Abstracts
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Analysis and Validation of Algorithms for Neutron Kinetics Simulation Based on Monte Carlo Methods

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This paper presents Monte Carlo calculations performed for transients. These calculations simulate the experiments conducted at SPERT III and CROCUS reactor facilities. This paper analyzes how the number of neutron generations affects the prediction accuracy for the kinetic parameters that determine the kinetic process development. The influence of the total number of neutron histories used by KIR-C program during Monte Carlo calculations is also examined.

***Key Words*:** adiabatic approximation, multi-zone approximation, direct method, number of neutron generations, statistics, Monte Carlo method, KIR-C program, kinetic parameters, SPERT III reactor, CROCUS reactor.

EDN: DNICLG

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Modeling of Neutron Kinetics in 3D Tests of the C5G7-TD Benchmark by the Code KIR

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The results of calculations of three-dimensional tests of the C5G7-TD kinetic benchmark by the KIR code using the MAGMA group physics module are presented. The code KIR solves the non-stationary neutron transport equation by the analog Monte Carlo method. The tests consider fast transients caused by changes in the position control rods.

The results of the code KIR are proposed as reference ones. The results of the SUHAM-3D-TD, RMS, MPACT, and PCI_nTRACER codes are compared with the results of the code KIR.

***Key Words*:** neutron kinetics, nonstationary neutron transport equation, benchmark C5G7-TD, Monte Carlo method, precision calculations, code KIR, MAGMA software module.

EDN: EFWMAM

UDC 621.039.524.2.034.3

Influence of Coated Fuel Particle Design Parameters on HTGR Characteristics in Transient Conditions

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This paper presents the results of computational studies on how the diameter of coated fuel particle kernels and the thermal resistance of protective coatings thereof influence the characteristics of High Temperature Gas-cooled (HTGRs) with spherical fuel elements in both steady-state and transient modes. It shows that the kernel size of coated fuel particles significantly affects the reactor steady-state characteristics, while the thermal resistance of coated particle claddings enables substantial prediction of reactor behavior and parameters throughout the entire transient.

Key Words: HTGR, coated fuel particle, diameter of the coated fuel particle kernel, heat transfer, steady-state process, transient, effective thermal resistance, neutronic constants.

EDN: IETGQR

UDC 621.039.514

Discrete Ordinates Calculations of 1D PIK Reactor Models Using ENDF/B-7.0 and ENDF/B-7.1 Libraries

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Comparative Monte Carlo and discrete ordinates calculations of the effective multiplication factor performed for five test configurations simulating the PIK reactor by cylindrical models of varying complexity show that effective multiplication factors calculated by KENO-6 and PARTISN codes based on ENDF/B-7.0 nuclear data library for 1D reactor models differ by no more than 0.03. Calculations using ENDF/B-7.1 and ENDF/B-7.0 nuclear data libraries yield close effective multiplication factor values. Calculated by two methods (using the XSDRN code of the SCALE software package and the SERPENT-2 code), the two sets of 238-group constants for the PARTISN code yield practically coinciding effective multiplication factor values for the test configurations. This shows that the PARTISN code with a 238-group constants system based on the ENDF/B-7.0 nuclear data library can be used in neutronic calculations of the PIK reactor.

Key Words: discrete ordinates method, evaluated nuclear data library, multiplication factor.

EDN: IRHEMQ

UDC 621.039.56

Analysis of Approaches to Reduce Power Nonuniformities while using Gray Control Rod Assemblies in Large VVERs

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This paper analyzes how gray control rod assemblies that serve to compensate xenon transients in order to exclude water exchange during reactor plant load following affect the in-core power distribution in large VVERs. It suggests technical solutions to both reduce power nonuniformities and increase the linear power margin to limit curve in steady states with inserted gray control rods for basic 18-month fuel cycles. Fuel burnup is also calculated. The results hereof allow load-following algorithms to be optimized as regards compliance with local parameters, and next steps to be taken towards implementing a unified algorithm to control both power and axial offset in load-following modes.

Key Words: load-following modes, AES-2006 project, load-following algorithms, axial offset, spatial kinetics, KORSAR/GP code, gray control rod.

