Uncertainty assessment of the experimental thermal-hydraulic parameters for CDTN TRIGA research reactor

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Abstract: Experimental studies have been performed in the TRIGA Nuclear Reactor at Nuclear Technology Development Centre (CDTN), Brazil, to find out its thermal hydraulic parameters. Fuel to coolant heat transfer patterns must be evaluated as function of the reactor power. The heat generated by nuclear fission in the reactor core is transferred from fuel elements to the cooling system through the fuel-cladding (gap), and the cladding to coolant interfaces. As the reactor core power increases the heat transfer regime from the fuel cladding to the coolant changes. This paper presents the uncertainty analysis in the results of the thermal hydraulics experiments performed. The uncertainty analysis on thermal hydraulics parameters is determined, basically, by the uncertainty of the reactor's thermal power.

Keywords: thermal-hydraulic; uncertainty; TRIGA; temperature; heat transfer; fuel rod.

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1 Introduction

The objective of the thermal and hydrodynamic projects of the reactors is to remove the heat safely, without producing excessive temperature in the fuel elements. The regions of the reactor core where boiling occurs at many different power levels can be determined from the heat transfer coefficient data. As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling.

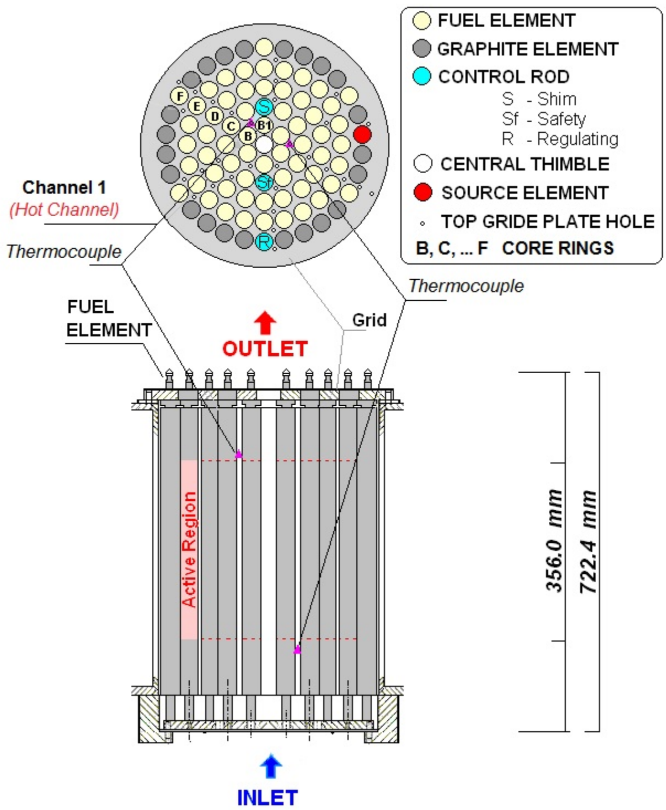
The TRIGA (Training, Research, Isotopes, General Atomic) reactor at CDTN in Belo Horizonte is a pool type nuclear research reactor, with an open water surface, and the core has a cylindrical configuration. The maximum core power is 250 kW cooled by light water. Experimental studies have been performed in the IPR-R1 reactor to find out the core thermal power, the temperature distribution as a function of the reactor power under steady-state conditions (Mesquita and Rezende, 2007; Mesquita and Souza, 2014). The heat generated by nuclear fission is transferred from fuel elements to the cooling system through the fuel-to-cladding gap and the cladding to coolant interfaces. The fuel thermal conductivity, and the heat transfer coefficient from the cladding to the coolant were evaluated experimentally. A correlation for the gap conductance between the fuel and the cladding was also found.

This paper presents the uncertainty analysis in the results of the thermal hydraulics experiments performed. The uncertainty analysis on thermal hydraulic parameters is determined, basically, by the uncertainty of the reactor's thermal power. The other parts of the propagation equation are negligible. The uncertainty in the value of the reactor thermal power is a result of the uncertainty in the value of the flow rate, the uncertainties in the values of the inlet and of outlet temperatures of the water in the coolant loop, and also in the estimations of the specific heat of the water obtained in function of its temperature.

