Application of Sub-cooled Boiling Model to Thermal-hydraulic Analysis Inside a CANDU-6 Fuel Channel

Manwoong Kim, Sang Kyu Lee , Hyun Koon Kim, Kun Joong Yoo
Korea Institute of Nuclear Safety, P.O.Box 114, Yuseong, Daejeon, Korea, mwkim@kins.re.kr
Hyoung-Chul Kang, Seong-Yeon Yoo
Dept. Mechanical Design Engineering, Chungnam Univ., 220 Gung -dong, Yuseong, Daejeon, Korea

1. Introduction

Forced convection nucleate boiling is encountered in heat exchangers during normal and non-nominal modes of operation in pressurized water or boiling water reactors (PWRs or BWRs). If the wall temperature of the piping is higher than the saturation temperature of the nearby liquid, nucleate boiling occurs. In this regime, bubbles are formed at the wall. Their growth is promoted by the wall superheat (the difference between the wall and saturation temperatures), and they depart from the wall as a result of gravitational and liquid inertia forces. If the bulk liquid is subcooled, condensation at the bubble-liquid interface takes place and the bubble may collapse. This convection nucleate boiling is called as a sub-cooled nucleate boiling.
As for the fuel channel of a CANDU 6 reactor, forced convection nucleate boiling models for flows along fuel elements enclosed inside typical CANDU-6 fuel channel has encountered difficulties due to the modeling of local effects along the horizontal channel. Therefore, the sub-cooled nucleate boiling has been modeled through temperature driven boiling heat and mass transfer, using a model developed at Rensselaer Polytechnic Institute [1].
The objectives of this study are: (i) to investigate a proposed sub-cooled boiling model developed at Rensselaer Polytechnic Institute and (ii) to apply against a experiment and (iii) to predict local distributions of flow fields for the actual fuel channel geometries of CANDU-6 reactors. The numerical implementation is conducted using by the FLUENT 6.2 CFD computer code.

2. Sub-cooled Boiling Model

Convective sub-cooled boiling occurs when heated walls are superheated while bulk liquid is sub-cooled under given operating pressure. The bubble interface mass transfer depends on several system variables, such as the liquid and saturation temperatures, interfacial heat transfer coefficient, bubble diameter, and vapor void fraction. Boiling at the wall depends on the evaporation heat flux and the wall heat transfer coefficient, which combines single phase heat transfer with the quenching effect of liquid filling the void near the wall after the bubble departs. Mechanistic prediction of this phenomenon is developed at Rensselaer Polytechnic Institute (Podowski, 1997) [2]. According to RPI model, the total heat flux from wall to liquid is partitioned into three components:

$$q''_w = q''_l + q''_Q + q''_E$$

which are liquid convective heat flux, quenching heat flux and evaporative heat flux. Under subcooled boiling conditions, the wall surface is subdivided into portion Ω $(0 \le \Omega \le 1)$ covered by nucleating bubbles and portion $1-\Omega$ covered by fluid. Therefore, convective heat flux is expressed as

$$q''_l = h_{lw} \cdot (T_w - T_l^{cell}) \cdot (1-\Omega)$$

where h_{lw} single phase heat transfer coefficient is derived from either log law if flow is logarithmic or Fourier law if flow is laminar. Liquid phase properties must be used while calculating h_{lw} for either turbulent or laminar flow.

Quenching heat flux q''_Q models additional energy transfer related to liquid filling the wall vicinity after the bubble detachment:

$$q''_Q = 2\pi^{-0.5}\Omega\left(f\kappa_l \rho_l C_{pl}\right)^{0.5}\left(T_w - T_l^{cell}\right)$$

Evaporative heat flux is given by

$$q''_E = \frac{\pi}{6} d_{vw}^3 fn\rho_v L$$

Here, correct prediction of bubble departure diameter d_{vw} is very important because evaporation heat rate depends strongly on this parameter. It can be calculated from the following relation (Unal, 1976, Wei and Morel, 2002):

$$d_{vw} = 2.42 \cdot 10^{-5} \cdot p^{0.709} \cdot a \cdot (b\theta)^{-0.5}$$

3. An Evaluation of a Frigg Assembly

Experiments for upflow of liquid Refrigerant-113 (R-113) through a vertical concentric annular channel were conducted at Arizona State University (1999). The test section consists of 15.8 mm ID, 38.1 mm OD, 3.66 m long, pressurized (269 kPa) annulus, oriented vertically and inner wall was heated. To model the experiment, a two dimensional axisymmetric computational domain was used, consisting of 3,900 quadrilateral cells with 15 cells in the radial. The experimental condition is shown in Table 1.

