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An Evaluation and Development of Drift-Flux Correlations for Enhanced Nuclear Reactor Simulations

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In the coupled neutronics and thermal-hydraulics analysis, the void fraction plays a significant role in determining various reactor parameters such as reactor coolant mixture density, neutron moderation, local power distribution, two-phase pressure drop, two-phase flow regimes, and heat transfer coefficient. Consequently, accurate prediction of the void fraction is crucial for nuclear reactor simulations in both steady-state and transient conditions. This research aims to evaluate a range of drift flux correlations frequently employed in the nuclear and oil industries. Initially, a simple and robust one-dimensional two-phase flow code, based on the drift flux model, was developed and validated to assess the performance of the selected drift-flux correlations. Subsequently, a comprehensive statistical evaluation was performed using over 1,600 experimental tests drawn from open literature, which encompassed various vertical and horizontal flow regimes and geometries, including pipes, annulus channels, and fuel assemblies. The evaluation results identified the Hibiki & Ishii correlation as the most accurate, with a mean absolute error of 16.2%, followed by Toshiba and Antonio correlations, with mean absolute errors of 17.45% and 17.69%, respectively. In addition, the same experimental dataset was utilized to derive a new drift-flux correlation for various vertical and horizontal flow regimes. The performance assessment of the newly developed correlation showed an overall improvement, with a mean absolute error of 14.6%, offering a more precise correlation for future nuclear reactor simulations

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