

BUNDLE AND ANNULUS CHF CORRELATIONS APPLICABLE FOR NEAR CRITICAL PRESSURE REGION

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ABSTRACT

Bundle and annulus CHF correlations are developed by using CHF experimental data from the Freon R-134a Thermal Hydraulic Experimental Loop developed at KAERI. The CHF data are in the range of the pressure of 3200 ~ 4030 kPa (Critical pressure of R-134a = 4059 kPa), the mass flux 150 ~ 1500 kg/m²s, and the inlet subcooling 40 ~ 70 kJ/kg. We obtained local T/H values using a subchannel analysis code and produced CHF correlations for the matrix subchannel, cold-wall subchannel and annulus channel, respectively. A subchannel code, The MATRA- α IBM PC version, is used to obtain the local conditions in a hot subchannel (CHF observed channel) in the 5x5 bundle geometry. All the CHF experimental data are successfully calculated by the present correlations with good prediction performance. The prediction accuracy of the correlations is not distorted by much in any of the ranges of the independent parameters. The bundle CHF correlations could help one to conceptually design a SCWR, by adopting a fluid-to-fluid modeling technique for a CHF near a critical pressure.

INTRODUCTION

The Critical Heat Flux (CHF) phenomenon is defined as an abrupt rise of a wall temperature due to a departure from a nucleate boiling occurring on a heated wall by a heat input. For the influence of a system pressure on CHF, a sharp decrease of the CHF is observed near the critical pressure ($p_r > 0.94$). Collier and Thome (1994) reported that this sharp decrease is closely related to the latent heat of a fluid; when the pressure reaches near to the critical point, the latent heat reduces sharply and becomes zero at the critical pressure. On the bases of this fact, Collier and Thome (1994) extrapolated a CHF curve to the critical pressure where the CHF converges to zero at the critical pressure. At a pressure which is higher than a critical pressure, a coolant becomes a supercritical state and it is heated without a phase transition; no CHF phenomenon occurs.

The supercritical-water-cooled-reactors (SCWR) currently being developed adopt a once-through direct cycle as a heat removal system (Nakatsuka, T. *et al.* 2000) similar to that of

the supercritical *fossil fire power plants* (FPPs). In the once-through direct cycle, the Critical Heat Flux must occur at some location during the reactor start-up and the associated temperature increase must be considered in the design. Because the CHF decreases fast near the critical pressure region, information on the CHF in this region is very useful for the protection of the nuclear fuels from melting. Many experimental studies have been performed on heat transfer in the boiler tubes of supercritical FPPs, since the 1960s (Jackson, J. D. *et al.*, 1979). However, the thermal-hydraulic conditions of the SCWR core are different from that of the FPP boiler where the heat is transferred from the tube wall to the supercritical water flowing inside tubes. In the SCWR core, the heat transfer to the supercritical fluid occurs on the outside surface of fuel rods in rod bundles with spacers. Recently, Hong *et al.* performed CHF experiments near the critical pressure ($P_r > 0.90$) for the annular and bundle geometries by using R-134a (Hong, S. D. *et al.*, 2003, 2004). In this study, we discussed an assessment of generalized CHF correlation of Katto & Ohno's correlation (1984) and Vijayarangan *et al.* correlation (2006) to predict their experimental results, and development of new CHF correlations for the bundle and annulus geometry.

NOMENCLATURE

D	diameter	[m]
D_h	hydraulic diameter	[m]
D_{he}	heated equivalent diameter	[m]
G	mass flux	[kg/m ² s]
h	enthalpy	[kJ/kg]
h_c, h_h	cold and hot side channel enthalpy	[kJ/kg]
h_{fg}	latent heat of vaporization	[kJ/kg]
L	heated length	[m]
P	system pressure	[MPa]
P_c	critical pressure	[MPa]
P_r	reduced pressure, P/P_c	[-]

q''	heat flux	[W/m ²]
Re	Reynolds number	[-]
RMS	root mean square	[-]
SD	standard deviation	[-]
Z	distance	[m]

Greek symbols

ρ	density of gas	[kg/m ³]
v	velocity	[m/s]
σ	surface tension	[N/m]

Subscripts

chf	critical heat flux
g	vapor phase
i	initial
in	test section inlet
f	liquid phase
out	test section outlet

CHF EXPERIMENTS

A series of CHF experiments with a 5x5 bundle test section is performed in the Freon Thermal Hydraulic Experimental Loop at the Korea Atomic Energy Research Institute (KAERI). The Experimental loop uses R-134a ($P_c = 4.059\text{MPa}$, $T_c = 374.2\text{K}$) as a working fluid and can be operated up to 4.50MPa. As shown in Figure 1, the Experimental loop mainly consists of two circulation pumps, two preheaters, two pressurizers, a test section, two condensers and two coolers.