2 Materials and methods

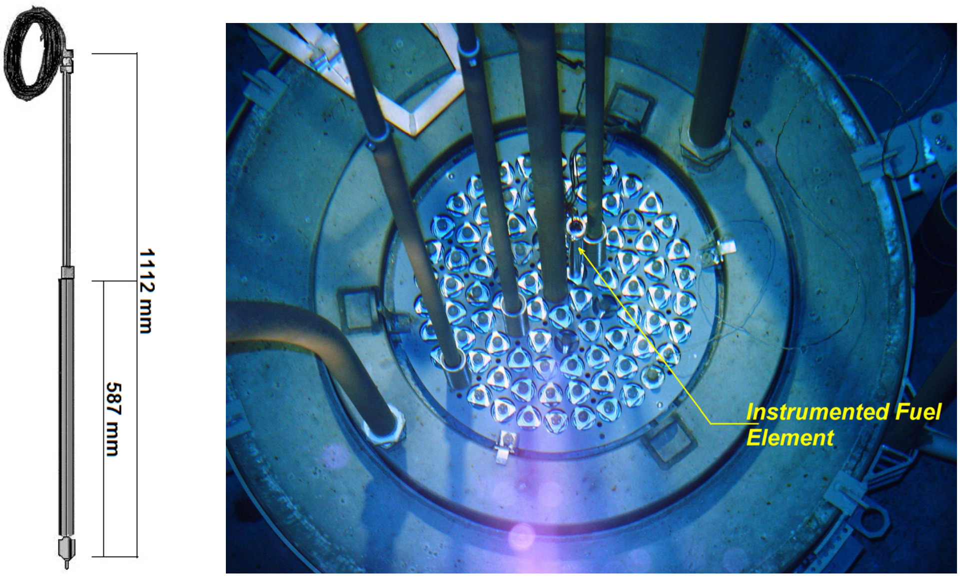
The TRIGA reactor at CDTN is a typical TRIGA Mark I reactor cooled by assisted natural convection with an annular graphite reflector. The core is placed at the bottom of a cylindrical open tank of 6.625 m deep and 1.92 m in diameter, able to assure an adequate shielding of radiation from the core. The cylindrical fuel elements are a homogeneous mixture of zirconium hydride and uranium 20% enriched in ^{235}U. The reactor core has 63 cylindrical fuel elements, 58 aluminium-clad fuel elements, and five stainless steel-clad fuel elements. A simplified diagram of the core configuration is shown in the Figure 1.

Figure 1 Simplified core diagram (see online version for colours)



The original fuel element at the hottest ring (B1 position) in the core was removed, and replaced by an instrumented fuel element. Two thermocouples were inserted into the core through some holes on the top grid plate. These thermocouples were placed near the instrumented fuel element and measured the inlet and outlet temperatures in the hot channel. The instrumented fuel element is in all aspects identical to standard fuel elements, except that it is equipped with three type K thermocouples (Chromel-Alumel), embedded in the fuel meat. Figure 2 shows the instrumented fuel element before and after it has been was been positioned in the core.

Figure 2 Instrumented fuel element, (a) outside and (b) inside the core



(a) (b)

An operational computer program, and a data acquisition and signal processing system were developed in order to facilitate the experiments. Besides showing the real-time performance of the plant, the system stores the information in a computer hard disk, with an accessible historical database.

2.1 Thermal power calibration

The reactor core is cooled by natural convection of demineralised light water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere through the primary cooling loop, the secondary cooling loop and the cooling tower. Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total power is determined by the thermal balance of cooling water flowing through the primary and secondary loops added to the calculated heat losses. These losses represent a very small fraction of the total power (about 1.5% of total). 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 loop is measured by an orifice plate and a differential pressure transmitter. The flow in the secondary loop is measured by a flowmeter. The pressure transmitter and the temperature measuring lines were calibrated and an adjusted equation was added to the data acquisition system. The steady-state is reached after some hours of reactor operation, so that the power dissipated in the cooling system added with the losses should be equal to the core power. The thermal power dissipated in the primary and secondary loops were given by (Mesquita et al., 2011b):

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

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 ΔT is the difference between the temperatures at the inlet and the outlet of the primary loop.

The data acquisition computer program calculates the power dissipated in the cooling loop with the $\dot{m}$ and c_p calculated as function of coolant temperature (Miller, 1989). To calculate the heat losses, one resistance thermometer (PT-100) was 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 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 these losses are calculated by the data acquisition system.

2.2 Overall thermal conductivity of the fuel elements

From Fourier equation it was obtained the expression of overall thermal conductivity (k_g), in [W/mK], for cylindrical fuel elements (Lamarsh and Baratta, 2001; Todreas and Kazimi, 2010).

$$k_g = \frac{q'''r^2}{4\left(T_o - T_{sur}\right)} \tag{2}$$

where q''' is the volumetric rate of heat generation [W/m^3], T_o and T_{sur} are the fuel central temperature and the surface temperature [°C] and r is the fuel element radius [m].

