Assessment of subcooled nucleate boiling models for sub-channel analysis of the PSBT experiment using SPACE code

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1. Introduction

Prediction of the void fraction under the subcooled boiling condition is important in nuclear power plant reactors. It causes a significant effect on the flow and heat transfer characteristics that influence the reactor system behaviors at both operating and accidental conditions. In series of research have been studied to estimate the point of net vapor generation and wall evaporation model for the nuclear reactor safety analysis with the best-estimate thermal-hydraulic system codes. In the present paper, four subcooled nucleate boiling models have compared its prediction capability of the void fraction by implementing into SPACE code [1].

2. Methods

In this section, the information on the benchmarking experiments and subcooled models will be provided. The results of the PSBT experiment [2] were chosen to reference data to compare simulation results. Four subcooled boiling models, implementing the version of SPACE code [1], Končar et al. [3], Ha [4], and Ha et al. [5], were selected as candidate models.

2.1 PSBT Experiment

The benchmark exercises were performed at the NUPEC test facility that contains a high-pressure and high-temperature recirculation loop. Three different test sections were used to perform the benchmarks for the single channel void distribution test, bundle void distribution and bundle DNB measurements. The effective heated length is 3.658 m for the bundle test assembly with the transmission method of gamma-ray to measure the density and converted to the void fraction of the two-phase flow. The measurements were performed at three axial elevations at lower (2.216 m), middle (2.669 m), and upper (3.177 m) elevation, respectively. The benchmark consists of two phases with seven exercises. In the present paper, the Phase I – exercise 2 cases, observation of the void distribution in steady-state bundle benchmark, have selected for the validation of subcooled nucleate boiling models.

2.2 Subcooled Nucleate Boiling Models

The original model of RELAP5/MOD3.2.2 code [6] was directly implemented the critical enthalpy from the Saha-Zuber correlation [7] which represents the onset of significant void (OSV) point. Končar et al. proposed a modification of the subcooled boiling model based on the implemented version of RELAP5/MOD3.2.2 code. The quenching factor is derived from the density of active nucleation sites and maximum bubble diameter.

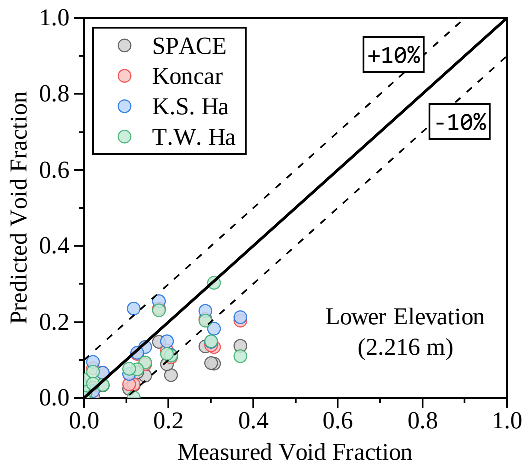
Ha modified the formulae of critical enthalpy from the Saha-Zuber correlation which contains the trend of the bulk liquid temperature at the OSV was opposite. And the quenching factor was also modified based on the model of Končar et al. A new term, the boiling number with a coefficient, is added to supplement the mass flux.

In SPACE code, the subcooled nucleate boiling model has implemented the hybrid version of Koncar and Ha model. For the PNVG prediction, the critical enthalpy is adopted from the model of Ha, and the quenching factor is adopted from the model of Končar with maximum bubble size that suggested by Kocamustafaogullari [8].

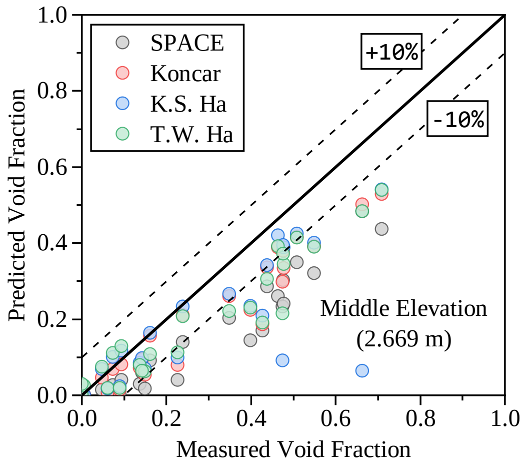
Ha et al. suggested a new model of NVG correlation based on the implemented version of the MARS code [9]. Their new model contains the effect of inlet liquid velocity and hydraulic diameter on axial void fraction development.

3. Results

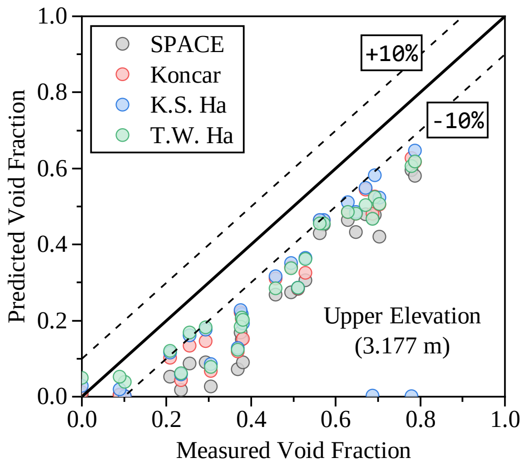
Fig. 1 represents the predicted void fraction comparing to the measured void fraction. The measuring points are located in three different axial positions for the lower, middle, and upper part of the heating section. As shown in fig. 1(a), all the models estimate the void fraction within 10% of measured data except the implemented version of SPACE code. The discrepancy increases as the measuring point increases. In fig. 1(b) and 1(c), most



(a) Lower elevation (2.216 m)



(b) Middle elevation (2.669 m)



(c) Upper elevation (3.177 m)

Fig. 1. Comparison of the measured and predicted void fraction at three axial locations.

of the models are under-estimated the void fraction especially in the low subcooled region. In the case of the model of Ha, the two calculation cases were not converged at the equivalent experimental conditions. The averaged RMS error for each location is summarized in table I.

Table I: Comparison of RMS error for all cases

	Lower Elevation	Middle Elevation	Upper Elevation
SPACE	5.70 %	13.41 %	19.45 %
Končar	4.56 %	8.66 %	15.80 %
K.-S. Ha	4.15 %	7.03 %	14.07 %
T.-W. Ha	4.43 %	9.75 %	15.39 %

4. Conclusions

In the present paper, four subcooled nucleate boiling model have assessed to the experimental data at high-pressure and high-temperature conditions. Most of the models produced reasonable results at the lower part of the heater section which is highly subcooled however, the discrepancies are increased at the upper part of the heater. The model of Končar and model of Ha et al. improve the accuracy of SPACE code in terms of void fraction. Further validations are necessary to ensure the coverage of assessed models with various sets of the experiments.

ACKNOWLEDGMENT

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