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Analysis of Critical Heat Flux Correlations for Small Modular Pressurized Water Reactors

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1. INTRODUCTION

The issue of finding other energy sources surrounds technology since the discovery that all existent today are based on limited oil. This year, the European Union has ceased the investments on alternative sources in order to throw them into the open market, hoping for their increase in value. In this way, nuclear energy is one that stands out for the reduced amount of fuel needed and because it does not produce carbon dioxide.

Small-scale pressurized water reactors (PWRs) are able to power small cities, large-sized industries, submarines and high-performance ships. Icebreaker ships, for example, would usually use 100 tons of gas per day if powered by a regular engine. Using a small nuclear reactor this number reduces to 1 pound (half a kilogram) of uranium.

The critical heat flux (CHF) or the departure from nucleate boiling (DNB) is a major thermal hydraulic parameter in the design of light water reactors (LWRs). The CHF occurs when a region of instability is reached during the change of heat transfer mechanism from a heated wall for a fluid, which would lead to a dramatic increase of wall temperature. Transients in a nuclear reactor can affect the rate of heat generation or the coolant flow in the core, damaging the removal of heat from the fuel rods. Knowledge of the CHF on these conditions is essential to prevent fuel rod damages and therefore the release of radioactive material.

Since the critical heat flux depends on several specific reactor parameters, as fuel geometry (diameter, cladding thickness, pitch, etc.), material properties (cladding, coolant, etc.), flow mass rate, pressure, temperature, heat transfer mechanisms (convection, boiling, etc.), there is still a lack of theoretical models able to predict it. Thus, the design of a new reactor depends on experiments to formulate a consistent correlation with its conditions and able to predict the CHF.

This paper aims to compare and analyze existing correlations for flow boiling crisis in order to check their validity in small modular reactors (SMR) and their applicability range. The results presented hereby are preliminary based on the CHF correlations implemented and studied.

2. LITERATURE REVIEW

The small modular reactor is an emerging energy technology that meets the demand of safety, efficiency and sustainability (Liu and Fan, 2014). Several SMR designs have been proposed in recent years. Among the light water reactors, there are the mPower reactor design, the Westinghouse SMR and the NuScale with 400, 800 and 160 thermal power, respectively.

Several empirical CHF models have been proposed in the last decades. Due to the complexity of boiling phenomenon, the development of many semi-empirical models has become necessary, as for example, the boundary layer separation model, the bubble crowding model, the sub-layer dry out model and the interfacial lift-off model (Guo et al, 2014).

In addition, understanding the dependency of the CHF on the flow pattern is useful to develop heat transfer models for the post dry out situation (Ishi and Denten, 1990; Babelli et al.,1994). Ishi and Denten (1990) carried out a flow visualization study of a simplified two-phase core film boiling flow geometry with the objective of determining and characterizing the various post dry out flow regimes considering various pre-CHF flow regimes as the initial condition. Furthermore, flow pattern dependent pressure drop models (Holt et al., 1999) have been developed to understand various phenomena dependencies, such as flow instability on the flow pattern (Nayak et al., 2003). Nayak et al. (2003) analyzed the flow pattern transition instability considering a flow pattern specific model for the pressure drop.

Moreover, simulations by computational fluid dynamics (CFD) are promising for predicting critical heat flux, instead of current simulations through subchannel balances and empirical correlations, such as the COBRA code (Bestion et al, 2009).

Critical heat flux correlations are still the most used method to predict boiling crisis by using subchanel codes. These correlation and their behaviors are presented in the next sections.

3. COMPARING CORRELATIONS

Eight of the most known CHF correlations, listed in Appendix A, were used according to the literature (Tong and Lang, 1997; Todreas and Kazimi, 2012).They are enough for an initial study, which is the purpose of this work. These correlations are functions that depend on the outlet pressure, mass flux, and inlet temperature, usually expressed as quality or enthalpy. ASME (ASME, 2003) tables for water properties were used.

A program was written using Fortran 90 language to help the analysis. It received the parameters of one test section and the variables used in correlation functions, and returned the critical heat flux value charts. A test to verify the applicability range of each correlation on pressure, quality and flow values was implemented.

4. RESULTS

The behavior of the CHF correlations is shown in Figures 1-3. Figure 1 shows the critical heat flux as a function of the inlet quality for seven correlations. Most correlations were formulated to high pressures and high mass fluxes. These characteristic are a challenge for small modular reactors, which operate at lower mass fluxes in general. The general behavior of these correlations is the same although the predicted value varies within a large range.