The temperature at the centre of the fuel was measured. The heat transfer regime at the power of 250 kW in all fuel elements is the subcooled nucleate boiling. The cladding outside temperature is the water saturation temperature (T_{sat}) at the pressure of 1.5 bar (atmospheric pressure added up of the water column of ~ 5.2 m), increased of the wall superheat (ΔT_{sat}). The superficial temperature (T_{sur}) in [°C] is found using the expression below, where T_{sat} is equal to 111.37°C (Wagner and Kruse, 1998).

$$T_{sur} = T_{sat} + \Delta T_{sat} \tag{3}$$

It is then:

$$k_g = \frac{q'''r^2}{4\left(T_o - T_{sut} - T_{sat}\right)} \tag{4}$$

with q''' in [W/m^2] and T_{sat} in [°C].

The wall superheat is obtained by using the correlation proposed by McAdams found in Tong and Weisman (1996).

$$\Delta T_{sat} = 0.81\left(q''\right)^{0.259} \tag{5}$$

The fuel element instrumented with three type K thermocouples was introduced into position B1 Two thermocouples were also placed in two core channels adjacent to position B1 (hot channel) (Figure 1 and Figure 3).

Some data from the instrumented fuel element used in the calculations are found in Table 1. The power supplied by this element is the core total power multiplied by its active area divided by the total heat transfer area of all fuel elements.

Table 1 Instrumented fuel element features (Gulf General Atomic, 1972)

Parameter	*Value*
Heated length	38.1 cm
Outside diameter	3.76 cm
Active outside area	450.05 cm^2
Fuel outside area ($U\text{-}ZrH_{1.6}$)	434.49 cm^2
Fuel element active volume	423.05 cm^3
Fuel volume ($U\text{-}ZrH_{1.6}$)	394.30 cm^3
Power (total of the core = 265 kW)	4.518 kW

2.3 Heat transfer regimes of the cladding to coolant

2.3.1 Heat transfer coefficient in turbulent single phase flow

Dittus-Boelter 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 (Collier and Thome, 1994):

$$h_{sp} = \frac{0.023k\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4}}{D_w} \quad \text{or} \quad h_{sp} = 0.023\frac{k}{D_w}\left(\frac{GD_w}{\mu}\right)^{0.8}\left(\frac{c_p\mu}{k}\right)^{0.4} \tag{6}$$

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^2s], c_p is the isobaric specific heat in [J/kgK], k is the thermal conductivity in [W/mK] and μ is the fluid dynamic viscosity in [kg/ms].

The water thermodynamic properties at the bulk water temperature on the sub-saturated at 1.5 bar were taken from Wagner and Kruse (1998) 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 hottest channel in the core is the Channel 1 (Figure 3). Table 2 gives the geometric data of Channel 1, and the percent contribution of each fuel element to the channel power.

The reactor was operated in steps of about 50 kW until 250 kW (linear neutronic channel) and data were collected in function of the power supplied to Channel 1. In this channel, from about 50 kW, the heat transfer regime changes from single-phase to subcooled nucleate boiling (Mesquita et al., 2011a).

Figure 3 Detail of the hottest channel in the core (see online version for colours)