The main parameters of the 5x5 bundle geometry (Test Section) are shown in Table 1 and Figure 2. For measuring the heater rod wall temperatures for a CHF, the ungrounded K-type thermocouples with a sheath diameter of 0.5 mm were attached to the inside rod surface. The radial power factor had about a 20% difference between the 9 centre rods and 16 side rods (Figure 2). Each of the 9 centre rods was instrumented with four thermocouples, while the other 16 rods contained two thermocouples. The heater rods were supported axially by six plain spacer grids.

The general procedure for performing CHF tests is as follows: After setting the mass flux, inlet subcooling and the exit pressure to the desired values, the power to the test section is increased gradually in small steps. At each power level, the test parameters are allowed to stabilize for several minutes to achieve a quasi-steady state condition before raising the power level again. This process is repeated until a CHF occurs, as indicated by a sharp and continuous rise in the surface temperature in any of the 25 heater rods. Based on the Manufacturer’s specification and calibration procedures, the estimated errors for the measurement of the temperature, pressure and mass flux were $\pm 0.5^\circ\text{C}$, $\pm 0.6\%$ and $\pm 0.1\%$, respectively. The experimental conditions of the CHF experiments were as follows:

- System pressure 3.20 ~ 4.03 MPa
- Mass flux 150 ~ 1500 kg/m²s
- Inlet subcooling 40 ~ 70 kJ/kg

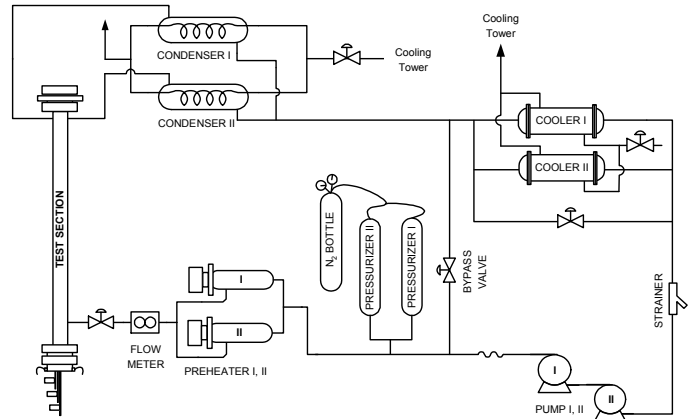


Figure 1. Schematic diagram of the Freon T/H experimental loop

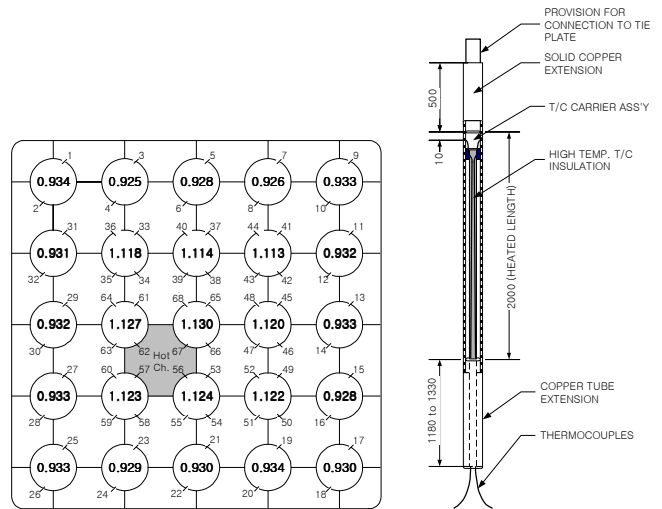


Figure 2. Test section details (heater rod, normalized radial power and 68 T/C locations)

