Thermal Hydraulics of High Heat Flux Components

by

Anthony Edward Hechanova

Submitted to the Department of Nuclear Engineering
on January 13, 1995, in partial fulfillment of the
requirements for the degree of
Doctor of Philosophy in Nuclear Engineering

Abstract

The thermal hydraulic phenomena, particularly the critical heat flux (CHF) limits, for highly-subcooled water in unobstructed pipe flow are investigated using experiments and computational models. These phenomena are important in the design of plasma facing components in fusion tokamak reactors. The experiments employ filtered and de-ionized water flowing through a 9.5 mm bore in a 19 mm x 19 mm x 130 mm copper monoblock. Single-sided heating of the block is achieved by direct electric heating of a 51 mm long plasma sprayed thin layer (0.4 mm) of tungsten overlaying a thin film (0.1 mm) of plasma sprayed ceramic on an outer wall. In the analysis, the heat transfer coefficient on the coolant-side wall relies on extrapolation of the existing Chen and Shah nucleate boiling correlations but is validated using outer wall temperature measurements and a heat conduction model.

A total of 33 test runs were conducted, of which 17 qualify as bench mark CHF data points. Fifteen of the bench mark runs are in a region where it is argued that bubble detachment cannot occur. The hydraulic boundary conditions for the 15 bench mark data points are: pressure between 2.2 and 3.0 MPa, coolant mass flux between 2.6 and 15 Mg/m²s, and equilibrium exit quality between -0.44 and -0.49. The critical heat flux ranges between 13 and 28 MW/m². A correlation is formulated in which the data is fit as a relation between Stanton and Peclet numbers.

Our results are combined with a CHF data base of 275 points from several sources to enhance the generality of the following proposed CHF correlation:

\[ St_{CHF} = 50 \frac{Pe}{Re} \left( \frac{1}{Ja} + 0.00216 Pr^{0.8} Re^{0.5} \right) \left( 1 + \frac{10}{20 + Lh/Dh} \right) Pr^{0.6} Pe^{-0.9} \]

The CHF data base parameter ranges are as follows: Pe [7 x 10⁴ to 3.2 x 10⁶], heated length/heated diameter ratio [5 to 78], pressure [1 to 7 MPa], coolant channel diameter [5 to 25 mm], and equilibrium exit quality [-0.49 to -0.07]. The proposed correlation bounds the CHF data base as a lower limit and, thus, is an appropriate conservative limit for design applications.

Thesis Supervisor: Mujid S. Kazimi, Professor of Nuclear Engineering

Thesis Supervisor: John E. Meyer, Professor of Nuclear Engineering