Analysis of a Subchannel Void Distribution in a 8x8 Rod Bundle Using the MATRA Code

Dae-Hyun Hwang, Kyong-Won Seo, Chung-Chan Lee

Korea Atomic Energy Research Institute, P.O.Box 105, Yuseong, Daejeon, 305-600, Korea, dhhwang@kaeri.re.kr

1. Introduction

Subchannel analysis codes have been widely adopted in the design calculation for an LWR core since it provides reasonably accurate results on the flow and enthalpy distributions inside the rod bundles with a pertinent computing time. A subchannel analysis code MATRA is under development at KAERI which is purposed to analyze the thermal hydraulic characteristics in rod bundles for several kinds of nuclear reactors; PWR, integral-type advanced reactor, supercritical light water reactor, and a gas cooled reactor. The applicability of the subchannel codes based on the mixture equations has usually been examined at subcooled or low quality conditions. Recently, the MATRA code was extended to higher quality conditions by employing the void drift model [1]. The subchannel exit quality and mass velocity were evaluated for 3x3 or 4x4 test bundles. In this study the subchannel void distribution data obtained from the 8x8 BWR fuel bundles has been evaluated by the MATRA code. The experimental data was collected from a OECD/NRC benchmark program [2].

2. Analysis

2.1 Description of the experimental data

Void fraction measurement has been performed by NUPEC in a vertical square 8x8 rod array, which simulates a BWR fuel assembly, using an X-ray CT scanner system. The tests were carried out in an out-ofpile test facility under high pressure and high temperature fluid conditions. The experimental data is available for the participants in the OECD/NRC BFBT benchmark program. As a part of this program, the steady-state subchannel void distribution data was assessed by employing a subchannel analysis code MATRA. Fifteen test series from five different test bundles were selected which includes a different number of unheated rods and radial/axial power distributions.



Figure 1. Cross-sectional view of the 8x8 test bundles.

The cross-sectional view of the test bundles and the subchannel nodalization scheme is provided in Fig. 1.

2.2 Subchannel void drift model

The void drift model was incorporated in the MATRA code. The net fluctuating mass velocity from channel-i to channel-j is expressed as,

$$w'_{i \leftrightarrow j} = \left(w'_{ij}\right)_{SP} \cdot \theta \cdot \left[\left(\alpha_{j} - \alpha_{i}\right) - K_{VD} \frac{\left(G_{j} - G_{i}\right)}{G_{avg}}\right]$$

The first term in the bracket represents the equal volume exchange due to the turbulent mixing between the subchannels. If the void fraction in channel-j is larger than that in channel-i, then the first term has a positive value and plays a role in reducing the void fraction at channel-j. The second term reveals the equilibrium void distribution due to the void drift phenomena. According to the Lahey's model [3], it was assumed that the equilibrium void distribution. This term has an effect to move the void towards the higher velocity channel. From the assessment of the experimental data, it was found that the void drift coefficient, K_{VD}, is dependent on the flow regime [1]. At the bubbly-slug flow regime ($\chi < \chi_c$),

$$K_{vD} = a_{1} \left(\frac{\chi - \chi_{OSV}}{\chi_{C} - \chi_{OSV}} \right)$$

and at annular flow regime ($\chi \ge \chi_c$),

$$K_{vD} = a_1 + a_2 \left(\frac{\chi - \chi_{OSV}}{\chi_C - \chi_{OSV}} - 1 \right)$$

where,

$$a_{1} = 0.72 \left(\frac{1 - P_{r}}{P_{r}}\right)^{1.33}$$
$$a_{2} = 10$$

2.3 Analysis result

The overall trend of the void distribution was examined for the test bundles by grouping the subchannels into five zones as shown in Fig. 1. The zone averaged void distribution calculated by the MATRA code, was compared with the experimental data at various exit quality conditions for the TS 0-1 as shown in Fig. 2. It revealed that the void fraction appeared at a maximum at INR2, and the void fraction at the CNTR is slightly larger than the surrounding zone. These trends were reasonably well reproduced by the MATRA code. It was also found that the MATRA code tends to underpredict the total void fraction as the average exit quality increases.



Figure 2. Radial void distribution in TS 0-1

The effect of unheated rods was examined for three different rod bundles which contain two, four, and nine unheated rods at the central region. It was found that the MATRA code tends to over-predict the void fraction at CNTR as shown in Fig. 3. Since the central channel has a higher mass velocity, it is conceived that the void in the surrounding region moves towards the central channel due to the void drift model. When we neglect the void drift model in the MATRA code, we obtained a more accurate result at the CNTR for this specific case.



Figure 3. Effect of the void drift model for a test bundle with four unheated rods

For the test bundle with a large unheated rod at the central region and a ferrule-type spacer grid, the

MATRA code reveals a good prediction accuracy as shown in Fig. 4.



Figure 4. Subchannel void distribution in TS 4.

3. Conclusion

A subchannel code MATRA was examined for the subchannel void distribution data obtained from 8x8 BWR rod bundles. As a result of the analysis, it was found that the MATRA code reveals a reasonable prediction capability for the subchannel void distribution up to the average exit quality of 25%. Due to the characteristics of the void drift model employed in the MATRA code, it tends to over-predict the void at the central region as the number of unheated rods increases. In addition, the MATRA code under-predicts the total void fraction as the quality increases. For the test bundle with a ferrule-type spacer and a central large unheated rod, the MATRA code revealed a good prediction accuracy for the subchannel void distribution.

ACKNOWLEDGEMENTS

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

REFERENCES

[1] D.H. Hwang, et. Al., Assessment of the interchannel mixing model with a subchannel analysis code for BWR and PWR conditions, Nucl. Engng. Des., Vol.199, p. 257, 2000.

[2] E. Satori, et. Al., The OECD/NRC BWR full-size fine-mesh bundle tests benchmark (BFBT) – general description, NUTHOS-6, Nara, Japan, 2004.

[3] R.T. Lahey, and F.J. Moody, The thermal hydraulics of a boiling water nuclear reactor, ANS, 1977.