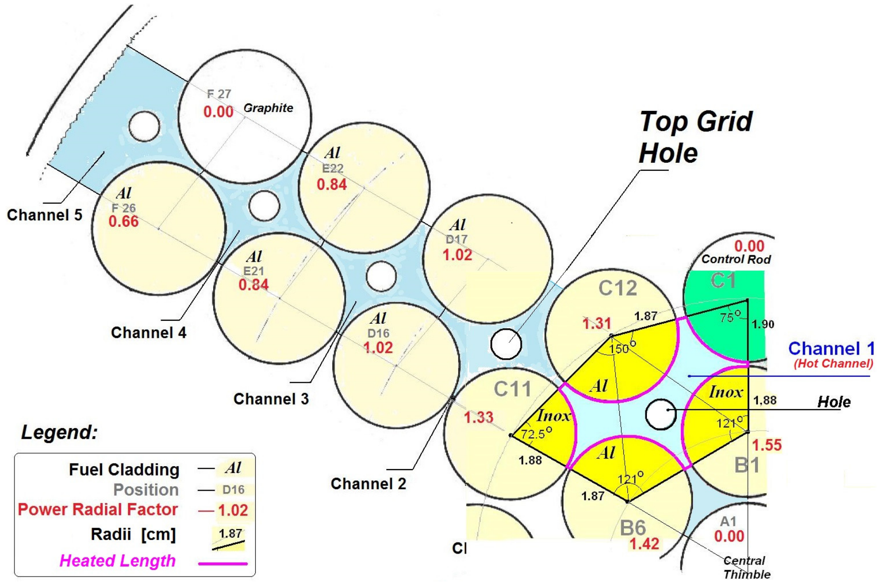


Table 2 Channel 1 characteristics (Mesquita et al., 2011a)

Parameter	*Channel 1*	*Unit*
Area (A)	8.214	cm^2
Wetted Perimeter (P_w)	17.643	cm
Heated Perimeter (P_h)	15.156	cm
Hydraulic Diameter (D_w)	1.862	cm
B1 and C1 Fuel Diameter (stainless)	3.76	cm
B6 and C12 Fuel Diameter (Al)	3.73	cm
C1 Control Rod Diameter	3.80	cm
Central Thimble	3.81	cm
Core Total Power (265 kW)	100	%
B1 Fuel Contribution	1.11	%
B6 Fuel Contribution	0.94	%
C11 Fuel Contribution	0.57	%
C12 Fuel Contribution	1.08	%
Total Power of the Channel	3.70	%

2.3.2 Heat transfer coefficient in subcooled nucleate boiling

For local boiling the Newton equation of cooling is used:

$$h_b = \frac{q''}{T_{sur} - T_f} \tag{7}$$

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] and T_{sur} is the surface temperature [°C], given by equation (3) ($T_{sur} = T_{sat} + \Delta T_{sat}$). The surface superheat was calculated by the McAdams correlation equation (5) ($\Delta T_{sat} = 0.81(q'')^{0.259}$) (Collier and Thome, 1994). 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 as modified by Kreith et al. (2011):

$$h_{sur} = q''/\Delta T_{sat} \tag{8}$$

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 ΔT_{sat} is the wall superheat [°C].

2.4 Heat transfer coefficient in the fuel gap

The instrumented fuel element is composed of 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. The thermocouples are fixed in the central rod. It is assumed that all heat flux is in the radial direction. Using the analogy with electric circuits, the resistance to the heat conduction from the fuel centre to the coolant (R_g) is given by the sum of the fuel component resistances (Figure 4). Table 3 presents the thermal resistance equations to fuel element geometry. Table 4 shows the thermal conductivity as a function of temperature.

Figure 4 Fuel element configuration (see online version for colours)

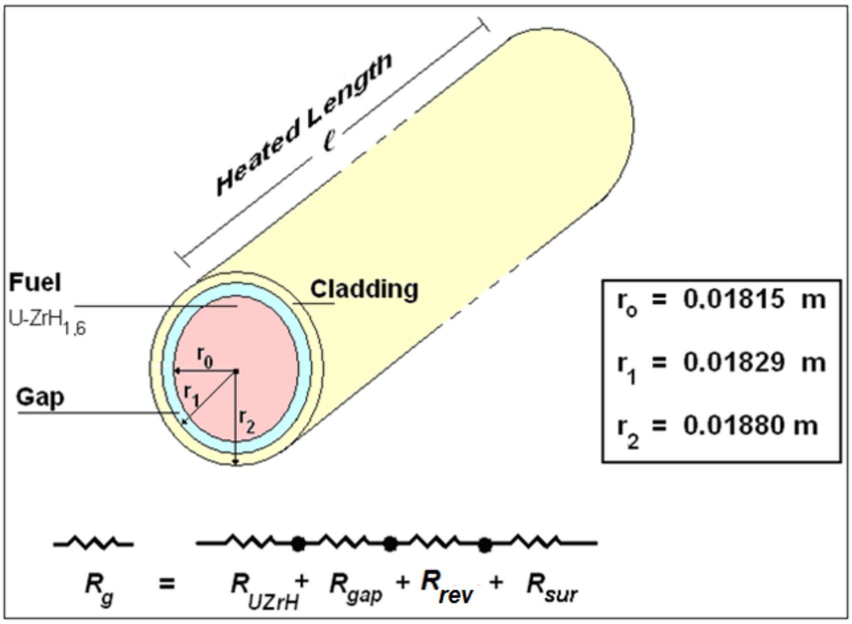


Table 3 Thermal resistance for conduction (Holman, 2002)

