Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor

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Abstract. The 250 kW IPR-R1 TRIGA Nuclear Research Reactor, installed at Nuclear Technology Development Center (CDTN) in Belo Horizonte, Brazil, is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U. 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 presented. 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. Results indicated that subcooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels above approximately 60 kW. 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. 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.

INTRODUTION

The IPR-1 TRIGA core contains 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented in the center with three thermocouples. Figure 1 shows the core upper view with the instrumented fuel element in ring B.

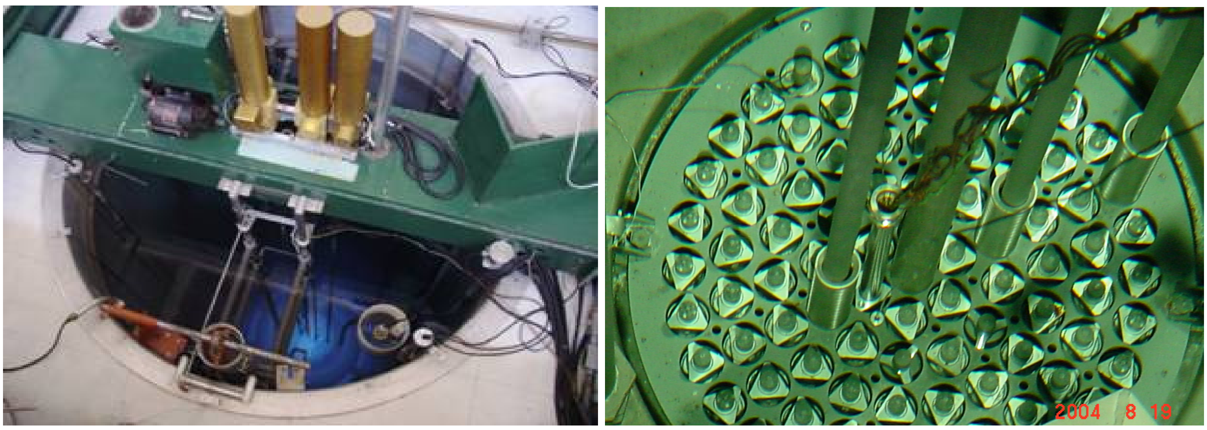

FIG. 1. Pool and core top view with the instrumented fuel element in ring B

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 various power levels can be determined from the heat transfer coefficient data.

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The thermal conductivity (*k*) of the 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 change 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 [1] to evaluate its value in power reactors fuels, which use mainly uranium oxide. The only reference found to TRIGA reactors fuel is General Atomic [2] that recommends the use of three hypotheses for the heat transfer coefficient in the gap. The heat transfer coefficient (*h*) is a property not only of the system but 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 [3] to allow on line monitoring of the operational parameters.

As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to sub cooled nucleate boiling. Results indicated that subcooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels in excess of 60 kW [4].

OVERALL THERMAL CONDUCTIVITY OF THE FUEL ELEMENTS

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

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

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 [^{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 2 show the instrumented fuel element and the TRIGA IPR-R1 core. Two thermocouples were also placed in two core channels adjacent to position B1.

