Abstracts
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Inelastic Scattering Effect on the Neutrons Transmission by a Single Crystal of Lead

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The results of the experiment and calculations aimed to studying the transmission of monochromatic neutrons by a single crystal of lead are presented. The temperature dependence of the total cross section obtained in the experiment is well described by calculation, taking into account one-, two- and three-phonon neutron scattering. An additional temperature-independent component of the total neutron interaction cross-section has been found. It is assumed that this component is associated with the presence in the studied lead sample of a small number of impurity atoms that strongly absorb neutrons. It is shown that in the case of absence of a neutron-absorbing impurity, a lead single-crystal can be used as a gamma radiation filter, without significant neutron loss.

Key Words: lead single crystal, neutron transmission, phonon spectrum, inelastic neutron scattering cross section, neutron absorption.

EDN: AOXTHQ

UDC 621.039.51...17, 539.125.523.4

Development of a Method for Taking into Account the Continuity of the Materials Density Distribution and Implementation to the KIR Code

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We consider a technique for taking into account the inhomogeneous density of a material when modeling neutron transport using the Monte Carlo method [1]. The inhomogeneity of the density is specified in the initial data based on piecewise continuous analytical functions of spatial coordinates. The approach involved is similar to the method of equalized cross-sections and it is implemented in the KIR calculation code [2, 3] as one of the possibilities. Despite some similarities with the delta-tracking method [4, 5] but the presented improved technique is a completely independent development. The obtained algorithm for taking into account the continuous density of the material is used in the usual zones of the geometric module NCG [6] with tracking the transition of particles through the boundary. The algorithm was tested by simple calculations of test models for WWER-type reactor cells with a sharp change in coolant density similar to systems with supercritical parameters of coolant, for example SCWR [7].

***Key Words*:** neutron transport equation, Monte Carlo method, precision calculations, KIR code, continuity of the density distribution in materials.

EDN: BVMCXX

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Algorithms for Calculating the Effective Parameters of the Point Kinetics Equations Based on the Monte Carlo Method, their Verification and Validation

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Simulating of transient modes and processes in nuclear reactors is carried out on the basis of point kinetics equations using effective functionals — the fraction of delayed neutrons β_{eff} and the prompt neutrons generation time Λ. The parameter β_{eff} also plays a key role in reactor measurements it is used in the inverted equations of point kinetics [1] and it is the main unit of reactivity measurement (dollar). Accurate calculation of the functionals of point kinetics equations in heterogeneous systems is a rather time-consuming task since it is necessary to solve adjoint transport equations to determine the neutron importance function. Especially calculations are difficult for cores with fuel containing a mixed nuclide composition of several main fissile elements or having a complex geometric structure that cannot be modeled by engineering calculation tools. In this case programs based on the Monte Carlo method are used in practice.

In this paper three algorithms for calculating the effective parameters of the point kinetics equations based on the Monte Carlo method implemented in the KIR code are considered. The algorithms were validated on a series of benchmark experiments and verified on the basis of the published results of the MCNP code and group test problems that were calculated using the discrete ordinate method.

Key Words: neutron transport equation, Monte Carlo method, precision calculations, KIR program, kinetic parameters, kinetic algorithms.

