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EXPERIMENTAL DETERMINATION OF HEAT TRANSFER COEFFICIENTS IN URANIUM ZIRCONIUM HYDRIDE FUEL ROD

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ABSTRACT

This work presents the experiments and theoretical analysis to determine the temperature parameter of the uranium zirconium hydride fuel elements, used in the TRIGA IPR-R1 Research Nuclear Reactor. The fuel thermal conductivity and the heat transfer coefficient from the cladding to the coolant were evaluated experimentally. It was also presented a correlation for the gap conductance between the fuel and the cladding. In the case of nuclear fuels the heat parameters become functions of the irradiation as a result of change in the chemical and physical composition. The value of the heat transfer coefficients should be determined experimentally.

1. INTRUDUCTION

The TRIGA IPR-R1 Nuclear Research Reactor, of the Nuclear Technology Development Center - CDTN, localized in Belo Horizonte (Brazil) is a Mark I type, manufactured by General Atomic, cooling by light water, open-pool design and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U. The heat, generated by the fissions is transferred from fuel elements to the cooling system through the interface fuel/cladding (gap) and the cladding to the coolant. The objective of the thermal and hydrodynamic project of the reactors is to remove safety the heat, without producing an excessive temperature in the fuels. Computational safety codes are used and are validated by experimental measures. On the other hand some parameters, assumptions and approximations adopted in the codes can be improved from the knowledge of experimental values. Uranium-zirconium hydride was chosen for the TRIGA fuel because of its inherent safety features. This fuel has the important property that its prompt temperature coefficient of reactivity is negative and large.

There are not much data in the literature on the uranium zirconium hydride fuel elements. The thermal conductivity of the metallic alloys depends on several factors like the value of the temperature. In the case of nuclear fuels, the situation is more complicated because k also becomes a function of the 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. After the fuel alloy, the second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist (Todreas and Kazimi, 1990) to evaluate its value in power reactors fuels, which use mainly uranium oxide. The only reference found to TRIGA reactors fuel is General Atomic (1970). That recommends the use of three hypotheses for the heat transfer coefficient in the gap.

These values are fixed and differ of about 50%. The heat transfer coefficient is a property not only of the system in study 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 the specific heat. All these parameters are temperature dependent and change while wall transfer heat to the fluid.

2. OVERALL THERMAL CONDUCTIVITY OF THE FUEL ELEMENT

From Fourier equation it is possible to obtain the expression of thermal conductivity (k_g), in [W/mK], for cylindrical fuel elements (Lamarsh and Baratta, 2001) and (Duderstadt and Hamilton, 1976):

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

where, q''' is the volumetric rate the heat generation in [W/m^3], T_o and T_{sup} are the central temperature and the fuel element surface temperature in [^{o}C] and r is the fuel element radius in [m]. The temperature in the center of the fuel was measured and the cladding outer surface temperature it should be found. At a power of 265 kW, the heat transfer in all fuel elements is realized in regime of subcooled nucleate boiling. So the cladding outer temperature (T_{sup}) is the water saturation temperature (T_{sat}) at the pressure of 1.5 bar, equal 111.37 ^{o}C (Wagner and Kruse, 1998) (atmospheric pressure added up of the water column of ~ 5.2 m), increased of the wall superheat (ΔT_{sat}). The wall superheat is found using the correlation proposal by McAdams (Tong and Weisman, 1996),

$$\Delta T_{sat} = 0.81 (q'')^{0.259}, \tag{2}$$

with q'' in [W/m^2] and T_{sat} in [^{o}C]. The fuel element data used in the calculations are found in (Mesquita, 2005).

