Abstracts
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Some Methods for Calculation of Perturbations in Nuclear Reactors

B.D. Abramov
FSUE “SSC RF-IPPE”, 1, Bondarenko Sq., Obninsk, Kaluga Region, 249033

Some methods for calculation of local perturbations of neutrons fields and effects of reactivity accompanying them in nuclear reactors are developed. Theorems of existence and uniqueness of the solutions are established, schemes of numerical realization of offered methods are brought.
Key Words: Neutron Transport Theory, Perturbations Theory, Reactivity Effects in Nuclear Reactors.

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The Monte Carlo Estimation of the Effect of Uncertainties in the Input Data for the Transport Equation Solving by the MCU Code

D.S. Oleynik
NRC “Kurchatov Institute”, 1, Kurchatov Sq., Moscow, 123182

Direct taking into account the initial data uncertainty is realized in the new version of tally module of the MCU code. This approach is recommended by international standard “A Guide to Expression of Uncertainty in Measurement” (ISO 13005). The new module allows calculating the effect of uncertainties in input quantities for neutron characteristics of core. These uncertainties are usually caused by technological tolerances. Developed code is adapted to parallel computing that drastically decreases calculation time. Testing is performed by means of a simplified model of the Godiva bare-sphere critical experiment and infinite lattices of various fuel assemblies (VVER-440, VVER-1000 and VVER-1200). The results of calculation is compared with published MCNP5 ones (for Godiva experiment) and with RADAR engineering code ones. A good agreement is achieved.
Key Words: MCU Code, Monte Carlo Method, Uncertainty.

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Analysis of Research Reactor Core Geometrical Model Parameters for the Calculation Using Monte Carlo Code

A.I. Radaev, M.V. Schurovskaya
National Research Nuclear University “MEPhI”, 31, Kashirskoe Sh., Moscow, 115409.

The choice of spatial nodalization for the calculation of power density and burnup distribution in a research reactor core with IRT-3M – and VVR-KN – type fuel assemblies using Monte Carlo code is described. The influence of the spatial nodalization on the basic neutronic characteristics calculation results and the time of calculation is examined.
Key Words: Research Reactor, Power Density Distribution, Burnup, Monte Carlo Code.

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Experimental Investigation of Neutron-Physical Characteristics of the IR-8 Reactor to Confirm the Results of Calculations by MCU-PTR Code

A.V. Surkov, V.N. Kochkin, Yu.E. Pesnya, V.A. Nasonov, V.I. Vihrov, D.Yu. Erak
NRC "Kurchatov Institute", 1, Kurchatov Sq., Moscow, 123182

A comparison of the measured and calculated of neutron-physical characteristics (fast neutron flux and the fission rate of ^{235}U) in the core and reflector of IR-8 reactor is presented. The irradiation devices equipped with neutron activation detectors were prepared. Determination of fast neutron flux was performed using reactions ^{54}Fe (n, p) and ^{58}Ni (n, p). The ^{235}U fission rate was measured using uranium dioxide with 10 % enrichment by ^{235}U. Determination of detectors specific activity was carried out by measuring the intensity of characteristic gamma peaks using ORTEC gamma-spectrometer. Neutron fields in the core and reflector of IR-8 reactor were calculated using the MCU-PTR code.

Key Words: IR-8 Reactor, Neutron Flux, Neutron Activation Measurement, MCU-PTR Code.

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The Adjustment of the Power Distribution in the VVER-1000 Core at the Very Low Power Levels Using the SPND Records

V.M. Chapaev, V.A. Khvatov
JSC Rosenergoatom Concern Affiliate – Kalinin Nuclear Power Plant, Udomlya, Tver' Region, 171841
A.Yu. Kurchenkov, N.V. Mil'to
NRC "Kurchatov Institute", 1, Kurchatov Sq., Moscow, 123182

The records of the self powered neutron detectors (SPND) at the very low power levels of the Kalinin-4 NPP VVER-1000 reactor are presented. It is shown that the SPND records are stable and representative at these conditions. The SPND records are compared with the results of the core simulation using the IR code.

Key Words: SPND, In-Core Diagnostic System.

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New Method of the Coolant Mixing Studies at the Operating VVER-1000 Units

G.L. Ponomarenko, M.A. Bykov, V.Ja. Berkovich
JSC OKB "GIDROPRESS", 21, Ordzhonikidze St., Podol'sk, Moscow region, 142103

Paper describes an unique method of the coolant mixing study. Method is characterized by the use of an emergency boron insertion system for creating the nonuniform distribution of indicator (tracer). Method showed high reliability, effectiveness, availability and flexibility. Present work supports the conclusions of the paper [1] and contains the more detailed additional information and results according to the mixing indices, and also to the dissipation of the temperature indicator in different sections of circulation loop. Some results of post-test modeling by the code KORSAR/GP are presented too. Spatial and temporal detailing relate also to the dynamics of appearance, stabilization, propagation and disappearance of boron and boron-free "spots" in the core with start-up and trip of the emergency boron insertion pumps. Appropriate patent on invention of new method [2] was gained.

