Transactions of the Korean Nuclear Society Spring Meeting
Chuncheon, Korea, May 25-26 2006

Pressure effects on critical heat flux in rod bundles

Dae-Hyun Hwang, Keung-Koo Kim, Chung-Chan Lee
Korea Atomic Energy Research Institute, P.O.Box 105, Yuseong, Daejeon, 305-600, Korea, dhhwang@kaeri.re.kr

1. Introduction

Critical heat flux (CHF) is characterized by a sharp reduction of the local heat transfer coefficient which may result in the failure of a fuel rod cladding in a short period of time. The CHF is a parameter of great importance since the thermal power capability of a light water reactor is mostly limited by the prevention of CHF in the fuel rod. In an advanced light water reactor the operating condition may extend to a wider range of pressures in a comparison with conventional PWR conditions. The system pressure may affect CHF through a change of the latent heat of the fluid, the surface tension, the average bubble diameter, and the wall superheat at the CHF condition, and so on.

It is well known that the CHF cannot vary by one variable without affecting another accompanying variable. If the pressure varies under a fixed coolant temperature at the channel inlet, then the CHF may be determined due to the combined effects of the pressure and the local enthalpy. Under this situation, the channel exit quality increases as the pressure decreases due to a decrease of the saturated liquid enthalpy and the latent heat. On the other hand, an increase of the liquid-to-vapor density ratio may activate a bubble movement near the heated wall which will result in the enhancement of the heat transfer between the bubble layer near the wall and the channel core region. The former plays a role of reducing CHF while the latter may improve the CHF. Thus a CHF may reveal a complicated behavior with respect to the pressure at a fixed inlet temperature condition.

If the pressure decreases under a fixed local quality condition, the coolant temperature at the channel inlet should increase in order to maintain the magnitude of the exit quality. In addition, the heat transfer near the wall may be enhanced due to an increase of the bubble buoyancy force. The combined effects of the pressure and inlet subcooling may result in an increase of CHF.

The purposes of this work are to investigate the pressure effects on a CHF in rod bundles, and to assess the applicability of various CHF correlations for a wide range of pressure conditions.

2. Analysis

2.1 Parametric trends of CHF for various rod bundles

The parametric behavior of the CHF with respect to the system pressure was investigated for various rod bundles as shown in Fig. 1. The square array rod bundle consisted of 25 heated rods with an outer diameter of 9.5 mm and a heated length of 3 m [1]. The non-square SSF (Self-sustained Square Finned) rod bundle has a heated length of 0.8 m. In order to compare the parametric behavior at a fixed inlet temperature condition, the experimental data has been adjusted by employing the first order gradient of the CHF which was calculated from the experimental data.

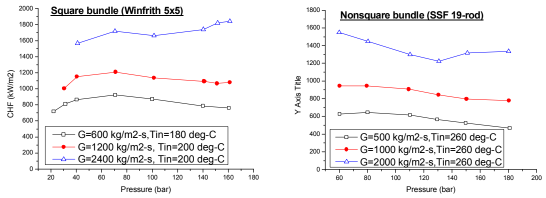

Figure 1. Parametric behavior of CHF with respect to the pressure at various mass velocity conditions.

From the experimental data it was observed that the parametric behavior of a CHF was altered as the mass velocity increased. It is known that the CHF mechanism is governed by a dryout of the liquid film near the heated wall at a high critical quality or a low mass velocity condition. Thus, the decreasing trend of CHF at low velocity conditions may be mainly attributed to a decrease of the latent heat which promotes an evaporation of the liquid film near the heated wall. As the mass velocity increases the CHF mechanism changes which results in a decrease of the critical quality. If the pressure increases at a higher mass velocity, it is supposed that a decrease of the average bubble diameter may interfere with the formation of a vapor blanket near the heated wall which results in an increase of the CHF.

2.2 Assessment of CHF correlations

Four empirical CHF correlations were employed for an assessment of the CHF data in rod bundles at various pressure conditions. The EPRI correlation [2] is a generalized CHF correlation based on the local conditions obtained from the COBRA-3C subchannel code by using the CHF data collected at the HTRF of CU, in large test sections simulating fuel assemblies and grid spacers in a PWR and a BWR. The SSF-1 correlation [3] was developed for the non-square array SSF rod bundles. The experimental data covers the pressure ranges from 60 to 180 bars. The KfK-3 correlation [4] was developed on the basis of the WSC-2 correlation. The geometry dependent parameters relating to the spacers were determined from the CHF data including a triangular array of the rod bundles with very tight lattices (1.02<p/d<1.36). The CHF lookup table method [5] provides CHF values for water-cooled tubes at discrete values of the pressure, mass flux, and critical quality. Linear interpolation between the table values gives the CHF for a specific condition, and several correction factors were introduced to extent the CHF table to various shapes of the boiling channels.

