Abstracts
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Algorithm for Solution of the Linear Cauchy Task for Large Systems of Ordinary Differential Equations Using Parallel Calculations

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This paper is devoted for the algorithm for solution of the linear Cauchy task for large systems of ordinary differential equations using parallel calculations. The algorithm for linear systems of first order equations was realized in *EDELWEISS* computer code. This code was developed especially for supercomputers that may use MPI technology to exchange of dates for parallel processes. The solution is presented as a row by orthogonal polynomials at [0, 1] segment. Features of this algorithm are simplicity, opportunity to get solution by parallel calculations and also possibility to get a solution for non-linear task by changing the operator using the solution from iteration process.

Key Words: Algorithm, Iteration Process, Program, System of Equations, Solution, Space, Vector.

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Program *LUCKY_TD* for Solution of Time Dependent Transport Equation Using Parallel Calculations

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This paper is devoted for the algorithm of solution of time dependent transport equation using parallel calculations. This algorithm was realized in *LUCKY_TD* code which was created especially for supercomputers with MPI technology of exchange dates for parallel processes.

Key Words: Algorithm, Transport Equation, P_mS_n Approximation, Iteration Process, Program, Supercomputer, Function, Solution, Space Domain.

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Effective Boundary Conditions for Neutron Flux Density at Axial Boundaries of the Core

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Analytical expressions for elements of the triangular matrix of effective conditions at boundary of the core with multizone reflector were derived in few group diffusion approximation. Verification of the developed procedure was carried out on the examples of calculations of fuel assembly of light water reactor with intermediate neutron spectrum.

Key Words: Neutron Flux, Neutron Current, Core, Axial Reflector, Boundary Condition.

УДК 621.039.55+621.039.5.001.24+621.039.566

Shaping of the Axial Power Distribution in the Core to Minimize the Vapor Volume Fraction at the Outlet of the VVER-1200 Fuel Assemblies

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The possibility of the vapor fraction decreasing at the VVER-1200 fuel assembly outlet is considered. The shaping of the axial power distribution in the reactor core is proposed to achieve this aim. The axial power field was shaped by axial re-distribution of gadolinium (burnable poison) concentration in Gd-containing fuel rods. The computer code NOSTRA was used for mathematical modeling of the VVER-1200 core.
Key Words: VVER-1200, Computer Simulation, Shaping of the Axial Power Distribution, Burnable Poison, True Volume Fraction of Vapor.

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Thermohydraulic Reactor Codes of 3rd Generation. Present Status and Problems of Implementation

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It is discussed the transition from the existing reactor thermal-hydraulic codes of 2ndgeneration, based on the non-equilibrium two-fluid flow model, to the reactor codes of 3rd generation – three-dimensional codes with taking into account the influence of turbulence (3D CFD). The transition to full application of the turbulent codes to evaluate the safety of existing nuclear power installations is expected only in the long term, after realizing the whole set of experimental research in the field of thermal physics and mathematical modeling of phase-change heat transfer medium. Such analysis is necessary fora clear understanding of the limits of the classical Navier – Stokes equations, taking into account the topology of real surfaces, on which a vapor phase is formed.
Key Words: Reactor Thermohydraulic Code, Computational Fluid Dynamics of Turbulent Multifluid Flows (3D CFD), Direct Numerical Solution (DNS) of the Navier–Stokes Equations, Verification and Validation of Code.

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Stochastic Model of the RBMK Graphite Stack Deformation

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A computational model of deformation of RBMK graphite stack caused by the formation of cracks in the graphite blocks is developed. Cracks orientation and block sizes dependence on the fluence are stochastic. Information about the measured values of the deflections of the individual channels is used to improve the accuracy of the calculations. Field of deflections is adjusted basing on the principle of maximum likelihood. An example of the optimization scheme of graphite stack repair is shown.
Key Words: RBMK, Graphite Stack, Stochastic Model, Fluence, Deflection of Channels, Principle of Maximum Likelihood, Repair Scheme.

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Registration Module of Expanded Set of Functionals in the Calculation of the Fragments and Full-Scale RBMK Cores by MCU Precision Code

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The description of the user registration module of MCU code [1] is stated. The module is designed for the registration of functionals in the 3D calculations of fragments and full-scale RBMK-1000 cores by Monte-Carlo method. The results which characterized the approbation of module for calculating of multicells and full-scale reactor cores by multiprocessor version of code MCU are stated.

Key Words: RBMK-1000, Computer Codes, Library of Constants, Precise Calculation, Code MCU, Registration, Functional, Multicell, Multiplication Factor, Monte-Carlo Method.

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The Integral System of Codes – Platform BREST

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The integral system of codes Platform BREST was developed for full-scale, precision, multiphysics calculations of BREST-OD-300 and BN-600 reactors. The codes for neutronic and thermohydraulic calculations of core are integrated into Platform BREST. The paper describes main functions and capabilities of the integral system of codes.

Key Words: BREST-OD-300 Reactor, BN-600 Reactor, Platform, FACT-BR, MCU-BR, TRIGEX, VEGA, MCNP, Cross-Verification, Neutronic Calculations, Cartogram, Project Codes.

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Calculation of the Time Dependent Neutron and Thermohydraulic Processes in the Lead-Cooled Reactor Core with the Control System

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The brief description of software package FACT-BREST [1, 2] for modeling non-stationary neutron-physical and thermohydraulic processes in reactor core with control system is provided in this paper. Results of calculation modeling transient of lead-cooled reactor normal operation are given.