EDN: CADHKG

UDC 621.039.5:519.876.5

Neutron Cross Section Library Creation using the Monte Carlo Code

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Currently, the Russian Federation is working on the creation and commissioning of various NPP: VVER-1200, RITM-200, KLT-40, BREST-OD-300, BN-1200 and others. Engineering calculation codes are used to justify their safety and to calculate operation support, using group neutron cross section libraries. Therefore, the aim of this work is to create a universal methodology for preparing neutron cross sections group libraries for engineering codes. OpenMC code, which implements the Monte Carlo method, was chosen as the main research tool in this work. When constructing a neutron library, group neutron cross sections are approximated using a cubic burn up spline and approximated by a multidimensional polynomial using the stepwise regression method. The dependence of the neutron constants of fuel assemblies on the burn up depth, fuel temperature, coolant temperature and density and boric acid solution concentration (in the case of a VVER reactor) is approximated: fission, fission production, scattering, absorption cross sections, xenon and samarium absorption micro cross sections, delayed neutron and yields of fission products of xenon, iodine and promethium. The proposed methodology effectiveness is shown on the neutron cross sections library creating example, including several types of fuel assemblies, and modeling measurements performed at the 3rd unit of Kalinin NPP with VVER-1000 reactor. SKETCH-N code was used for neutron calculations. At the same time, calculations were carried out of critical states at the hot zero power (HZP), integral and differential characteristics of control groups at the HZP, temperature and density reactivity coefficients at the HZP, reactivity coefficients based on the concentration of boric acid at the HZP, energy release fields at power. The obtained results were compared with the measurements results at nuclear power plants and showed good agreement.

***Key Words*:** Monte Carlo method, neutron cross section library, cross section approximation, stepwise regression, cubic spline, uniformly distributed mesh.

EDN: DZBSAR

UDC 621.039.54, 621.039.516.22

Analytical Construction of Grid Diagrams for Burnup of Nuclear Fuel of Different Compositions in Water-Cooled Reactors

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The paper presents the possibility of analytical estimates of nuclear fuel burnup depending on its composition, thermal load, enrichment, periodicity and multiplicity of fuel reloading, discovered by the authors. The comparison of analytical and neutron-physical calculations of uranium fuel burnup for specific reactors shows that the maximum deviations range from –4% to +8.5% in a wide range of fuel cycle parameters (fuel enrichment 3.5—10%, burnup 8—100 MW·day/kg U, fuel reloading multiplicity 1—8, reactor campaign 250—800 days). One of the variants of visualization of fuel burnup dependence on the specified parameters, based on the construction of grid diagrams, is shown. As an illustration of the approach, a number of grid diagrams of burnup of various fuels differing in thermal load, including those compositions considered as promising tolerant fuels (UO_2, U_3Si_2, UN, U—Mo) are attached. According to calculations, U—Mo fuel due to its lower thermal load has the highest potential for use in prolonging the fuel campaign in case there are restrictions on fuel enrichment and burnup.

Key Words: nuclear fuel burnup depth, burnup grid diagram, enrichment, thermal load, reloading multiplicity, reactor campaign, accident tolerant fuel, analytical approximation.

EDN: HEXNIK

UDC 621.039.5

Two-Group Benchmark K3_1_T for Simulation of Control Rod Dropping in VVER-1000-type Reactor

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The test is proposed for cross verification of codes used to calculate spatial kinetics of VVER-type reactors. It is based on the model of the first fuel loading of unit 3 of Kalinin NPP with simplification of the structure of axial and radial reflectors and enlargement of groups of control rods (CRs). The test allows to simulate gradual lowering of both groups of CRs and two single CRs. Thirteen physical zones and nine types of fuel assemblies were used, including four types of CRs. Kinetic parameters were composed for the test along with two-group cross sections prepared using the HELIOS code. The presented set of input data is sufficient to perform independent cross-validation calculations of spatial kinetics. The results of calculations in the environment of the intelligent software system ShIPR are given. Motivation for the creation of the test were significant differences of the reactivity values obtained during the motion of the CRs in direct calculations K_{eff} and using the inverse solution of the equation of kinetics. The acceptability of the two-group approximation for spatial kinetics calculations is questioned.

Key Words: VVER-1000, spatial kinetics, ORUK, OR efficiency, reactivity, delayed neutrons, two-group diffusion approximation, 3D triangular geometry, ShIPR, HELIOS.

EDN: INKISG

UDC 629.039.58

Beyond Design Basis Accident with Complete Blackout of the RBMK

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The accident is examined using the three-dimensional STEPAN-T code, developed specifically for simulating an accident with a complete blackout of the RBMK. Also, for the option with water supply to the control system circuit, data obtained using the KLADKA-2 code are presented, in which the core is described in maximum detail. The temperature time dependences in the core and surrounding metal structures are given. Quantitative estimates of the hydrogen generated during the accident are shown. The possibility of re-criticality and the release of radioactive substances during the accident are discussed. Possible measures to mitigate the consequences of the accident are given.