Table 1. 5x5 bundle geometry data

Parameter	Reference bundle	Unheated Rods bundle
Number of heated rods	25	21
Number of unheated rods	0	4
Rod pitch (mm)	12.85	12.85
Rod diameter (mm)	9.5	9.5
Heated length (mm)	2000	2000
Rod to wall gap (mm)	2.55	2.55
Corner radius (mm)	2.5	2.5
Between spacer grids (mm)	564	564
Axial power distribution	Uniform	Uniform
Radial power distribution	Non-uniform	Non-uniform

GENERATION OF CHF CORRELATIONS

Subchannel analysis of the 5x5 bundle data

The hot subchannel is found in a “matrix subchannel” at the reference test section and a “cold-wall subchannel” for the 4-unheated rods test section. The “matrix subchannel” is a normal subchannel that is composed of all the heater rods in a subchannel represented in Figure 3. The “cold-wall subchannel” has one or more cold-walls in a subchannel. The structure of a cold wall can be a guide tube or a burnable poison rod instead of a heater rod. We used the MATRA- α code for the analysis of local conditions from the 5x5 bundle experimental data. The MATRA (Multichannel Analyzer for steady state and Transients in Rod Arrays) is a thermal-hydraulic analysis code based on a subchannel approach for calculating the enthalpy and flow distribution in rod bundle nuclear fuel elements and reactor cores for both steady-state and transient conditions (Yoo et al., 1999). The R-134a property routine in REFPROP version 7.0 is incorporated into MATRA- α to calculate the subcooled properties directly. The base routine of the R-134a property in REFPROP version 7.0(2002) is from the data of Tillner-Roth and Baehr (1994).

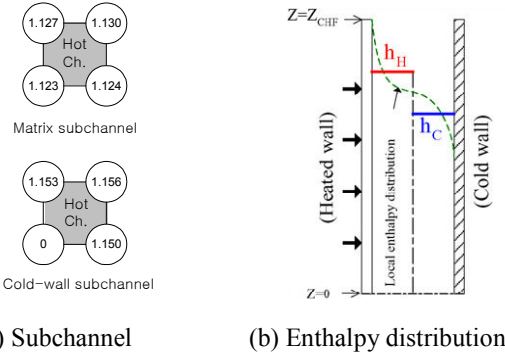


Figure 3. Type of subchannels and schematics for an enthalpy distribution in a cold-wall subchannel

Optimization of CHF correlations

The CHF trends are investigated in all of the 5x5 bundle test sections. Figure 4 shows the dependence of the CHF on the following independent parameters: the latent heat of vaporization, exit enthalpy, exits mass flux, reduced pressure, vapor/liquid density ratio and critical quality. The Figure show that CHF decreases with the system pressure, the quality and the exit enthalpy, and increases with the mass flux and the latent heat of vaporization. In general, the CHF is less for a high vapor/liquid density ratio (ρ_g / ρ_f), and the significant effect of the vapor/liquid density ratio is seen clearly as it approaches 0.7 (Figure 4). This density ratio plays an important role in the RNCP generation of a CHF correlation. When the density ratio is simply correlated, the CHF predictions cannot prevent an over-shooting close to the critical pressure. This problem is solved to mitigate the sensitivity of the CHF to the density ratio as is shown below:

$$q_{CHF}'' \propto f(h_{out}, h_{fg}, G_{out}, \sigma) \times \left(1 - a \left(\frac{\rho_g}{\rho_f} \right)^b \right)$$

This density ratio term has three sensitivity factors obtained by regression analysis, but the updated correlation predicts the critical heat flux very well throughout overall pressure range, especially in the vicinity of the critical pressure. The updated correlation still has the problem that the distribution of the prediction error tends to tilt with pressure. This secondary effect is corrected by considering the reduced pressure as follows:

$$q_{CHF}'' \propto f \left(h_{out}, h_{fg}, G_{out}, \sigma, \frac{\rho_g}{\rho_f} \right) \times \left(\frac{p_{out}}{p_c} \right)^a$$

This reduced pressure term (which has a sensitivity factor) is added to correct the tilt problem of the prediction error. A basic form of the successful CHF correlation is derived via regression analysis, as shown below:

$$q_{CHF, BU}'' = 10.84 (h_{out} \sigma)^{0.316} \left(\frac{G_{out} h_{fg}}{1000} \right)^{0.6} \times \left(1 - 3.2 \left(\frac{\rho_g}{\rho_f} \right)^{5.0} \right) \times \left(\frac{p_{out}}{p_c} \right)^{1.18} \quad (1)$$

