x_3, ...., x_n$. Thus,

$$R = R(x_1, x_2, x_3, .....x_n) \qquad (7)$$

Let $U_R$ be the uncertainty in the result and $U_1$, $U_2$, $U_3$,........., $U_n$ be the uncertainties in the independent variables. The uncertainty in the result is given as:

$$U_R = \left[ \left( \frac{\partial R}{\partial x_1} U_1 \right)^2 + \left( \frac{\partial R}{\partial x_2} U_2 \right)^2 + .... + \left( \frac{\partial R}{\partial x_n} U_n \right)^2 \right]^{1/2} \qquad (8)$$

### 3.1 Uncertainty in the Thermal Power of the Reactor *q*

The calculation of the thermal power is subject to the uncertainties of the measures of the flow rate and its temperatures, and also to the estimations of the specific heat of the water obtained in function of its temperature. All the uncertainties are determined taking in consideration the results of the calibrations of the measurement instruments. The uncertainty in the value of power $q$ is a combination of the uncertainty of the flow rate, the uncertainty in the value of the specific heat ($c_p$) and the uncertainty of the difference between the inlet and outlet temperatures of the water in cooling loop ($T = T_{in} - T_{out}$). The thermal power $q$ dissipated in the heat exchanger was given by Equation 1.

Using Equation 8, the uncertainty in the thermal power is

$$\frac{U_q^{'}}{q} = \sqrt{\left( \frac{U_{\dot{m}}}{\dot{m}} \right)^2 + \left( \frac{U_{c_p}}{c_p} \right)^2 + \left( \frac{U_{T_{in}}}{T_{in} - T_{out}} \right)^2 + \left( \frac{U_{T_{out}}}{T_{in} - T_{out}} \right)^2} \qquad (9)$$

Where $U_{\dot{m}}$, $U_{c_p}$, $U_{T_{in}}$ e $U_{T_{out}}$ are respectively the consolidated uncertainties of the primary variables $\dot{m}$, $c_p$, $T_{in}$ e $T_{out}$. To the found value should be added the standard deviation $S_q$ of the thermal power found during the time of data acquisition. The value of the uncertainty is,

$$\frac{U_q}{q} = \sqrt{\left( \frac{U_q^{'}}{q} \right)^2 + \left( \frac{S_q}{q} \right)^2} \qquad (10)$$

The uncertainty in the value of the reactor thermal power is a result of the uncertainty in the value of the flow rate and, mainly, the uncertainties in the values of the inlet and of outlet temperatures of the water in the coolant loop. Using the expressions above, we finally meet an uncertainty of 7.2% in the thermal power supplied by the core.

### 3.2 Uncertainty in the Overall-Thermal Conductivity of the Fuel Element $k_g$

The overall-thermal conductivity $k_g$ of the fuel element is:

$$k_g = \frac{q^{'''} r^2}{4(T_o - T_{sur})} \qquad (11)$$

Where the superficial temperature is given by $T_{sur} = T_{sat} + \Delta T_{sat}$. The saturation temperature of the water at 1.5 bar is 111.37 °C, with a very low value of relative uncertainty that could be negligible. The superheating $\Delta T_{sat}$ is found using the correlation of McAdams (Eq. 5). Using Equation 8 to determine the relative uncertainty of $k_g$ and the relative uncertainty of $\Delta T_{sat}$, we have as a result the following expression for the relative uncertainty in the overall-thermal conductivity of the fuel element:

$$\frac{U_{k_g}}{k_g} = \sqrt{\left(\frac{U_{q'''}}{q'''}\right)^2 + \left(\frac{2U_r}{r}\right)^2 + \left(\frac{U_{T_o}}{T_o - T_{out} - \Delta T_{sat}}\right)^2 + \left(\frac{U_{T_{sat}}}{T_o - T_{sat} - \Delta T_{sat}}\right)^2 + \left(\frac{0.259 U_{q''} \Delta T_{sat}}{q''(T_o - T_{sat} - \Delta T_{sat})}\right)^2} \quad (12)$$

The uncertainty in the overall-thermal conductivity of the fuel element depends, mainly, on the reactor's thermal power uncertainty. Using Equation 12 above, we finally find an uncertainty 7.3% for $k_g$.

