EXPERIMENTAL INVESTIGATION OF THE ONSET OF NUCLEATE BOILING IN THE IPR-R1 TRIGA NUCLEAR RESEARCH REACTOR

Amir Zacarias Mesquita, Hugo César Rezende

Centro de Desenvolvimento da Tecnologia Nuclear/Comissão Nacional de Energia Nuclear, CDTN/CNEN,
Campus da UFMG - Pampulha, Caixa Postal, 941, CEP: 30.161-970, Belo Horizonte,
amir@.cdtn.br, hcr@cdtn.br

ABSTRACT

The IPR-R1 TRIGA Nuclear Research Reactor at the Nuclear Technology Development Center (CDTN), in Belo Horizonte (Brazil)), is a pool type reactor cooled by natural circulation. Fuel to coolant heat transfer patterns must be evaluated as function of the reactor power in order to evaluate 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. Experimental results indicated that subcooled pool boiling occurs at the cladding surface in the reactor core central channels at power levels in excess of 60 kW. However, due to the high heat transfer coefficient in subcooled boiling the cladding temperature is quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. An operational computer program and a data acquisition and signal processing system were developed as part of this research project to allow on line monitoring of the operational parameters.

INTRODUCTION

The IPR-R1 TRIGA (*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 kWth, cooled by light water and with graphite reflectors. The fuel is an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U. The reactor core, showed in Fig. 2, has 63 cylindrical fuel elements, 58 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 with three thermocouples along its center axis (Fig. 3).

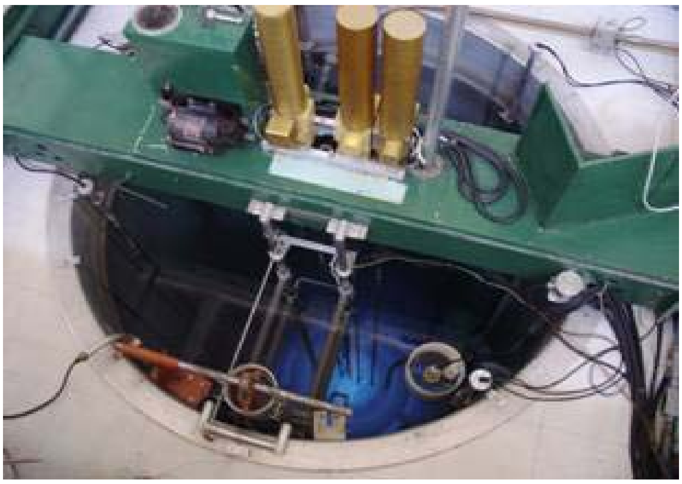

Figure 1 – Upper view of the IPR-R1 TRIGA Reactor

Experimental studies have been performed in the IPR-R1 Reactor [1] 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.

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.

The thermal conductivity (k) of metallic alloys is mainly a function of temperature. In nuclear fuels, this relationship is more complicated because k also becomes a function of irradiation as a result of the changes in the chemical and physical composition (porosity changes due to temperature and fission products). Many factors affect the fuel thermal conductivity. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO_2 content, pellet cracking, and burnup. The second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist to evaluate its value in power reactors fuels, which use mainly uranium oxide [2]. The only reference found to TRIGA reactors fuel is General Atomic [3] that recommends the use of three hypotheses for the heat transfer coefficient through the gap. The heat transfer coefficient (h) is a property not only of the system but it also depends on the fluid properties. The determination of h is a complex process that depends on the thermal conductivity, density, viscosity, velocity, dimensions and specific heat. All these parameters are temperature-dependent and change when heat is being transferred from the heated wall to the fluid. An operational computer program and a data acquisition and signal processing system were developed as part of this research project to allow on line monitoring of the operational parameters [4].

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.