This correlation is a local parameter correlation of R-134a fluid based on a subchannel approach for the 5x5 bundle geometry. In the case of a cold-wall subchannel, the above correlation is still useful if an additional cold-wall correction factor is considered, as in the following:

$$q_{CHF, BU-CW}'' = q_{CHF, BU-CW}'' \times \left(\frac{D_{he}}{D_h} \right)^{-m} \quad (2)$$

1/3 or 1/2 is proposed as the power factor of m (Doroshchuk et al., 1975) in developing CHF look-up tables (Groeneveld, 1986). Doroshchuk et al. (1975) concludes that a power factor m of 1/2 gives better prediction results than those of 1/3. The present study also obtains same results (Table 2).

It is proved that the form of the developed correlation considers the physical trends of CHF when it is applied to producing an annulus CHF correlation. When the sensitivity factor of the correlation is obtained by regression analysis for the annulus CHF data without a change of the bundle correlation form, the Annulus CHF correlation is generated as follows:

$$q_{CHF, AN}'' = 43.5 (h_{out} \sigma)^{0.43} \left(\frac{G h_{fg}}{1000} \right)^{0.54} \times \left(1 - 1.56 \left(\frac{\rho_g}{\rho_f} \right)^{3.85} \right) \times \left(\frac{p_{out}}{p_c} \right)^{1.36} \quad (3)$$

Table 2. CHF prediction results

Correlations		Mean	SD	n
Matrix subchannel		1.032	0.169	157
Cold-wall subchannel	m=-1/2	1.010	0.184	67
	m=-1/3	1.059	0.193	67
Annulus channel		1.006	0.114	51

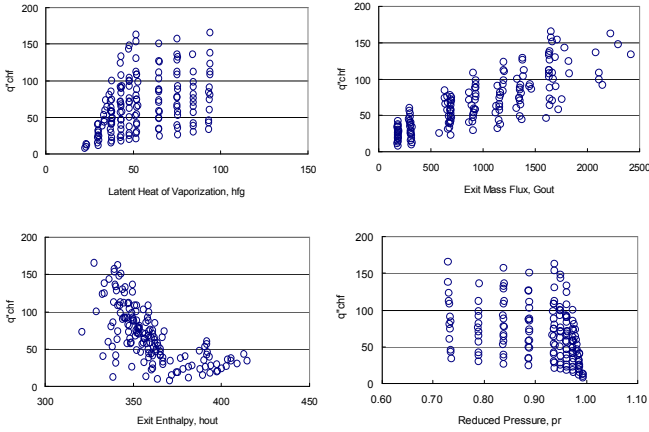


Figure 4. Measured CHF values with the local parameters (5x5 bundle)

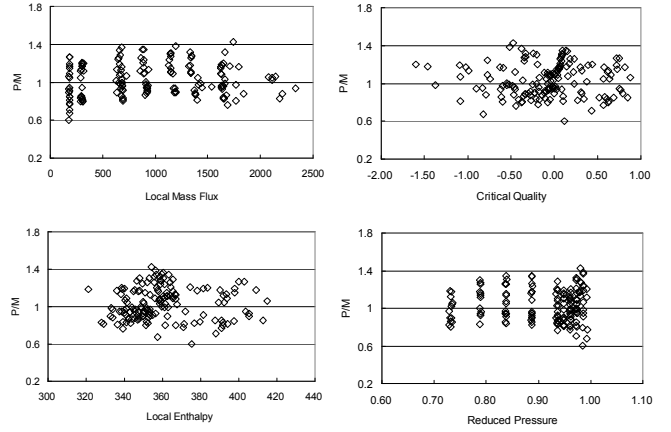


Figure 5. Predicted over measured (P/M) CHF prediction results along the parameters (5x5 bundle)

The new CHF correlation shows very good predictability for the measured CHF data, as listed in Table 2. Figure 6 help to shows that the prediction accuracy of bundle correlation is not distorted by much in any of the ranges of the independent parameters. All the CHF experimental data are successfully calculated by the bundle and annulus correlations with good prediction performance as shown in Figure 6.