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

Table 4 Thermal conductivity as a function of temperature

Material	*k (T)*	*Reference*
Zirconium (Zr)	4.0×10^{-3} T + 21.23	Glasstone and Sesonske (1994)
Uranium/zirconium hydride ($U\text{-}ZrH_{1.6}$)	0.0075 T + 17.58	Simnad (1981)
Stainless steel AISI 304	$3.17 \times 10^{-9} \times T^3 - 6.67 \times 10^{-6}\, T^2 + 1.81 \times 10^{-2}\, T + 14.46$	ASME (1992)

The value of R_{gap} is the value of the overall resistance of the fuel element (R_g) less the values of other component resistance. It is found with the values of k_g and h_{sur} obtained previously and with the values of k for the fuel alloy and for the cladding corrected in function of temperature. The heat transfer coefficient in the gap is:

$$h_{gap} = \frac{2}{r_0}\left(\frac{k_g k_{UZrH} k_{rev}}{k_{UZrH} k_{rev} - k_g k_{rev} - 2k_g k_{UZrH}\, \ell n\left(r_2/r_1\right)}\right) \tag{9}$$

2.5 Calibration of instruments and uncertainties analysis

The primary calibration of measurement instruments was performed comparing all the measurement chain (sensor, cables, and acquisition system) with reference standards. It was determined the standard deviations of measures average (S_x), the correction equations, the determination coefficients (R^2) (R = correlation coefficient), and error standards of the adjusted curves. The expressions found in the regression were added to the data acquisition program. The primary parameter uncertainties were used in their propagation within the experimental results. Was also taken into account the uncertainty associated in the calibration standard.

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 (Holman, 1998). In the found expressions, the contributions of the uncertainties associated with the geometry of the fuel element are negligible owing to the rigorous tolerances specified in the maker's drawings (Gulf General Atomic, 1972). 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 method adopted to calculate the propagation of uncertainty was proposed by Kline and McClintock (1953) described by Holman (1998). 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. It is wished 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 $x_1, x_2, x_3, \ldots, x_n$. Thus,

$$R = R(x_1, x_2, x_3, \ldots x_n) \tag{10}$$

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:

$$U_R = \sqrt{\left(\frac{\partial R}{\partial x_1}U_1\right)^2 + \left(\frac{\partial R}{\partial x_2}U_2\right)^2 + \cdots + \left(\frac{\partial R}{\partial x_3}U_3\right)^2} \tag{11}$$

3 Results

3.1 Thermal power calibration

Table 5 presents the results of the reactor thermal power as described in Section 2.1. Other calibration data is also shown in this table (Mesquita and Souza, 2014). The uncertainty in the thermal power of the reactor was 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 estimations of the specific heat of the water obtained in function of its temperature were also considered. 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) ($q = \dot{m}\, c_p\, \Delta T$).

Table 5 The TRIGA reactor thermal power (Mesquita and Souza, 2014)

Average flow rate (from 28 m^3/h to 33 m^3/h)	32.7 ± 0.41 m^3/h (±1.1%)
Average inlet primary temperature	41.7 ± 0.3°C
Average outlet primary temperature	34.8 ± 0.3°C
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
Power uncertainty	**±19 kW (±7.2%)**
Heat power dissipated in the secondary loop	248 kW

Using equation (11), the uncertainty in the thermal power is:

$$\frac{U_q}{q} = \sqrt{\left(\frac{U_{\dot{m}}}{\dot{m}}\right)^2 + \left(\frac{U_{c_p}}{c_p}\right)^2 + \left(\frac{U_{T_{in}}}{T_{in} - T_{out}}\right)^2 + \left(\frac{U_{T_{out}}}{T_{in} - T_{out}}\right)^2} \tag{12}$$

where $U_{\dot{m}}$, U_{c_p}, $U_{T_{in}}$ and $U_{T_{out}}$ are respectively the consolidated uncertainties of the primary variables $\dot{m}$, c_p, T_{in} and T_{out}. To the found value should be added the standard deviation S_q of the thermal power registered by data acquisition system. Then the value of the uncertainty is,

$$\frac{U_q}{q} = \sqrt{\left(\frac{U'_q}{q}\right)^2 + \left(\frac{S_q}{q}\right)^2} \tag{13}$$

The power calibration is performed when the system is in thermal equilibrium with the environment. Therefore, the heat losses are very small compared to the total power (about 1.5%). Uncertainties in the amount of heat loss are thus insignificant. It considered the uncertainty in the thermal power to be equal the power dissipated in the heat exchanger. The uncertainty in the value of specific heat of water is very low and can be neglected also (Miller, 1989). Using equation (13) it found the value of 7.2% for the uncertainty in the thermal power supplied by the core.

