Abstracts
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Sixty Years without Kurchatov

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Analysis of Covariation Data for Uranium-235

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Uncertainties in the neutron data and correlations between them — covariances — are necessary both for estimating errors of neutron-physical characteristics of reactors and shielding, and for reducing these errors to an acceptable level by adjusting based on an analysis of the results of macro experiments. Up-to-date information on uncertainties of neutron data is contained in three relatively independent libraries of evaluated neutron data — American ENDF / B-VII, Japanese JENDL-4.0, and Russian BROND-3.1. In this article the covariance data from these libraries for the most studied fuel material, uranium-235, are compares among themselves and with a scattering of evaluated cross-sections, recommended in these libraries. It was found that in the fast neutron region, the uncertainties attributed to such important characteristics as the number of secondary fission neutrons and the fission cross section are comparable in magnitude with the scattering of evaluated values of these characteristics. Since the evaluation of neutron data in all libraries are based almost on the same set of experimental results, only differently averaged, this indicates that the uncertainties assigned to them are too optimistic. It was also revealed that in all considered data the balance of dispersions attributed to the total cross section and to its components is roughly disturbed. A method of correcting data, which eliminates the noted inconsistency, is proposed. The revision concerns the uncertainties of the components of the scattering cross section — elastic, inelastic, the reaction (n, $2n$) — and leads to a decreasing of the uncertainties attributed to the cross sections of these reactions. The proposed technique can be used to correct the covariance data for other reactor materials.
Key words: uncertainties, covariances, correlations, neutron data, neutron cross-sections, libraries of evaluated neutron data, neutron cross-sections adjusting.

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The Method of the Operator Linear Perturbation for Solution of the Cauchy Problem for the Large Systems of Ordinary Differential Equations Using Parallel Calculations

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This method of solution of the Cauchy problem for large systems of the first order ordinary differential equations using parallel calculations is considered. Estimations of parameters for the iteration process optimization are presented. Advantages of this method are simplicity, opportunity to get solution by parallel calculations and also possibility to resolve solution for non-linear problems by changing the operator using of the solution from iteration process.
Key Words: Cauchy problem, algorithm, iteration process, program, computer, system of equations, solution, space, vector function, parallel calculations.

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"Avalanche" Method of the Numerical Solution for the Linear Cauchy Task for Systems of Ordinary Differential Equations

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This paper is devoted for the algorithm of the numerical solution for the Cauchy task for linear systems of ordinary differential equations. The algorithm for first order systems was realized in the AVALANCHE computer code. This one was developed especially to get accurate solution for large time intervals. The parallel calculations are used to calculate inverse operator of the task.

Key Words: Cauchy task, algorithm, code, system of equations, solution, inverse operator, vector.

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Comments to the "Avalanche" Method of the Numerical Solution for the Linear Cauchy Task for Systems of Ordinary Differential Equations

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This paper is devoted for the mathematical base of algorithm proposed by A.V. Moryakov for the numerical solution for the Cauchy task for linear systems of ordinary differential equations [1]. This base shows that the Moryakov's algorithm may be applied to more general class of the equation right parts.

Key Words: Cauchy task, algorithm, code, system of equations, solution, phase flux, one parameter group.

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Calculation of Neutron Kinetics Parameters for 2D Tests of International Benchmark C5G7-TD Using SUHAM-TD Code

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The international benchmark C5G7-TD is developed for cross verification of calculations of spatial neutron kinetics using the codes which does not use the spatial homogenization and diffusion approximation. In specification of the benchmark C5G7-TD, in addition to the spatial distribution of energy release, it is proposed to compare the timedependent parameters of neutron kinetics, namely reactivity, the fraction of delayed neutrons and the lifetime of prompt neutrons. Calculations of neutron kinetics parameters by SUHAM-TD code performed using different formulas are presented. The impact of using a plane function as an adjoint function on the values of the parameters of neutron kinetics is studied. Since all tests of the C5G7-TD benchmark are "blind", there are no results of comparison with calculations by other codes.

Key Words: surface harmonics method, neutron transport equation, SUHAM-TD code, benchmark C5G7-TD, neutron kinetics parameters.

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Evaluation of the Efficiency of the In-Reactor Control System of the Neutron Flux in the Core of a Fast Neutron Reactor

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The method of calculation evaluation of the efficiency of the in-reactor control system (IRCS) of neutron flux in the core of a nuclear reactor is considered. There are presented the results of numeric analysis of the effectiveness of the (IRCS) regarding to the control of the neutron power in normal and emergency modes with different number and location of the neutron sensors in the core.
Key Words: neutron power, reactivity, IRCS, efficiency, methods of estimating, dynamic process, delayed neutrons, feedback.

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Adaptation of the CTART4 Code for the Calculation of Fast Non-Stationary Processes in a Research Reactor

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The article presents the results of computational studies of non-stationary processes without feedback in a research reactor. The obtained results show a large degree of uncertainty of the calculated functionals depending on the parameters of the time, spatial and energy grids. A brief description of the parallel version of the CTART4 code is given.
Key Words: beyond design basis accidents, dispersal on the instant neutrons, a multigroup diffusion approximation, *R-Z* geometry, parallel calculations.