Key Words: Neutron Detectors, Thermal Detectors, Tracer Dissipation, Measurements and Treatment of Weak Signals, Mixing Factors, Swirl of Loop Flows.

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Results of the Verification of the Computer Codes Used for Analysis of the BN-1200 Reactor Core Neutronics

S.B. Belov, A.V. Kiselyov, E.V. Marova, M.R. Farakshin, V.M. Frolov
JSC “Afrikantov OKBM”, 15, Burnakovskiy Proezd, Nizhniy Novgorod, 603074
I.V. Malysheva, A.A. Peregudov, M.Yu. Semyonov, V.Yu. Stogov, A.M. Tsibulya
FSUE “SSC RF-IPPE”, 1, Bondarenko Sq., Obninsk, Kaluga Region, 249033
P.N. Alekseev, V.F. Boyarinov, M.N. Zizin, V.A. Nevinitsa, A.V. Timoshinov, P.A. Fomichenko
NRC “Kurchatov Institute”,1, Kurchatov Sq., Moscow, 123182

The paper presents the results of the BN-1200 benchmark neutronic calculations by the engineering codes and codes based on the Monte Carlo method. A full-scale test model of the reactor core is used for calculations. The benchmark model reflects basic features of the BN-1200 reactor core and facilitates coordinated calculations with homogeneous and heterogeneous presentation of subassemblies with nitride and MOX fuel.
Key Words: BN type Reactor, Benchmark Calculations, Neutronic Characteristics, Multiplication Factor, Reactivity Effects, Control Rod Worth, Deviations.

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Results of the Benchmark Calculations of the Neutronic Processes Caused by Movement of Single Control Rods in a BN-1200 Type Reactor

S.B. Belov, A.V. Kiselyov, E.V. Marova, M.R. Farakshin
JSC “Afrikantov OKBM”, 15, Burnakovskiy Proezd, Nizhniy Novgorod, 603074
M.N. Zizin, P.A. Fomichenko
NRC “Kurchatov Institute”, 1, Kurchatov Sq., Moscow, 123182
K.F. Raskach, M.Yu. Semyonov
FSUE “SSC RF-IPPE”, 1, Bondarenko Sq., Obninsk, Kaluga Region, 249033

The paper presents the results of the benchmark calculations of transients caused by movement of single control rods in a BN-1200 type core. Calculations were performed by RADAR3D and ShIPR codes with the use of a full-scale test core model. This benchmark model reflects the basic design features of the BN-1200 core. Additionally, steady-state calculations were carried out by the JARFR code.
Key Words: BN Type Reactor, Neutron-Physics Characteristics, Space-Dependent Neutronic Kinetics, Single Rod Movement, Neutron Flux Perturbation, Neutron Flux Measurement, Benchmark Calculations.

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The Radiation Characteristics of the Transport Packages with the Vitrified High-Level Waste

S.A. Bogatov
JSC “VNIPIpromtechnologii” 33, Kashirskoe Sh., Moscow, 115409
E.F. Mitenkova, N.V. Novikov
Nuclear Safety Institute of the Russian Academy of Sciences, 52, Bol'shaya Tul'skaya St., Moscow, 115191

The calculation method of neutron yield in (α, *n*)-reaction for a homogeneous material of arbitrary composition is represented. It is shown that the use of the code ORIGEN 2 without the real elemental composition of vitrified high-level waste leads to significant underestimation of neutron yield in the (α, *n*)-reaction. For vitri-

fied high-level waste and spent nuclear fuel from VVER the neutron fluxes are analyzed. The thickness of the protective materials for transfer cask and transport containers with the vitrified high-level waste are estimated.
Key Words: Spent Nuclear Fuel, High-Level Radioactive Wastes, Transport Cask, Radiation Safety.

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Temperature Calculation in the Dewatered RBMK Fuel Storage

A.V. Krayushkin, I.N. Geraskin, G.B. Davydova, L.N. Zaharova
NRC “Kurchatov Institute, 1, Kurchatov Sq., Moscow, 123182

For the calculation of the temperature in the dewatered fuel storage during the beyond design accident it is necessary to solve 3D task of the heat exchange of fuel assemblies with each other and constructions surrounding them – the wall, the bottom and the upper ceiling of the storage. The main mechanism is the heat exchange by radiation. The description of the calculation model used in investigation is given. Receiving of the coupling coefficients of separate elements with each other is carried out by means of the MCNP code. Some results of temperatures calculations are given.
Key Words: Temperature Calculation, Dewatered Fuel Storage, Heat Exchange by Radiation.