3.2 Overall thermal conductivity of the fuel elements (k_g)

Table 6 shows in column 7 the k_g found from equation (2), as a function of power, with the instrumented fuel element inserted in the core position B1. The dissipated power in instrumented fuel element is shown in Table 6 (column 2). This power level was presented in Table 1. In Table 6 this power was multiplied by the radial and axial distribution on the fuel rod factors (radial factor = 1.551, axial factor = 1.25) (Mesquita et al., 2011a). In Table 6, T_o corresponds to the measured temperature in the fuel centre (three thermocouples average).

Table 6 Thermal parameters of the fuel element in subcooled boiling regime (Mesquita et al., 2011a)

q_{core}	q_{B1}	T_o	q'	q''	q'''	ΔT_{sat}	T_{sur}	k_g	h_{sur}
[kW]	*[W]*	*[°C]*	*[W/m]*	*[W/m²]*	*MW/m³*	*[°C]*	*[°C]*	*[W/mK]*	*[kW/m²K]*
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

The values of k_g are shown in Table 6 (column 7). They were found by linear regression of the expression for the fuel element overall thermal conductivity k_g in [W/mK] as a function of total reactor power q (column 1) in [kW]:

$$k_g = 0.0157\, q + 6.5386 \tag{14}$$

with coefficient of determination R^2 = 0.9936, and standard error of the fitted curve: $U_{y,x}$ = 0.104 W/mK.

The overall-thermal conductivity k_g of the fuel element was given by the equation (4). 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 ΔT_{sat} is found using the correlation of McAdams (equation 5) ($\Delta T_{sat} = 0.81(q'')^{0.259}$). Using equation (11) to determine the relative uncertainty of k_g, and the relative uncertainty of ΔT_{sat}, we have as a result the following expression for the relative uncertainty in the overall-thermal conductivity of the fuel element:

$$U_{k_g} = \sqrt{\left(\frac{\partial k_g}{\partial q'''}U_{q'''}\right)^2 + \left(\frac{\partial k_g}{\partial r}U_r\right)^2 + \left(\frac{\partial k_g}{\partial T_o}U_{T_o}\right)^2 + \left(\frac{\partial k_g}{\partial T_{sat}}U_{T_{sat}}\right)^2 + \left(\frac{\partial k_g}{\partial T_{\Delta T_{sat}}}U_{\Delta T_{sat}}\right)^2} \tag{15}$$

Solving the differential equation comes the expression for the uncertainty of k_g:

$$\frac{U_{k_g}}{k_g} = \sqrt{\left(\frac{U_{q'''}}{q'''}\right)^2 + \left(\frac{2U_r}{r}\right)^2 + \left(\frac{U_{T_o}}{T_o - T_{sat} - \Delta T_{sat}}\right)^2 + \left(\frac{U_{T_{sat}}}{T_o - T_{sat} - \Delta T_{sat}}\right)^2 + \left(\frac{U_{\Delta T_{sat}}}{T_o - T_{sat} - \Delta T_{sat}}\right)^2} \tag{16}$$

Dimensional fuel uncertainties are negligible (Gulf General Atomic, 1972), compared to the uncertainty in thermal power. Therefore, it was considered the uncertainty in power per unit of volume (q'''), and the uncertainty in the heat flux at the surface (q), equal the power uncertainty. The water saturation temperature at 1.5 bar is equal to 111.37°C as Wagner and Kruse (1998). Then it has a very low uncertainty, and can also be neglected. Thus the ΔT_{sat} uncertainty is given by:

$$\frac{U_{\Delta T_{sat}}}{\Delta T_{sat}} = \frac{0.259 . U_{q''}}{q''} \tag{17}$$

Replacing $U_{\Delta T_{sat}}$ it has the expression for the relative uncertainty in the fuel element overall thermal conductivity k_g:

$$\frac{U_{k_g}}{k_g} = \left[\left(\frac{U_{q'''}}{q'''}\right)^2 + \left(\frac{2U_r}{r}\right)^2 + \left(\frac{U_{T_o}}{T_o - T_{out} - \Delta T_{sat}}\right)^2 + \left(\frac{U_{T_{sat}}}{T_o - T_{sat} - \Delta T_{sat}}\right)^2 + \left(\frac{0.259 U_{q''} \Delta T_{sat}}{q''\left(T_o - T_{out} - \Delta T_{sat}\right)}\right)^2\right]^{1/2} \tag{18}$$

The uncertainty in the overall-thermal conductivity of the fuel element depends, mainly, on the reactor's thermal power uncertainty. Other components of equation (18) contribute very little to the total uncertainty of this parameter. It reveals no information on the uncertainty in McAdams correlation. Using equation (18) we find an uncertainty of 7.3% for k_g.