RESULTS & DISCUSSION

Katto and Ohno’s correlation (1984) method has been known to agree reasonably well with the available CHF data for flow boiling of water, R-12, R-22, and the liquid helium for tube geometry. They correlated the data as follows:

$$q_{CHF}'' = q_{co}'' \left(1 + K_K \frac{h_f - h_{in}}{h_{fg}} \right)$$

where q_{co}'' , K_K are obtained from an algorithm listed in their paper. The CHF predicted by the correlation of Katto and Ohno (1984) shows a discontinuous behaviour because they proposed different CHF correlations in different ranges of parameters. Vijayarangan et al. (2006) obtained tube CHF data at near-critical pressures by using R-134a as the working fluid and refined the second form of Katto and Ohno’s correlation (1984) to extend its prediction range to near-critical pressures as follows:

$$\frac{q_{CHF,Tube}''}{G h_{fg}} = 0.0051 \times \left(\frac{\rho_g}{\rho_f} \right)^{0.133} \left(\frac{\sigma \rho_f}{G^2 L} \right)^{1/3} \times \left(\frac{1}{1 + 0.003(L/D)} \right) \times P_r^{0.147} Re_f^{0.25}$$

It improved the prediction performance at high reduced pressures. The predicted CHF values obtained via the correlations of the Katto & Ohno (1984) and Vijayarangan et al. (2006) are compared with the measured CHF data (Figure 7 & Table 3). A significant deviation is found with both methods. The correlation of Katto & Ohno (1984) shows that the CHF is

not affected by pressure. It is clear from Figure 7 that the correlation suffers from a large over-prediction at the “Region of Near Critical Pressure (RNCP)”, with part of this overestimation being due to the low value of the CHF near a critical pressure. The correlation of Vijayarangan et al. (2006) underestimates the CHF very large but indicates that the predicted values follow a CHF-reducing trend at the RNCP. Those facts demonstrate that the existing tube correlation is not suitable for annulus or bundle CHF prediction. The principal variables affected by the CHF are the system pressure, mass flux, exit enthalpy, thermophysical properties of the fluid, and geometric dimensions of a test section.

Table 3. Comparison of CHF prediction performance

Test Section	Correlation	Mean	SD	RMS
Annulus (51 data)	Katto & Ohno	1.511	0.553	0.753
	Vijayarangan	0.248	0.063	0.755
	Present	1.006	0.114	0.114
Bundle (157 data, matrix)	Katto & Ohno	5.504	2.022	4.937
	Vijayarangan	0.275	0.118	0.735
	Present	1.032	0.169	0.172

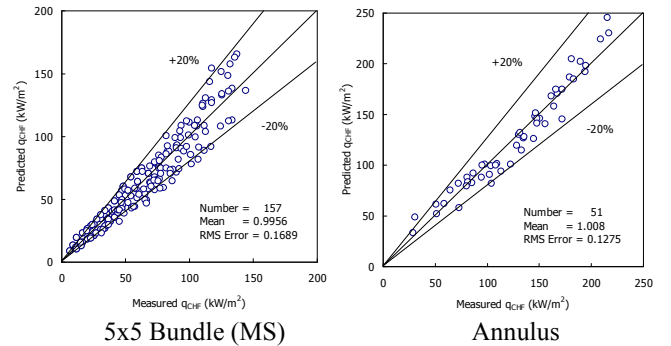


Figure 6. Prediction performance of the bundle CHF correlation for the matrix subchannel

CONCLUSION

CHF correlations are produced from the present R-134a CHF experimental data, based on the local parameter approach. A basic form of the successful CHF correlation is derived by regression analysis (Eq. 1). In the case of a cold-wall subchannel, the above correlation is still useful through considering an additional cold-wall correction factor. When the sensitivity factor of the correlation is obtained by regression analysis for the annulus CHF data without a change of the bundle correlation form, the annulus CHF correlation is generated with very good predictability for the 51 annulus CHF data. The bundle CHF correlations could help one to conceptually design a SCWR, by adopting a fluid-to-fluid modelling technique for a CHF near a critical pressure. In addition, the present experimental work will be used to understand the core thermal hydraulic characteristics of a SCWR and to develop a SCWR safely.

ACKNOWLEDGEMENTS

This study has been carried out under the Nuclear R & D Program supported by the Ministry of Science and Technology (MOST) of Korea.

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