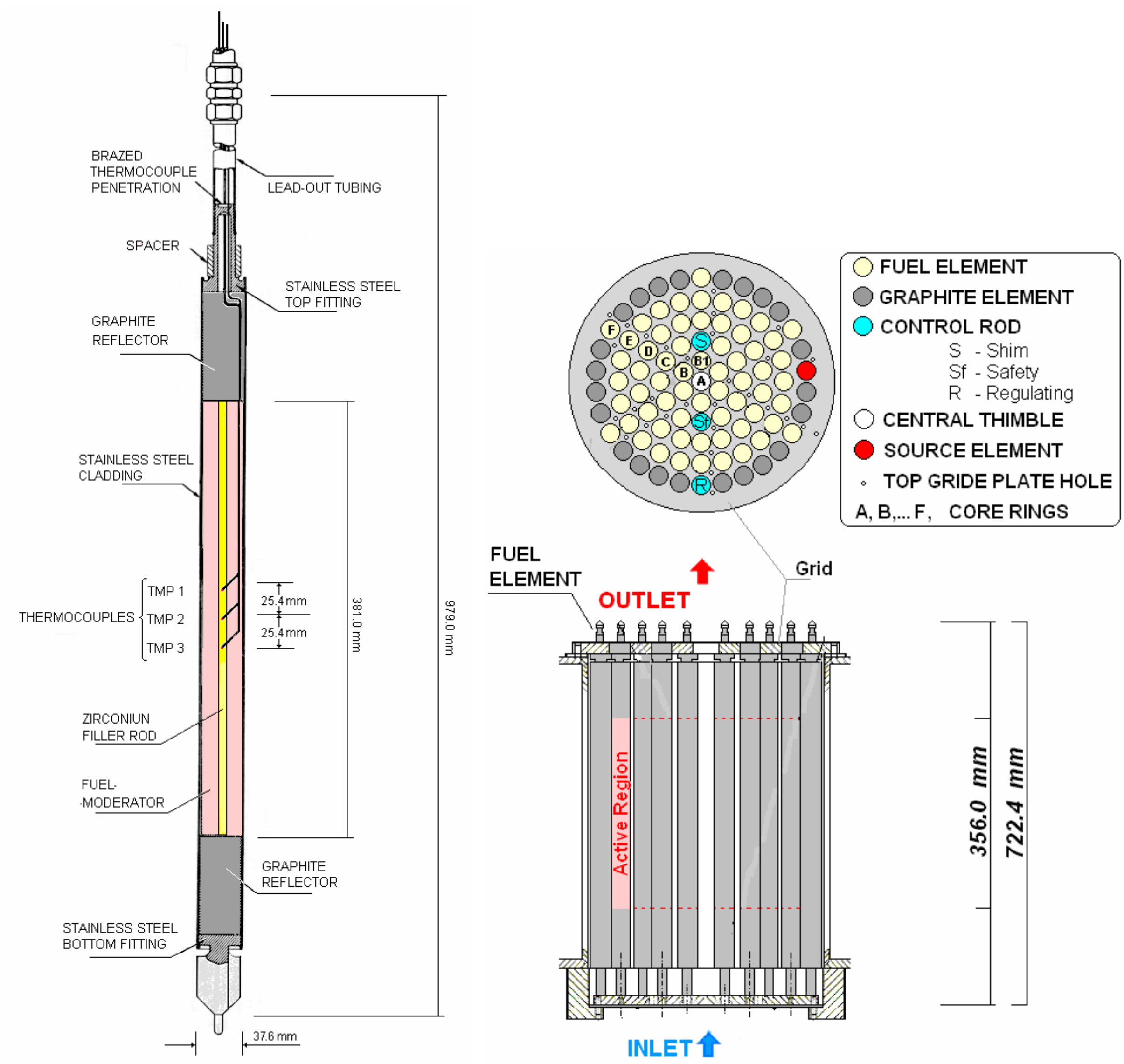

FIG. 2. The instrumented fuel element and the core of the IPR-R1 TRIGA Reactor

HEAT TRANSFER IN THE REACTOR CORE

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:

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

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^{2}s]; 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. 3). 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 isn´t hole. The inlet and outlet temperatures in Channel 0 were considered as being the as in Channel 1'. Table 1 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. 4.

