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On the Formation of the Asymptotic Spectrum of Delayed Neutron Emitters at Measuring Scram System Effectiveness of VVER-1000

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This article is devoted to illustrate the process of establishing the asymptotic distribution of the neutron flux density in the reactor system after the introduction of various negative reactivity. The impact of two points after the introduction of reactivity is evaluated: 1) non-uniformity of the core properties of the disturbance on the one hand, and 2) a sharp decrease in the density of prompt neutrons, which prevents the appearance of new delayed neutron emitters distributed in accordance with the "new" prompt neutrons distribution on the other hand. The calculations showed that the errors of measuring the scram system effectiveness by ORUK method the fact that after the introduction of negative reactivity sources of prompt and delayed neutrons have a different spatial distribution. In case of large negative reactivity this distinction is retained to the moment when the system still has neutrons, which can be measured.
Key Words: VVER-1000, ORUK, Reactor Scram System Effectiveness, Delayed Neutrons, Two-Group Diffusion Approximation, 3D Triangular Geometry, ShIPR.

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Results of the GT-MHR 3D Model Neutronic Calculation by the MCU-HTR Code

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Paper presents the development of a neutronic model of the high temperature gas cooled reactor GT-MHR for MCU-HTR Monte-Carlo code, as well as results of benchmark calculations including reactivity margin, core power distribution and efficiency of control systems. A 3D model of GT-MHR developed to perform the calculations includes specific design features, parameters of the fuel composition, zoning of fuel and burnable poison loading, axial displacement of fuel blocks and their refueling scheme assumed in the actual reactor design. Performed work is a part of planned research investigations aimed at verification of codes, nuclear data libraries and approximations.
Key Words: Neutronic, 3D Model, GT-MHR, MCU-HTR Code.

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On the Possibility of the Uranium-Beryllium Oxide Fuel Using in VVER

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The possibility of the enhanced thermal conductivity (UO_2-BeO) composite fuel use in VVER-type reactors is considered with respect to thermal-physical fuel properties. The results of the estimation of neutron-physical characteristics of the VVER-type fuel assemblies with the (UO_2-BeO) composite fuel are presented.

Key Words: VVER-type Reactor, Enhanced Thermal Conductivity (UO_2-BeO) Composite Fuel, Fuel Pellets.

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Evolution of Pores in the Fuel Cladding of the Electrogenerating Channel

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The results of reactor tests of carbonitride fuel with monocrystalline cladding of molybdenum-based alloy can be used for support of operational reliability of the fuel elements for the drafting megawatt nuclear power plant for space vehicle. These tests was held previously in the experimental facility Ya-82 for 8 300 hours at temperature of ~ 1500 $^{\circ}$C. On the raster image of the sample cladding surface we found that pores decorated the boundaries between layers. This result is explained in this paper using the theory of coalescence. The mechanisms of pores evolution at the parameters of experiment on the Ya-82 are considered. An explanation is done for the effect of pore decorating of boundaries between layers in the sample cladding of electro generation channel during the reactor tests. The dependence between the average pore radius and the time of the experiment was found. Calculation of mean pore size at the experimental conditions yields a value of ~ 2 mm, which is consistent with experimental data. Calculation study of swelling of the cladding material in the process of irradiation was made. Forward-looking assessment of the behavior of the pore system and the swelling of the cladding material for a nuclear power plant of megawatt class was made.

Key Words: Electro Generation Channel, Pores, Coalescence, Swelling of the Material under Irradiation.

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The VVER-1000 Maneuvering Area at the Operation in the Daily Load Chart

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The possibility of the reactor VVER-1000 operation in the mode of maneuvering capacity at the most conservative approximations with regard to the management of the in-core power distribution is analyzed.

Key Words: VVER-1000, Maneuvering, In-core Power Distribution.

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The Calculation-Experimental Validation of the Fuel Assemblies Heat Transfer Model, Used in TIGR-1 Code, in Accidents with Heat Transfer Crisis

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It is estimated the conservatism of fuel rod cladding temperature model, used in TIGR-1 code, during transition from pre-CHF (Critical Heat Flux) to post-CHF area and back. The experimental data received for fuel assembly models in modes with CHF are used.
Key Words: Heat Transfer Crisis**,** Critical Heat Flux, Advanced Boiling Curve, Peaking Cladding Temperature.

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Uncertainty Analysis of Main Steam Line Break Calculations for AES-2006 Power Unit by KORSAR/GP Code with LINQUAD Program Using

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The paper deals with uncertainty analysis of main steam line break transient for AES-2006 unit (project V-392M). The calculations are provided by the LINQUAD program based on response surface construction. Method concerned is compared with widely used GRS method. Calculated safety criteria realistic boundary values are analyzed taking into account the uncertainties of most essential physical parameters.
Key Words: Uncertainty Analysis, Response Surface Method, Transient Emergency Process, Steam Line Break, Calculation Model Parameters, GRS Method, LINQUAD Program.

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Up-to-date Designs of Passive Heat Removal Systems of Water Cooled Reactors

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Review of the passive heat removal systems intended for the core cooling in the case of the accident is given. On the basis of common features the classification of passive residual heat removal systems is created. Fulfilled review allowed showing variety of design and scheme decisions used at the NPP passive heat removal systems.
Key Words: Core, Residual Heat, Passive Systems.

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Reactor Facilities with Horizontal Steam Generators

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The analysis of the application of horizontal steam generators in reactor facilities with lead and lead-bismuth coolantsispresented.
Key Words: Fast Reactor, Horizontal Steam Generator, Lead Coolant, Intercontour Leak of a Steam Generator.

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The Comparative Analysis of Models of Nuclear Power Development in Russia Using CYCLE and MESSAGE Codes

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The computations are carried out on simplified models and on scenario of nuclear power development in Russia using MESSAGE and CYCLE codes. It was analyzed the modeling accuracy of the nuclear materials balance conducted in the total energy optimization code MESSAGE. In the scenarios simulated operation of nuclear power on fast, thermal reactors and related front-end and back-end nuclear fuel cycle infrastructure. The average characteristics of the CYCLE calibration code were used as the input data for the MESSAGE. Code CYCLE implements simulation of physical processes at different stages of the fuel cycle.
Key Words: MESSAGE, CYCLE, Scenario Modeling, Calculations Analysis, Nuclear Power Development Scenario.

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Paradoxes of Responsibilities Concerning NPP Safety

Letter to the editors

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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 127 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 seminar, information on them was published in journal issue No.1 for 2013.

This issue contains the information about 123 – 127 seminar sessions and abstracts of reports provided by speakers