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EVALUATION OF THERMAL-HYDRAULIC PARAMETER UNCERTAINTIES IN A TRIGA RESEARCH REACTOR

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ABSTRACT

Experimental studies had been performed in the TRIGA Research Nuclear Reactor of CDTN/CNEN to find out the its thermal hydraulic parameters. Fuel to coolant heat transfer patterns must be evaluated as function of the reactor power in order to assess the thermal hydraulic performance of the core. 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 from single-phase natural convection to subcooled nucleate boiling. This paper presents the uncertainty analysis in the results of the thermal hydraulics 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. The uncertainty analysis on thermal hydraulics parameters of the CDTN TRIGA fuel element is determined, basically, by the uncertainty of the reactor's thermal power.

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. Understanding the behavior of the operational parameters of nuclear reactors allow the development of improved analytical models to predict the fuel temperature, and contributing to their safety.

The natural disaster that caused damage in four reactors at the Fukushima nuclear power plant shows the importance of studies and experiments on natural convection to remove heat from the residual remaining after the shutdown. Experiments, 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 IPR-R1 TRIGA (*Instituto de Pesquisas Radioativas*, Reactor nr. 1, Training, Research, Isotopes, General Atomic), showed in Fig. 1 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 and with graphite reflectors. 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.

Figure 1: The TRIGA reactor at CDTN.

Experimental studies had 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 [1]. 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 thermo-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.

2. THE CDTN TRIGA REACTOR

The IPR-R1 TRIGA reactor at CDTN in Belo Horizonte is a typical TRIGA Mark I light-water reactor cooled by assisted natural convection with an annular graphite reflector. It is a research pool reactor and the core is placed at the bottom of an open tank of about 6 m height and 2 m 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 235U with a cylinder of metallic zirconium inside and aluminum or stainless steel cladding. The reactor core has 63 cylindrical fuel elements, 58 aluminum-clad fuel elements, and 5 stainless steel-clad fuel elements. One steel-clad fuel element was instrumented in the center with thermocouples, and was placed in the core to the experiments. The moderating effects are entrusted both to the light water coolant, and to the zirconium hydride in the mixture. The hydride fuel possesses a very high negative prompt temperature coefficient that is the main reason for the high inherent safety behavior of the TRIGA reactors. This temperature-reactivity coefficient allows great freedom in steady state and transient operations.

2.1. The data acquisition system

An operational computer program, and a data acquisition and signal processing system were developed in order to facilitate the experiments performed in the reactor [2]. The information of reactor operation is displayed on the computer screen. The color graphic monitors display real-time operation data in concise, accurate, and easily understood formats. Besides showing the real-time performance of the plant, the system stores the information in a computer hard disk, with an accessible historical database. Some of the parameters monitored are: the control rod positions and their reactivity, the reactor power, the fuel and water temperatures, the radiation levels, the primary cooling system flow, the water pool level and so on. About forty variables are registered by the data acquisition system. The system responds to the IAEA recommendations on the monitoring and recording of the operational variables [3]. Figure 2 shows one of the video-screen displays of the digital monitoring system computer that consolidates information for the reactor power status in real time

Figure 2: One of the user interface of the data acquisition system developed for IPR-R1 reactor.

2.2 The instrumented fuel element

The original fuel element at the hottest location in the core (largest thermal power production) 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 chromel-alumel thermocouples (K type), embedded in the fuel meat. The sensitive tips of the thermocouples are located along the fuel centerline. Their axial position is one at the half-height of the fuel meat and the other two 2.54 mm above and 2.54 mm below. Figure 3 shows the instrumented fuel element before and after it was been positioned in the IPR-R1 reactor core.

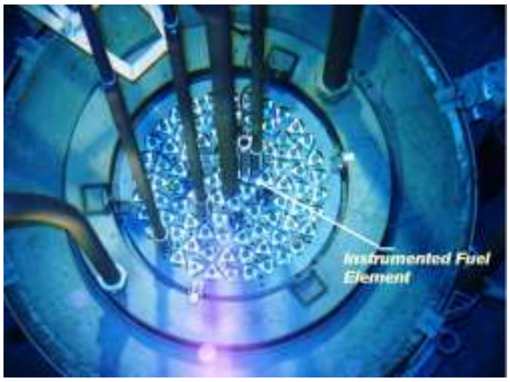

Figure 3: Instrumented fuel element before and after it was been positioned in the IPR-R1 reactor core.

3. THERMAL-HYDRAULIC EXPERIMENTS

The reactor power was calibrated and monitored using thermal techniques, and after the channels of the neutron power measurements were adjusted. The basic safety limit for the TRIGA reactor system is the fuel temperature, both in steady-state and pulsed mode operation. Several experiments were carried out with the measurement of the temperature inside the fuel element, in the reactor core, and at different height of the reactor pool. During these experiments the reactor was set in many different power levels up to 250 kW. The mass flow rate through the core coolant hot channel was indirectly determined from the heat balance across the channel. This was done using measurements of the water entrance and exit temperatures, with the forced cooling system switched off and on. The thermal conductivity of the fuel and the heat transfer coefficient from the cladding to the coolant were also evaluated experimentally. It was also presented a correlation for the gap conductance, between the fuel and the cladding, using the instrumented fuel element [4].

3.1 Power calibration

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, according to Equation 1 [5],

$$q = \dot{m} \; . \; c_p \; . \; \Delta T \quad . \qquad (1)$$

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. Table 1 presents the results and some calibration data.