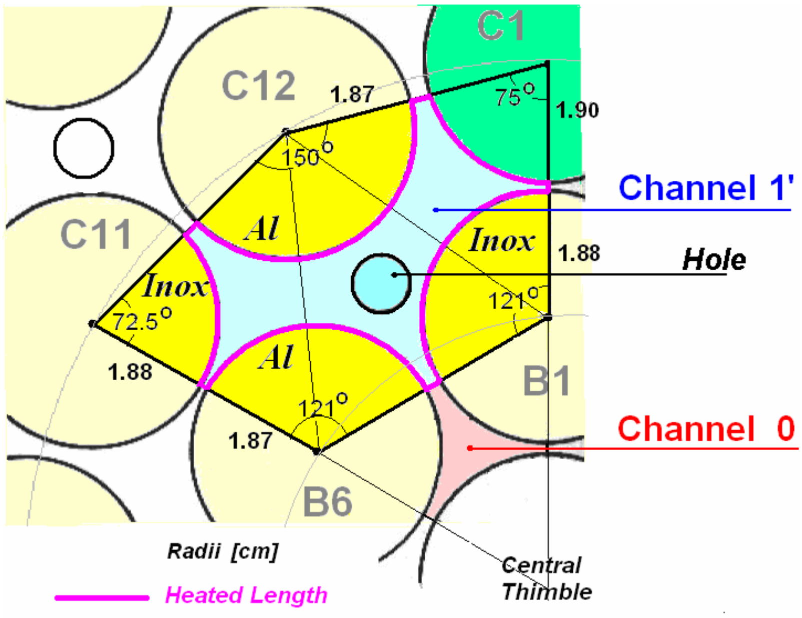

FIG. 3. The two hottest channels in the core

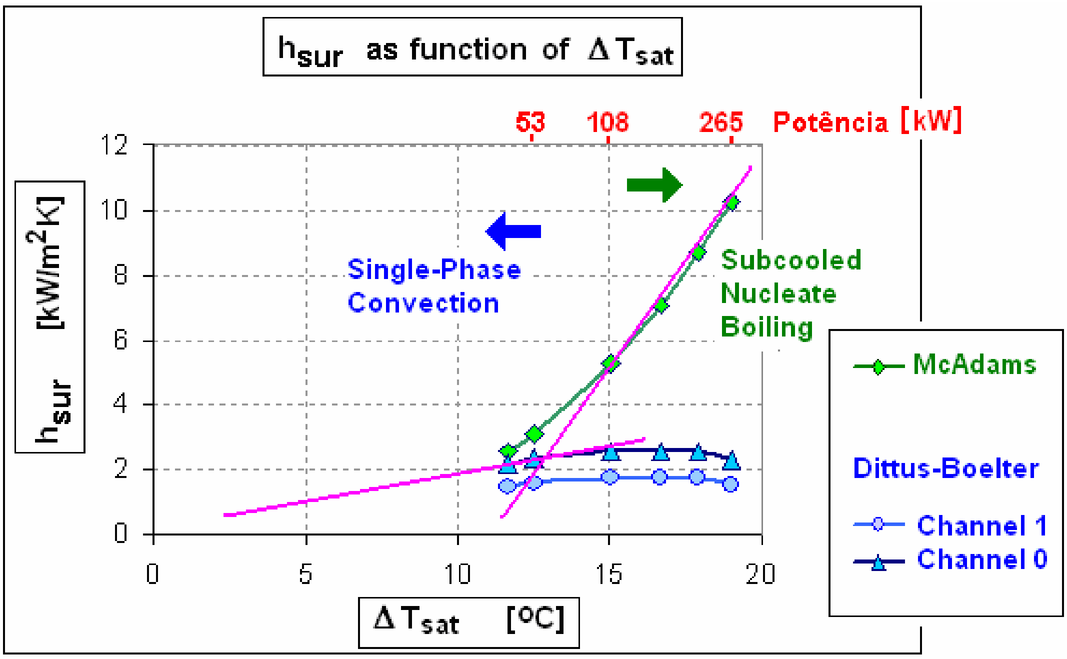

FIG. 4. Heat-transfer regimes in the fuel element surface of the IPR-R1 TRIGA Reactor

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

$$q = \dot{m} c_p \Delta T \ , \tag{5}$$

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 [^{o}C].

Table 1. **Channel 0 and Channel 1' Characteristics [4]**

	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 (265 kW)	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	%
Total Power of the Channel	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 [9]. The curve for heat transfer coefficient (h_{sur}) in the single-phase region is shown in Fig. 5 as function of the power.

Subcooled Nucleate Boiling Region

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

$$h_{sur} = q'' / \Delta T_{sat} , \tag{6}$$

where h_{sur} is the convective heat-transfer coefficient from the fuel cladding outer surface to the water [kW/m^2K]; q'' is the fuel surface heat flux [kW/m^2]; and, ΔT_{sat} is the surface superheat in contact with the water [^{o}C].