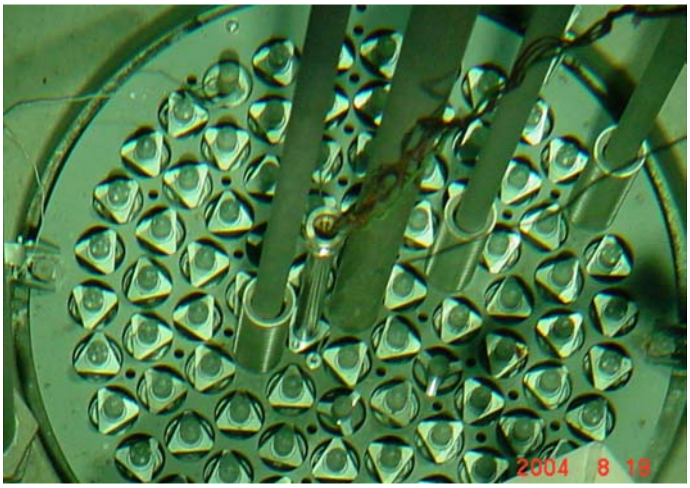

Figure 2 - Core top view with the instrumented fuel element in ring B

OVERALL THERMAL CONDUCTIVITY OF THE FUEL ELEMENTS

From Fourier equation described in [5, 6], it was obtained the expression for overall thermal conductivity (k_g), in [W/mK], for cylindrical fuel elements

$$k_g = \frac{q''' r^2}{4(T_o - T_{sur})} \quad , \tag{1}$$

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

The temperature at the center of the fuel was measured. The heat transfer regime at the power of 265 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 [^{o}C] is found using the expression below, where T_{sat} is equal to 111.37 ^{o}C [7].

$$T_{sur} = T_{sat} + \Delta T_{sat} \quad . \tag{2}$$

The wall superheat is obtained by using the correlation proposed by McAdams found in [8],

$$\Delta T_{sat} = 0.81(q'')^{0.259} , \tag{3}$$

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

A fuel element instrumented with three type K thermocouples was introduced into position B1 of the core. Figure 3 show the diagram of the instrumented fuel element and the Table 1 presents its mean characteristics. Figure 4 shows the core configuration, two thermocouples were placed in two core channels adjacent to ring B.