Key Words: severe accident, temperature dependences, reactivity effects, hydrogen formation, radionuclide release, RBMK, computer program.

EDN: JJTDZR

UDC 621.039:519.23

Uncertainty Evaluation in Inventory Calculations of the Uranium-Plutonium Nitride Fuel Samples Irradiated in BN-600

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The results of analysis of the impact of nuclear data and design parameters uncertainties on the calculation of the experiment on irradiation of the uranium-plutonium fuel in the experimental assembly KETWS-1 in the fast reactor BN-600 are presented. We evaluated uncertainty of the inventory calculations, caused by uncertainties of the major design parameters: initial composition of the irradiated samples, decay time before radiochemical analysis carrying, neutron flux. Uncertainties in nuclear data (one-group neutron cross-sections, branching ratios and half-life) are calculated based on the actual versions of the evaluated data libraries. Uncertainty evaluation of the nuclear data and design parameters uncertainties is conducted with deterministic and stochastic methods, implemented in SUN and NUCLEX codes. It is shown that the difference between the calculation and experimental results is explained by nuclear data and design parameters uncertainties for almost all nuclides (exception is ^{243}Am).

Key Words: uncertainty evaluation, sensitivity analysis, nuclear data, KETWS-1, isotopic kinetics, nuclear data uncertainty, design parameters.

EDN: KDOBOH

UDC 621.039.4

Multiphysics Analysis of Molten Salt Reactors

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Calculation complex (CC) MULTIMSR, created at the National Research Center "Kurchatov Institute" for modeling a wide-purpose MSR's, consists of several software tools (ST) built in a single program, with the ability of information exchange between them. CC MULTIMSR uses ST SERPENT to perform neutron physics calculations using the Monte Carlo method. In addition, the CC includes a number of ST's designed to solve thermal hydraulics, neutron transfer using diffusion approximation and auxiliary problems, implemented within the framework of the library of computational operations OpenFOAM, a database of physical properties of fuel salts, intermediate coolants and structural materials, as well as a unifying subroutine necessary for the coordinated work of the elements of the CC.

A description and a block diagram of the CC are given, as well as the results of an analysis of the influence of the number of groups used in the diffusion approximation on the correctness of the results obtained, as applied to the MSR burner (MSR-B) of transuranium elements from spent nuclear fuel of VVER-1000/1200 reactors with Li,Be,A_n/F_m fuel salt. Multiphysics modeling of the development of an accident in the event of loss of forced circulation of fuel salt and failure of protection operation in MSR-B with a thermal power of 2.4 GW with a core of cavity type was carried out. Some conclusions about the stability of the reactor in a number of transient processes are presented.

Key Words: molten salt nuclear reactor, multiphysical modeling, calculation methods, melts of metal fluoride salts, circulating liquid fuel.

EDN: LGPNPV

UDC 621.039.51...17

Nuclide Kinetics in the Thorium Blanket of a Thermonuclear Neutron Source

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The article discusses the processes of transmutation of nuclides in the blanket of a thermonuclear reactor containing metal ^{232}Th in the form of a ball filling. The geometric and physical characteristics of the DEMO-TIN thermonuclear installation (Thermonuclear Neutron Source) with an installed capacity of 100 MW were used for computational studies. Based on the developed model, neutron fluxes distributed over the blanket volume were obtained, which were used in solving the problem of nuclide kinetics. The results of computational studies of the accumulation of fissile material and radioactive fission products in the DEMO-TIN blanket are presented, depending on both the irradiation time and the distance from the center of the installation. The results of calculations of energy consumption in the blanket, when cooling the irradiated ^{232}Th, are also presented. The presented results show the potential for effective accumulation of ^{233}U with significantly lower energy emissions in irradiated fuel compared to traditional schemes for the accumulation of fissile nuclides in nuclear reactors.

Key Words: evaluated nuclear data, thermonuclear reactor, blanket, energy release, thermonuclear neutron source, nuclide kinetics, activity.

EDN: LNDXRX