3. HEAT TRANSFER REGIMES IN THE REACTOR CORE

As the 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. The hottest measured temperature in the core channel was of 65 ^{o}C (Channel 1), therefore below of 111.4 ^{o}C that is the water saturation temperature to 1.5 bar. So the saturated nucleate boiling regime is not reached. The heat transfer coefficient in single phase region (h_{cm}) was calculated with the Dittus-Boelter correlation (Glasstone and Sesonske, 1994), valid for turbulent flow in narrow channels, given for:

$$h_{cm} = \frac{0{,}023 k\, Re^{0,8}\, Pr^{0,4}}{D_w}, \tag{3}$$

where, Re = Reynolds number (Re= GD_w/μ) and Pr = Prandtl number (Pr=$c_p\mu/k$) e D_w = $4A/P_w$ is the hydraulic diameter of the channel based on the wet perimeter, A is the flow area in [m^2]; P_w is the wet perimeter in [m]. G is the mass flow in [kg/m^{2}s], c_p is the isobaric

specific heat in [J/kgK], k is the thermal conductivity in [W/mK] and μ is the fluid dynamic viscosity in [kg/ms]. The properties for the TRIGA IPR-R1 core are estimated as the average local temperature of the fluid sub-saturated at 1.5 bars. Channel 1' in the Figure 1a is showed in prominence had its temperature monitored. The Channel 0, located close to the center, there is the largest density of neutron flux. There is no hole in the superior plate in the direction of the Channel 0; so it was not possible to measure its temperature. For comparison it was made the estimation of the heat transfer coefficient, using the Dittus-Boelter correlation, in the two channels. The inlet and outlet temperatures in Channel 0 were considered as being the same to the one of Channel 1'. The geometric data of Channel 0 and Channel 1' are given in Mesquita (2005).

For the subcooled nucleated boiling region (local or surface boiling), the used expression is shown below (Kreith, 2002), (Tong and Tang, 1997):

$$h_{sup} = q'' / \Delta T_{sat} \quad , \tag{4}$$

where h_{sup} is the convective heat-transfer coefficient between the fuel cladding outer surface to the water, in [kW/m^{2}K], q'' is the fuel surface heat flux in [kW/m^2] and ΔT_{sat} is the surface superheat in contact with the water in [^{o}C]. The Figure 1b presents the graph of the fuel element surface heat transfer coefficient for the coolant as function of the superheat, in the two regimes. This is a specific curve for the TRIGA IPR-R1 reactor conditions. The correlation used for subcooled nucleate boiling is not valid for single phase convection region, as well as the Dittus-Boelter correlation is not valid to the boiling region. The transition among the two regimes starts from approximately 100 kW as shown in the graph.

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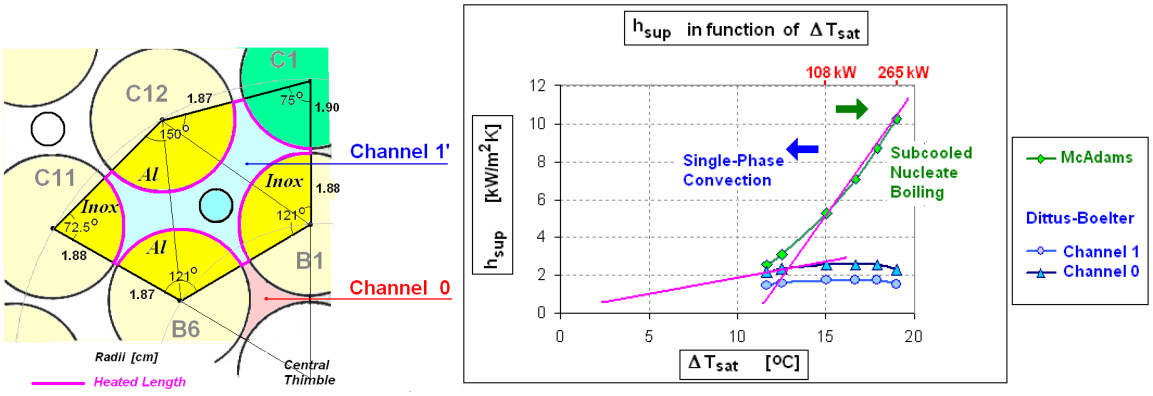

a) b)

Figure 1. a) The two hottest channels in the core, b) Heat-transfer regimes in the fuel element-fluid interface.

Figure 2 presents the curve for the fuel element surface heat transfer coefficient (h_{sup}) and of the overall fuel element thermal conductivity (k_g), as function of the power, obtained for the instrumented fuel at the B1 core position. With the k_g values it was obtained the following expression, by linear regression, for the overall fuel element thermal conductivity k_g, in [W/mK] as function of the total reactor power q, in [kW]:

$$k_g = 0.0157\, q + 6.5386 \quad , \tag{5}$$

with the determination coefficient R^2 = 0.9936, standard error of the adjusted curve $U_{y,\,x}$ = 0.104 W/mK and uncertainty of ±7.3%.