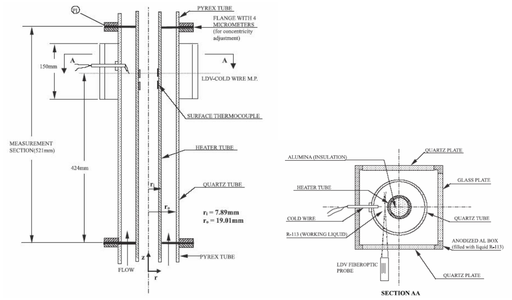


Figure 1 Sub-cooled refrigerant annular flow experiment

Table 1 Parameters of experiments used in validation.

Parameter	Exp. 1	Exp. 2	Exp. 3	Exp. 4
Heat flux, kW/m2	980	980	600	300
Mass flux, kg/m2/sec	436.4	351.0	218.2	208.7
System pressure, bar	1.55	1.55	1.55	1.55

The experiments were compared with SCB mode using based on the same operating conditions.

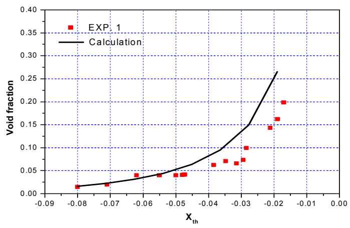


Figure 2 Void fraction comparisons for Exp. 1

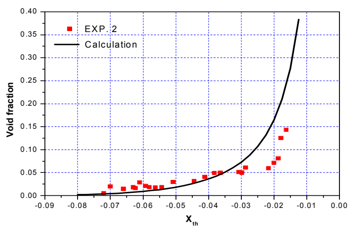


Figure 3 Void fraction comparisons for Exp. 2

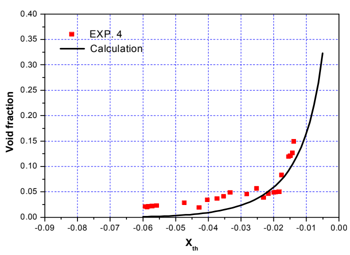


Figure 4 Void fraction comparisons for Exp. 4

These experiments were also compared to FLUENT 6.2 predictions based on the same operating conditions as shown in Fig. 5 and Table 2.

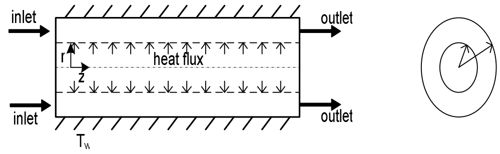
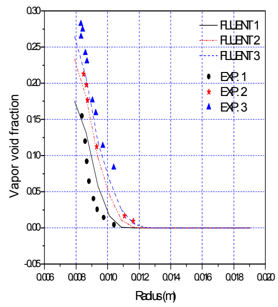


Figure 5 Sub-cooled boiling experiment by Roy

Table 2 Parameters of experiments used in validation.

Parameter	CASE 1	CASE 2	CASE 3
Inner wall heat flux, W/m2	80,000	95,000	116,000
Fluid mass velocity, kg/m2/sec	565	787	785
Mean liquid sub-cooling at test section inlet, °C	37.8	30.3	30.3



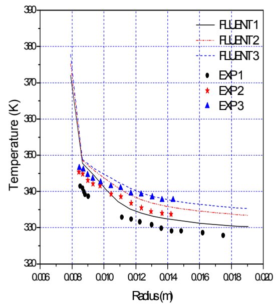


Figure 6 Void fraction Figure 7 Temperature

4. Conclusion

To investigate the convective sub-cooled boiling occurs inside the fuel channel of the CANDU-6 reactor, the RPI subcooled boiling model has been implemented in FLUENT 6.2 using the Eulerian multiphase framework. The model was able to adequately predict the behavior of subcooled convective boiling in a vertical annulus. The results are encouraging for CANDU-6 fuel channels that involve the complex physics of phase change in a multi-fluid environment.

REFERENCES

[1] Podowski, M.Z. (1997) "Towards next generation multiphase models of nuclear thermal-hydraulics" *Proceedings of Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Kyoto, Japan,* Vol. I, pp. 53-68.

[2] Anglart, H. et al. (1997) "CFD prediction of flow and phase distribution in fuel assemblies with spacers" *Nuclear Engineering and Design* Vol. 177, pp. 215-228.

[3] R. P. Roy, V. Velidandla, S. P. Kalra, 1997, ASME J. of Heat Transfer, 119, pp 754-766