Abstracts
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Simulation of the Kinetics of a Nuclear Reactor Using the Monte Carlo Method

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The computer code KIR intended for calculations of nuclear reactors kinetics using Monte Carlo method is described. The algorithm realized in the code is exhaustively described. Some results of test calculations are given.
Key Words: Calculation, Kinetics, Nuclear Reactor, Algorithm, Monte Carlo Method, Super Computer.

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Features of MCU-FR

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On the basis of the modules of the MCU-6 package a software tool MCU-FR is developed. Its main purpose is to perform by means of the Monte Carlo method precision calculations of parameters of nuclear reactors with a fast neutron spectrum using pointwise cross-sections based on the evaluated nuclear data files for the entire range of energies. This new software tool has all the main features of the MCU code family and a number of distinctive features which are described in the paper.
Key Words: Transport Equation, MCU-FR Code, Monte Carlo Method, Precision Calculations.

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MCU -FR Verification for Fast Nuclear Reactors Criticality Calculations

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MCU-FR is designed for high-precision calculations by means of the Monte Carlo method of parameters of nuclear reactors with fast neutron spectrum. To check the accuracy of the code 197 benchmark experiments from the International Data Bank ICSBEP with fast neutron spectrum with highly enriched uranium, plutonium, or their mixture as a fuel were calculated.
Key Words: Fast Nuclear Reactor, Benchmark Experiments, Precision Calculations, Monte Carlo Method, MCU-FR Code, MDBFR60 Data Bank.

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Nuclear Heating Calculations for Reactor Structural Materials by Monte Carlo Method

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The method of nuclear heat deposition calculation in reactor materials by MCU Monte Carlo code is described. The method takes into account the decay of radioactive isotopes accumulated during reactor

operation. The accepted model for describing the sources of delayed radiation allows to obtain the spatial distribution of heat in simultaneously with the assessment of effective neutron multiplication factor.
Key Words: MCU Code, Monte Carlo Method, Nuclear Reactor, Nuclear Heat Deposition.

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Finite Difference Equations for Neutron Flux and Importance Distributio in 3D Heterogeneous Reactor with Unstructured Mesh

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The paper develops algorithms of surface harmonics method for deriving finite difference (algebraic) equations to describe neutron field in a heterogeneous reactor. Neutron transport equation is used as an initial equation. The step of deriving the diffusion equation in differential form is omitted. The paper contains no assumption on symmetry of elementary reactor cells (unstructured mesh acceptable) and on possibility to describe neutrons distribution at cell boundaries in diffusion approximation for deriving the equations. It is computationally shown that refusal from diffusion approximation at boundary cells makes solution of test problems sufficiently more accurate.
Key Words: Method of Surface Harmonics, Unstructured Mesh, Neutron Distribution, Neutron Importance, Finite-Difference Equations, Abandonment from Diffusion Approximation.

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Generalized Correction Algorithm for the Finite-Difference Diffusion Equations in Askew-Takeda Method

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This paper describes the correction algorithm for the finite-difference diffusion equations, which increases the coarse mesh accuracy. The equations are brought to the well-known discretization method for Askew diffusion equation (Askew's Coarse Mesh Method); however, the new obtained expressions for correction coefficients provide higher accuracy of finite-difference approximation.
Key Words: Neutron Diffusion, Finite-Difference Equation, Coarse Mesh Correction, Testing, Software.

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The Results of the Research Works for Implementation of the Method and Means of Subcriticality Control of the “Cooling” Ponds of the Spent-Fuel Storage (SFS) of Smolensk NPP

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The paper presents the results of research obtained during the implementation of the technique of subcriticality control of the “cooling” ponds of SFS of Smolensk NPP based on the use of pulsed neutron experiment (α-method). The method includes stationary and non-stationary neutron-physics calculations and measurements of the main characteristics of subcritical system that is “cooling” pond. The description of the software package STEPAN-SFS specially made for the estimated tracking pulsed experiment is given. Using

this method, calculations were carried out, and measurements actual downloads of the “cooling” ponds were performed which are also given in this paper.
Key Words: RBMK SFS, “Cooling” Pond, Subcriticality Control, Pulsed Neutron α-Method, Prompt Neutron Flux Decrease Constant, Neutron-Physics Calculations, STEPAN-Code, Neutron Field.