EDN: LXTOMI

UDC 621.039.5

The Influence of U—Gd Rod Locations on Microfield Power Distribution in VVER-440 Fuel Assemblies

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This paper discusses fuel enrichment profiling across VVER-440 fuel bundles, and analyzes how the arrangement of differently enriched uranium-gadolinium and uranium fuel rods affects the intra-assembly power microfield non-uniformity. New options for optimized fuel enrichment profiling in second- and third-plus-generation VVER-440 assemblies, as well as the key neutronic characteristics of fuel cycles developed using the new RK-2 and RK3+ assemblies, are presented. This paper also describes comparative thermohydraulic calculations performed for RK3+ fuel assemblies with new 468JA and RK-2 47W profiles. Calculated results demonstrate the advantages of 468JA RK3+ fuel assemblies in terms of stationary thermal hydraulics.

Key Words: VVER-440, fuel assembly, fuel rod, U—Gd rod, power peaking factor, microfield power distribution, fuel enrichment, multiplication factor, RK-2, RK3+, JA enrichment profiling type.

EDN: OWJXMH

UDC 004.896

Development of the Computational Model for Noise Filtration in Data Reserved from Sensors of I&C System on the Base of Artificial Neural Network for Implementation in the Framework of Accident Management Strategy

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Data received from sensors are always accompanied by noise and interferences. This is due to a number of objective external and internal reasons. To reduce the noise level in the data, many different solutions are used: optimization of the sensor location ant its type, physical protection of sensors from external noises, hardware and software filtering.

The study considers a number of traditional methods of software noise filtering (arithmetic mean, median mean filter, exponential running average with an adaptive coefficient, Kalman filter). It is shown that use of the considered filters is accompanied by data distortion and bias, loss of essential features. It could be the reason of low accuracy (or even errors) of diagnostic models to support the operator in conditions of beyond design basis accidents at nuclear power plants.

To solve the problem, an alternative noise filtrating method using the AANN architecture is considered and a corresponding model is developed, the use of which, with the same time costs for data filtering, allows solving the problems of data bias and distortion with significant increase in the quality of filtering (values of the quality metrics MAE were reduced by at minimum 4 times and MSE — by at minimum 22 times compared to the considered traditional approaches to software noise filtering with the comparable computational costs).

Key Words: noise, noise software filtering, median filters, Kalman filter, neural network.

EDN: RIERQA

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Daily Load-Following Range of VVER-TOI

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This paper discusses how the primary makeup-blowdown system’s process constraints involved with boron control, as well as the limitation of the highest admissible power increase rate, do affect the reactor operation in daily load-following modes.

Key Words: VVER-TOI, IR program, algorithm, offset, load following.

EDN: RSRQJU

UDC 504.064.3, 551.46.077, 621.039.764

Study of Radioactivity Release from the “Komsomolets” Nuclear Submarine using Underwater Gamma-Spectrometric Equipment

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This paper presents an overview of basic results yielded by long-term study of radioactivity released from the K-278 Komsomolets submarine into the surrounding waters using underwater gamma spectrometers of the REM series developed by NRC “Kurchatov Institute”. It analyzes the dynamics of radioactivity releases from the submarine reactor compartment into the marine environment, and concludes about the status of protective barriers. Particular attention is paid to the latest results obtained during the 2022—2024 survey, which can underlie an assessment of threats to the Norwegian Sea ecosystem and highlight the need for long-term monitoring due to ongoing nuclear fuel corrosion in the reactor core. The paper also suggests a preliminary design of an autonomous underwater station for continuous monitoring.

Key Words: radiation survey, underwater gamma spectrometers, “Komsomolets” nuclear submarine.

EDN: SOHJKR

UDC 621.039

Kinetics of Fission Gas Accumulation in the Buffer Layer of HTGR Coated Fuel Particles

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A theoretical analysis of mechanisms of fission gas releases from fuel kernels into the buffer layer of HTGR coated fuel particles has been carried out, including thermally activated diffusion, radiation-induced diffusion, direct recoil, and knockout. A computational code was developed to evaluate the kinetics of excess xenon pressure buildup in the temperature range of 1000—1300 °C. It was shown that at low temperatures of about 1000 °C, the recoil effect contribution to pressure buildup is comparable to that of diffusion, while at higher temperatures the diffusion begins to dominate. An analysis of how the diffusion coefficient depends on burnup established the largest contribution to fission gas releases to source from thermally activated diffusion, which increases with fuel burnup.

Key Words: coated fuel particle, xenon, krypton, diffusion, knockout.