Key Words: BREST-OD-300 Reactor, FACT-BREST, Neutronic Calculations, Thermohydraulic Calculations, Cartogram.

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Simulation of the First Fuel Loading and Initial Operating Cycles of MBIR

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The paper presents the results of a computational simulation for the first fuel loading and initial operating cycles of the MBIR reactor under design, as part of justifying its neutronics in the design and during the operating modes. It has been shown that monitoring of the subcriticality level at all first criticality stages requires a neutron source of the capacity 10^8 s^{-1} installed in the core center. Procedures have been proposed for the loading of fuel assemblies (FA) to compose the critical configuration of the core which is achieved with 63 loaded FAs when control rods inserted partially. The initial operating cycles of MBIR prior to the equilibrium operating mode being achieved were simulated based on the strategy of forming the core's design configuration and reaching the rated operating power as fast as possible. It has been shown that the total neutron flux density in experimental devices is close to the equilibrium values of $(2…5)\cdot10^{15}$ $cm^{-2}\cdot s^{-1}$ as from the early operating cycles.

Key Words: MBIR, Fast Reactor, Research Reactor, MCU, TRIGEX.

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Use of the PRISET-MBIR Code for Study of the Transients and Emergencies of the MBIR Reactor Installation

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The calculation code PRISET-MBIR for study of the dynamics, transients and emergency processes of the multipurpose fast reactor MBIR is discussed. The code block-diagram and basement design scheme of thermal-hydraulic model of the reactor have been described. It has been presented description of the three-dimensional kinetics model of the core. The results of the studies of emergencies are presented also.

Key Words: PRISET-MBIR Software Package, Multipurpose Fast Research Reactor, Three-Dimensional Model of Kinetics Reactor Core, Models of Circuits Technological Scheme, Transients and Emergencies, Safety.

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Using the Parallel Calculations on the Multiprocessor Computers to Accelerate the PRISET-MBIR Code Run

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The article demonstrates the possibility of parallel computing to accelerate calculation with PRISET-MBIR software package. Various mathematical methods for solving matrix equations of higher dimensions are discussed. It has been presented the parallelization of these methods using OpenMP directives. It has been shown that the application of these methods allows increasing speed of calculation. The results given in the article show the effectiveness of the use of parallel computing in the PRISET-MBIR software package and the possibility of their use in the calculation codes with three-dimensional models of neutron kinetics.

Key Words: Software Package, Parallelization, Mathematical Methods, Processing Speed, Acceleration of Calculation.

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Analysis of the Applicability Range of Thermodynamic Calculations at the Design of Nitride Fuel

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The paper presents the analysis of the applicability areas of the thermodynamic calculations in the development of nitride fuel. Characteristic values of parameters that directly affect the time of establishment of the equilibrium concentration were evaluated: rate of nuclides generation; characteristic times of the establishment of local equilibrium in the temperature range; the characteristic time of the stationary temperature profile; the characteristic time for the establishment of a quasi-stationary concentration field on the dimensions comparable to the size of the fuel pellet. It is shown that the equilibrium thermodynamic calculations can be used to evaluate the chemical and phase composition of the fuel, however, to describe transport processes condensed and in gaseous phases it is necessary to develop two-layer kinetic model. In the hot part, in the center of fuel, for determining the composition diffusive transport should be taken into account.
Key Words: Nitride Fuel, Thermodynamics, Fission Products.

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Some Thermodynamic Features of Nitride Uranium-Plutonium Fuel during Burnup

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In the paper the effect of carbon and oxygen impurities in the chemical and phase compositions of the nitride uranium-plutonium fuel during burnup is calculated using IVTANTERMO code. It is shown that with increasing burnup, the number of moles of UN decreases, and $UN_{1,466}$, $UN_{1,54}$ и $UN_{1,73}$ noticeably increases. The presence of oxygen and carbon impurities increases number of $UN_{1,466}$, $UN_{1,54}$ и $UN_{1,73}$ phases in the original fuel especially at relatively low temperatures. However, the presence of impurities leads to a drastic reduction of the free uranium amount in unburned fuel. Plutonium forms following compounds in the system: Pu, PuC, PuC_2, Pu_2C_3 and PuN. Plutonium and uranium carbides are produced in small quantities. Most of the plutonium is in the form of nitride PuN, and unbound Pu is present only in areas of low burnups and high temperatures.
Key Words: Nitride Fuel, Thermodynamics, Fission Products.

Seminar "Physics of Nuclear Reactors"

The seminar "Physics of Nuclear Reactors" is working.in the NRC "Kurchatov Institute" since 1999 under the direction of the head of the Nuclear Reactors Physics Department S. M. Zaritskiy.

By the time of this journal issue there were 147 seminar meetings, the theme of which is not limited by the fact stated in seminar title.

The speakers and participants of the seminar are the scientiests from NRC KI and other Institutions.

The information about the seminar is located on the site of NRC "Kurchatov Institute" (www.nrcki.ru), and is sending to the participants.

In 2014 there were 7 meetings of seminars (from 134 till 140). Information on 134-137 meetings was published in issue No.1-2 for 2014, information on 138 – 140 seminar meetings was published in issue No.4 for 2014.

This issue contains the information about 141 – 147 seminar meetings and abstracts of reports provided by speakers.