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Boiling heat transfer on a simulated nuclear fuel rod with annular fins

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ABSTRACT

This work describes an improved design for a nuclear fuel rod, which was developed using an experimental circuit that simulates the nuclear rod environment in a pressurized water reactor (PWR). The heat exchanger circuit runs atmospheric to high-pressure boiling water through an annular finned tube (153 fins), which is used to enhance confined heat transfer rates and guide the nucleate boiling process. Volumetric electrical generation (Joule effect) is used to simulate nuclear heating. As much as 48 kW of electrical power is dissipated at different pressure levels up to 40 bar. Under these experimental conditions, heat transfer coefficients of the order of 73,000 W m^{-2} K^{-1} are achieved. Our experimental results show that critical safety levels can be easily reached using this design approach. Several nucleate boiling correlations were compared with the controlled experimental data sets, and a modified functional form for the boiling heat transfer coefficient is proposed.

1. Introduction

The nuclear fuel is the consumable part of the fuel rod, which is typically formed by a column of sintered uranium dioxide pellets encapsulated in zirconium alloy tubes. Fuel rods used in commercial power reactors are generally considered to be very robust when operating under critical temperature levels because only a very small number of failures (which may or not imply leakage of radioactive material into the cooling circuit) are expected. The high level of engineering reliability of such rods was obtained through extensive research carried on by universities, research centers and by industrial manufacturers. Recent works studied the physical behavior of real and simulated fuel rods under the severe operational conditions found in newly proposed designs for nuclear power reactors, e.g. [1–3]. The main operational goal is to ensure that there is no dispersion of radioactively contaminated water into the cooling circuit. Two critical conditions may determine operational failure: melting by over-heating and over-pressuring of the water line [3].

Fig. 1 is a schematic view of a nuclear fuel rod simulator, and Table 1 shows the main components of the power circuit. Understanding the heat transfer process in the region between the rod, pressurized water and the pressure vessel is critical to the overall operation of the system. This region involves phase-changing water (predominantly nucleate boiling), as can be seen in the Fig. 1.

Many reduced order models for the complex physics of nucleate boiling have been suggested. Mikic and Rohsenow [4] micro convective model, the microlayer model of Cooper and Lloyd [5] and the line contact model of Stephan and Hammer [6] and Mitrovic [7] are good examples of such models. Demiray and Kim [8] conclude that micro convection is the main heat transfer mechanism during nucleate boiling, highlighting that transient conduction during the re-wetting process is also a critical contributor to the overall heat transfer rates.

A quantitative analysis of the boiling phenomenon follows pool boiling and flow boiling experiments. Flow boiling phenomena plays an important role in conventional PWR nuclear rods, because of the presence of forced convection [9]. In this specific experiment, the guide discs affect the flow boiling, causing a confined pattern, and pool boiling can be used accurately for determining the HTC. Rops et al. [10] have discussed pool boiling in confined conditions in greater details, observing an enhancement effect due to confinement. Fig. 2 represents a typical pool boiling curve during water phase change considering the heat flux and temperature difference (ΔT, between wall and saturated fluid), and a confined correspondent curve (as shown by Rops et al. [10]). When ΔT is higher than 5 °C the process of nucleate boiling takes place. As the heat flux increases in the nucleate boiling range, the number of nucleate sites increase highly for a relatively low variation on temperature, generating a higher heat transfer coefficient. Rops et al. [10] indicate that inside a confined recipient, heat flux can be 10 times greater than in unconfined recipients considering the

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Nomenclature

β	contact angle, rad
Bo	Bond number, $\frac{s}{\sqrt{\frac{\sigma}{g(\rho_l-\rho_v)}}}$
c	specific heat capacity, J kg^{-1} K^{-1}
C	constant
C_1	constant from modified-Rohsenow relation, −0.151
C_2	constant from modified-Rohsenow relation, 0.533
C_g	boiling number
C_{sf}	Rohsenow's C_{sf} constant, 0.0133
C_{sf}^*	Pioro's C_{sf}^* constant, 1.503
Δ	finite difference
D	diameter, m
ϕ	heat flux, W m^{-2}
Fr	Froude number, $\frac{\phi^3}{\rho_l^2 h_{lv}^3 \mu_l g}$
G	mass flux, kg m^{-2} s^{-1}
HTC	heat transfer coefficient
k	conductivity, W m^{-1} K^{-1}
L	thickness, m
m	Pioro's m constant, −1.1
μ	viscosity, Pa s
n	Rohsenow's n constant, 1
P	pressure, kPa
Pr	Prandtl number
q''	heat flux, W m^{-2}
R	thermal resistance, W^{-1} m^2 K
r	roughness, 1 μm
Re	Reynolds number
ρ	density, kg m^{-3}
s	space between discs, 6 mm
σ	surface tension, N m^{-1}
T	temperature, K
w	uncertainty
We	Weber number, $\frac{\phi \mu_l}{\rho_l h_{lv} \sigma}$