3.3 Heat transfer regimes of the cladding to coolant

3.3.1 Heat transfer coefficient for cladding to coolant in single-phase regime (h_{sp})

The regime of heat transfer in the single-phase natural convection region occurs only until about 50 kW in the hottest channel. Table 7 shows the coolant properties as a function of power to the channel beside the position B1 of the core (the hottest channel). In Table 7, G is the mass flux given by $G = \dot{m}$ */channel area* and u is the velocity given by $u=G/\rho$, where ρ is the water density (995 kg/m^3). The water thermodynamic properties are calculated at the bulk water temperature on the sub-saturated at 1.5 bar (Wagner and Kruse, 1998). Table 7 shows, in last column, the heat transfer coefficient in the single-phase flow (h_{sur}) calculated by the Dittus-Boelter's correlation (Mesquita et al., 2011a).

Table 7 Coolant properties and the single-phase heat transfer coefficient (Mesquita et al., 2011a)

qCore [kW]	*qChannel [kW]*	ΔT [^oC]	c_p [kJ/kgK]	$\dot{m}$ [kg/s]	G [kg/m^{2}s]	U [m/s]	μ [10^{-3}kg/ms]	k [W/mK]	*Re*	*Pr*	h_{sur} [kW/m^{2}K]
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**

3.3.2 Heat transfer coefficient in subcooled nucleate boiling (h_{sur})

Table 6 shows in the last column h_{sur} starting from values found from equation (7), as a power function (column 1). The instrumented fuel element was positioned in the B1 position. The power dissipated in Channel 1 is shown in column 2. The following expression was found by linear regression to the heat transfer coefficient in subcooled nucleate boiling (h_{sur}) as function of the power:

$$h_{sur} = 0.0317q + 1.9144 \tag{19}$$

with coefficient of determination R^2 = 0.9988, and standard error of the fitted curve: $U_{y,x}$ = 0.179 kW/m^{2}K (Mesquita et al., 2011a).

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 equation (8) ($h_{sur} = q''/\Delta T_{sat}$). Using equation (11) the uncertainty in $U_{h_{sur}}$ is given by:

$$U_{h_{sur}} = \sqrt{\left[\frac{\partial h_{sur}}{\partial q''} U_{q''}\right]^2 + \left[\frac{\partial h_{sur}}{\partial \Delta T_{sat}} U_{\Delta T_{sat}}\right]^2} \tag{20}$$

Solving the partial differential equation find the following expression for the relative uncertainty of h_{sur}:

$$\frac{U_{h_{\sup}}}{h_{\sup}} = \sqrt{\left[\frac{U_{q''}}{q''}\right]^2 + \left[\frac{U_{\Delta T_{sat}}}{\Delta T_{sat}}\right]^2} \tag{21}$$

The relative uncertainty of ΔT_{sat} was found previously (equation 17). Substituting in this equation the relative uncertainty of h_{sup} as a function of the heat flux at the surface:

$$\frac{U_{h_{sur}}}{h_{sur}} = \sqrt{\left[\frac{U_{q''}}{q''}\right]^2 + \left[\frac{0.259 U_{q''}}{q''}\right]^2} \tag{22}$$

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, finally gives an uncertainty of 7.4% for h_{sur}.

