Prediction of Low Pressure Subcooled Burnout Test using General Prediction Methods

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1. Introduction

Among 220 research reactors currently in operation worldwide, more than half of the reactors incorporate core structure submerged in the open pool[1]. In most of the cases, these pool type reactors operate at the pressure range much lower than that of commercial power plants. During normal operation, the core of the reactor is cooled by convection of the coolant keeping thermal-hydraulic parameters well below the safety limits such as the critical heat flux ratio (CHFR). However, during abnormal events where the decrease of the flow or the increase of the power is followed, the CHFR is decreased and may approach the limit. Utilizing every possible measure, this should be prevented to preserve the fuel integrity. In designing and analyzing the safety margin of the reactor core, the appropriate CHF correlation is selected concerning its applicability with respect to reactor's thermal-hydraulic operating conditions. The best selection strategy will be developing dedicated correlation and its limit from well controlled CHF experiment data. This usually guarantees relatively low safety limit value but with cost of money and time[2]. In carrying out preliminary design calculations which must be done in timely manner, developing dedicated correlation is not an option. In this situation, one can adopt general correlation which has wide applicable range to estimate thermal hydraulic margin. Therefore, it is worthwhile to assess the predictive capability of the general prediction methods on the above-mentioned research reactor core design conditions. In this study, the low pressure burnout test results from Mirshak et al. (1959) is predicted by two general CHF prediction methods (Hall-Mudawar correlation and Groeneveled 2006 Lookup Table)[3,4,5].

2. Methods and Results

In this section, the tests performed by Mirshak et al. (1959) is briefly described along with utilized CHF prediction methods and results.

2.1 Mirshak et al.'s Burnout Experiment

In order to assess the effect of various thermal-hydraulic and geometric parameters on CHF, Mirshak et al.(1959) have carried out a series of burnout tests and presented experimental data with correlation[5]. As summarized in Table I and depicted in Fig.1, the test section geometry and thermal-hydraulic parameter ranges include or close to those of typical open pool type research reactor core[6,7]. The burnout heat flux is achieved by increasing the outlet temperature while keeping the heater power constant. From the experiment, total 65 results are obtained, and it was concluded that the critical value depended upon three thermal hydraulic variables (velocity, subcooling, and pressure). The authors also proposed the empirical correlation as shown in Eq. (1) which showed standard deviation of 8%.

$$q_{\mathrm{CHF}} = 266{,}000(1 + 0.0365v)(1 + 0.00914T_s)(1 + 0.0131P) \quad (1)$$

where, q_{CHF}, v, T_s, and P correspond to critical heat flux [pcu/hr-ft^2], velocity [ft/s], subcooling [^{o}C], and pressure [psia], respectively.

2.2 General CHF Prediction Methods

In this study, two prediction methods are utilized. First, Hall and Mudawar (2000) developed two version (inlet and local condition) of correlation by analyzing more than 5,000 subcooled CHF data selected from PU-BTPFL database[3]. By observing parametric trends of selected thermal-hydraulic and geometric conditions, the authors proposed correlations in nondimensionalized forms. The local condition correlation as shown in Eq. (2) is utilized in this study. This correlation is applicable to equivalent diameter between 0.25~15.0 mm, mass flux between 300~30,000 kg/m^{2}s, pressure between 1~200 bar, and quality between -1.00~-0.05.

$$Bo = 0.0722We^{-0.312}\left(\frac{\rho_f}{\rho_g}\right)^{-0.644}\left[1 - 0.900\left(\frac{\rho_f}{\rho_g}\right)^{0.724} x_o\right] \quad (2)$$

where, Bo, We, ρ_f, ρ_g, and x_o corresponds to boiling number [-], Weber number [-], saturated liquid density [kg/m^3], saturated vapor density [kg/m^3], and outlet thermodynamic quality [-], respectively.