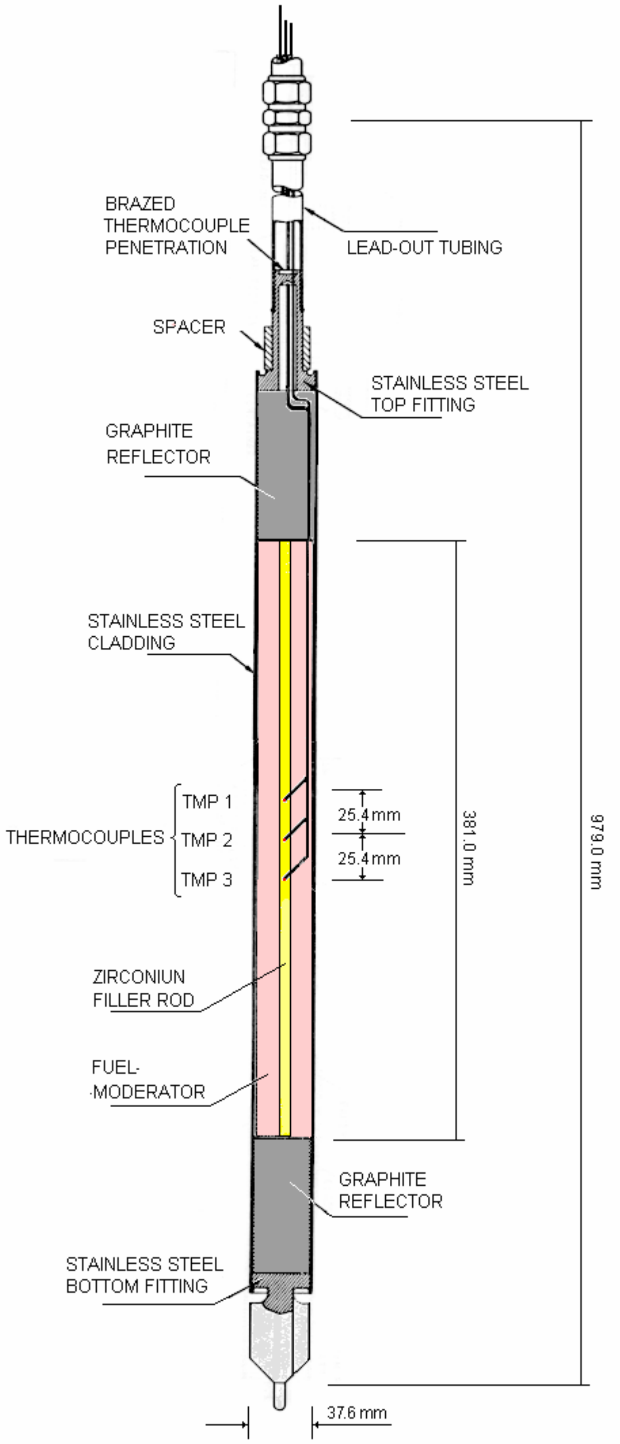

Figure 3 - The instrumented fuel element

Table 1 – Instrumented fuel element features

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

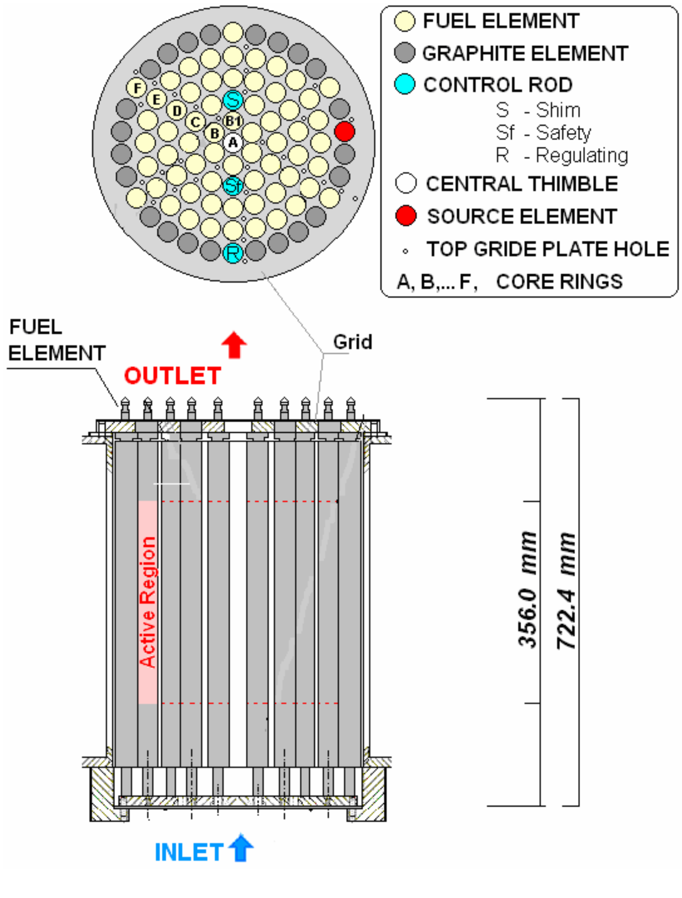

Figure 4 - The core of the IPR-R1 TRIGA Reactor

Single-Phase Region

The heat transfer coefficient in single-phase region (h_{sp}) was calculated with the Dittus-Boelter correlation described in [9] valid for turbulent flow in narrow channels, given for:

$$h_{sp} = 0.023 \frac{k}{D_w} \left(\frac{GD_w}{\mu} \right)^{0.8} \left(\frac{c_p \mu}{k} \right)^{0.4}, \qquad (4)$$

where $D_w = 4A/P_w$ is the hydraulic diameter of the channel based on the wetted perimeter; A is the flow area [m^2]; P_w is the wetted perimeter [m]; G is the mass flow [kg/m^2s]; c_p is the isobaric specific heat [J/kgK]; k is the thermal conductivity [W/mK]; and, μ is the fluid dynamic viscosity [kg/ms]. The fluid properties for the IPR-R1 TRIGA core are calculated for the bulk water temperature at 1.5 bar.