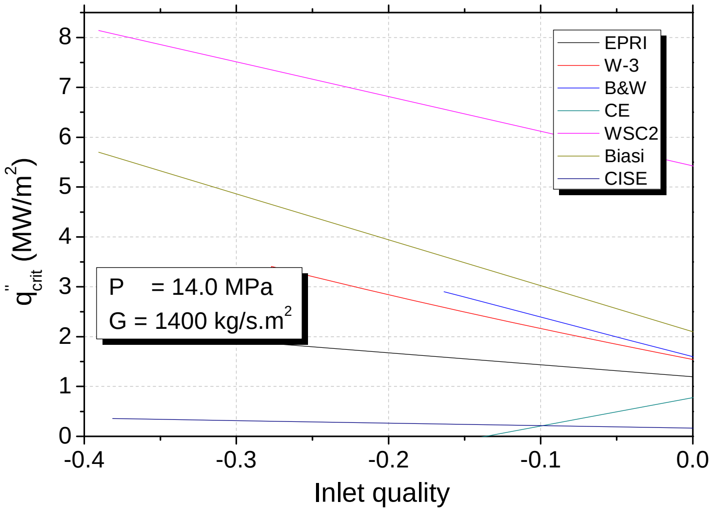

Figure 1 - Critical heat flux correlations as a function of the inlet quality.

Figure 2 shows four critical heat flux correlations as a function of pressure for an inlet temperature equal to 290 ºC and a mass flux equal to 1200 kg/s.m^2. One can note that the EPRI and W-3 correlations presented the same CHF value in this range. The qualitative behaviors were the same for all correlations although the Biasi one shows a slight slope with pressure increase.

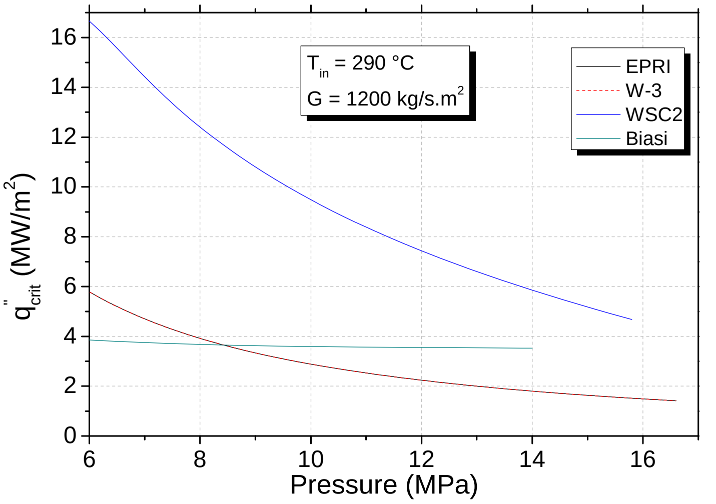

Figure 2 - Critical heat flux correlations as a function of outlet pressure (for large mass flux).

Finally, Figure 3 shows the critical heat flux as a function of the mass flux for outlet pressure equal to 14 MPa and an inlet temperature equal to 290 °C. The WSC2 correlation predicted the larger CHF for all situations.

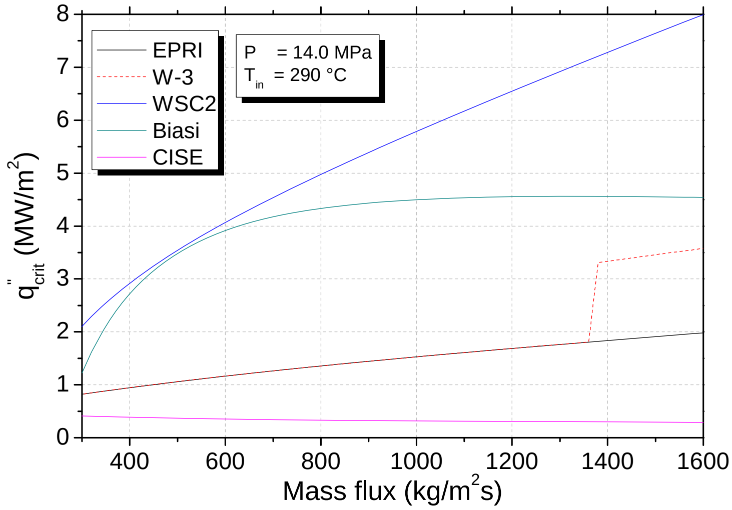

Figure 3 - Critical heat flux correlations as a function of mass flux.