Second, Groeneveld et al. (2007) have proposed a rather unique way of predicting CHF values for desired thermal hydraulic conditions[4]. They have analyzed more than 30,000 data and came up with organized CHF look-up table (hereafter, AECL 2006 LUT) in terms of local pressure, mass flux, and quality. Since the table is a collection of values which has been normalized for 8 mm pipe geometry, additional correction factor is applied, as shown in Eq. (3). This correlation can be used for pressure between 1~200 bar, mass flux between 0~8,000 kg/m^2-s, and quality between -0.5~1.0. When compared with the thermal-hydraulic parameter ranges of the experiment, the correlation cannot be applied to test conditions with velocity higher than ~8 m/s. In order to overcome this problem, Kalimullah et al. (2012) have extended the AECL 2006 LUT by adding extra factors as shown in Eq.(4) for mass flux values higher than 8,000 kg/m^2-s up to 30,000 kg/m^2-s[8]. They even adjusted the exponent values of the diameter correction factor (0.5 to 0.312) to further improve the overall predictive capability.

$$q_{CHF} = CHF_{8mm}(P,G,x)\left(\frac{8}{D_e}\right)^{0.5} \quad (3)$$

where, q_{CHF}, CHF_{8mm}, P, G, x, and D_e corresponds to critical heat flux [kW/m^2], pressure [kPa], mass flux [kg/m^2-s], thermodynamic quality [-], and equivalent diameter [mm], respectively.

$$q_{CHF} = CHF_{8mm}(P,G,x)\left(\frac{8}{D_e}\right)^{0.312}\left\{\left(\frac{G}{8000}\right)^{0.376} for\ G > 8{,}000\right\} \quad (4)$$

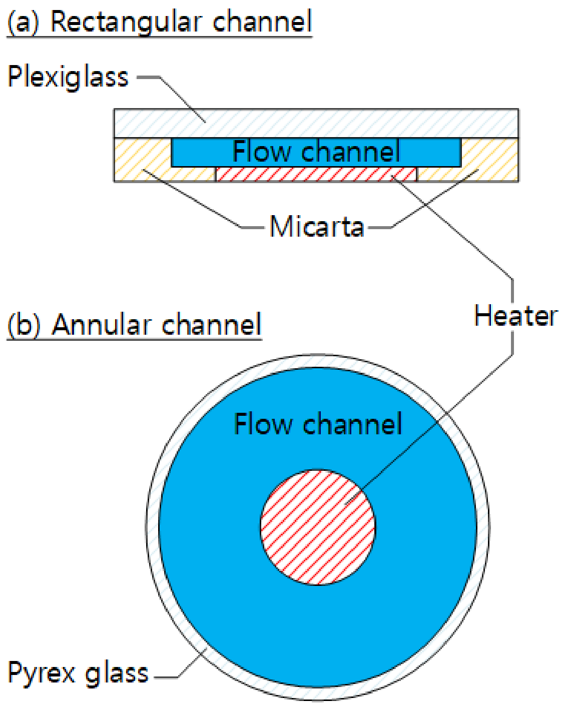

Fig. 1. Cross section view of the test sections of burnout experiment by Mirshak et al. (1959) (not to scale).

Table I: Summary of Experiment Conditions

	Test value	Typical Research Reactor value[6][1]
Equivalent diameter [mm]	5.3~11.7	~5
Heated length [mm]	489.0~609.6	~615
Velocity (outlet) [m/s]	1.6~12.7	~8
Pressure (outlet) [bar]	1.7~5.9	~2
Subcooling (outlet) [K]	6.0~74.0	~70

2.3 Prediction Results

In this study, the thermal-hydraulic simulation of the experimental test section is carried out using CORAL 1.1 (Code Optimized for Research Reactor Thermal Hydraulic Analysis) which is a steady-state research reactor thermal-hydraulic safety margin analysis code developed by Korea Atomic Energy Research Institute[9]. The code solves energy and momentum equations for given inlet temperature, pressure and mass flow boundary conditions. Since the thermal hydraulic conditions given in the test results are outlet conditions, iterative search has been performed to yield inlet values. In the simulation, the heated length of the test section is axially divided into 10 elements and the heat structure with constant power is attached. For each run, constant temperature, pressure, and mass flow rate boundary conditions are applied to the inlet and the code is run until the convergence is reached. Then, the CHF values are evaluated by embedded correlations.