The two hottest channels in the core are Channel 0 and Channel 1' (Fig. 5). The heat transfer coefficient was estimated using the Dittus-Boelter correlation. In the top gride plate above the Channel 1' there is a hole to insert thermocouples. Above the Channel 0 there is not any hole. The inlet and outlet temperatures in Channel 0 were considered as being the same as in Channel 1'. Table 2 gives the geometric data of Channel 0 and Channel 1' and the percent contribution of each fuel element to the channel power. The curves of single-phase heat transfer, as function of ΔT_{sat}, are presented in the Fig. 6.

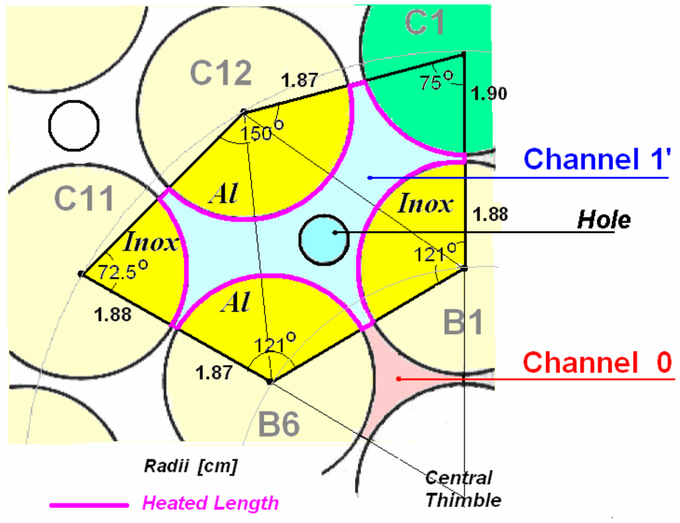

Figure 5 - The two hottest channels in the core

The mass flow rate is given indirectly from the thermal balance along the channel using measurements of the water inlet and outlet temperatures:

$$q = \dot{m} c_p \Delta T , \qquad (5)$$

where q is the power supplied to the channel [kW]; $\dot{m}$ is the mass flow rate in the channel [kg/s]; c_p is the isobaric specific heat of the water [J/kgK]; and, ΔT is the temperature difference along the channel [°C].

Table 2. Channel 0 and Channel 1' characteristics [1]

	Channel 0	Channel 1'	Unit
Area (A)	1.574	8.214	cm^2
Wetted Perimeter (P_w)	5.901	17.643	cm
Heated Perimeter (P_h)	3.906	15.156	cm
Hydraulic Diameter (D_w)	1.067	1.862	cm
B1 and C1 Fuel Diameter (stainless)	3.76	3.76	cm
B6 and C12 Fuel Diameter (Al)	3.73	3.73	cm
C1 Control Rod Diameter	3.80	3.80	cm
Central Thimble	3.81	3,81	cm
Core Total Power (265kW)	100	100	%
B1 Fuel Contribution	0.54	1.11	%
B6 Fuel Contribution	0.46	0.94	%
C11 Fuel Contribution	-	0.57	%
C12 Fuel Contribution	-	1.08	%
Channel Total Power	1.00	3.70	%

The reactor was operated on steps of about 50 kW until 265 kW and data were collected in function of the power supplied to Channel 1' and Channel 0. The values of the water thermodynamic properties at the pressure 1.5 bar as function of the bulk water temperature at the channel were taken from Wagner and Kruse [7]. The curve for heat transfer coefficient (h_{sur}) in the single-phase region is shown in Fig. 7 as function of the power and Table 3 presents the water proprieties.

