ISSN 0040-6015, Thermal Engineering, 2013, Vol. 60, No. 5, pp. 331–337. © Pleiades Publishing, Inc., 2013.
Original Russian Text © V.M. Alipchenkov, V.V. Belikov, A.V. Davydov, D.A. Emel'yanov, N.A. Mosunova, 2013, published in Teploenergetika.

NUCLEAR POWER STATIONS

Recommendations on Selecting the Closing Relations for Calculating Friction Pressure Drop in the Loops of Nuclear Power Stations Equipped with VVER Reactors

V. M. Alipchenkov[a], V. V. Belikov[b], A. V. Davydov[b], D. A. Emel'yanov[b], and N. A. Mosunova[b]

[a] *Joint Institute for High Temperatures of the Russian Academy of Sciences, Izhorskaya str., 13 bd. 2, Moscow, 125412 Russia*

[b] *Nuclear Safety Institute of the Russian Academy of Sciences, Bol'shaya Tul'skaya str., 52, Moscow, 115191 Russia*

Abstract—Closing relations describing friction pressure drop during the motion of two-phase flows that are widely applied in thermal–hydraulic codes and in calculations of the parameters characterizing the flow of water coolant in the loops of reactor installations used at nuclear power stations and in other thermal power systems are reviewed. A new formula developed by the authors of this paper is proposed. The above-mentioned relations are implemented in the HYDRA-IBRAE thermal–hydraulic computation code developed at the Nuclear Safety Institute of the Russian Academy of Sciences. A series of verification calculations is carried out for a wide range of pressures, flowrates, and heat fluxes typical for transient and emergency operating conditions of nuclear power stations equipped with VVER reactors. Advantages and shortcomings of different closing relations are revealed, and recommendations for using them in carrying out thermal–hydraulic calculations of coolant flow in the loops of VVER-based nuclear power stations are given.

Keywords: thermal–hydraulic computation code, wall friction, closing relation, nuclear power station, VVER-type reactor, verification, mathematical simulation

DOI: 10.1134/S0040601513050029

Quite significant experience has been gained in Russia and abroad in the development and verification of thermal–hydraulic computation codes used for simulating flows of water coolant in the loops of reactor installations, such as RATEG [1], KORSAR [2], RELAP [3], and CATHARE [4]. In addition, development of new-generation computation codes is seen as an undoubted tendency [5]. The notion "a new-generation computation code" includes not only transition to 3D simulation (RANS, LES, and DNS), but also development of 1D codes for carrying out fast assessment calculations with numerical algorithms, verification basis, and physical models based on the modern level of knowledge.

The aim of this work is to carry out a comparative analysis of correlations for calculation of friction pressure drop that are used for determining the parameters characterizing the flow of water coolant in thermal power installations and to work out recommendations on applying the closing correlations describing friction pressure drop in two-phase flows intended for use in simulating the flow of water coolant in the loops of nuclear power plants (NPPs) equipped with VVER-type reactor installations.

CORRELATIONS FOR MECHANICAL INTERACTION BETWEEN THE PHASES AND CHANNEL WALL

The dynamics of two-phase flows in the loops of reactor installations is described in so-called best-estimate computation codes using the suitable models. The correctness of a model is mainly determined by its ability to take into account the specific features relating to mechanical interaction of the phases with the channel walls in a particular flow regime of a two-phase mixture with the use of correlations serving as closing relations for the system of phase conservation equations. The suitable correlations are selected using the maps of two-phase flow regimes.

This section gives a brief description of the closing relations for calculating friction pressure drop that are used in the majority of thermal–hydraulic codes and in the scientific literature. In addition, a new correlation developed by the authors of this paper is presented. In all, four correlations are given.

—The empirical correlations of Lockhart–Martinelli were initially developed by Martinelli et al. [6, 7] in 1944–1949 for calculating horizontal annular flows, which were afterwards modified [8] and extended for all flow regimes. At present, they are used with modifications in the majority of thermal-hydraulic codes, including RATEG [1], KORSAR [2], RELAP [3],