Table 1: The IPR-R1 TRIGA Reactor thermal power [5]

Average flow rate	32.7 ± 0.4 m^3/h
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
Average power uncertainty	±19 kW (±7.2%)
Heat power dissipated in the secondary loop	248 kW

3.2 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 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 [6]:

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

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^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]. 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 ρ 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 [7]. 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	q Channel	Δ *T*	c_p	$\dot{m}$	*G*	*u*	*μ*	*k*	Re	Pr	h_{sur}
[kW]	[kW]	[ºC]	[kJ/kgK]	[kg/s]	[kg/m²s]	[m/s]	[10^{-3}kg/ms]	[W/mK]			[kW/m²K]
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:

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

Where h_b is the coefficient for nucleate boiling heat transfer; q'' is the heat transfer rate per unit of surface area [W/m²]; T_f is the bulk fluid temperature [ºC]; T_{sur} is the surface temperature [ºC], given by:

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

The surface superheat was calculated by the McAdams correlation [6];

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

With q'' in [W/m²] and ΔT_{sat} in [ºC]. This correlation reproduces experimental data for subcooled water from 11 to 83 ºC, 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 [8]:

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

Where h_{sur} is the heat transfer coefficient for local pool boiling between the cladding surface and the coolant [kW/m²K], q'' is the heat flux in fuel surface [kW/m²] and ΔT_{sat} is the wall superheat [ºC]. 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}	q_{B1}	T_o	q'	q''	q'''	Δ T_{sat}	T_{sur}	k_g	h_{sur}
[kW]	[W]	[ºC]	[W/m]	[W/m²]	MW/m³	[ºC]	[ºC]	**[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**

4. 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 [9]. 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 [10]. 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 $U = \pm 0.4$ ºC for resistance temperature detectors, and $U = \pm 1.0$ ºC for thermocouples [5].

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%) [5]. The method adopted to calculate the propagation of uncertainty was proposed by Kline and McClintock [11]. 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:

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

4.2 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

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

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,

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

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.

4.3 Uncertainty in the overall-thermal conductivity of the fuel element k_g

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

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

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 ºC, with a very low value of relative uncertainty that could be negligible. The superheating ΔT_{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 ΔT_{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.

4.4 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:

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

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}.

4.5 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 ($U\text{-}ZrH_{1,6}$), 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	$R = ln(r_{ext}/r_{int})/2 \ \pi \ \ell$	$\Delta T = q''' \ r_o^2 \ ln(r_{ext}/r_{int})/2k$
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

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

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

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

The substitution of the numerical values gives an uncertainty to the heat-transfer coefficient in gap (h_{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).

3. CONCLUSIONS

The uncertainties 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 taking into account the law-physical 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 [10]. 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 = ±0.4 ºC for resistance temperature detectors, and U = ±1.0 ºC 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 = ±0.41 m^3/h (±1.1%).

The uncertainty analysis on thermo-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 % and the uncertainty in the heat-transfer coefficient of the fuel cladding outer surface to the water (h_{sur}) was 7.4 % .

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REFERENCES

1. Mesquita, A.Z.; Palma, D.A.P.; Costa, A.L; Pereira,C., Veloso, M.A.F., Reis, P.A.L. *Experimental Investigation of Thermal Hydraulics in the IPR-R1 TRIGA Nuclear Reactor*. Editor: Amir Z. Mesquita & Petra Zobic. Book: Nuclear Reactors. 1 ed. InTech Publisher: Rijeka, Croatia, v. 1, p. 23-58. (2012). DOI: 10.5772/25869.
2. Mesquita, A.Z.; Silva, V.V.A; Santos, A.A.C.; Campolina, D.A.M.; Schweizer, F.L.A; Palma, D.A.P. *Real-Time Monitoring of Dynamic Behavior in the CDTN TRIGA Research Reactor*. International Journal of Advanced Engineering Applications, v. 1, p. 15-24. (2013).

3. IAEA - International Atomic Energy Agency. *Specifications of Requirements for Upgrades Using Digital Instrumentation and Control Systems*, IAEA-TECDOC-1066, IAEA, Vienna. (1999).
4. Mesquita, A.Z.; Souza, R.M.G.P. *Thermal-hydraulic and neutronic experimental research in the TRIGA reactor of Brazil*. Progress in Nuclear Energy (New Series), v. 76, p. 183-190. (2014). doi:10.1016/j.pnucene.2014.05.022.
5. Mesquita, A.Z.; Rezende, H.C.; Souza, R.M.G.P. *Thermal Power Calibrations of the IPR-R1 TRIGA Reactor by the Calorimetric and the Heat Balance Methods*. Progress in Nuclear Energy (New series), v. 53, p. 1197-1203. (2011). doi:10.1016/j.pnucene.2011.08.003.
6. Collier, J.G.; Thome, J.R. *Convective Boiling and Condensation*. 3rd. Ed. Clarendon Press, Oxford. (1994).
7. Wagner, W. and Kruse, A. *Properties of Water and Steam - The Industrial Standard IAPWS-IF97 for the Thermodynamics Properties*. Springer, Berlin, 354p. (1998).
8. Tong, L.S. and Weisman, J. *Thermal Analysis of Pressurized Water Reactors*, Third Edition, American Nuclear Society. Illinois. (1996).
9. Holman, J.P, *Experimental Methods for Engineers.* 7th ed. McGraw-Hill: Boston. 689p. (1998).
10. Gulf General Atomic, 15 *SST Fuel Element Assembly Instrumented Core*. San Diego, CA.. Drawing Number TOS210J220. (1972).
11. Kline, S.J.; McClintock, F.A. *Describing Uncertainties in Single-Sample Experiments*. Mechanical Engineering. 3-8. (1953).