Abstracts
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Issues of Development of VVER-TOI Reactor-Based Three-Loop Plant

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The article deals with the possible ways of improvement of VVER reactors with power range 1150…1300 MW (el). It is proposed an alternative three-loop plant the implementation of which is to reduce specific quantity of metal, the number of equipment and pipelines and eliminate a part of high-cost safety system components. The proposed solutions will enable to reduce the capital cost, the terms of construction of the Units and the temporary expenses for maintenance thereby resulting in enhancement of competitiveness of domestic NPP at the inter-national market.
Key Words: water-cooled water-moderated power reactor, simplification of reactor plants, generation III+.

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Conseptual Proposals of China CSR-1000 with Supercritical Parameters (Overviev)

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It is considered the water-cooled reactor with supercritical coolant parameters China CSR-1000 referred to the next reactor generation “Generation- IV”.
Key Words: China, reactor, water, supercritical parameters, next generation.

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Software package SKALA for precision calculations of neutron and gamma fields based on the combined constant library COLIBRI

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The article deals with description of software package and libraries for neutronics calculation of reactors and protection by Monte-Carlo method. The description of the calculated functionals and the data used for this purpose is given. It is also described the usage of calculation program by Monte-Carlo method with KOLIBRI constants in SKALA code for implementation of the SKALA system available possibilities. As regards, in particular, calculations of isotope kinetics, injurious doses, errors of neutronics calculation results, linear functionals of neutron flux. The system is a computational tool which allows to carry out calculation of neutron spectra and associated gamma-quantum spectra, distributions of rates of interaction processes of neutrons with substance, power distributions, kinetic characteristics, calculation of the self-shielded neutron cross-sections.
Key Words: Monte-Carlo method, neutronics calculations, isotope kinetics, neutron flux, neutron constants, neutron data libraries.

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New Calculation Procedures Used in the SKALA-ROCOCO Package

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The program of calculation by Monte-Carlo method with geometrical module MMK and constant support system ROCOCO (hereinafer MMK-ROCOCO) is a part of SKALA package. The article is devoted to describing the developed original procedures which have been implemented in the package and allow to solve some problems of neutronics calculation of this package.

Key Words: Monte-Carlo method, neutronics calculations, subgroup parameters, software package, neutron flux, neutron constants, neutron data libraries, subgroups, self-shielding, neutron spectrum, neutron lifetime

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Using a Submodeling Method to Solve the Problem of Calculation of Thermal Neutron Flux at the Ionization Chambers

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The article deals with an approach to calculation of thermal neutron flux at the ionization fission chambers with the use of the submodeling method.

Key Words: submodeling, reactor plant, thermal neutron flux, fission chamber, PMSNSYS-II.

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Determination of Computer Code Error by the Experimental Results

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The article deals with determination of a computer code error considering an experiment error. Using the statistical simulation methods, in determination of an error it is taken into account both an error of measurement of the parameter under study and an error of the initial and boundary conditions of the experiment.

Key Words: error, uncertainty, experiment, Monte Carlo method, accident.

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Experimental studies of two-phase coolant mixing using the full-scale FA model

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The article presents the description of experimental facility, the methods of experiments, and also the results of experimental studies simulating the TVS-2M inter-subchannel mixing processes in two-phase coolant flow. The experiments were performed at OKB “GIDROPRESS” at the FA coolant mixing test bench using the cross section-full scale fuel assembly model. By the experimental results we performed a comparative assessment of inter-subchannel mixing intensity in the one-phase and two-phase flow. The influence of steam generation rate in a bundle and coolant flowrate per a void fraction value in the jet was analyzed.

Key Words: coolant mixing, two-phase flow, fuel assembly.

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Experimental studies of condensation-induced water hammers in the VVER reactor pipelines

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The article deals with information on the OKB "GIDROPRESS" experimental studies of condensation-induced water hammers in the different geometry pipelines of an internal diameter up to 100 mm which were performed on demand of "Concern Rosenergoatom". The obtained experimental data are used to analyze the conditions of condensation-induced water hammers in the WWER reactor plant pipelines and to validate the computer codes.
Key Words: VVER reactor, two-phase flow, condensation induced water hammer, pressure, experimental study.