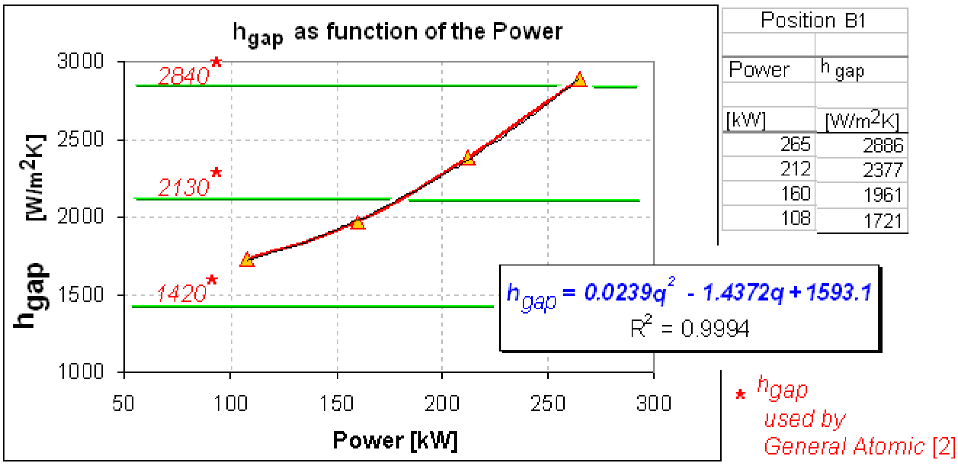
3.4 *Heat transfer coefficient in the fuel gap (h_{gap})*

The estimate for the heat-transfer coefficient in gap was done by the equation (9). The results were plotted on a graph as a function of the reactor power (Figure 5). This figure also shows three theoretical values recommended by General Atomic (1970) for the heat transfer coefficient. By regression it found the polynomial for h_{gap} value in [W/m^{2}K], as a function of the reactor power q in [kW]:

$$h_{gap} = 0.0239\, q^2 - 1.4372\, q + 1593.1 \tag{23}$$

with coefficient of determination R^2 = 0.9994, and standard error of the fitted curve: $U_{y,x}$ = 15 W/m^{2}K (Mesquita et al., 2011a).

Figure 5 Heat transfer coefficient through the gap as a function of the power (see online version for colours)



Using equation (11) the expression for uncertainty in the coefficient h_{gap} is:

$$U_{h_{gap}} = \sqrt{\left(\frac{\partial h_{gap}}{\partial r_0}U_{r_0}\right)^2 + \left(\frac{\partial h_{gap}}{\partial k_g}U_{k_g}\right)^2 + \left(\frac{\partial h_{gap}}{\partial k_{UZrH}}U_{kUZrH}\right)^2 + \left(\frac{\partial h_{gap}}{\partial k_{rev}}U_{k_{rev}}\right)^2 + \left(\frac{\partial h_{gap}}{\partial r_1}U_{r_1}\right)^2 + \left(\frac{\partial h_{gap}}{\partial r_2}U_{r_2}\right)^2} \tag{24}$$

Solving the partial differential equation gives the following expression for the uncertainty in the value of h_{gap}:

$$\frac{U_{h_{gap}}}{h_{gap}} = \left[\left(\frac{U_{r_0}}{r_0} \right)^2 + \left(\frac{U_{k_g}}{k_g} \right)^2 + \left(\frac{U_{k_{UZrH}}}{k_{UZrH}} \right)^2 + \left(\frac{U_{k_{rev}}}{k_{rev}} \right)^2 + \left(\frac{U_{r_1} r_0 h_{gap}}{r_1 k_{rev}} \right)^2 + \left(\frac{U_{r_2} r_0 h_{gap}}{r_2 k_{rev}} \right)^2 \right]^{1/2} \tag{25}$$

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

4 Conclusions

Understanding the behaviour of the operational parameters of nuclear reactors allows the development of improved analytical models to predict the fuel temperature, and contributes to their safety. Developments and innovations used for research reactors can be later applied to larger power reactors. Their relatively low cost allows research reactors to provide an excellent testing ground for the reactors of tomorrow.

The uncertainty analysis associated with values of the experimental measurement and the expressions deduced to calculate propagation of uncertainties in thermal power and heat-transfer coefficients, always take into account the physical law equations used in theoretical calculations. 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 (Gulf General Atomic, 1972). 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 $U = \pm 0.4$°C for resistance temperature detectors, and $U = \pm 1.0$°C for thermocouples. 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 uncertainty analysis on thermal hydraulic parameters is determined, basically, by the uncertainty of the reactor's thermal power (q). The other parts of the propagation equation are negligible. 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 and also to the estimations of the specific heat of the water obtained in function of its temperature. It was found an uncertainty of 7.2% in the thermal power supplied by the core. The uncertainty in the fuel element overall-thermal conductivity (k_g) was 7.3%. The uncertainty in the heat-transfer coefficient of the fuel cladding outer surface to the water (h_{sur}) was 7.4%. The uncertainty in the heat-transfer coefficient in gap (h_{gap}) was 7.5%.

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