5. CONCLUSIONS

The result of the paper is an assessment of CHF correlations considering the operation conditions of small modular reactors. A more detailed analysis is needed in order to check the validity of these correlations.

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REFERENCES

ASME, **Steam Properties for Industrial Use**, Professional Version Software, Version 1.1. US, 2003.

Babelli, I., Revankar, S.T. and Ishii, M., “Flow visualization study of post, critical heat flux in inverted flow”, *Nuclear Engineering and Design*, vol. 146, 15-24, 1994.

Bertoletti, S., G. P Gaspari, C. Lombardi, G. Peterlongo, M. Silvestri, and F. A. Tacconi, Heat Transfer Crisis with Steam-Water Mixtures, *Energia Nucleare* 12(3): 121-172, 1965.

Bestion, D., Anglart, H., Caraghiaur, D., Péturaud, P., Smith, B., Andreani, M., Niceno, B., Krepper, E., Lucas, D., Moretti, F., Galassi, M.C., Macek, J., Vyskocil, L., Koncar, B. and Hazi, G., “Review of Available Data for Validation of Nuresim Two-Phase CFD Software Applied to CHF Investigations”, *Science and Technology of Nuclear Installations*, 2009.

Biasi, L., G. C. Clerici, R. Sala, and A. Tozzi, Studies on Burnout: Pt. 3, *Energia Nucleare* 14: 530, 1967.

Bowring, R. W, **Physical Model, Based on Bubble Detachment and Calculation of Steam Voidage in the Subcooled Region of a Heated Channel**, Inst. for Atomenergie Rep. HRP- I O, Oslo, Norway, 1962.

EPRI, Electric Power Research Institute, Parametric Study of CHF Data, V**olume 2: A Generalized Subchannel CHF Correlation for PWR and BWR Fuel Assemblies**, NP-2609, Final Report, January 1983.

GE Report, **BWR Blowdown Heat Transfer**, Tenth Quarterly Progress Rep., Oct. I-Dec. 3 1 , 1 974, GEAP- 1 33 1 9- 1O, San Jose, CA, 1975.

Gellerstedt, 1. S., R. A. Lee, W 1. Oberjohn, R. H. Wilson, and L. 1. Stanek, “Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, in Two Phase Flow and Heat Transfer in Rod Bundles”, pp. 63-7 1, ASME, New York, 1969.

Guo, R., Kuang, B. and Cheng, X., “A theoretical CHF model for subcooled flow boiling in curved a channel at low pressure”, *Annals of Nuclear Energy*, vol. 69, 196–202, 2014.

Holt, A. J., Azzopardi, B. J. and Biddulph, M. W., “Calculation of Two-phase Pressure Drop for Vertical Upflow in Narrow Passages by Means of a Flow Pattern Specific Model”, *Trans IChemE*, vol. 77, Part A, January 1999.

Ishii, M. and Denten, J. P., “Two-phase Flow Characteristic of Inverted Bubbly, Slug and Annular Flow in Post-critical Heat Flux Region”, *Nuclear Engineering and Design*, vol. 121, 349-366, 1990.

Janssen, E., and S. Levy, 1 962, **Burnout Limit Curves for Boiling Water Reactors**, General Electric Co., Rep. APED-3892.

Liu, Z. and Fan J., “Technology readiness assessment of Small Modular Reactor (SMR) Designs”, *Progress in Nuclear Energy*, vol.70, 20-28, 2014.

Nayak, A. K., Vijayan, P.K., Jain, V., Saha, D. and Sinha, R.K., “Study on the flow-pattern-transition instability in a natural circulation heavy water moderated boiling light water cooled reactor”, *Nuclear Engineering and Design*, vol. 225, 159–172, 2003.

Todreas, N.E. and Kazimi, M.S., **Nuclear Systems, Volume 1 - Thermal Hydraulic Fundamentals**, 2nd ed., CRC Press, 2012.

Tong, L. S. and Tang, L. S., **Boiling Heat Transfer and Two-phase Flow,** 2nd ed., Taylor & Francis, 1997.

