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Experimental Determination of the Effective Resonance Capture Integrals of the ^{238}U, ^{158}Gd in Urania-Gadolinia Isolated Rods

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The technique and results of experimental determination of the ^{238}U, ^{158}Gd effective resonance capture integrals of isolated (UO_2-Gd_2O_3)-rods are presented. Measurements were performed by an activation method. Irradiation of (UO2-Gd2O3)-rods carried out in the neutron field with the spectrum of slowing-down neutrons (the Fermi spectrum), which is formed in the center of the core of a research reactor Ф-1 NRC “Kurchatov Institute”.
Key Words: VVER-type Reactor, Urania-Gadolinia Fuel, Fuel Pellets, ^{238}U Neutron Resonance Capture.

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Study into Peculiarities of Experiments with Pulsed Neutron Source in RBMK Spent Fuel Storage Facility

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The paper presents the results of calculations which were conducted for the adaptation of the pulse technique to Spent Fuel Storage Facility (SFSF) conditions of the Leningrad NPP. The developed analytical experimental technique for monitoring the subcriticality of LNPP’s storage facility is described.
Key Words: Leningrad NPP, SFSF, Storage Pool, SFA, Experiments with Pulsed Neutron Source, Neutron Flux Decay Decrement, Analytical Experimental Technique for Monitoring the Criticality.

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Calculation-Experimental Technique for Monitoring a Criticality in the Spent Fuel Storage Facility of the Leningrad NPP

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This paper presents the analytical experimental technique of monitoring the criticality in RBMK Spent Fuel Storage Facility (SFSF) of Leningrad NPP using the SAPFIR_95&RC_SFSF code package and facility for measuring the neutron flux decay decrement. The results of practical implementation of this technique are given in the paper.
Key Words: Leningrad NPP, SFSF, Storage Pool, SFA, Experiments with Pulsed Neutron Source, Neutron Flux Decay Decrement, Analytical Experimental Technique for Monitoring the Criticality.

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New Benchmark for Cross-Verification of the Deterministic Time-Dependent Codes for Neutron Transport Calculations without Spatial Homogenization

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The new space-time neutron kinetics benchmark for cross-verification of the deterministic time-dependent codes solving the time-dependent neutron transport equation without spatial homogenization was developed. The well-known stationary benchmark C5G7 was chosen as the basis for benchmark. Kinetics parameters for new benchmark were calculated and given in this paper. The proposed benchmark was calculated by SUHAM-TD code, which realizes the surface harmonic method (SHM). Authors hope to attract the attention of other researchers to participate in calculations of the proposed benchmark.
Key Words: Neutron Transients, Space-Time Benchmark C5G7-TD, SHM, SUHAM-TD Code.

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The Dependence of the Accuracy of the Calculation of the Mmultiplication Factor of the Critical Assembly Astra on Characteristics Tolerances of the Spherical Fuel Elements with Microfuel

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Microfuel is the fuel of double heterogeneity. One of uncertainties, complicating its modeling, – the parameters tolerances of spherical fuel elements, such as the diameter of the grains of fissile material in a graphite matrix. This paper presents the results of calculations on the MCU program, performed in order to study this problem on an example of the parameters of the spherical fuel element of the critical assembly Astra. Results show that under these parameters tolerances should not have major influence on the accuracy of calculation of the multiplication factor.
Key Words: Microfuel, Monte Carlo, Critical Assembly Astra, MCU, Characteristics Tolerances of the Fuel Elements, the Multiplication Factor.

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Effective Heat-Transfer Coefficient in a Flat Parallel-Plates Duct with Inhomogeneous Heating

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We presented a justification of a simple, physically clear, method of calculation for heat transfer in a flat parallel-plate duct with inhomogeneous heating on the basis of experimental data [1]. We gave an explanation of the relationship, found in the classic experiment, for the effective heat-transfer coefficient of the caliber number of the heated area. We tested the correlation relationship used for the closure of the Navier – Stokes equation in the code complex Ansys CFX. Based on the results of the study, we recommend using the SST model for the tasks of asymmetrical heating of the parallel-plate ducts.
Key Words: Asymmetrical Heating of the Parallel-Plate Duct, Turbulent Streams, Modeling and Calculation, CFD, SST.