Table 3 – Cooling proprieties and single phase convective coefficient

q Core [kW]	q Channel [kW]	ΔT [°C]	c_p [kJ/kgK]	$\dot{m}$ [kg/s]	G [kg/m²s]	u [m/s]	μ [10^{-3}kg/ms]	k [W/mK]	Re	Pr	h_{sur} [kW/m²/K]
Canal 1'											
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**
Canal 0											
265	2.65	13.9	4.1809	0.046	289.71	0.29	0.549	0.639	5630	3.6	**2.300**
212	2.12	9.6	4.1800	0.053	335.65	0.34	0.575	0.638	6228	3.8	**2.537**
160	1.6	7.0	4.1795	0.055	347.45	0.35	0.596	0.636	6220	3.9	**2.566**
108	1.08	4.6	4.1793	0.056	356.91	0.36	0.620	0.634	6142	4.1	**2.576**
53	0.53	2.5	4.1789	0.051	322.31	0.32	0.638	0.632	5390	4.2	**2.342**
35	0.35	1.8	4.1780	0.047	295.68	0.30	0.642	0.630	4914	4.3	**2.176**

Subcooled Nucleate Boiling Region

For the subcooled nucleated boiling region (local or surface boiling), the expression used is shown below, according to [10, 11]:

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

where h_{sur} is the convective heat-transfer coefficient from the fuel cladding outer surface to the water [kW/m²K]; q'' is the fuel surface heat flux [kW/m²]; and, ΔT_{sat} is the surface superheat in contact with the water [°C].

Table 4 presents the thermal parameter of the fuel element. Figure 6 shows the fuel element surface heat transfer coefficient for the coolant as a function of the superheat, in both regimes. This curve is specific for the IPR-R1 TRIGA reactor conditions. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling - ONB) is approximately 60 kW as shown in the graph.

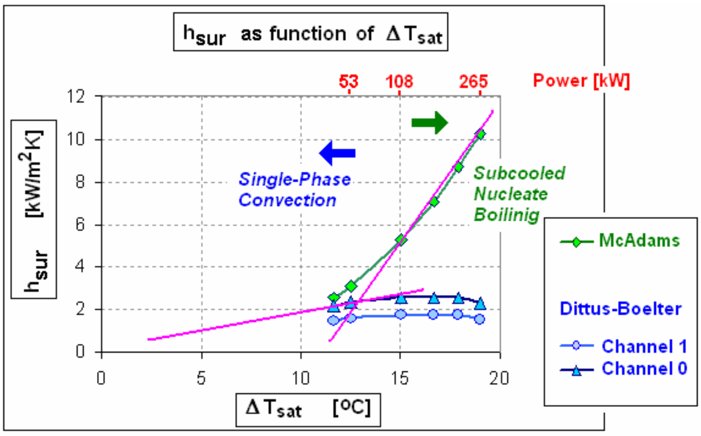

Figure 6 - The onset of nucleate boiling in the fuel element surface of the IPR-R1 TRIGA Reactor

Table 4 – Fuel element thermal parameter

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

Figure 7 presents the curves for the heat transfer coefficient (h_{sur}) on the fuel element surface and for the overall thermal conductivity (k_g) in fuel element as function of the power, obtained with the instrumented fuel at core ring B.

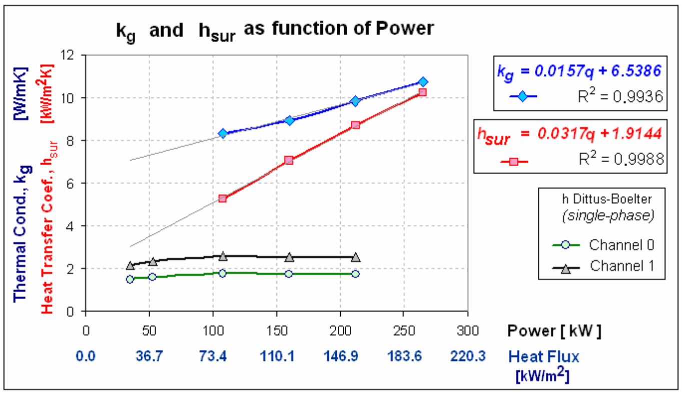

Figure 7 - Overall fuel element thermal conductivity and cladding heat transfer coefficient to the coolant