EDN: TIYGBA

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Dynamic Bending of Fuel Rods in Pulsed Reactors

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This paper presents the dynamic bending effect, which is typical for pulsed reactors due to periodic power pulses and nonuniform neutron flux occurring in the reactor core. It describes a heat transfer model developed for a single fuel element, and considers a respective time-dependent heat equation in its isotropic approximation. The resulting time-dependent function of temperature distribution in fuel pellets and cladding, set as an input parameter with a coefficient corresponding to the heat yield gradient, serves in the fuel element mathematical model to calculate the dynamic cross-bending of fuel cladding. Dynamic bending assessment results for fuel elements of the NEPTUNE reactor that is under design are also discussed.

Key Words: pulsed reactor, fuel element, dynamic bending.

EDN: TWDOJI

UDC 621.039.536.2

Shell Metal Heterogeneity Evaluation in Reactor Pressure Vessels for Generation 3 and 3+ VVERs

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An analysis of impact bending test results for specimens made of standard VVER-1000/VVER-1200 reactor pressure vessel shell metal has been carried out. It shows that the ductile-to-brittle transition temperature (T_k) values determined after heat treatment in "minimum" and "maximum" cycles can be considered as a single dataset. An assessment of differences in T_k values determined by the two methods specified in PNAE G-7-002-86 and GOST R 50.05.12-2018 shows that the T_k values determined as per GOST R 50.05.12-2018 are systematically lower by 19—39 °C than those determined as per PNAE G-7-002-86. Even with account of the methodological correction, the T_k values determined for the sample ring metal are consistently lower than those determined for surveillance specimens cut from the ring adjacent directly to the shell metal and at a distance of at least 70 mm from the thermal barrier. Heterogeneity of the test ring metal is shown to be higher than that of the surveillance specimen sample rings. An assessment of heterogeneity magnitude in VVER-1000/1200 reactor pressure vessel shell metal in the azimuthal direction has been also performed.

Key Words: VVER, reactor pressure vessel shell, heat treatment, ductile-to-brittle transition temperature, standard deviation.

EDN: UKHYSG

UDC 532.529

Engineering Methodology for Estimating Pressure Increase during Condensation-Induced Water Hammer

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This article addresses the current challenge of analyzing condensation-induced water hammers (CIWHs) in pipelines of VVER reactor. Since CIWHs are highly non-equilibrium processes that occur within unstable two-phase flows, they cannot be reliably predicted with currently available methods, especially for complex geometry of real pipelines. Therefore, design evaluations sometimes require postulating CIWH occurrence, which leads to another important task of estimating the shock pressure, i.e., the pressure surge during CIWH. This article describes the engineering methodology developed by the author for computational assessment of CIWH pressure increase in pipelines, considers several optional methodological assumptions, compares calculated data for each option based on a test problem, and concludes on the applicability of this engineering methodology for calculated reactor plant design evaluation.

Key Words: condensation-induced water hammer, pipeline, shock pressure, water plug velocity, engineering computing methodology, VVER.

EDN: VMVHMR

UDC 621.039.58

About Technologies and Equipment for Coolant Water De-ironing in Single-Loop VVER-SKD. Part 2. Expediency of Injecting Zinc into VVER-SKD Coolant

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First part of this article [1] examines feedwater de-ironing systems for a single-loop VVER-SKD reactor using high-temperature filters filled with porous titanium, as well as electromagnetic de-ironing filters. The second part of this article analyzes the possibility of using zinc compound dosing into the coolant to reduce radiation fields, as well as the intensity of general corrosion and stress corrosion cracking in VVER-SKD operating conditions. It considers relevant Russian experience obtained while studying neutral and oxygen modes at the domestic VK-50 nuclear power facility, and also presents foreign experience of zinc injection into coolant water. The authors hereof show that dosed injections of zinc acetate will inevitably increase the specific electrical conductivity of VVER-SKD coolant. Radiation and thermal effects on the dosed zinc acetate will cause the formation of acetic acid and other acidic substances in the coolant, thus increasing its corrosiveness. Zinc acetate decomposition can also trigger the buildup of hazardous carbon deposits on fuel cladding surfaces. All technologies of zinc dosing into the coolant of direct-flow VVER-SKDs will face a critical challenge of removing radioactive corrosion products displaced from metal surface films after zinc injection, as well as dealing with activity spikes of radionuclides (primarily ^{60}Co and ^{58}Co). For these reasons, dosed zinc injections into coolant water cannot be recommended for VVER-SKD reactors.

Key Words: pressure vessel single-loop water-cooled nuclear power reactor, supercritical parameters, water chemistry, zinc injection into coolant water.

EDN: XSNPOM