The parametric trends of the CHF correlations with respect to the pressure were compared with the experimental data under fixed inlet temperature conditions as shown in Fig. 2. For the square array rod bundle, it was revealed that the EPRI correlation provides a good result over the entire pressure ranges from 20 to 160 bars. At lower pressure conditions the CHF data reveals a decreasing trend as the pressure decreases. It is presumed that the larger bubble diameter and the higher surface tension may deteriorate the CHF at a lower pressure condition. This trend was reasonably reproduced by the CHF lookup table and the SSF-1 correlation. For the non-square rod bundle the decreasing trend of CHF at lower pressure conditions was not observed since the experimental data was obtained above 60 bars.

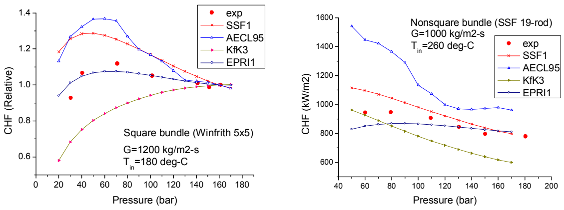

Figure 2. Comparison of the parametric trends for various CHF correlations.

2.3 Prediction of CHF data for rod bundles

The CHF data for rod bundles has been collected from the square array test bundle, non-square array in a circular barrel, and non-square array SSF test bundles. Several CHF correlations were used for the prediction of CHF by using the cross-sectional averaged local conditions in the test bundles. Figure 3 summarizes the analysis results for the four different CHF correlations. The SSF-1 correlation provides the best results for the non-square bundles, while it tends to over-predict CHF at pressures lower than 60 bars. The KfK-3 correlation reveals a reasonable prediction of CHF over the whole pressure range. However the prediction error for the SSF bundle is about two times larger than the SSF-1 correlation, and it tends to over-predict CHF at low pressure conditions. The EPRI correlation provides about 20% higher values of CHF for the square array rod bundle and it tends to over-predict CHF at higher pressures for the non-square array SSF test bundles. The CHF table method tends to over-predict the CHF in rod bundles by employing the same hydraulic diameters with the test bundles. However, the standard deviation was compatible with that calculated by the bundle correlations (for all of 337 data points, the standard deviations by KfK-3, EPRI, and CHF table were calculated as 16.3%, 17.0%, and 17.6%, respectively). Thus, by employing an appropriate bundle correction factor, the CHF table method can be applied to rod bundles over a wide range of pressures.

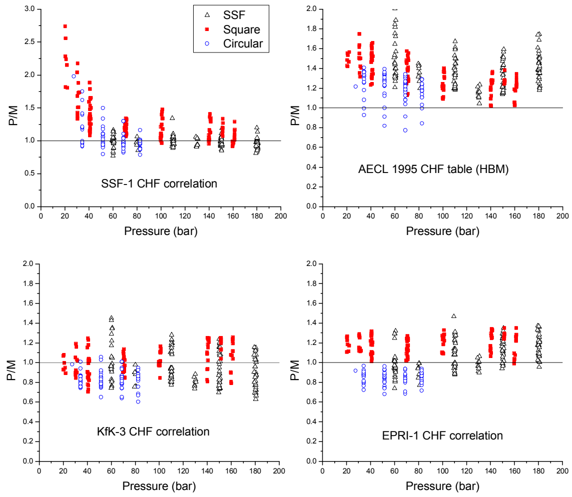

Figure 3. Prediction of CHF in rod bundles at various pressure conditions.

3. Conclusion

The influence of the pressure on CHF was investigated by employing experimental data for square and non-square array rod bundles and four different empirical CHF correlations. From the results of the correlation assessments, it was found that the CHF lookup table method can be applied to the rod bundles over the whole pressure range with an appropriate bundle correction factor.

ACKNOWLEDGEMENTS

This study has been carried out under the R&D program sponsored by the Ministry of Science and Technology of Korea.

REFERENCES

[1] F.H. Bowditch, and D.J. Mogford, An experimental and analytical study of fluid flow and critical heat flux in PWR fuel elements, AEEW-R 2050, 1987.

[2] D.G. Reddy, and C.F. Fighetti, Parametric study of CHF data Vol. 2. A generalized subchannel CHF correlation for PWR and BWR fuel assemblies, EPRI-NP 2609, 1983.

[3] D.H. Hwang, et. al., Development of CHF correlation systems for SMART-P fuel assembly, KAERI/TR-2943/2005 (proprietary), 2005.

[4] M.D. Donne, CHF-KfK-3: A critical heat flux correlation for triangular arrays of rods with tight lattices, KfK-4826, 1991.

[5] D.C. Groeneveld, et. al., The 1995 look-up table for critical heat flux in tubes, Nucl. Engng. Des. 163, 1-23, 1996.