WALL SUPERHEAT PREDICTION IN NARROW RECTANGULAR CHANNELS UNDER FULLY DEVELOPED BOILING OF WATER AT LOW PRESSURES

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ABSTRACT

The modeling of two-phase heat transfer is a crucial issue in the safety analysis of nuclear reactors. The thermal-hydraulic correlations employed in this kind of simulations are usually derived from experiments that were carried out over limited ranges of parameters and for specific geometries. Therefore their applicability to systems with different characteristics has to be carefully scrutinized.
In this paper, an assessment study of wall superheat correlations under fully developed boiling is presented. This is a contribution to the validation and improvement of the thermal-hydraulic modeling of the Jules Horowitz Reactor, which is a research reactor under construction at CEA-Cadarache (France).
The SULTAN-JHR experiments are used. These tests were performed at CEA-Grenoble with upward water flow in a vertical uniformly heated narrow rectangular channel with gap of 2.16 mm. The experimental conditions ranged between 0.2 and 0.9 MPa for the pressure and between 0.5 and 4.4 MW/m^2 for the heat flux.
It is shown that the correlations of Thom and Jens-Lottes significantly overestimate the wall superheat. The correlation of Belhadj and Qiu, which were developed for narrow channels at low heat fluxes, cannot accurately predict the experimental data. On the other hand, satisfactory results can be obtained with Gorenflo (standard deviation of 11.9%) and a simplified version of the Forster-Greif (standard deviation of 10.1%) correlations. In conclusion, considering the validity range of the above correlations along with the outcomes of the current assessment, the simplified Forster-Greif correlation is thus recommended for the analysis of the JHR.

KEYWORDS
Narrow rectangular channel, Wall superheat, Fully Developed Boiling, SULTAN-JHR, Research reactor

1. INTRODUCTION

The Jules Horowitz Reactor is a material testing reactor under construction in France, at CEA-Cadarache [1]. In the JHR core, cylindrical concentric fuel plates are arranged in such a manner that the coolant flows through narrow channels, under large heat fluxes (up to 5.5 MW/m^2), with high velocities (up to 15 m/s).

The safety analysis of the reactor is performed with the thermal-hydraulic system code CATHARE [2], which consists of a transient 2-fluid 6-equation model, complemented with proper closure laws for single-phase and two-phase flow. These closure laws have been extensively validated for the modeling of conventional Light Water Reactors (LWRs) [3]. However, the extension to the simulation of JHR needs further validation work due to the peculiar core design and operational conditions. For this purpose, the SULTAN-JHR experimental database is used. The database collects experiments carried out at CEA-Grenoble, in narrow rectangular channels under conditions which may be encountered in the JHR. The design of the test section aims at being representative of the JHR channels as already discussed in [4], where an analysis of the single-phase forced convection heat transfer correlation can be also found.

The objective of this paper is to assess the predictive capabilities of wall superheat correlations in Fully Developed Boiling (FDB) against the SULTAN-JHR experimental data.

The paper is organized as follows: in the next section a brief description of the SULTAN-JHR experimental campaign is given; in Section 3 the methodology applied is explained; in Section 4 the selected correlations are described along with their validity ranges and compared with the experimental data; in Section 5 conclusions are drawn.

2. THE SULTAN-JHR EXPERIMENTS

The SULTAN-JHR experimental campaign was carried out at CEA Grenoble (France) during the years 2001-2008, with the objective of providing a reliable set of data for system code validation. Several experiments were performed in vertical, narrow rectangular channels with demineralized and degassed water flowing upward. In this paper, the analysis is focused on a uniformly heated channel with gap equal to 2.16 mm. Figure 1 shows a schematic of the rectangular test section, whose geometric dimensions are reported in Table I.

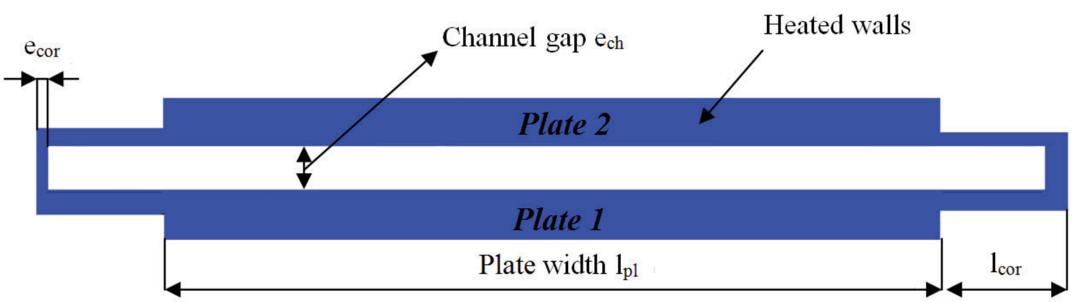

Figure 1. Geometry of the SULTAN-JHR test section (top view).

Table I. Geometric dimensions of the test section (in mm).

Gap size (e_{ch})	2.16 ± 0.05	**Thickness of the corner (e_{cor})**	0.5
Heated height (H_{ch})	599.7 ± 0.1	**Averaged thickness of plate 1 ($\bar{e}_{pl1}$)**	1.003 ± 0.002
Plate width (l_{pl})	47.15 ± 0.1	**Averaged thickness of plate 2 ($\bar{e}_{pl2}$)**	1.004 ± 0.002
Corner width (l_{cor})	2.85 ± 0.1	**Surface roughness**	0.4×10^{-3}

The test section is made of two Inconel 600 plates, which are heated by direct current power and encapsulated by an electrical mica-based insulation (Cogetherm®), and two pressure steel plates which keep the channel geometry constant during all tests. The heat losses were significantly reduced with a 200 mm-thick thermal insulation layer made of rock wool. The axial geometry and instrumentation layout of the test section is shown in Figure 2. The central part of the channel is heated with an approximately uniform heat flux, while two 70 mm-long adiabatic zones are present at the extremities of the test section. A smooth entrance in the test section was used in order to minimize the entrance effects.

