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ANALYSIS OF VARIOUS APPROXIMATIONS IN NEUTRONIC CALCULATIONS OF TRANSIENT IN FAST REACTORS

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A module of transient neutron transport problem solution is an essential part of complex codes designed for the analysis of nuclear safety in different modes of operation of a nuclear reactor. The neutron transport problem can be solved with a variety of approximate schemes, such as point kinetics, adiabatic, quasi-static and improved quasi-static approximations or direct numerical solution of the original equations. Approximate schemes of solution because of their inherent assumptions have limited application area. Therefore, it is necessary to study a possibility of application of the approximate scheme of solution for each specific task.
The paper considers the kinetic calculations of several tests associated with movement of control rods in fast reactors and characterized by the maximum local perturbation of a reactor environment. Each test has been calculated by a direct numerical solution of the transient neutron transport equation based on a diffusion theory and by different approximate schemes of solution of the original equation.
Error of numerical solution obtained using each approximate scheme has been obtained by comparison of calculated reactor parameters (e.g. power or reactivity) with direct numerical solution. The calculation results demonstrate that some approximations that are successfully used for the calculation of similar problems in the thermal reactor are not able to provide acceptable solution accuracy for the fast reactor. Results of the analyses of the different solution schemes are presented. It is found that solutions obtained using combined schemes based on an improved quasi-static approximation are preferred because of high solution accuracy and admissible time expenditure.

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