### 3.3 Uncertainty in the Heat-Transfer Coefficient of the Cladding to Coolant $h_{sur}$

In the subcooled nucleate boiling regime the heat-transfer coefficient of the cladding to coolant $h_{sur}$ as a function of the reactor power $q$ is given by Eq. 6. Using Equation 8 to determine the relative uncertainty of $k_g$ and the relative uncertainty of $\Delta T_{sat,}$ previously analyzed, we arrive in the following expression of the relative uncertainty in the overall-thermal conductivity of the fuel element $k_g$:

$$\frac{U_{h_{sur}}}{h_{sur}} = \sqrt{\left(\frac{U_{q''}}{q''}\right)^2 + \left(\frac{0.259 U_{q''}}{q''}\right)^2} \quad (13)$$

The uncertainty in the heat-transfer coefficient of the external surface of the cladding for the water depends, mainly, on the reactor's thermal power uncertainty. Using the expression above, we finally find as a result an uncertainty 7.4% for $h_{sur}$.

### 3.4 Uncertainty in the Heat-Transfer Coefficient in the Gap $h_{gap}$

The instrumented fuel element is composed by a central zirconium filler rod where the thermocouples are fixed, a fuel active part formed by an alloy of zirconium hydride (U-$ZrH_{1,6}$), an interface between the fuel and the external cladding (gap) and a 304 stainless steel cladding. Using electrical analogy we find the equations presented in Table 4 to the fuel element geometry.

**Table 4. Thermal resistance for conduction**

| Geometry | Thermal Resistance, R | Temperature Difference, ΔT |
|---|---|---|
| Cylinder | $R = 1/4\pi \ \ell \ k$ | $\Delta T = q''' \, r^2 / 4\, k$ |
| Hollow Cylinder | $R = ln(r_{ext}/r_{int})/2\, \pi \ \ell \ k$ | $\Delta T = q''' \, r_o^2 \ ln(r_{ext}/r_{int})/2k$ |
| Convective Resistance in Cylinder | $R = 1/2\, \pi \ \ell \ r\, h$ | $\Delta T = q''' \, r / 2\, h$ |

The estimate value for the heat-transfer coefficient in gap is

$$h_{gap} = \frac{2}{r_0}\left(\frac{k_g k_{UZrH} k_{rev}}{k_{UZrH} k_{rev} - k_g k_{rev} - 2k_g k_{UZrH} \ell n(r_2 / r_1)}\right) \quad (14)$$

Using Equation (8), the expression for relative uncertainty in the coefficient $h_{gap}$ is:

$$\frac{U_{h_{gap}}}{h_{gap}} = \sqrt{\left(\frac{U_{r_0}}{r_0}\right)^2 + \left(\frac{U_{k_g}}{k_g}\right)^2 + \left(\frac{U_{kUZrH}}{k_{UZrH}}\right)^2 + \left(\frac{U_{k_{rev}}}{k_{rev}}\right)^2 + \left(\frac{U_{r_1} r_o h_{gap}}{r_1 k_{rev}}\right)^2 + \left(\frac{U_{r_2} h_{gap} r_0}{r_2}\left(\frac{1}{k_{rev}}\right)\right)^2} \quad . (15)$$

The substitution of the numerical values gives an uncertainty to the heat-transfer coefficient in gap of 7.5%. This value indicates that the uncertainty depends only on the uncertainty in the overall-thermal conductivity of the fuel element ($k_g$).

## 4. CONCLUSION

The uncertainty analysis on thermohydraulic parameters of the IPR-R1 TRIGA fuel element is determined, basically, by the uncertainty of the reactor's thermal power. The other parts of the propagation equation are negligible. Thus, the reactor's thermal power uncertainty is determined mainly by the uncertainty in the coolant flow measure and by the uncertainty in the value of the temperature difference in the heat exchanger.

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