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Some Features of a Mathematical Model of the Dynamics of Space Thermionic Nuclear Power System (Case SNPS “Enisey”)

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We analyze the important features of space nuclear power system (SNPS), which have to be described in detail in the mathematical model for full-scale simulation of all operation modes. The technology of the model development based on the available experimental data is considered. The accuracy of the physical model is confirmed by comparison of calculated and measured parameters, giving possibility to calculate parameters not measured. Quantitative accuracy assessment is provided.
Key Words: SNPS, Thermionics, Mathematical Model, Dynamics, Simulation, Control.

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A Possible Mechanism for the Formation of Nonwettable “Dry Spots” on the Heated Surface during Nucleate Pool Boiling. Part II. “Feed Water Stop” Regime

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It is considered the modeling of heterogeneous nucleate boiling on a horizontal surface on the ascending branch of the boiling curve – from the formation of steam lens (“dry patch”) up to the pool boiling crisis. The proposed hypothesis can, in some cases, give a logically consistent interpretation of experiments and outline the organizational principle of transforming the wall-liquid-vapor system to the mode of nonwettable “dry spots” formation. The model includes the following modes of nucleate boiling: a) a cyclic mode with contact line reverse to the bubble bottom center and a bubble departure from the surface (at low heat flux q and the contact angle $\theta<90°$); b) the mode of a single steam bubble conversion to the vapor lens state, i.e. to the local film boiling mode with the possibility of spreading of a single “dry spot” at the contact angle $\theta\geq 90°$ and a significant increase of the departure diameter D_d and lifetime τ_d; c) the cluster of 4 steam lenses (dry patch) formation at a given pressure, the liquid underheating and average wall overheating ΔT_{wall}. The critical heat flux (CHF) is triggered due to an instability of dry spots on the heating surface.
Key Words: Nucleate Pool Boiling, Transition Boiling, Contact Angle, “Dry Spot”, Vapor Cluster, Coalescence, Boiling Crisis.

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Radiation Degradation of RBMK Graphite

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Analysis of experimental data on physical and mechanical properties of graphite has been carried out. The data was obtained by irradiation of samples in the research reactors and also by testing the samples cut out from graphite of the operating RBMK. It is possible to stand out three stages of degradation. The first stage is characterized by increasing of density, strength and other properties. It results from closing of oriented porosity (Mrozowski cracks), formation of radiation defects and secondary oriented porosity along basis planes. The second stage starts close to secondary swelling, which is caused by crack formation on boundaries between filler and binder, resulting from tensile stress along basis planes. The third stage is caused by crack formation on boundaries of single crystallites. The stresses, responsible for cracking, have been evaluated. The time, necessary to reach critical values of the stresses, has been also estimated. The model, allowing to make evaluations of the stresses and deformations, taking into account relaxation due to creep has been suggested. It is shown that physical and mechanical condition of graphite during irradiation is different from its condition after irradiation.

Key Words: Graphite, Influence of Radiation, Radiation Degradation, RBMK.

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Method to Enrich Reprocessed Uranium in a Cascade of Gas Centrifuges with Simultaneous Reducing of Content of Isotopes $^{232, 234, 236}U$

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It is shown that it is possible to enrich the reprocessed uranium with simultaneous dilution of $^{232, 234, 236}$U isotopes in a gas centrifuge cascade, which has three external feed flows (depleted uranium, low-enriched uranium and reprocessed uranium). Computational experiments were carried out for different content of ^{235}U isotope in low-enriched uranium, which is used as one of the cascade feed flows. It has been demonstrated that the selected combination of diluents can simultaneously reduce the value of the separation work and the consumption of natural uranium, not only in comparison to the previously used multi-flow cascades, but also to the standard cascade used for enrichment of natural uranium.

Key Words: Reprocessed Uranium, Isotope Separation, Separation Cascade.