Figure 4 presents 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 correlation used for the subcooled nucleate boiling is not valid for single-phase convection region, as well as the Dittus-Boelter correlation is not valid for the boiling region. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling) is approximately 60 kW as shown in the graph.

Figure 5 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 for the instrumented fuel at core position B1.

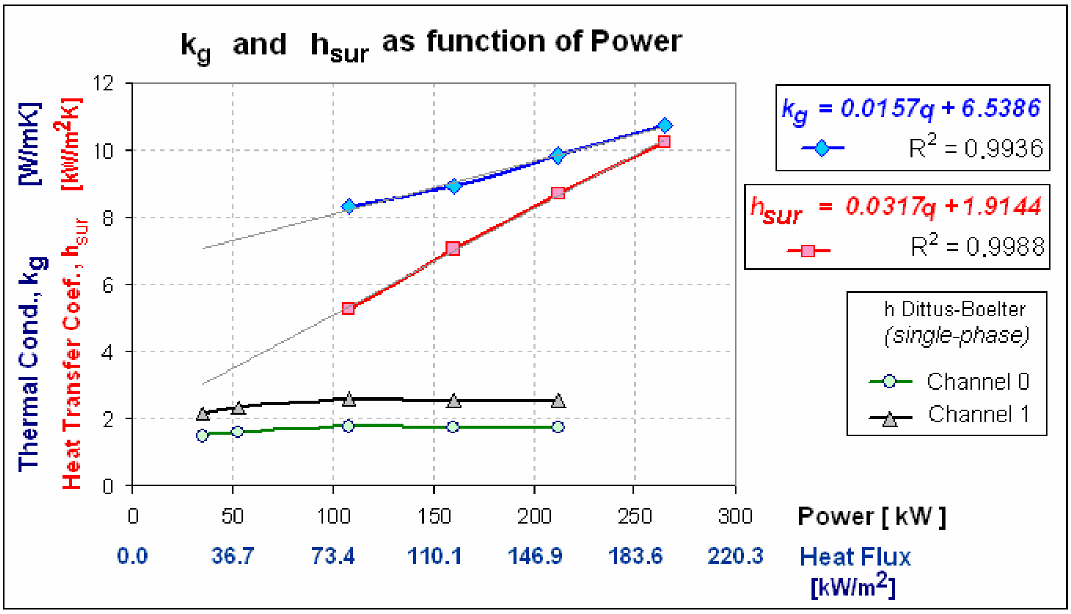

FIG. 5. Overall fuel element thermal conductivity and cladding heat transfer coefficient to the coolant

HEAT TRANSFER COEFFICIENT IN THE FUEL GAP

The instrumented fuel element is composed of a central zirconium filler rod with 6.25 mm in diameter, the active part of the fuel, formed by uranium zirconium hydride alloy (U-$ZrH_{1.6}$), an interface (gap) between the fuel and the cladding, and the 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 center to the coolant (R_g) is given by the sum of the fuel components resistances.

The fuel element configuration is shown in Fig. 6. The axial heat conduction and the presence of the central pin of zirconium were not considered. The thermal conductivity of the U-$ZrH_{1.6}$ fuel is given by [12]:

$$k_{UZrH} = 0.0075\, T + 17.58, \tag{7}$$

with T in [°C] and k_{UZrH} in [W/mK]. The thermal conductivity of the AISI 304 steel cladding is given by [13]:

$$k_{rev} = 3.17x\ 10^{-9}\, T^3 - 6.67\ x\ 10^{-6} T^2 + 1.81\ x 10^{-2} T + 14.46, \tag{8}$$

with T in [°C] and k_{rev} in [W/mK].

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:

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

The graph of heat transfer coefficient through the gap is shown in Fig. 7, as a function of the reactor power. This figure also shows three theoretical values recommended by General Atomic for the heat transfer coefficient [2].