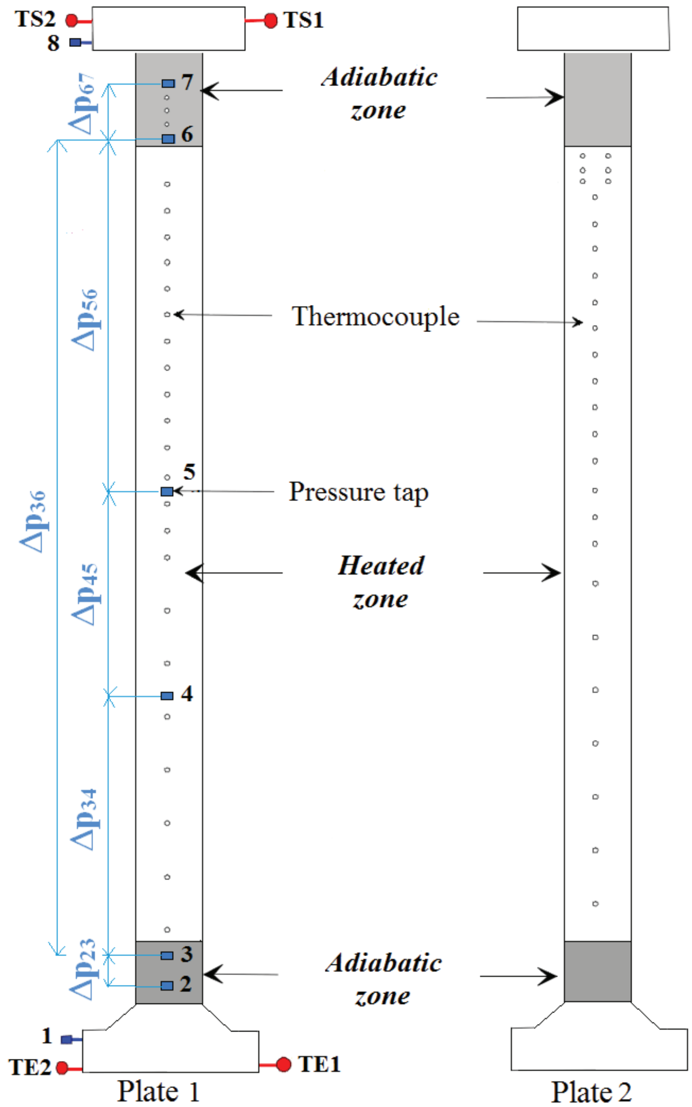

Figure 2. SULTAN-JHR axial geometry and instrumentation layout.

Several quantities were measured during the experiments, including the mass flow rate, the electrical voltage and current, the water and dry wall temperatures, the absolute pressures and the pressure drops along the channel. The signals from the sensors were integrated over a 20 ms time range and the final measurements were obtained as an average of 100 acquisitions, in order to increase the stability and reliability of the measurements. The voltage ΔV between the pressure taps P3 and P6 and the electrical current I were measured, so that the electrical power supplied to the test section could be estimated according to the formula $P = \Delta V \times I$. The water temperatures at the inlet (TE1 and TE2) and at the outlet (TS1 and TS2) were measured with a platinum probe.

Several thermocouples and 8 pressure taps were placed on the two heating plates, as shown in Figure 2. The pressure taps were located at the inlet (PE1), at the outlet (PS8), in the adiabatic zones (P2, P3, P6 and P7) and in the heated zone (P3 and P4). The 42 insulated K-thermocouples measured the dry wall temperatures at different axial locations and were placed in the insulation layer, as shown in Figure 3.

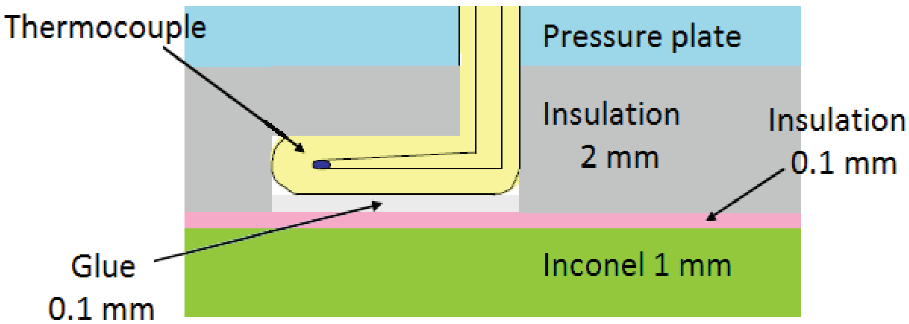

Figure 3. Simplified thermocouple layout.

The estimated uncertainties on the measurements are reported in Table II. A more detailed description of the experimental campaign and facility may be found in [4] and [5].

Table II. Measurements uncertainties on the main parameters.

Flow rate	± 1 %	**Electric power**	± 1.4 %
Absolute pressure	± 0.8 %	**Differential pressure**	± 0.8 %
Fluid temperature	± 0.25 °C	**Dry wall temperature**	± 1.5 °C

3. FDB MODELING AND EXPERIMENTAL DATA REDUCTION

The boiling is a complex heat transfer mechanism and it was studied for many decades. Extensive analysis and available correlations may be found, for example, in [6] and [7]. The two-phase heat transfer is usually modeled according to a superposition hypothesis, which combines a nucleate boiling term expressed with a FDB correlation together with a forced convective term estimated with a single-phase turbulent heat transfer correlation (e.g. Dittus-Boelter). However, in the fully developed region of the boiling curve, the contribution of the forced convection may be neglected, especially at high heat fluxes [6]. The FDB occurs at the end of the sub-cooled boiling region and it may be extended also in the first part of the saturated boiling region where the nucleate boiling term is the dominant one. A similar modeling strategy is used in CATHARE [2], where the heat transfer, in the saturated region, is determined exclusively by a FDB correlation (i.e. Thom correlation as discussed in sub-section 4.1) and, in the sub-cooled region, by a combination of the single-phase and nucleate boiling heat transfer correlations.

The most influential parameters in FDB are the system pressure and the heat flux, but other quantities may also play a role, such as the fluid properties, the surface properties (e.g. nucleation sites, roughness), the channel geometry, the dissolved gases content, etc. However all these variables can be hardly controlled in a reactor and the effect was observed to be relatively small when small variations are considered (e.g. in [8]).

