Abstracts
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To the Issue of the Pulsed Fast Reactor "NEPTUN" Dynamics — Comparison of the Calculation Models

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In the paper generation of a power pulse in periodically pulsed reactor is considered with account for fast reactivity feedback. Calculation method includes coupled neutronic-thermoelastic equations based on one-point kinetics of the reactor and on two models of elastic behavior of reactor fuel rods: approximation of a reactor core as one-dimensional oscillator, and more realistic accounting for elastic waves propagation through a single rod. It is shown that reasonably chosen parameters of the oscillator model gives results close to that of the wave model.
Key Words: pulsed neutron source, neptunium-237, nuclear reactor, kinetics, feedback of reactivity, subcriticality, nuclear safety.

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The Development of Reduced 3D Models for Neutron-Physical Calculations of ASTRA Critical Facility

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The paper presents a technique of the development of reduced 3D effective computational models for calculations of neutron-physical parameters of the ASTRA critical facility (intended for studies of physics of high-temperature reactors) with the use of numerical approach. The effective (extrapolated) height of the critical assembly model with significant difference between the heights of inner and lateral reflectors is determined based on experimental data on axial distributions of ^{235}U fission rates. Due to the presence of the bottom reflector and the absence of the top reflector in some assembly configurations the traditional approach based on 2D models with an effective buckling value is not applicable. Instead, we developed an approach based on the 3D effective model with the extrapolated height. Several examples of implementation of these models confirmed their successful applicability to computational analysis of experimental data on control rods mockup effectiveness obtained at the ASTRA critical facility.
Key Words**:** HTGR, ASTRA critical facility, axial fission rates reaction distribution, buckling, 3D model.

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The Use of the UNK Complex in Cell Calculations with Heterogeneous Distribution of Fuel along the Fuel Rod

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There are described the algorithms of the code for calculating the probability matrices of the first-flight neutron collisions (FFCP) in three-dimensional geometry and the code for calculating temperature field in a cell in *R*—*Z* geometry. These codes are included in the UNK complex. The results of the work of these programs are presented at calculating the change of the nuclide composition and the neutron multiplication coefficient during the burnout at homogeneous and heterogeneous distribution of fuel along the fuel rod.

Key Words: neutron, FFCP, heterogeneous distribution of fuel, nuclide composition, thermal conductivity, *R*—*Z* geometry.

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Effect of a Hybrid Fusion Plant Operating in a Fission and Fusion Reactor System on the Fuel Cycle

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The paper suggests and discusses the prospects of an alternative way to close fuel cycle, excluding the reprocessing of high-level radioactive spent fuel. The essence is the potential for the production of a fissile isotope ^{233}U from an isotope ^{232}Th in a hybrid fusion reactor (HFR). In present study, the authors evaluated the possibility of using a hybrid fusion plant as a fuel accumulator in the nuclear fuel cycle, and the potential need of the system for hybrid fusion reactors necessary for its closing. A system with two technologies-fission and fusion, working together, avoid the difficulties that arise as a result of the independent implementation of each of the technologies under consideration was modeled.

Key Words: fission reactor, fusion, thorium cycle, closed fuel cycle, radioactive waste, thorium.

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Light Water Reactors with the Thorium-Uranium Heterogeneous Fuel in a Nuclear Energy System

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The fuel of modern light-water reactors contains a large amount of ^{238}U. On the one hand, it makes it possible to breed a certain amount of fuel even in a thermal reactor. On the other hand, it absorbs neutrons and is a source of minor actinides generation. This work focuses on the concept of heterogeneous fuel. According to this concept fissile and fertile materials are placed in different fuel pellets which are arranged inside the same fuel pin of a conventional reactor. With this approach, a large number of fuel nuclides will be concentrated in fuel pellets of high enrichment. When using fuel pellets containing microfuel in graphite matrix, it will be possible to achieve high burnups, which will allow them to be unloaded from the core and sent for direct disposal without reprocessing and fission products extraction. At the same time, the amount of fission products will be reduced in pellets with raw materials, which will facilitate the process of their reprocessing. On the one hand, this approach will allow production of a certain amount of fissile material and will allow using a simplified reprocessing technology for its extraction. On the other hand, within the framework of this concept, it is possible to significantly reduce the risks of radiation exposure to the environment due to the practical complete burning of fissile material in fuel pellets and the refusal to reprocess them.

The work presents the nuclide balances for the fuel lifetime in the uranium-plutonium and thorium-uranium fuel cycles of the LWR reactor with heterogeneous fuel.

Key Words: heterogeneous fuel, light-water reactors, uranium-plutonium NFC, thorium-uranium NFC.

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Local Disturbance of the Water-Chemical Regime as a Cause of Increased Oxidation of VVER-1000 Fuel Elements Cladding

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It is shown that a sharp local decrease of the hydrogen concentration in a coolant and a corresponding increase of the radiolytic oxygen concentration can cause an increase of the oxide layer on the surface of the cladding of WWER fuel rods. This is due to two successive phase transitions in the reactor coolant — the decomposition of a solution of hydrogen in the coolant near the saturation temperature and subsequent liquid-vapor phase transformation. In this case, the coolant in some areas can leave the hard water-chemical regime of radiolysis suppression. A detailed study of the process of hydrogen release into gas-vapor bubbles near the saturation temperature of the coolant was performed. The distributions of hydrogen concentration in the coolant in the presence of bubbles under WWER core conditions are obtained. The processes of dissolution and diffusion are taken into account. It is shown that the hydrogen concentration in the coolant can decrease by a few orders of magnitude in a short time. In this case, the partial pressure of hydrogen in the bubbles remains constant.

Key Words: water chemistry mode, coolant, solubility, hydrogen, diffusion, radiolysis.

UDC 620.193.4

Corrosion of Fuel Element Cladding Made of Austenitic Steel when Interacting with Nitride Fuel

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It is demonstrated, that carbon present in the composition of the (U, Pu)N fuel can be dissolved in the fuel in the unbound form. In this case, the fuel turns out to be a carbon source, which can directly affect the state of the steel cladding of the fuel element and its mechanical characteristics. The effect of unbound carbon on the steel cladding of a fuel element is most pronounced in the low-temperature part of the temperature range under consideration at maximum burn-ups and maximum concentrations of carbon and oxygen impurities, i.e. in the same areas at which the fraction of unbound carbon reaches its maximum values. It is shown, that one of the possible ways to reduce the effect of carbon on the strength characteristics of steel cladding of fuel rods is to block its penetration into the cladding. Binding of free carbon in fuel by introducing into its composition a certain amount of effective carbide formers makes it possible to suppress intergranular corrosion of steel fuel cladding.

Key Words: nitride nuclear fuel, thermodynamics, carbon and oxygen impurities, chemical interaction, carbides, intergranular corrosion, steel shells.

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Analytical Solutions of Thermo-Visco-Elastic Equations for Cylindrical Bodies as Applied to Predicting the Behavior of Fuel Elements of Power Reactors

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On the basis of the developed method of inverse Laplace transformation and theory of the Cauchy-type integrals, the solution of linear equations of thermoviscoelasticity is considered in relation to the axisymmetric problem of an infinitely long coaxial cylinder. Obtaining solutions to linear equations when used for computing on a computer, along with the well-known advantages of analytical expressions, open up fundamental possibilities for solving complex systems of nonlinear equations of thermoviscoelasticity. The results obtained can be used to analyze and predict the behavior of fuel elements of power reactors.

Key Words: fuel element, linear and nonlinear systems of equations of thermoviscoelasticity, Maxwell body model, power reactor, analytical solution.