Tong, L. S., “Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution”, *J Nuclear Energy*, 21:24 1-248, 1967.

Appendix A

Table A1 - Critical heat flux correlations

Correlation	Parameters	Validity Range
EPRI (EPRI, 1983) $q''_{CHF} = \frac{1}{0{,}0036} \frac{A F_A - x_{in}}{C \cdot F_c \cdot F_g \cdot F_{nu} + \left(\frac{x - x_{in}}{0{,}0036 \cdot q''} \right)}$ $A = 0{,}5328 \cdot P_r^{0{,}1212} \cdot (0{,}0036 \cdot G)^{-0{,}3040 - 0{,}3285 \cdot P_r}$ $C = 1{,}6151 \cdot P_r^{1{,}4066} \cdot (0{,}0036 \cdot G)^{0{,}4843 - 2{,}0749 \cdot P_r}$ q’’$_{CHF}$ is the critical heat flux in Btu s^{-1} ft^{-2}; q’’ is the local heat flux in Btu s^{-1} ft^{-2}; P$_r$ is the ratio between system pressure and critical pressure; F$_A$, F$_C$, F$_G$ e F$_{nu}$ are optional critical heat flux correction factors due to cold wall, spacer grids and non-uniform heat flux.	Local mass flux (G) Pressure Local quality (x) Inlet quality (x$_{in}$) Hydraulic diameter Heated diameter Length Rod diameter Number of rods Radial profile Axial profile Subchannel type	0,2 -4,1 Mlbs hr^{-1} ft^{-2} 200 to 2450 psia -0,25 to 0,75 -1,10 to 0,0 0,35 to 0,55 in 0,25 to 0,55 in 30 to 168 in 0,38 to 0,68 in 9(3×3) to 25(5×5) Uniform and non-uniform Uniform Matricial only
W-3 (Tong, 1967 *apud* Tong and Tang, 1997) $q''_{crit} = \{(2{,}022 - 0{,}06238\,p) + (0{,}1722 - 0{,}01427\,p)$ $\exp[(18.177 - 0.5987\,p)x_e]\}$ $[(0{,}1484 - 1{,}596 x_e + 0{,}1729 x_e \lvert x_e \rvert)$ $2{,}326G + 3271][1{,}157 - 0{,}869 x_e]$ $[0{,}2664 + 0{,}8357 \exp(-124{,}1 \mathrm{D_e})]$ $[0{,}8258 + 0{,}0003413(h_f - h_{in})]$ q’’$_{CHF}$ is the critical heat flux in kW m^{-2}	Pressure (p) Mass flux (G) Equivalent diameter (D$_e$) Quality (x$_e$) Rod length (L)	5,5 to 16 Mpa 1356 to 6800 kg m^{-2} s^{-1} 0,015 to 0,018 -0,15 to 0,15 0,254 to 3,70 m

B &W-2 (Gellerstedt et aI., 1969 *apud* Tong and Tang, 1997) $q''_{crit} = \frac{A'(A-BX)\times 10^6}{C}$ Btu/hr ft^2 $A' = 1{,}1551 - 0{,}4070\,d$ $A = 0{,}3702 \cdot 10^8 (0{,}5914\,G)^{0{,}8304 + 0{,}6848 \cdot 10^{-3}(p-2000)}$ $B = 0{,}1521\,GH_{fg}$ $C = 12{,}7100\,(3{,}0545\,G)^{0{,}7119 + 0{,}2073 \cdot 10^{-3}(p-2000)}$	Pressure (p) Local quality (X) Local mass flux (G) Hydraulic diameter (d) Heated length (L)	2000 to 2400 psia (13,8 to 16,5MPa) -0,03 to 0,20 0,75 to 4,00×10^6 lb hr^{-1} ft^{-2} (1000 to 5420 kg s^{-1}m^{-2}) 0,20 to 0,50 in (0,5 to 1,3 cm) 72 in (1,83 m)
CE-l CHF (C-E Report, 1975, 1976 *apud* Tong and Tang, 1997) $q''_{crit} = \frac{A'(A-BX)\times 10^6}{C}$ Btu/hr ft^2 $A' = 2{,}8922 \cdot 10^{-3}\left(\frac{d}{d_m}\right)^{-0{,}50749}$ $A = (405{,}32 + -9{,}9290 \cdot 10^{-2}\,p)G^{-0{,}67757 + -6{,}8235 \cdot 10^{-4}\,p}$ $B = GH_{fg}$ $C = G^{3{,}1240 \cdot 10^{-4}\,p + -8{,}3245 \cdot 10^{-2}\,G}$	Pressure (p) Local quality (X) Local mass flux (G) T_{in} Inlet temperature Subchannel equivalent diameter(d) Channel length Channel equivalent diameter (d_m)	1785 to 2415 psia (12,3 to 16,7Mpa) -0,16 to 0,20 0,87 to 3,21×10^6 lb hr^{-1} ft^{-2} (1180 to 4350 kg s^{-1}m^{-2}) 382 to 644 °F (194 to 340 °C) 0,36 to 0,55 in (0,9 to 1,4 cm) 84 to 150in (2,1 to 3,8 m)
Bowring Correlation (Bowring, 1962 *apud* Tong and Tang, 1997) $\frac{q''_{crit}}{10^6} = \frac{\mathrm{A - Bh_{fg}}\,x}{\mathrm{C}}$ W/m^2 $A = \frac{2.317 \cdot F_1 \cdot h_{fg}\,DG}{(1 + 0.0143\,F_2 D^{0.5} G)}$ $B = \frac{DG}{4}$	Pressure (p) Mass flux (G) Equivalent diameter (D) Rod length	0.2 to 19MPa 136 to 18,600 kg/m^{2}s 0.002 to 0.045 m 0.15 to 3.7 m