The predictions of the FDB correlations are compared to the experimental wall superheat, which is defined as:

$$\Delta \mathrm{T_{sat}} = \mathrm{T_w} - \mathrm{T_{sat}} \tag{1}$$

The wet wall temperature T_w is obtained from the measured dry wall temperatures T_{dw}, applying the one-dimensional Fourier's law of conduction. As described in [5], the wet wall temperatures reads:

$$T_w = \frac{1}{\mathrm{a}}\left(\sqrt{\mathrm{b}^2 - \mathrm{a}\left(\mathrm{e_{pl}}(\phi - 2\phi_{\mathrm{loss}}) - \mathrm{a}{T_{iw}}^2 - 2bT_{iw}\right)} - \mathrm{b}\right) \tag{2}$$

where ϕ_{loss} is the heat loss experimentally evaluated, e_{pl} is the thickness of the Inconel-600 plates. The parameters a and b comes from the relationship of the Inconel conductivity ($k_{inconel}$ [W/m°C] = b + a × T [°C]) and they are equal to 0.0178 W/m°C^2 and 12.12 W/m°C, respectively. The temperature at the interface between the electrical insulation and the heated plate T_{iw} reads:

$$T_{iw} = T_{dw} + \phi_{\mathrm{loss}}\left(\frac{\mathrm{e_{co}}}{\mathrm{k_{co}}} + \frac{\mathrm{e_{gl}}}{\mathrm{k_{gl}}}\right) \tag{3}$$

where e_{co}, e_{gl} are the thicknesses of the Cogetherm® and glue layer respectively, and k_{co}, k_{gl} are the thermal conductivities. The Corsan conductivity [9] for Inconel 600 is used, as discussed in [4].

The saturation temperature T_{sat} was estimated with the system code CATHARE [10], whose single-phase friction and heat transfer correlations were optimized for the simulation of the SULTAN-JHR experiments as suggested in [4]. The test section was modeled in CATHARE as a one-dimensional channel with hydraulic diameter equal to:

$$D_h = \frac{4A}{P_w} = \frac{2\ e_{ch}\ l_{ch}}{e_{ch}+l_{ch}} \tag{4}$$

where A indicates the flow area, P_w is the wet perimeter, e_{ch} is the gap size and $l_{ch} = l_{pl} + 2\ (l_{cor} - e_{cor})$ is the channel width. The heated length of the channel was divided in 150 meshes of 4 mm each. The mesh independence of CATHARE results was proven. The use of the saturation temperatures calculated by CATHARE may be justified by comparing the experimental pressures measured at the locations P5 and P6 with CATHARE predictions. In fact, the saturation temperature is exclusively dependent on the system pressure and the majority of the experimental points used in this paper are located between P5 and P6. The comparison shows that the residuals between CATHARE and the experimental values have a standard deviation equal to 2.5 %. This demonstrates that CATHARE can accurately predict the pressure profile (and therefore the saturation temperature) for the selected tests in this paper.

A careful review of the SULTAN-JHR database led to the selection of 32 tests, where FDB could be clearly identified. From these experiments, the points of interest were chosen in such a manner that the wall superheat is approximately constant along the channel (see Figure 4), which indicates that the single-phase forced convection heat transfer mechanism becomes negligible. The FDB region is assumed to start when the experimental wall temperature profile becomes flat (or slightly decreasing). A total of 227 points was then collected, and they are always found to be beyond the Net Vapor Generation (i.e. $z > z_{NVG}$). In Figure 4, the red curve represents the wall temperature $T_{w,FG}$ predicted with the Forster-Greif correlation as given in Eqn. (14). The measurements of the last four thermocouples were removed in order to avoid possible undesirable effects due to the axial conduction towards the adiabatic zones.

The range of variation of the physical parameters is reported in Table III.

Table III. Range of physical parameters in the selected tests.

Mass flux G [kg/m^{2}s]	500 - 5364	**Liquid sub-cooling ΔT_{sub} [°C]**	0 - 38.5
Pressure p [MPa]	0.23 - 0.9	**Heat flux ϕ [MW/m^2]**	0.46 - 4.41
Steam quality x	-0.08 - 0.08		

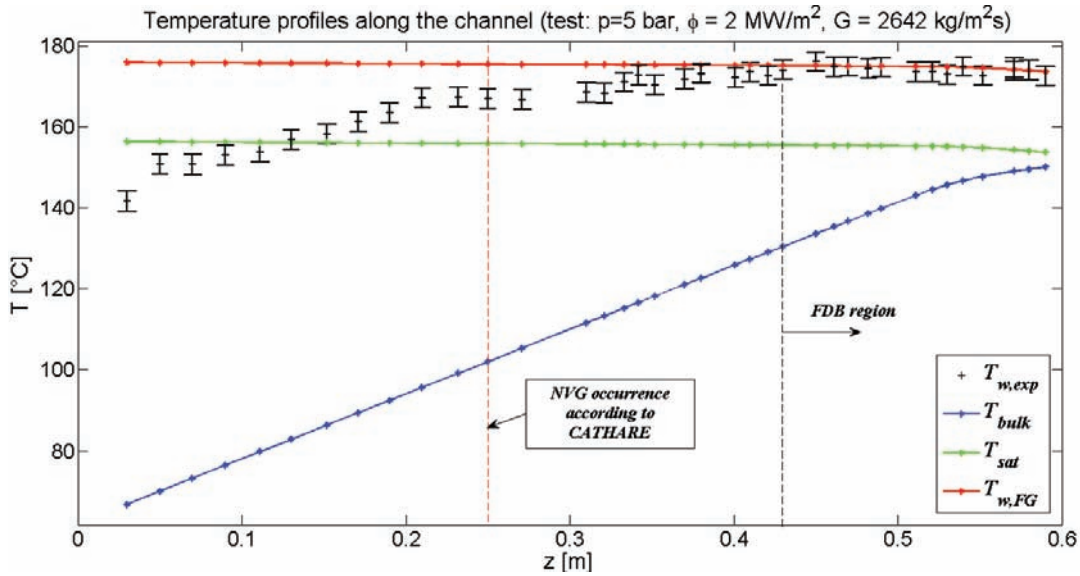

Figure 4. Determination of the FDB region according to the experimental temperature profile.

4. RESULTS

Several correlations for fully developed boiling are presented along with their validity ranges and eventual drawbacks. The wall superheat predictions are then compared to the experimental data. All the quantities in the paper are expressed in SI units. In particular, the pressure p and the heat flux ϕ in the correlations are in Pa and in W/m^2, respectively.