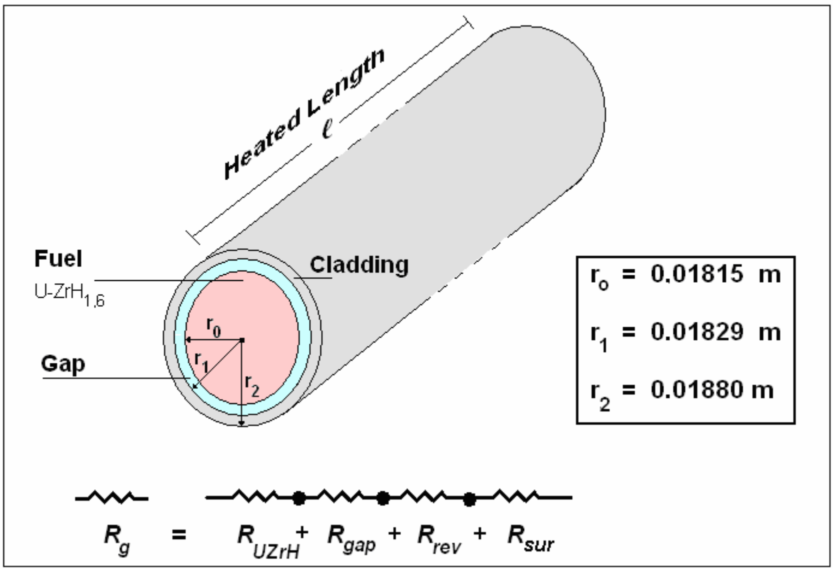

FIG. 6. Fuel element configuration

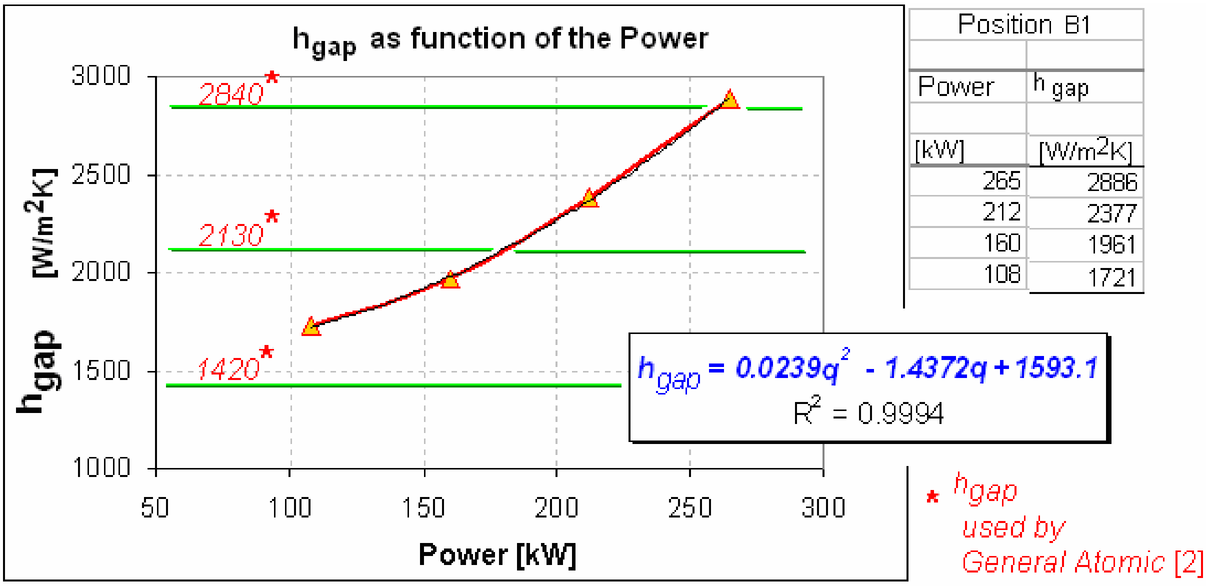

FIG. 7. Heat transfer coefficient through the gap as a function of the power

FUEL ROD TEMPERATURE PROFILE

From the temperature in the center of the fuel and using the equations of conduction for the fuel element geometry, it is possible to obtain the radial temperature distribution in the fuel element. Figure 8 shows the experimental radial profile of maximum fuel temperature in position B1 and it is compared with the PANTERA code results [14]. The instrumented fuel element was used to measure the fuel temperature at several reactor powers, the results are shown in Fig. 9.