$C = \dfrac{0.077\,F_3 DG}{\left(1 + 0.347\,F_4 \left(\dfrac{G}{1356}\right)^n\right)}$ F_1, F_2, F_3, F_4 are functions of $p'_r = 0.145\,p$		
<u>ARS CHF (or Biasi)</u> (Biasi *et al*, 1967 *apud* Tong and Tang, 1997) $q''_{crit} = \dfrac{1.883\times10^3}{D_a G^{1/6}}\left(\dfrac{\mathrm{y(P)}}{\mathrm{G}^{1/6}} - X_e\right)$ for low-quality region $q''_{crit} = \dfrac{3.78\times10^3 h(P)}{D^a G^{0.6}}\,(1 - X_e)$ for high-quality region $y(P) = 0.7249 + 0.099\,P \exp(-0.032\,P)$ $h(P) = -1.159 + 0.149\,P \exp(-0.019\,P) + \left(\dfrac{8.99\,P}{10 + P^2}\right)$	Pressure (P) Equivalent diameter (D) Mass flux (G) Heated length (L) Quality (X)	0.27 to 14MPa 0.003 to 0.0375 m 100 to 6,000 kg/m^{2}s 0.2 to 6.0 m $1/(1+\rho_f/\rho_g)$ to 1
<u>GE</u> (Jenssen and Levy, 1962 *apud* Tong and Tang, 1997) $\dfrac{q''_{crit}}{10^6} = 0.705 + \dfrac{0.237G}{10^6}$ for $X < X_1$ $\dfrac{q''_{crit}}{10^6} = 1.634 - \dfrac{0.270G}{10^6} - 4.710X$ for $X_1 < X < X_2$ $\dfrac{q''_{crit}}{10^6} = 0.605 - \dfrac{0.164G}{10^6} - 0.653X$ for $X_2 < X$	Pressure (p) Mass flux (G) Quality (X) Equivalent diameter (D) Rod length (L)	600 to 1 ,450 psia (4.1 to 9.7 MPa) 0.4 to 6.0 MIb/hr ft^2 (540 to 8,108 kg/m^{2}s) -0.45 to 0.45 0.245 to 1.25 in. (0.62 to 3.1 8 cm) 29 to 108 in. (0.95 to 3.5 m)
<u>CISE</u> (Bertoletii *et al*, 1965 *apud* Tong and Tang, 1997) $q''_{crit} = \dfrac{0.794H_{fg}}{\left[\left(\dfrac{p_{cr}}{p}\right) - 1\right]^4 D^{1.4}}\left[\dfrac{1 - \left(\dfrac{p}{p_{cr}}\right)}{\left(\dfrac{G}{100}\right)^{1/3}} - X_0\right]$	Diameter (D) Mass flux (G) Latent heat of vaporization (H_{fg}) Pressure (p)	0.38 to 0.68 in (1.0 to 1.7 cm) 0.2 to 4.1 Mlb/ft^2.hr 200 to 2.450 psia