4.1. Jens-Lottes and Thom Correlations

The Jens-Lottes and Thom correlations are standard correlations employed in the modeling of conventional nuclear reactors and boilers [6]. Jens-Lottes correlation [8] was developed using experiments with water in electrically heated vertical tubes of small diameters (i.e. 3.63, 4.57 and 5.7 mm). The mass fluxes were between 11 and 10400 kg/m^{2}s, the pressures between 0.59 and 17.24 MPa, and heat fluxes between 0.8 and 7.8 MW/m^2. The correlation reads:

$$\Delta T_{sat} = 25\left(\frac{\phi}{10^6}\right)^{0.25} e^{\left(-\frac{1}{62}\frac{p}{10^5}\right)} \tag{5}$$

Thom correlation [11], which is the standard model in CATHARE, is a modification of the Jens-Lottes correlation and was obtained from experimental data in a tube of internal diameter equal to 12.7 mm. The correlation is valid only for water, flowing upwardly with velocities between 1.5 and 6.1 m/s, pressures between 5.17 and 13.78 MPa, and heat fluxes between 0.284 and 1.58 MW/m^2. Thom observed that the Jens-Lottes correlation may under-estimate the wall superheat and suggested an improved version:

$$\Delta T_{sat} = 22.65\left(\frac{\phi}{10^6}\right)^{0.5} e^{\left(-\frac{1}{87}\frac{p}{10^5}\right)} \tag{6}$$

Figure 5 shows that these correlations significantly under-predict the heat transfer in FDB and lead to higher wall superheat in comparison with the experiments. The reason of this poor performance may be that both the correlations were developed for relatively high pressure applications, while the SULTAN-JHR experiments are at lower pressure (0.2 to 0.9 MPa). Therefore their use for the JHR modeling is not recommended.

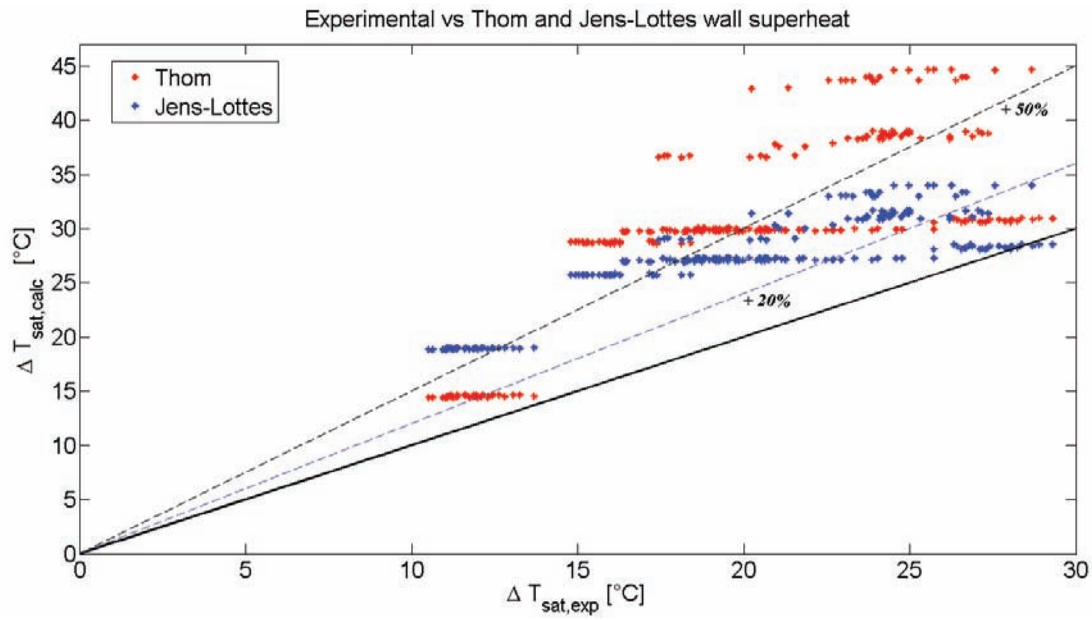

Figure 5. Comparison of the experimental data with Jens-Lottes and Thom correlations.

4.2. Qiu Correlation

Qiu correlation [12] was developed for vertical narrow annuli, using experimental data obtained from two tests sections with gaps of 1.0 and 1.5 mm. It was observed that the boiling heat transfer is enhanced due to the restricted growth of bubbles in the narrow channel. In particular, a decrease of the channel gap led to a decrease of the wall superheat. A modified version of the Jens-Lottes correlation was suggested:

$$\Delta T_{sat} = A \left(\frac{\phi}{10^6}\right)^{0.25} e^{\left(-\frac{1}{62}\frac{p}{10^5}\right)} \tag{7}$$

The pressure and the heat flux coefficients were kept constant while the multiplicative factor A was obtained with a best-fit, and it is equal to 6.08 and to 7.24 for a gap size of 1.0 and 1.5 mm respectively. In order to apply this correlation to the SULTAN-JHR modeling, the coefficient A = 8.77 was computed with a linear extrapolation. It was observed that Qiu correlation strongly under-predict the wall superheat, as shown in Figure 6. The under-prediction may be due to the fact that the correlation is not used within its range of validity (including possible effects of the linear extrapolation on A). In fact, the correlation was developed with experiments at higher pressures (1.2 - 4.0 MPa) and very low heat fluxes (< 0.1 MW/m^2). These low heat fluxes implied low wall superheat (2 - 4 °C) that seems to be comparable with the experimental uncertainties. Thus the correlation may also be affected by this issue.