CONCLUSION

The IPR-R1 TRIGA Reactor normally operates in the range from 100 kW until a maximum of 250 kW. On these power levels the heat transfer regime between the clad surface and the coolant is subcooled nucleate boiling in the hottest fuel element. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling - ONB) is approximately 60 kW on the cladding surface in the central channels of the IPR-R1 TRIGA core. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core. The results can be considered as typical of pool-type research reactor.

Pool temperature depends on reactor power as well as on the external temperature since it affects the heat dissipation rate in the cooling tower. The data showed the efficiency of the natural circulation to remove the heat generated by the fissions in the core.

ACKNOWLEDGEMENTS

The author would like to express their special thanks to the IPR-R1 TRIGA operator team for their help during the experiments and the *Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG)* for the financial support.

NOMENCLATURE

Symbol	Quantity	SI Unit
A	Area	m^2
c_p	Isobaric specific heat	J/kgK
D_w	Hydraulic diameter	m
D_w	Hydraulic Diameter	cm
G	Mass flow	kg/m^2s
h	convective heat-transfer coefficient	kW/m^2K
k	Thermal conductivity	W/mK
$\dot{m}$	Mass flow rate	kg/s
P_h	Heated Perimeter	cm
P_w	Wetted perimeter	m
q	Power	kW
q''	surface heat flux	W/m^2
q'''	Volumetric rate of heat	W/m^3
r	Radius	m
T	Temperature	°C
μ	Fluid dynamic viscosity	kg/ms

REFERÊNCIAS

[1] A.Z. Mesquita A. Z., Experimental Investigation on Temperatures Distributions in a Research Nuclear Reactor TRIGA IPR-R1, Ph.D Thesis, Universidade Estadual de Campinas, São Paulo, (in Portuguese), 2005.

[2] N.E. Todreas and M.S. Kazimi, *Nuclear Systems I: Thermal Hydraulic Fundaments,* Hemisphere Publishing Corporation, New York, 1990.

[3] General Atomic, Safeguards Summary Report for the New York University TRIGA Mark I Reactor. (GA-9864). San Diego, 1970.

[4] A.Z. Mesquita and H.C. Rezende, Data Acquisition System for TRIGA Mark I Nuclear Research Reactor of CDTN. *Proc. of the America Nuclear Energy Symposium (ANES 2004)*, Miami Beach, Flórida, USA. America Nuclear Energy, 2004.

[5] J.R. Lamarsh and A.J. Baratta *Introduction to Nuclear Engineering,* 3° ed., Upper Saddler River: Prendice Hall, 2001.

[6] S. Glasstone and A. Sesonske, *Nuclear Reactor Engineering,* 4 ed., Chapman and Hall, New York, NY, (1994).

[7] W. Wagner and A. Kruse, Properties of Water and Steam – The Industrial Standard IAPWS-IF97 for the Thermodynamics Properties, Springer, Berlin, 1998.

[8] L.S. Tong L.S. and J. Weisman, *Thermal Analysis of Pressurized Water Reactors*, Third Edition, American Nuclear Society. Illinois, 1996.

[9] J.G. Collier and J.R. Thome, *Convective boiling and condensation*, Oxford Science Press, New York, 1994.

[10] L.S. Tong and Y.S. Tang, *Boiling Heat Transfer and Two-Phase Flow*, 2nd. Ed. Taylor & Francis, Washington, 1997.

[11] F. Kreith and M.S. Bohn, *Principles of Heat Transfer*, 6th ed., Brooks/Cole, New York, 2001.