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Method for Determining the Time Before Startup to Minimum Controllable Power for VVER

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Startup to minimum controllable power (criticality approach) is one of the most hazardous nuclear operations during operation. In particular, the spontaneous and unauthorized startup to minimum controllable power is very dangerous, and it occurs as a result of some technological operations or changes in technological regimes. Currently, there are codes for neutron-physical calculations at NPPs with VVER, such as reactor simulator (IR) and BIPR-7A. These codes calculate the boric acid critical concentration without relying on ex-core ionization chamber data, which may result in inaccuracies in determining the critical concentration. In addition, feeding the primary circuit with clean condensate must be stopped at least 15 minutes before is reached, and these codes do not calculate the time to reach the critical state. As a result, the idea arose to develop a code that would predict the time to reach the critical state and the critical concentration of boric acid only using the measuring equipment readings without reliance on additional calculations.
***Key Words*:** VVER-1000, MCL, IR, BIPR-7A.

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Diagnostics of the Coolant Flow Area Communications Reduce (Clogging) in the RBMK-1000 Fuel Channels

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The NPP safety and reliability problem solution requires the use of a continuous diagnostics of the state of the reactor and software support of operation to identify possible or occurred faults (errors) at a stage where it is possible to quickly and easily eliminate the cause of the fault with minimal damage. For this purpose, the system of calculation and experimental diagnostics of the RBMK-1000 state ECRAN 3D (Experimental & Computational Reactor ANalisys) was developed. However, the basic algorithm of ECRAN 3D system diagnostics, based on the use of short time queues without taking into account the history, does not allow for the diagnosis of slow processes, such as coolant communications flow area reducing in fuel channels, which are possible during the operation of the reactor. To solve this problem, it was decided to develop a new diagnostic algorithm based on machine learning, taking into account the history of changes in thermal hydraulic parameters of the reactor.

***Key Words*:** diagnostics, RBMK-1000, clogging of water communications of fuel channels, ECRAN 3D system.

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Comparative Parameters of Space Gas-Cooled Reactors with Dioxide and Carbonitride Fuel

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A design concept of the space high-temperature gas-cooled reactor on the basis of carbonitride fuel is considered. Composition of a modular reactor is proposed. It is possible due to a considerably higher coefficient of thermal conductivity of carbonitride fuel in comparison with uranium dioxide fuel. Possible dimensionality of the fuel element module is determined. The neutron-physical characteristics of a reactor with the modular structure of the core are investigated and their comparison with the characteristics of a reactor with the monoblock structure of the core is executed.

Key Words: space reactor, carbonitride fuel, modular structure of core, safety.

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Analysis of VVER-440 Materials Embrittlement Under High Fluences Irradiation

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The results of the data analysis of a radiation embrittlement research for the weld metal of WWER-440 (mechanical characteristics, microstructure and composition of solid solution) are presented. The database on radiation embrittlement for VVER-440 materials in channels for surveillance specimens is considered, the principles of the database analysis are described. Influence of chemical elements on radiation embrittlement in the different ranges of fast neutron fluences is revealed. The dependence of copper content in solid solution from fast neutron fluence is offered. Result of work is analytical model of radiation embrittlement for VVER-440 RPV weld metal considering features of radiation embrittlement in the different ranges of fast neutron fluences and microstructural changes in material. The suggested model describes experimental values well and it is applicable for the broad range of copper and phosphorus content in weld metal.

Key Words: weld metal, radiation embrittlement, ductile-to-brittle transition temperature, transition temperature shift, WWER-440 RPV materials, radiation induced precipitates, copper content in solid solution, fast neutron fluence, embrittlement mechanism.

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Concerning the List of Normative and Controlled Radionuclides in Airborne Discharges of NPPs

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The task of well-proved reduction of the list of normative and controlled radionuclides in airborne discharges of NPPs is one of the high-priority one in the field of regulation of materials, contaminating the environment. This article describes one of the possible ways of the solution of the above-mentioned problem. It is based on the analysis of the long-term observation of gas-aerosol discharges of European NPPs with WWER and PWR power units. Besides, it shows created and analyzed List 1, which includes 63 normative and controlled radionuclides in the discharges of the European NNPs of the Soviet design with WWER power units, and List 2, which includes 82 radionuclides, registered in the airborne discharges of the European NPPs with PWR power units.

Key Words: nuclear power plant, WWER type reactor, PWR type reactor, radionuclide, airborne discharge, environment.

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Analysis of the Influence of Restrictions on Isotopes ^{232,234,236}U in Marketable LEU on the Choice of Methods for Enriching Uranium Regenerate in Cascades of Centrifuges

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The study addresses the problem of reprocessed uranium re-enrichment. The problem statement is as follows. To produce reactor-grade low-enriched uranium (LEU) that meets all the requirements on $_{232,234,236}$U presence and, at the same time, to spend the prerequisite amount of reprocessed uranium per unit of product. This task is impossible for an ordinary (three-flow) cascade, standard for natural uranium enrichment. Double cascade fails to accomplish the assigned task too. Though, all the requirements could be satisfied if we apply the modified double cascade that uses LEU as a diluent. The calculations showed that such an approach makes it possible to produce reactor-grade LEU from the allocated amount of reprocessed uranium within the constraints on the $_{232}$U (even within the currently accepted strict ones).

Key Words: reprocessed uranium, isotope separation, separation cascade, closed nuclear fuel cycle, dual cascade, uranium enrichment, uranium recycling.