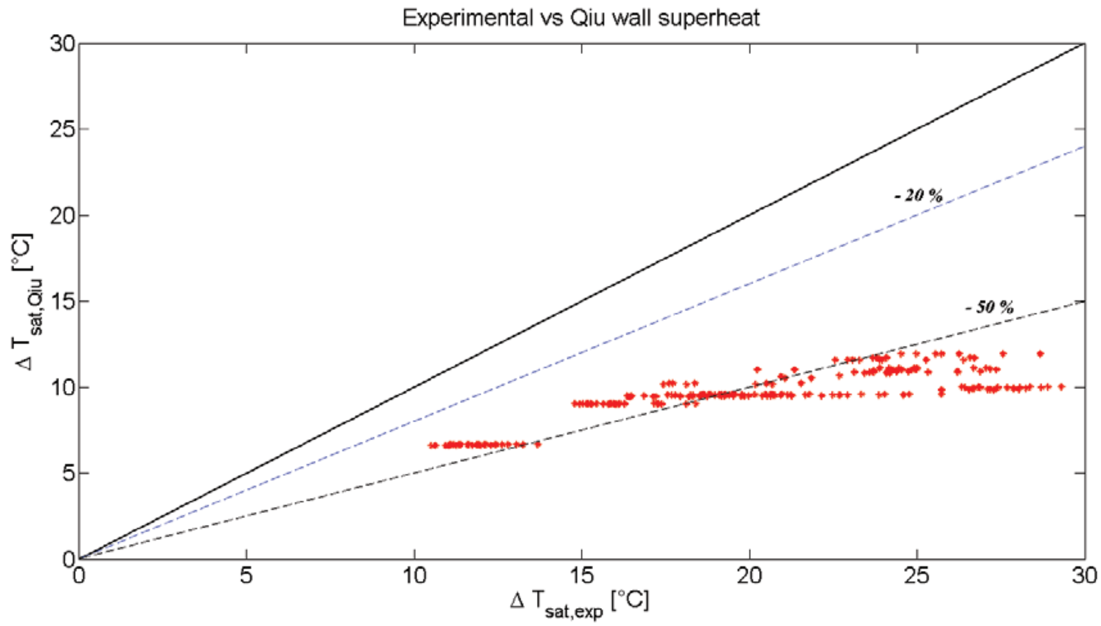

Figure 6. Comparison of the experimental data with Qiu correlation.

4.3. Belhadj Correlation

Belhadj et al. [13] studied the wall superheat under fully developed nucleate boiling in vertical, narrow rectangular channels with gap sizes of 2.0, 3.0 and 4.0 mm. The experimental study focused on upward flows at low velocities (< 0.15 m/s), low heat fluxes (< 0.12 MW/m^2), and low pressures (0.139 - 0.145 MPa). The boiling phenomenon was investigated in natural convection conditions that are typical of accidental transients in plate-type research reactors. The authors showed that the conventional correlations for fully developed nucleate boiling (e.g. Jens-Lottes and Thom correlations) overestimate the wall superheat when compared to their experimental data. The heat transfer enhancement was explained with the influence of the narrow gap: the bubbles growing on the two opposite walls can touch each others, thus the rate of bubble detachment from the walls is increased. This effect was taken into account in a correlation that includes the gap size e_{ch} and the bubble diameter D_b via a dimensionless number. The final correlation was obtained from a best-fit of the experimental data that gives:

$$\Delta T_{sat} = 0.484\,(\phi)^{0.25}\left(\frac{e_{ch} - 1.13\,D_b}{e_{ch}}\right)^{0.26} \tag{8}$$

$$D_b = 1.5\,\times 10^{-4}\left[\frac{\sigma}{g\,(\rho_l - \rho_g)}\right]^{0.5}\left(\frac{\rho_l\,C_{pl}\,T_{sat}}{\rho_g h_{lg}}\right)^{5/4} \tag{9}$$

The vapor tension and the steam properties are calculated at the saturation temperature T_{sat}, which is expressed in Kelvin. In case of the SULTAN-JHR experiments, this correlation provides reasonable results at low wall superheat, but several experimental points are poorly predicted, as shown in Figure 7. It should be noticed that Belhadj is the only correlation of the reviewed ones which fails to predict the dependency of the wall superheat from the pressure, as shown in Figure 8. The reason for this may be that the experimental range of pressure variation is quite limited (0.139 - 0.145 MPa). In addition, the accuracy of the correlation may be affected by the use of low heat fluxes, as discussed for Qiu correlation.

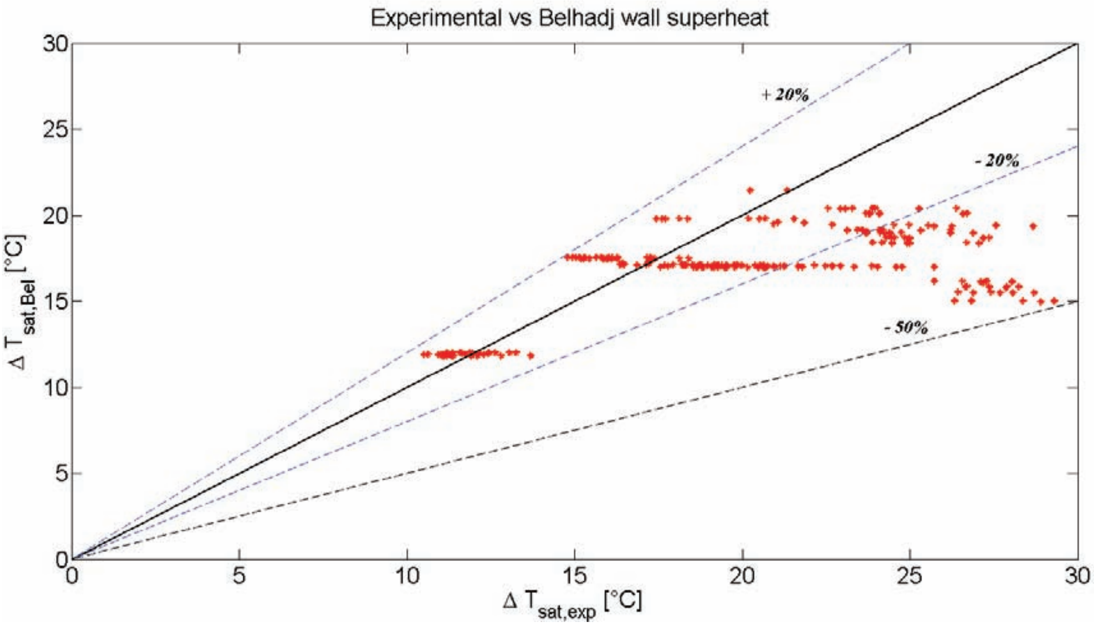

Figure 7. Comparison of the experimental data with Belhadj correlation.