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Statistical Analysis of Parameters of the Reactor Plant with the Direct Closed Gas-Turbine Cycle

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Presented is a description of the mathematical model for the closed gas-turbine cycle integrated with a high-temperature gas-cooled reactor (HTGR). A methodology is developed for Monte Carlo calculations of the static parameters of the reactor plant with the gas-turbine cycle with taking into consideration the range of possible deviations in characteristics of the main equipment and systems from their nominal values. A statistical analysis is carried out for probabilistic distribution laws of reactor plant parameters in the power operation range. The analysis is essential for preparing technical requirements for main components of equipment and for developing control algorithms that guarantee variation of parameters in the specified range.
Key Words: HTGR, Direct Closed Gas-Turbine Cycle, Monte Carlo Method.

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Minimum Calculation Error of Gaseous Fission Products Accumulation in the Metal Fuel Irradiated in Fast Neutron Spectrum

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The comparative experimental and calculated results of post-irradiation fuel assembly (FA) with experimental U-Pu-Zr fuel (eFA) in BOR-60 reactor are presented in the paper. Based on the analysis of the calculated and experimental values for eFA with burnout 9,7 % h.a. the errors, caused by individual calculation components, are given. The lower bound calculation errors are presented for gaseous fission products. Calculations are executed using MONTEBURNS−MCNP5−ORIGEN2 codes and nuclear data libraries being compiled from ENDF/B-VII.0, JEFF 3.1 and files with more detailed energy grids for fission yield libraries compilation that is especially important in accurate burn-up calculations of fuel, irradiated in a fast spectrum.
Key Words: Burnout, FA, Metal U-Pu-Zr Fuel, BOR-60.

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The Calculations of Parameters of the VVER-1000 Fuel Rod High-Temperature Testing in the MIR Reactor the Simulated Using the MUZA Code

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Presented are the data of post-test computations of parameters of the VVER-1000 fuel rod high-temperature testing in the MIR reactor using the MUZA code. The parameters were computed for the fuel rod testing at the second and third stages of the maximum design-basis LOCA. During the test the temperature on the fuel rod claddings achieves the value up to 1 100 ^{o}C and there is a large portion of thermal radiation in the overall heat transfer. The data show that the peak cladding temperature calculated values are close to the measured values.
*Key Words***:** Computations, Test, Fuel Rod VVER-1000, Research Reactor MIR, High-Temperature Testing.

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Influence of the Degree of MOX Fuel Pyroelectrochemical Purification on the BN Neutronics and Radiation Characteristics of Fuel Assemblies

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Influence of degree of Pu purification from minor actinides (MA) on the basic physical characteristics of BN reactor and fuel assemblies handling is considered. Pu extracted from the MOX fuel irradiated in this reactor and reprocessed by pyroelectrochemical method is used for fabrication of the fuel assemblies to be loaded to the reactor core. Analysis has shown that, from the standpoint of preservation of such important characteristics of BN-800 type reactor as criticality, sodium void reactivity effect, Doppler effect, and control rods worth, there is no need for MA removal from reprocessed MOX fuel to be used for the reactor core makeup. Effect of additional heat generated by fission products and MA, as well as neutron radiation by MA on the fuel assemblies handling is also insignificant. Only strong γ-radiation of such fuel assemblies would require measures for personnel protection. Recommendations on such protection measures are given.

Key Words: Sodium Cooled Fast Reactor, MOX Fuel, Pyroelectrochemical Reprocessing, MA, Physical Characteristics, Neutron and Gamma Radiation.

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On Stochastic Theory of Neutron Transport in Reactor Linear Stochastic Distributed Model Equations

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We derive the forward and backward time-dependent linear stochastic equations for probability density of the integer number of neutrons and delayed neutron precursors in distributed model of nuclear reactor.

Key Words: Time-Dependent Markov Process, Stochastic Equations, Nuclear Reactor, Fluctuations.