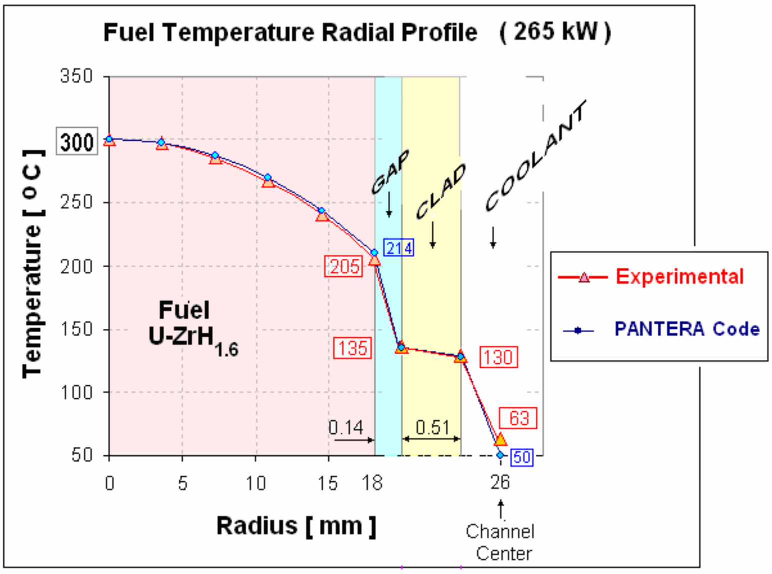

FIG. 8. . Experimental fuel rod radial temperature profile in position B1 at 265 kW

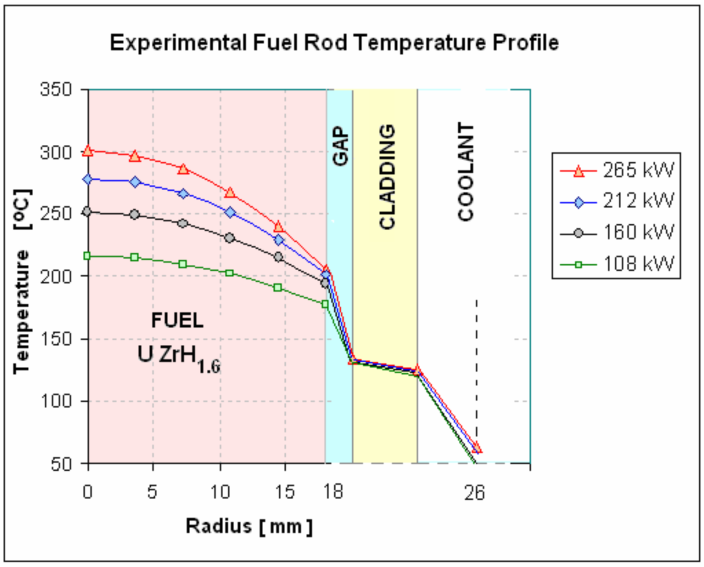

FIG. 9. Experimental fuel rod radial temperature profile in position B1 at several reactor powers

CRITICAL HEAT FLUX AND DNBR

In the fully developed nucleate boiling regime, it is possible to increase the heat flux without, an appreciable change in the surface temperature until the point of DNB. At this point, the bubble motion on the surface becomes so violent that a hydrodynamic crisis occurs with the formation of a continuous vapour film in the surface and the critical heat flux (CHF) is reached. In subcooled boiling the CHF is a function of the coolant velocity, the degree of subcooling, and the pressure. There are a lot of correlations to predict the CHF. The used equation is given by Bernath [15] [16]. This correlation predicts CHF in the subcooled boiling region and is based on the critical wall superheat condition at burnout and turbulent mixing convective heat transfer. Bernath's equation gives the minimum results so it is the most conservative. It is given by:

$$q''_{crit} = h_{crit}(T_{crit} - T_f) \quad , \qquad (9)$$

where,

$$h_{crit} = 61.84\frac{D_w}{D_w + D_i} + 0.01863\frac{23.53}{D_w^{0.6}}u \quad , \qquad (10)$$