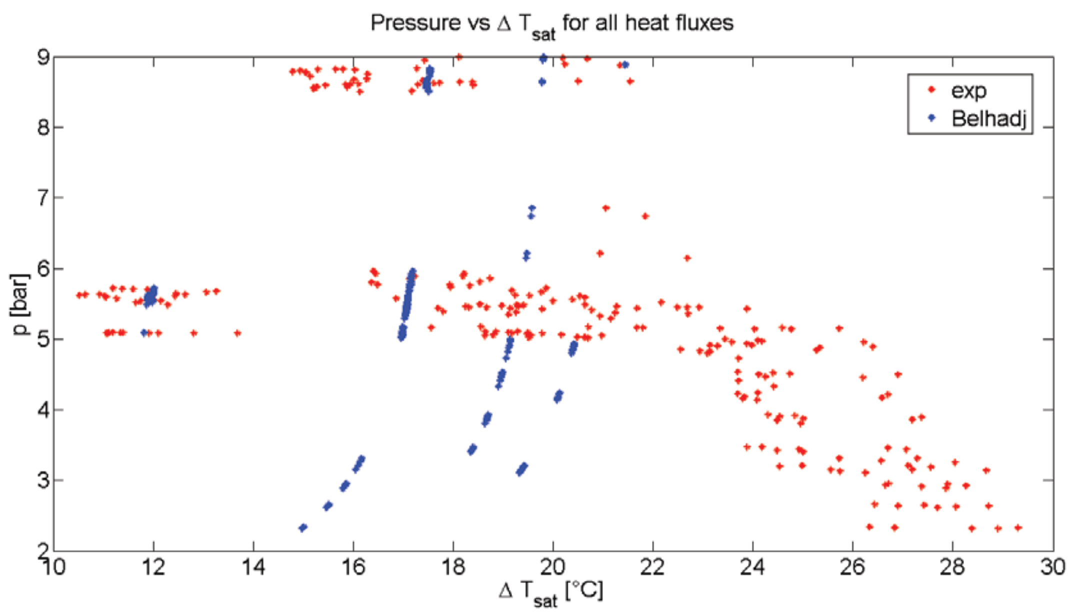

Figure 8. Pressure dependency of Belhadj correlation compared to the experimental results.

4.4. Gorenflo Correlation

The Gorenflo correlation was developed for water pool boiling at pressure between 0.022 and 2.2 MPa [14]. The predictive model is based on a reference heat transfer coefficient h_0 = 5600 W/m^{2}K obtained with a surface roughness R_{p0} = 0.4 μm, a heat flux ϕ_0 = 20000 W/m^2 and a reduced pressure P_{r0} = 0.1. The reduced pressure is defined as $P_r = p/p_{crit}$, where the critical pressure for water is equal to 22.064 MPa. The correlation is written in term of normalized quantities as:

$$h_{PB} = h_0 F_P \left(\frac{\phi}{\phi_0}\right)^n \left(\frac{R_p}{R_{p0}}\right)^{0.133} \tag{10}$$

where the coefficients n and F_P are derived for water as [14]:

$$n = 0.9 - 0.3\, P_r^{\,0.15} \tag{11}$$

$$F_P = 1.73\, P_r^{\,0.27} + \left(6.1 + \frac{0.68}{1-P_r}\right) P_r^{\,2} \tag{12}$$

The surface roughness is usually set to 0.4 μm if the properties of the surfaces are unknown, as suggested in [6] and [15]. This correlation is used in the system code TRACE for the modeling of the nucleate boiling heat transfer term in flow boiling [15]. The use of this closure law in system codes was however criticized in [16], because the surface roughness is usually unknown and difficult to model. The use of this correlation provides a good comparison with the experiments, as shown in Figure 9. The standard deviation of the residuals is equal to 11.9% and the bias is equal to 0.01%. The residuals are computed as:

$$Residual\,[\%] = 100\, \frac{\Delta T_{sat,calc} - \Delta T_{sat,exp}}{\Delta T_{sat,exp}} \tag{13}$$

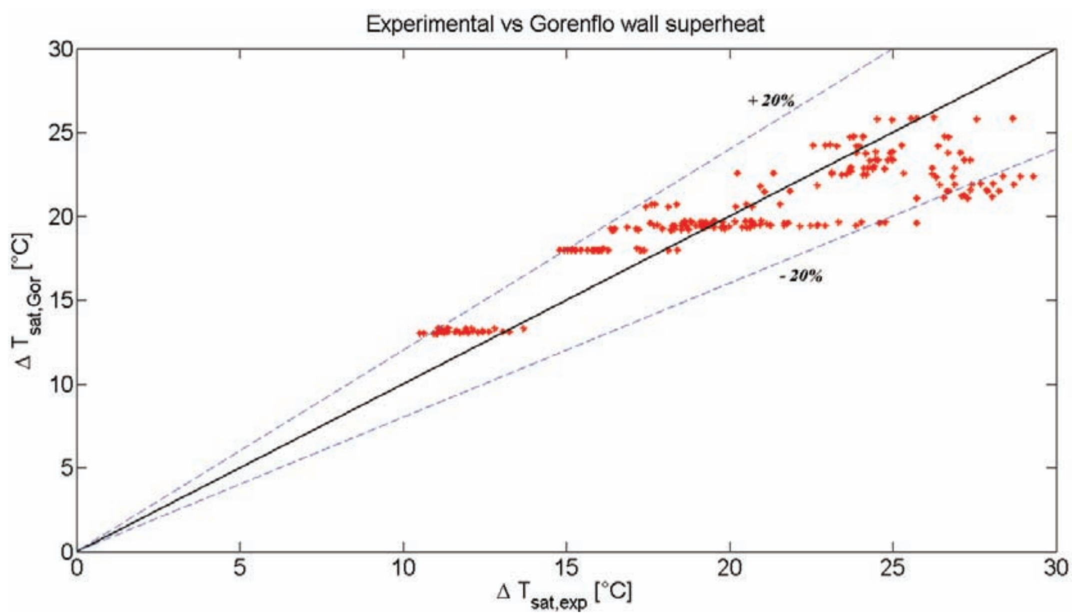

Figure 9. Comparison of the experimental data with Gorenflo correlation.

4.5. Forster-Greif Correlations

The Forster-Greif correlation is used for prediction of the heat transfer in sub-cooled flow boiling of water, especially at low pressure. It has been employed in the thermal-hydraulic modeling for research reactors with fuel flat plates [17]. Different versions can be found in the literature and a short review of them is presented below.