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Development of the Basic Concepts of the Methodology for Calculation of Fast Reactor Fuel Elements Performance in the Design-Basis Accident Conditions

A.A. Proshkin, A.V. D′yakov, A.S. Stepanov
NRC “Kurchatov Institute”, 1, Kurchatov Sq., Moscow, 123182

There are discussed the current situation in the development of computational methods for evaluation of the fuel elements reliability and availability of the experimental data on the material properties of fast reactor fuel elements under the design-basis accident conditions. The statement of the problem is considered for determination of the thermo-mechanical condition of fuel elements in the course of the accident. Recommendations are provided regarding experimental investigations required for obtaining data on fuel element reliability criteria.
Key Words: Fuel Element, Accident Conditions, Fuel Element Performance in Design-Basis Accident Conditions.

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The Possibility of the Nuclear Power Fuel Cycle with Minimal Radioactivity

E.A. Andrianova, V.D. Davidenko, V.F. Tsibul'skiy
NRC “Kurchatov Institute”, 1, Kurchatov Sq., Moscow, 123182

Practical implementation of the closed fuel cycle assumes two main tasks solution. The first task is creation of the ecologically affordable conditions of the fuel cycle realization. The second task is creation of effective and economically appropriate conditions of the raw isotopes involving into the fuel cycle. The hardest problem is a creation of reliable technologies of the high-level radioactivity spent fuel’s management.
Key Words: Hybrid Thermonuclear Systems, Thermal Nuclear Reactors, Fuel Cycle, Burnup, Disposal of Spent Fuel, an Acceptable Level of Radioactivity.

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The Multiple Recycle of the REMIX-Fuel at work the VVER-1000 Operation in the Closed Fuel Cycle

P.N. Alekseev, E.A. Bobrov, A.V. Chibinyaev, P.S. Teplov, A.A. Dudnikov
NRC "Kurchatov Institute", 1, Kurchatov Sq., Moscow, 123182

The basic features of the VVER-1000 fuel loadings with a new variant of the REMIX-fuel during multiple recycle in the closed nuclear fuel cycle are described. Such fuel composition is produced on a basis of an uranium and plutonium mixture allocated at processing the spent fuel after the irradiation in the VVER-1000 core, depleted uranium and fission material such as: ^{235}U as a part of high-enriched uranium from the warheads superfluous for defense or ^{233}U, accumulated in thorium blankets of fusion (electronuclear) neutron sources or fast reactors. At manufacturing of such fuel it is not planned to use a natural uranium in addition. While including parts of VVER-1000 reactors in the closed fuel cycle on the basis of the REMIX-technology the consumption of natural uranium essentially decreases and there is no degradation of isotope composition of plutonium and change of characteristics of reactor safety from recycle to recycle.

Key Words: REMIX-Fuel, VVER-1000, High Enriched Uranium, Degradation of the Fuel Isotopic Composition, the Spent Fuel.

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Heavy Oil + Small Nuclear Power Plants: Diversification Factor of Nuclear Power Risks and Harmonization of Fuel-Energy Complex

S.A. Subbotin, T.D. Schepetina
NRC "Kurchatov Institute", 1, Kurchatov Sq., Moscow, 123182
D.Yu. Chumak
State University of Management, 99, Ryazanskiy Pr., Moscow, 109542

The concept of synergia interactions of nuclear and extracting branches on an example of extraction and processing of heavy oil is presented. The epoch of readily available and cheap organic power resources comes to end; power inputs grow and return decreases. In the Russian environmental conditions and low oil extraction factor from chinks, the role of power supply crafts grows, especially of thermal methods for intensification of layers raises. It is offered to consider basic possibility of association of an exhausted oil craft, on which there are still at least 65…75 % of heavy oil, with the small power NPP in complex extracting-reprocessing plant. At such plant energy the NPP will allow to carry out also both primary, and deep processing of heavy oil at preservation and use of all household, transport and industrial infrastructure, essentially having prolonged service life of existing deposits.

Key Words: Small Nuclear Power Plants, Synergy, Heavy Oil, Fuel-Energy Complex, Nuclear Power-Technological Industrial Complex, Risks.

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 140 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 2012 there were 13 meetings of seminars, information on them was published in journal issue No.1 for 2013.

In 2013 there were 11 meetings of seminars (from 123 till 133). Information on 123-127 meetings was published in issue No.2 for 2013, information on 128-133 meetings – in issue number 4 for 2013.

Information on 134-137 meetings was published in issue No.1-2 for 2014.

This issue contains the information about 138 – 140 seminar meetings and abstracts of reports provided by speakers.