and,

$$T_{crit} = 57\ln(p - 54)\frac{p}{p + 0.1034} + 283.7 - \frac{u}{1.219} \quad , \qquad (11)$$

q''_{crit} is the critical heat flux [W/m^2], h_{crit} is the critical coefficient of heat transfer [W/m^{2}K], T_{crit} is the critical surface temperature [^{o}C], T_f is the bulk fluid temperature [^{o}C], p is the pressure [MPa], u is the fluid velocity [m/s] (u = $\dot{m}$ /*channel area/water density*), D_w is the wet hydraulic diameter [m], D_i is the diameter of heat source [m]. This correlation is valid for circular, rectangular and annular channels, pressure of 0.1 to 20.6 MPa, velocity between 1 to 16 m/s and hydraulic diameter of 0.36 to 1.7 cm.

In reactor power of 265 kW operating in steady state, the core inlet temperature was 47 ^{o}C. The critical flow for the Channel 0 is about 1.6 MW/m^2, giving a DNBR of 8.5. Figure 10 shows the values of critical flow and DNBR for the two channels. The theoretical values for reactor TRIGA of the University of New York [2] and calculated with the PANTERA code [14] for the IPR-R1 are also shown. The two theoretical calculations gave smaller results than the experiments. These differences are due to the temperature used in the models.

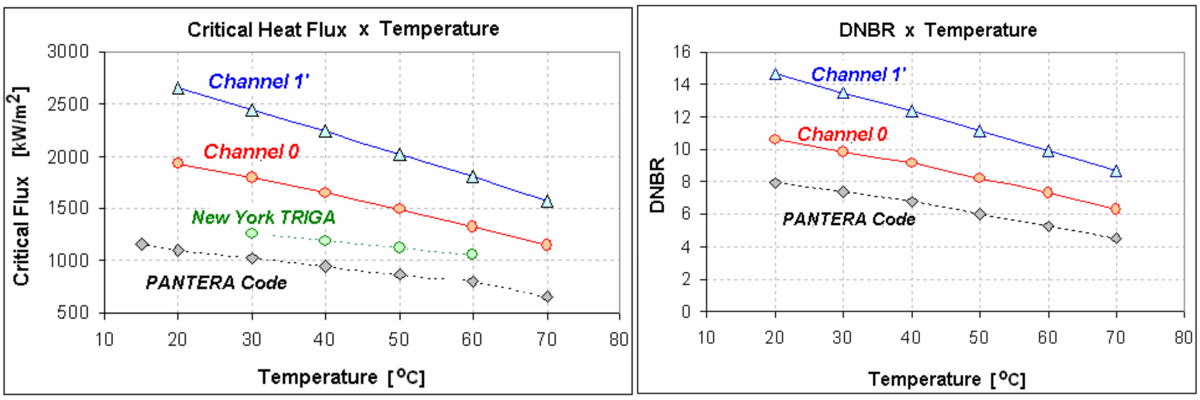

FIG. 10. Critical heat flux and DNBR as a function of the inlet coolant temperature

CONCLUSION

Pool temperature depends on reactor power, as well as on the external temperature because it affects the heat dissipation rate in the cooling tower. Subcooled pool boiling occurs above 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. 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. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core [6]. The results can be considered as typical of pool-type research reactor.

The minimum DNBR for IPR-R1 TRIGA (DNBR=8.5) is much larger than other TRIGA reactors. The 2MW McClellen TRIGA [17] has a DNBR=2.5 and the 3 MW Bangladesh TRIGA has a DNBR=2.8 [18]. The power reactors are projected for a minimum DNBR of 1.3. In routine operation they operated with a DNBR close to 2. The IPR-R1 reactor operates with a great margin of safety at its present power of 250 kW, the maximum heat flux in the hottest fuel is about 8 times lesser than the critical heat flux that would take the hydrodynamic crisis in the fuel cladding. This investigation indicates that the reactor would have an appropriate heat transfer if the reactor operated at a power of about 1 MW. The data showed the efficiency of the natural circulation to remove the heat generated by the fissions in the core.

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