Figure 2 compares the predicted CHF values over the measured ones. The comparison shows that the dedicated correlation proposed in the test (Eq. (1)) gives good prediction capability with average M/P (measured to predicted) value of 0.997 and NRMSE (normalized root mean square error) of 8.36%. The general prediction methods tends to over predict the test data which gives average M/P of 0.931 and 0.871 for Hall and Mudawar (2000) and AECL 2006 LUT (2007), respectively. Their NRMSE values were 30.1% and 19.0%, respectively, which exhibits relatively more scattered predictions from Hall and Mudawar (2000) correlation. This trends is also reported by Kalimullah et al. (2012)[8]. They have mentioned that some of the test cases showed deviations from correct trends from literatures. In this study, the original AECL 2006 LUT is also used to predict the test results. Figure 3 compares the prediction results from

[1] Approximate estimation based on available literatures

two extended versions of AECL 2006 LUT with different exponent value for diameter correction factor ($8/D_e$). The results show that the original version (solid square dots) gives relatively better predictions in terms of average M/P of 1.017 and NRMSE of 13.5%.

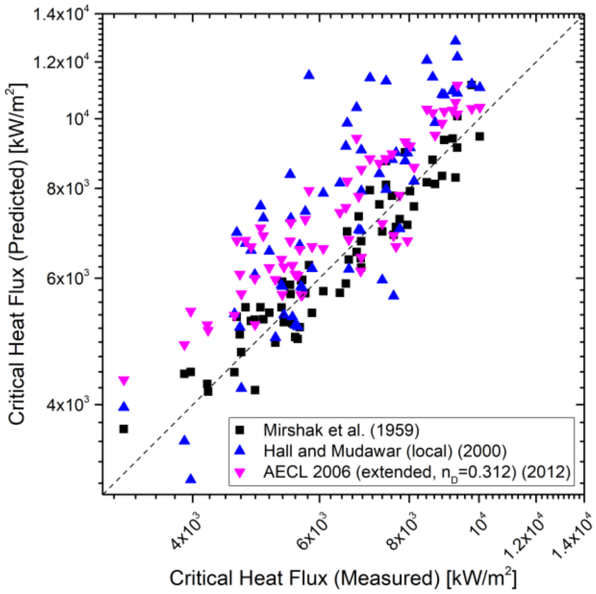

Fig. 2. Comparison of predicted/measured CHF values

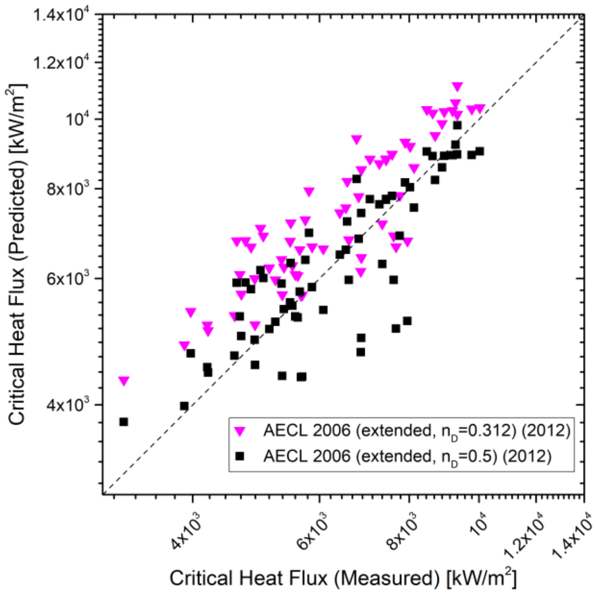

Fig. 3. Effect of diameter correction factor of AECL LUT

3. Conclusions

In this study, the low pressure subcooled burnout experiments were simulated by thermal margin analysis code CORAL using embedded general type correlations. The analysis showed that the general prediction methods tend to over-predict the test results and scatter was wider with respect to the dedicated correlation. For AECL 2006 LUT, using original exponent for the diameter correction factor and adding extension factor for high mass fluxes gave better prediction accuracy over one proposed from the literature. The study shows that more work is needed to clarify the applicability of the general prediction methods for low pressure and subcooled conditions.

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