4.5.1. The original Forster-Greif correlation [18]

The original Forster-Greif correlation [18] is based mainly on theoretical considerations (dimensional analysis and appropriate hypothesis) complemented with a limited set of experimental pool boiling data for water at 1 and 50 atm. No reference was reported about the origin of the experimental data, but, from the figures in [18], it is possible to deduce a heat flux validity range between 0.16 and 6.3 MW/m^2. Bergles [19] noticed that this correlation was developed for pool boiling, but it may be also applicable to flow boiling.

Nevertheless, the complexity of the correlation led to the development of simplified formulations for the implementation in system codes, such as CATHARE.

4.5.2. A simplified version of the Forster-Greif correlation

This simplified version is usually reported in the literature (e.g. [17] and [20]) as the correlation of Forster-Greif. The correlation reads:

$$\Delta T_{sat} = 4.57 \left(\frac{p}{10^5}\right)^{-0.23} \left(\frac{\phi}{10^4}\right)^{0.35} \tag{14}$$

The exclusive dependency of the wall superheat on the system pressure and the imposed heat flux (i.e. known quantities) makes this correlation very simple to use and implement in system codes. No original reference could be identified for this relationship. To the authors' knowledge, the first documented application is in an internal CEA report on the preliminary safety analysis of the research reactor OSIRIS (1963). In this report, Eqn. (14) is reported as the correlation of Forster-Greif simplified and approximated for water between 1 and 50 atm. As a matter of fact, this correlation predicts relatively well Forster-Greif's experimental points [18]. In the paper written by Ricque and Siboul [20], it is reported that the correlation was developed at CEA-Grenoble based on experimental data at low pressures (between 0.1 and 0.8 MPa). Unfortunately, the name of the test loop, the channel geometry and the flow conditions were not documented and could not be found in any other available source. From the extensive literature review, it may be concluded that this correlation was developed at CEA-Grenoble before 1963 and it is based on the work of Forster-Greif [18]. It is valid for pressures between 0.1 and 0.8 MPa, but the range might be extended up to 5.0 MPa. The validity of this correlation was also verified for fully developed nucleate boiling in small-diameter tubes (between 2 and 4 mm), at high heat fluxes (between 5.6 and 20.5 MW/m^2), and low pressures (approximately between 0.13 and 0.5 MPa) [20]. The comparison with the experimental data from the SULTAN-JHR database gives excellent results, as shown in Figure 10. The standard deviation of the residuals is equal to 10.1% and the bias is equal to 1.32%.

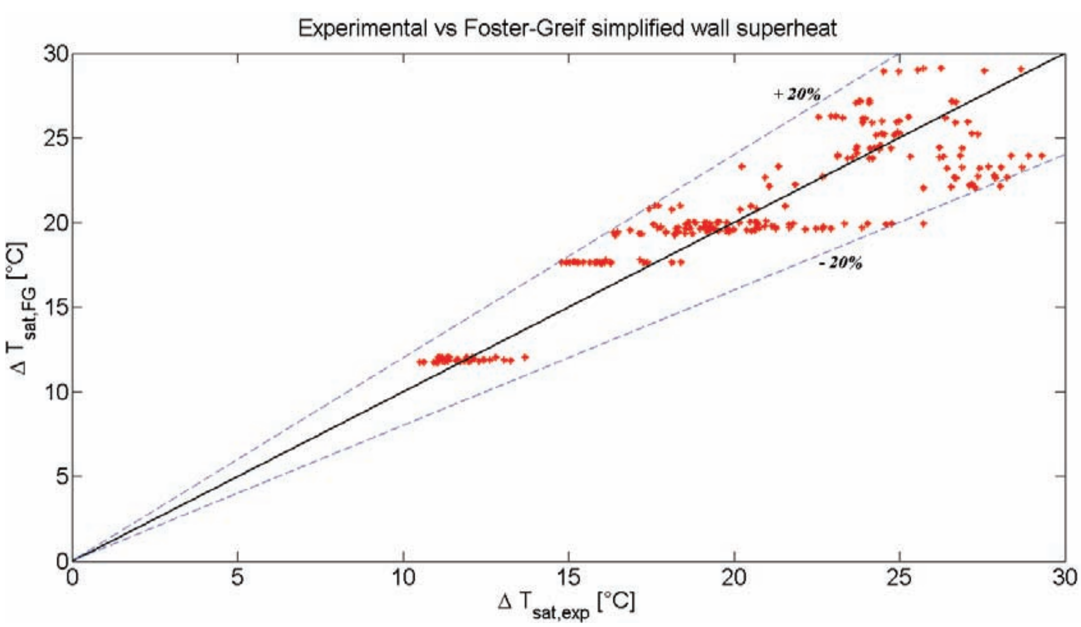

Figure 10. Comparison of the experimental data with the simplified Forster-Greif correlation.

The largest discrepancies between experiments and predictions are observed at low pressures (0.2 - 0.4 MPa) and low heat fluxes (≈ 0.5 MW/m^2), as shown in Figure 11.

This effect is expected at low pressure because it is more difficult to stabilize the boiling phenomenon, and because larger uncertainties are associated to the estimation of the saturation temperature.