The following expression was obtained by linear regression for the heat transfer coefficient for the subcooled nucleate boiling h_{sup} in [kW/m^{2}K], as function of the reactor power q, in [kW]:

$$h_{sup} = 0.0317q + 1.9144 \quad , \tag{6}$$

with determination coefficient R^2 = 0.9988, standard error of the adjusted curve = 0.179 kW/m^{2}K and uncertainty of 7.4%.

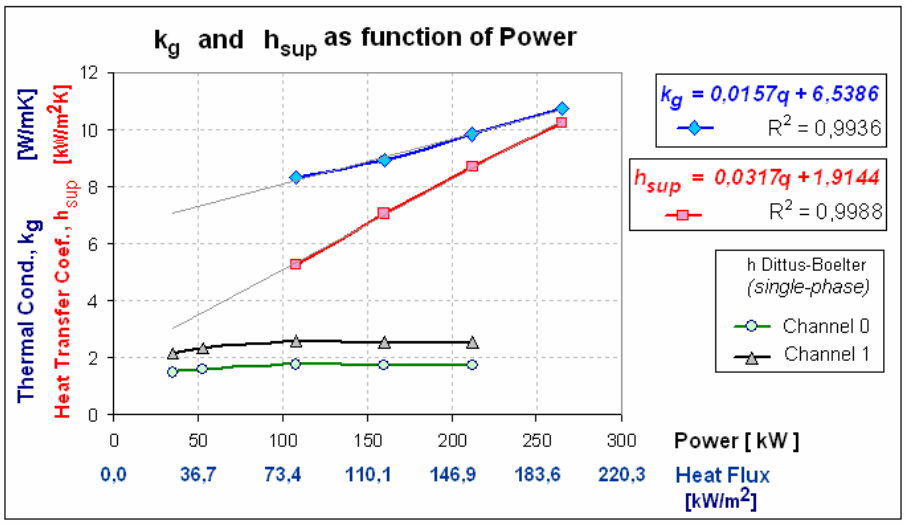

Figure 2. Overall fuel element thermal conductivity and cladding heat transfer coefficient to the coolant.

4. HEAT TRANSFER COEFFICIENT BETWEEN THE FUEL AND THE CLADDING (GAP)

The instrumented fuel element is composed of: a central zirconium filler rod with 6.25 mm diameter where are fixed the thermocouples, the fuel active part formed by an alloy of uranium zirconium hydride (U-ZrH$_{1.6}$), an interface between the fuel and the external cladding (gap) and the 304 stainless steel cladding. It is assumed that all the heat flux is in the radial direction. Using the analogy with electric circuits, the resistance to the heat conduction from the center of the fuel to the coolant is given by:

$$R_g = R_{Zr} + R_{U\text{-}ZrH} + R_{gap} + R_{rev} + R_{sup} \quad , \tag{7}$$

where R_g is the overall resistance of the fuel element, R_{Zr} is the resistance of the zirconium central pin; $R_{U\text{-}ZrH}$ is the resistance of the fuel alloy; R_{gap} is the resistance of the gap; R_{rev} is the cladding resistance; and, R_{sup} is the wall convective resistance to the coolant. The axial heat conduction and the presence of the central pin of zirconium were not considered. The fuel thermal conductivity of U-ZrH$_{1.6}$, in [W/mK], is given for: $k_{U\text{-}ZrH} = 0.0075\ T + 17.58$

(Simnad et al., 1976) with T in [°C]. The steel cladding thermal conductivity AISI 304, in [W/mK], is given for: $k_{SS\ 304} = 3.17 \times 10^{-9}\ T^3 - 6.67 \times 10^{-6}\ T^2 + 1.81 \times 10^{-2} T + 14.46$, with T in [°C] (ASME, 1992). The value of R_{gap} will be the value of the overall resistance of the fuel element (R_g) less the values of other component resistance. Substituting for the fuel element geometry's (Holman, 2002):

$$\frac{1}{2\pi\ell r_o h_{gap}} = \frac{1}{4\pi\ell k_g} - \frac{1}{4\pi\ell k_{UZrH}} - \frac{ln(r_2 / r_1)}{2\pi\ell k_{rev}} - \frac{1}{2\pi\ell r_2 h_{sup}} \tag{8}$$

