Assessment of Critical Heat Flux Data Base for Rod Bundles

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1. Introduction

In the nuclear reactor core, a CHF(Critical Heat Flux) is characterized by a sharp increase of the cladding temperature that ultimately results in a fuel failure. Since the thermal power capability of a PWR core is usually limited by the CHFR margin, accurate prediction of CHF in rod bundles is essential to ensure the safety of fuel rods in the reactor cores. In the framework of developing advanced reactors, both theoretical and experimental investigations of a CHF in rod bundles have been continuously conducted at KAERI. An extensive assessment of CHF data base for rod bundles has been performed for PWR and ALWR conditions by employing various CHF prediction models. The parametric effects were also investigated for rod bundles.

2. Assessment of rod bundle CHF data base

2.1 Rod bundle CHF data base

The rod bundle CHF data base has been collected from various sources[1-5]. Available rod bundle CHF data base are summarized in Table 1.

Table 1. Description of available rod bundle CHF data base

Parameters	CHF data base	
Rod array	Square	Non-square
Heater rod diameter, mm	9.1～11.2	6.4～19.8
Pitch-to-diameter ratio	1.30～1.34	1.05～1.41
Unheated rod dia., mm	10.9～28.8	-
Heated length, m	1.2～4.3	0.4～2.7
Number of heater rods	21～36	7～55
Number of test bundles	161	59
Number of data points	10,162	2,492
Pressure, MPa	1.4～17.1	2.8～18.4
Mass flux, kg/m^2-s	400～5500	140～5470
Heat flux, MW/m^2	0.45～4.38	0.14～5.05
Critical quality	-0.41～0.88	-0.35～1.0

2.2 CHF prediction for nominal conditions

For the core thermal hydraulic design calculations of an advanced PWR, a local parameter CHF correlation, named SR-1, has been developed according to the following procedure[6]: (i) Rod bundle CHF data was selected by considering appropriate bundle geometry and the range of thermal hydraulic parameters. 3980 data points from 61 test bundles were selected which covered uniform and non-uniform axial power shapes. (ii) Subchannel analyses were performed for the test bundles by employing the MATRA code. Important input parameters for determining the local parameters at the CHF locations were the turbulent mixing factor and pressure loss coefficients of spacer grids. (iii) The basis of the correlation form implied a linear relationship between a CHF and the local quality. A velocity correction factor was introduced to account for the rapid decreasing rate of a CHF at lower velocity conditions. (iv) The correlation coefficients were optimized by a non-linear least square fitting method. (v) The validity of the correlation was confirmed by examining the residual trends for the coolant conditions such as the pressure, mass flux, and local quality. The statistical characteristics of P/M were also examined. (vi) The correlation limit DNBR was determined from the 95% tolerance limit of a DNBR with a 95% confidence level.

The enhancement of CHFR margin during normal operating conditions could be achieved by: devising new spacer grids to improve turbulent mixing and heat transfer, improving thermal hydraulic field analysis code for accurate calculation of local conditions, and reducing the uncertainties of the CHF correlation including various correction factors as well as the overall procedure for DNB design. Specifically, experimental investigation on the spacer grid effects for non-uniform axial power shapes may contribute to reduce the conservatism of the existing Tong's F-factor method.

2.3 CHF prediction for accident conditions

In order to provide reasonable predictions of a CHF during reactor transient conditions, it is required that the CHF prediction model could deal with a wide range of thermal hydraulic conditions such as the pressure and mass velocity, also it should properly account for the transient effects on a CHF. During DNB-limiting transients, the minimum DNBR should be calculated by employing the design CHF correlation. If the reactor conditions exceeded the applicable range of the design CHF correlation during non DNB-limiting accidents, a CHF model with a wide applicable range is required to determine the transition of heat transfer mode from pre-CHF to post-CHF conditions.

The accuracy of several CHF prediction models including the CHF lookup table have been assessed for the CHF in square- and nonsquare-array rod bundles[7,8]. As shown in Figure 1, the CHF lookup table method revealed relatively large deviations of the P/M at lower velocities (red symbols) and over-predictions at lower pressures (blue symbols) for the square-array bundle data. As a result, a simple correction factor model was proposed for the CHF lookup table method in order to compensate for the tendency of an over-prediction with decreasing pressures[7].

It is known that a CHF during flow and power transients can be predicted using steady-state CHF correlations and local parameters calculated by an appropriate thermal hydraulic analysis code. However, a very early CHF has been reported during blowdown experiments with flow reversal or mixed inlet flow conditions [9]. It was also indicated that the quasi-steady state method is not adequate for very rapid pressure transients even for subcooled or low quality conditions [10].

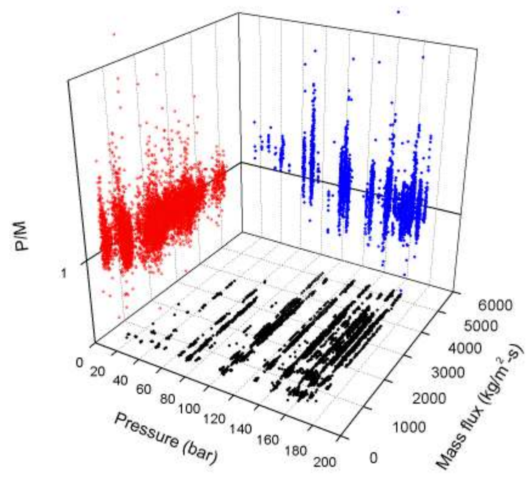

Figure 1. Prediction of rod bundle CHF by 1995-CHF lookup table method

2.4 Parametric effects on a CHF in rod bundles

The parametric effects in tightly spaced nonsquare array rod bundles have been investigated for the heated length, the unheated rods, and nonuniform axial power shapes[11]. Several CHF prediction models were employed to observe CHF characteristics in various configurations of test bundles. Fluid-to-fluid modeling in tightly spaced rod bundles was examined by employing CHF data from Freon and water test loops[2].

The influence of unheated rods as well as the CHF at low velocity conditions has been investigated with CHF data for 5x5 square array test bundles from a Freon-loop. The upstream memory effects due to non-uniform axial power shapes have been evaluated with a mechanistic model [12]. The basic formulation of the correction factor was devised from the mass balance equation in the bubble layer. The key parameters representing the influence of the upstream heat flux profile were revealed as the bubble layer thickness, the mixture velocity of the bubble layer directed parallel to the heated wall, and the lateral mass velocity from core to bubble layer caused by the turbulence. The validity of the model is examined on the CHF data with various axial power shapes for round tubes and rod bundles.

3. Conclusions

- Since the effects of bundle geometry on a CHF are very complicated, suitable experimental data as well as an appropriate thermal hydraulic field analysis code is essential to establish a reliable CHF prediction system for rod bundles.
- Furthermore, a systematic utilization of an existing rod bundle CHF data base, as well as mechanistic approaches in connection with a CFD, would facilitate the enhancement of the thermal margin and the optimization of the CHF test matrices for newly developed rod bundles.

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