G. Sharaevsky

Institute for Safety Problems of Nuclear Power Plants NAS of Ukraine, Chornobyl, Ukraine

Problems of Reliability Indicators Increase of Critical Heat Flux Calculations in the Water-Cooled Nuclear Reactors Based on the Computer Thermal-Hydraulic Codes

The analysis of the current state of research and developments in the field of creation of thermal-hydraulic computer codes has been performed. The experience of creation of foreign versions of best-estimate codes was analyzed. Considerable attention is paid to the issue of critical heat flux calculation of nuclear reactors channels. It is demonstrated that now the efficiency of application of modern computer codes for estimating of the heat transfer crisis in the water-cooled nuclear reactors requires further improvement. Calculation methods for accuracy increase of predicting this thermal-hydraulic phenomenon in reactor channels are considered. The well-known methods of critical thermal flux in nuclear reactors channels have been analyzed. Peculiarities of determination of the heat transfer crisis in the forced of the vapor-water steam motion have been reviewed. Adequacy of software computer codes designed to calculate the main safety parameters of water-cooled nuclear reactors was analyzed. The idea of the physical mechanism of the heat transfer crisis under forced motion of a two-phase flow in heated channels was considered. Particular attention has been paid to analysis of experimental and calculated data on conditions of initiation of a heat transfer crisis in fuel assemblies rods.

Keywords: thermal-hydraulic computer codes, critical heat flux, heat transfer crisis.