Subscripts

a	average
b	bubbles
ext	external
f	function
i	independent variables
int	internal
l	saturated liquid
p	constant pressure
pw	pressure vessel wall
red	reduced
sat	saturation
sim	simulator
v	saturated vapor

same temperature. As represented schematically in the Fig. 2, the unconfined curve tends to be convex and the confined one tends to be concave [10]. The theoretical physical explanation to this fact is the difficulty for re-wetting process by liquid flow due to the vertical bubble chaotic flow motion in unconfined recipients, generating a lower coefficient of heat transfer than in a confined one. For confined recipients, there is a tendency of circular cyclic flow, facilitating the re-wetting process because colder liquid is pumped towards the heated surface, which generates a higher heat flux. When ΔT approaches the critical value, the quantity of nucleate sites becomes so large that the pattern observed in the unconfined flow becomes very similar to the confined flow patterns. Using similar criteria, the angular discs between the rod and the pressure vessel tend to organize the bubble flow, thus enhancing the process of heat exchange.

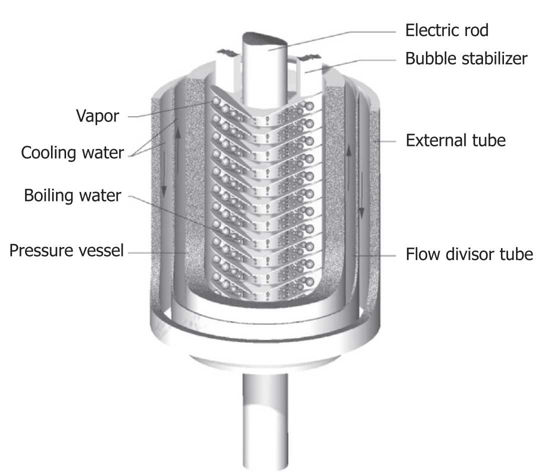

Fig. 1. Schematic view of a fuel rod, pressurized water, pressure vessel and cooling water circuit in the research simulator highlighting the flow divisor and the bubble stabilizer sections.

Table 1
Main components of the experimental loop.

System	Main function
Pressurized water	To enable heat transfer from the rod by nucleate boiling (Fig. 3)
Cooling water	To enable heat transfer from pressurized water by forced convection (Fig. 3)
Simulated fuel rod	To deliver Joule effect energy for simulating a nuclear fuel rod (Fig. 4)

On the one side, it is not expected a decreasing on the heat transfer coefficient (HTC, W m^{-2} K^{-1}) as consequence of the rod's vertical surface orientation [11]; on the other side, the vertical heated plate settles the flux upwards, parallel to the pressure vessel's wall surface, where it is located a great portion of the heat exchange surface area, and a lesser amount of heat (W) is transferred. On the other hand, the guiding fins emulate a confined environment; the annular discs force the flow in the direction of the pressure vessel wall, where bubbles condensate due to the lower temperature. Fig. 1 shows that the process of nucleate boiling in the present experiment occurs along the horizontal axis.

An experimental apparatus was built for studying the phenomena described above, with the goal of accomplishing the following objectives:

1. maintain the temperature of the pressure vessel for different ranges of pressurization (allowable temperature is 650 °C [12]);
2. determine the heat transfer coefficient for pressure levels of 1 bar, 5 bar, 10 bar and 40 bar;
3. compare different heat transfer coefficient relations with the data set, and suggest improvements in these correlations if necessary.

1.1. Experimental features

Fig. 3 presents the basic design of the experimental apparatus. Table 2 shows a list of experimental conditions for the fuel rod