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Calculation of the Drop Time of VVER Reactor Rod Control Assemblies Taking into Account the FA Distorted State in the Core. Code CORE_1

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Code CORE_1 developed in OKB "GIDROPRESS" is considered. It is used to model the thermal-mechanical behaviour of all fuel assemblies (FA) in the core under the normal operating conditions. We also consider the calculation of the drop time of the reactor rod control assemblies (RCCA) which includes: calculation of FA bowing during operation, calculation of friction forces during RCCA motion in the guiding channels of the bowed FA.
Key Words: computer code, FA thermomechanical behavior, resistance to absorbing rod motion.

UDC 539.3

Method of Calculation of the Stress Intensity Factors for Postulated Underclad Cracks for Assessment of VVER Pressure Vessel Structural Integrity within the Framework of the NPP Safety Analysis

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The structural integrity of the VVER reactor pressure vessel referred to the first class equipment operating under the high-temperature irradiation conditions has to be justified for the whole service life period to assure the NPP safety. The method for determining the stress intensity factors has been developed for the postulated underclad cracks in justification of the reactor pressure vessel brittle fracture resistance. The use of singular equation methods, approximations and special function theory made it possible to develop an approach that is characterized by high accuracy and universality and adapted to fracture mechanics problems. The developed method allows elaborating a solution of a wide class of the applied continuum mechanics problems. The method efficiency has been demonstrated for the model of an inclined underclad crack under the mixed loading conditions.
Key Words: underclad crack, stress intensity factor, brittle fracture.

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Calculated Relations for Determination of the Lead Coolant Thermodynamical Properties

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The article was prepared based on the results of the work of the Thermodynamic Properties Data Center (TPDC) State Atomic Energy Corporation “Rosatom”. The evaluation of the thermodynamic and transport properties of liquid lead is based on experimental data from 67 experimental studies published in the open literature. The present paper is devoted to the numerical analysis of data obtained for the period from 1913 to 2017 for the transport and thermodynamic properties of lead in the liquid phase.

The analysis was performed for the following properties: heat capacity coefficient, density, thermal conductivity coefficient, electrical resistivity, dynamic viscosity coefficient, surface tension coefficient, and local sound velocity. The paper presents the errors of the proposed relations and the range of their applicability. The basis of the data evaluation methodology was a modified least squares method that allowed for the errors of experimental data to be considered.

Key Words: bismuth, coolant, density, thermal conductivity, heat capacity, viscosity, thermodynamic properties, transport properties.

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Investigation of Thermomechanical Behavior of VVER Fuel Rod Claddings with SOCRAT/B1 Code

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The paper deals with the fuel rod cladding ballooning processes which can result in loss of integrity of fuel rods and possible release of radioactive fission products into the reactor coolant during the initial stage of beyond-design basis accidents and severe accidents. The OKB "GIDROPRESS" experiments were numerically modeled using SOCRAT/B1 computer code. A good agreement is demonstrated between the calculated values of fuel rod clad collapse time and the results of measurements. The importance of state-of-the-art experiments on fuel rod cladding ballooning and burst is demonstrated.

Key Words: SOCRAT/B1 code, cladding, fuel rod, VVER.

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Assessment of the Marginal Value of Neutron Multiplication Factor for the Radioactive Waste Receiver Tank

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The paper deals with the modification of the Rossi-α method to determine the multiplying properties of subcritical multiplying media with unknown composition and localization of ionizing radiation source. A measurement facility based on the high-efficiency fast neutron detectors made it possible to verify the proposed method by determining values of k_{eff} in the range of 0.09 - 0.45 for different multiplying media with known composition and geometry. The method was tested to determine values of k_{eff} in the range of 0.75–0.97 in a fast assembly with uranium, steel and sodium. This method made it possible to determine the marginal value of k_{eff} for multiplying receiver tank of radioactive waste - the medium of unknown composition and geometry, which did not exceed-0.1.

Key Words: Rossi-α, neutron multiplication, subcritical multiplying media, media with unknown composition and localization of ionizing radiation sources, radioactive waste receiver tank