In the expression above the only unknown parameter is the h_{gap}. With the k_g value and h_{sup} obtained previously and with the k values of the fuel alloy and cladding corrected in function of the temperature it was obtained the value of h_{gap}. It was considered, for the fuel temperature, the values predicted by the theoretical calculation in function of the radius. For the cladding temperature, it was considered the value in its surface T_{sup}, that was obtained with the McAdams correlation. So it was not used, in the Equation 8, the convective thermal resistance of the interface wall/coolant (R_{sup}). The heat transfer coefficient in the gap is:

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

The graph of the heat transfer coefficient of the gap as function of the reactor power is shown in the Figure 3. The figure also shows three theoretical values for the heat transfer coefficient recommended by General Atomic (1970).

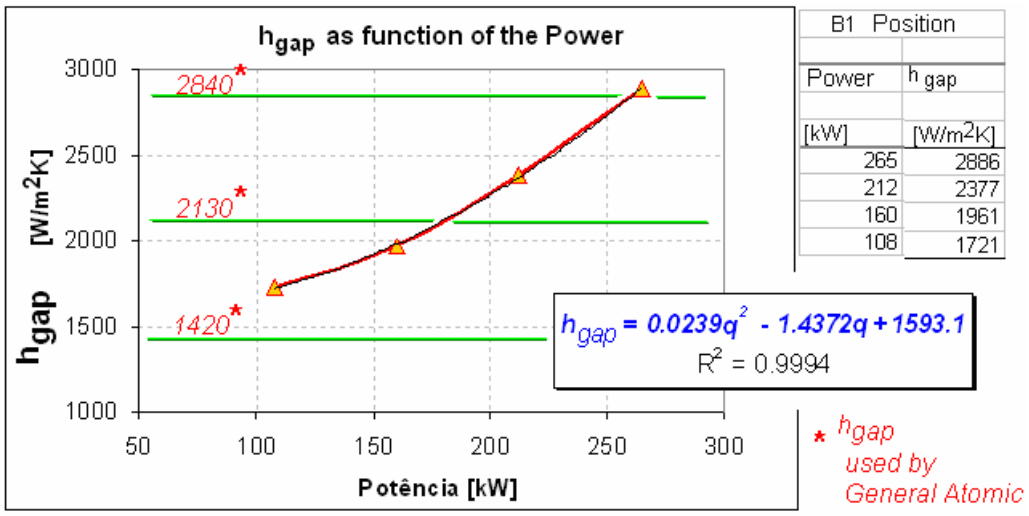

Figure 3. Heat transfer coefficient in the gap as function of the power.

The following polynomial expression for the h_{gap} in [W/m^{2}K] was obtained, by regression, as function of the reactor power q in [kW]:

$$h_{gap} = 0.0239\ q^2 - 1.4372\ q + 1593.1 \quad , \tag{10}$$

with a determination coefficient R^2 = 0.9994, a standard error of the adjusted curve = 15 W/m^{2}K and an uncertainty of ±7.5%.

5. CONCLUSION

Fixed values recommended by reactor builder (General Atomic, 1970), have been used for the heat transfer coefficient in the fuel element gap, or this parameter is not considerate (Feltus and Miller, 2000). At the power of 265 kW the h_{gap} value is practically the same recommended by General Atomic. This value was used in the theoretical simulation (PANTERA code) giving thus a good agreement between the two curves. The expressions for the heat transfer coefficient obtained by this investigation will be used for optimize the thermal hydraulics codes (Veloso, 1999). The studies were realized with a fuel used for the first time. The value of the heat transfer coefficient in the gap changes with the fission gas products. During the time it should also appear an additional thermal resistance in the cladding outer wall, due to the oxidation. It is recommended to monitor the evolution of the thermal parameters periodically with the fuel burnup.

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