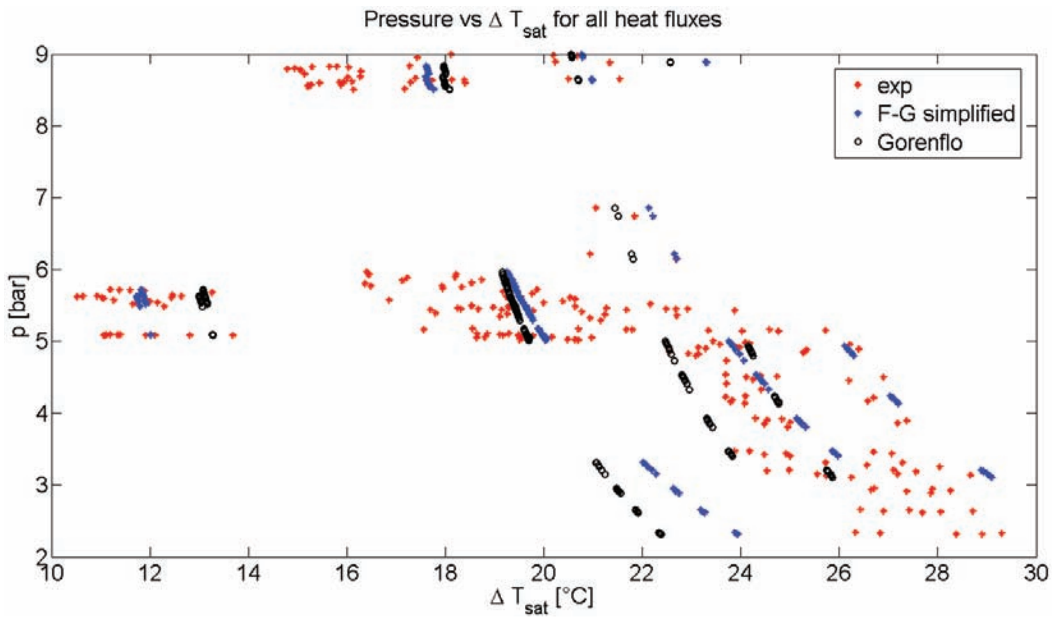

Figure 11. Pressure dependency of the simplified Forster-Greif and Gorenflo correlation compared to experimental data.

4.5.3. The Forster-Greif correlation according to Fabrega [21]

This correlation was obtained from experiments carried out in the CF4 test loop at CEA-Grenoble, and it was developed by Fabrega [21].

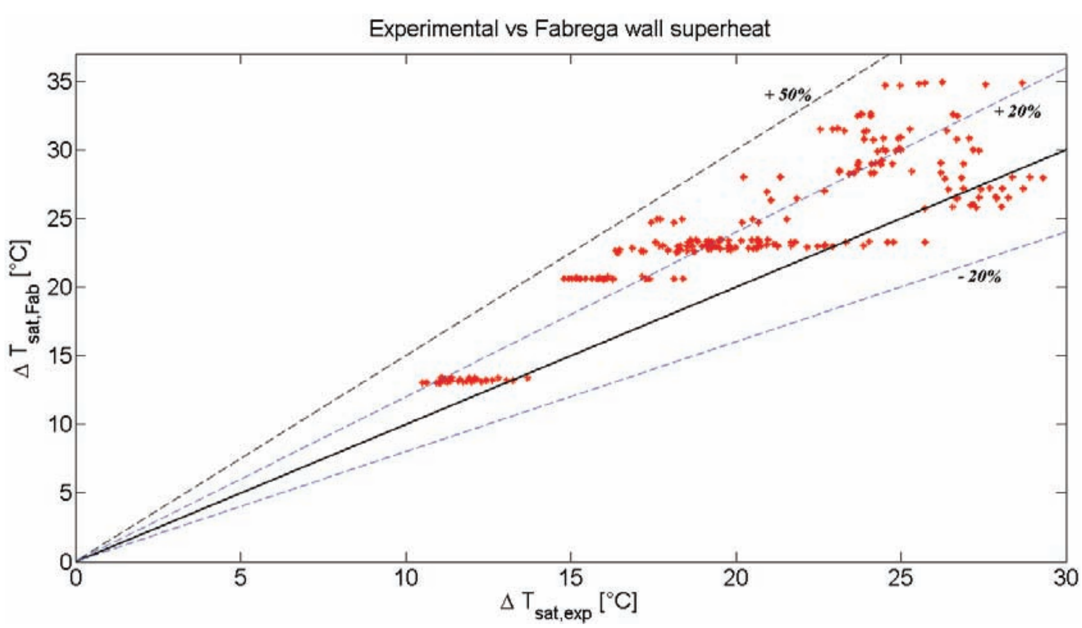

Figure 12. Comparison of the experimental data with Fabrega correlation.

The correlation is quite similar to Eqn. (14), but the fitting coefficients are slightly different. It reads:

$$\Delta T_{sat} = 4.44\left(\frac{p}{10^5}\right)^{-0.23}\left(\frac{\phi}{10^4}\right)^{0.385} \tag{15}$$

The test section consisted of a circular tube of internal diameter equal to 6 mm, and it was operated at a pressure of approximately 0.8 MPa. The measurements were in conditions of upward flow boiling. No clear validity range for the correlation is given. For the SULTAN-JHR experiments, Eqn. (15) predicts higher wall superheat if compared to Eqn. (14) and provides poorer performances as shown in Figure 12.

5. CONCLUSIONS

This paper present an assessment study of correlations for fully developed boiling, with respect to the SULTAN-JHR experiments in a vertical, narrow rectangular channel with gap of 2.16 mm. The flow conditions resemble the design conditions of JHR.

The analysis shows that the correlation of Jens-Lottes, Thom, Belhadj, Qiu and Fabrega cannot accurately predict the experimental data, which were obtained at low pressures (between 0.23 and 0.9 MPa) and high heat fluxes (between 0.46 and 4.41 MW/m^2). Conversely, the relationship of Gorenflo and the simplified formulation of Forster-Greif provide good results. A summary of the predictive capabilities of the FDB correlations with respect of SULTAN-JHR experiments is reported in Table IV. The columns "95 per." and "Min - Max" provide the range in which 95% of the residuals fall and the minimum and maximum values of the calculated residuals, respectively.

Table IV. Summary of the analysis for the FDB correlations.

Correlation	Bias [%]	2σ [%]	95 per. [%]	Min - Max [%]
Jens-Lottes (5)	39.0	39.3	0.7 - 71.4	-2.6 - 78.7
Thom (6)	52.8	50.7	9.1 - 101.0	5.6 - 112.2
Qiu (7)	-51.2	13.8	-64.7 - -39.8	-65.8 - -37.3
Belhadj (8)	-12.9	32.1	-44.1 - 14.8	-48.8 - 18.6
Gorenflo (10)	0.01	23.8	-23.0 - 19.6	-24.6 - 23.5
Forster-Greif (14)	1.3	20.2	-19.1 - 17.9	-22.6 - 19.2
Fabrega (15)	18.4	24.9	-5.4 - 39.3	-9.5 - 41.8

For the modeling of JHR, it is therefore suggested that the simplified Forster-Greif correlation is used, as given in Eqn. (14). The related standard deviation of the residuals is equal to 10.1%. The simplified Forster-Greif formula also has two attractive features: first, its implementation is straightforward; second, it is independent from the surface conditions and roughness (that may not be accurately known for complex systems, such as a reactor core).

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