Abstracts
Journal "Problems of Nuclear Science and Engineering. Series: Physics of Nuclear Reactors", issue No.1, 2018

UDC 519.622

The Convergence Estimation of the Parallel Algorithm of the Linear Cauchy Problem Solution for Large Systems of First Order Ordinary Differential Equations Using the Solution as Expansion Over Orthogonal Polynomials

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This paper is devoted for the algorithm of the linear Cauchy problem solution for large systems of the first order ordinary differential equations using parallel calculations. The proof of the convergence of the iteration process using the solution as expansion over orthogonal polynomials for [0,1] interval is presented. Features of this algorithm are simplicity, opportunity to get solution by parallel calculations and also possibility to get solution for non-linear problems by changing the operator using the solution from iteration process.
Key Words: Algorithm, Iteration Process, Code, Computer, System of Equations, Solution, Space, Vector.

UDC 621.039

Determination of Neutron Flux by Multi-Point Kinetics Method

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The equations for calculation of a neutron flux together with multi-point kinetics equations are presented. The calculational algorithm of exchange coefficients is realized within the user module for MCU code. The correctness of user module for calculation of exchange coefficients is shown on the example of steady-state calculation of full-scale model of the reactor KLT-40S core. For the solution of the differential equations the implicit (3,2)-method is chosen. The method is realized in the MRNK code. Code was demonstrated for RPCEU235 test problem.
Key Words: Multi-Point Kinetics, Exchange Coefficients, Coupling Coefficients, Monte Carlo Method.

UDC 621.039

Simulation of Neutron Kinetics of KLT-40S Reactor Core by Monte-Carlo Method

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The results of the neutron kinetics simulation by MRNK and KIR codes for full-scale model of the KLT-40S reactor core are presented. Processes are considered with the introduction of positive reactivity, as well as with the movement of individual groups of rods under the condition of maintaining stationary criticality. In calculations, in addition to the integral heat power, neutron fluxes were determined in ionization chambers located outside the core.
Key Words: kinetics, calculation, multipoint kinetics, direct Monte Carlo method, nuclear reactor.

UDC 621.039

Benchmark VVER-VN for Verification of Nonstationary Computer Codes

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It is presented the mathematical benchmark worked out for the verification of computer codes which are intended for transient processes in water-water nuclear reactors calculations. Results of calculations using codes KIR, MRNK, START-UNK are given.
Key Words: Calculation, Kinetics, VVER, Nuclear Reactor, Monte Carlo Method, Diffusion Approximation, Multi-Point Kinetics, Supercomputer.

UDC 621.039

Concerning Measurement of Power Reactors Subcriticality by Statistical Methods

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It is discussed the possibility of the power reactors subcriticality measurement by statistical methods according to regulatory rules NP-082-07. By statistical methods, by Feynman method in particular, it is possible to measure the subcriticality of the stationary reactors in the required range (0,01…0,02) at the level of the neutron detectors count ~ 1 of pulse per second. The Feynman method was explored on the critical facility Proteus (Paul Scherrer Institute, Switzerland). The results of the subcriticality measurements are given. The conditions, which must be ensured for obtaining the acceptable result of experiment, are formulated.
Key Words: Subcriticality, Feynman's Method, Nuclear Safety.

UDC 631.039.56

Program of Control Rods Movement During Nuclear Reactor Start-up

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The paper describes the effect of absorber rods withdrawal on the behavior of the reactor period during the start-up. The results of two versions of the program performed at full-scale prototype of marine reactor are given.
Keywords: Reactivity, Reactor Start-up, Reactor Period, Shim Rod Bank

UDC 621.039.56

Numerical Simulation of Measurements Performed During the Novovoronezh-2 Unit 1 Reactor Physical Startup Tests

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The article represents the results of numerical simulation of experiments during the unit 1 Novovoronezh-2 NPP reactor physical start-up. The list of specified parameters includes: critical concentration of boric acid and scram efficiency. The calculations were performed using code RAINBOW-TPP, providing a combined neutron-physical and thermal-hydraulic calculation of the reactor, and by precision neutron-physical Monte Carlo codes. The scram efficiency was defined from steady-state calculations and using the equation of the inverse point kinetics based on the numerical simulation of the neutron flux at the location of the ionization chambers. The results of the numerical simulation of scram efficiency were compared with measured data defined in the process of the physical start-up.

Key Words: VVER, safety, reactivity, ionization chamber, minimum controlled power level, reactimeter, scram efficiency, simulation, unit 1 of Novovoronezh-2 NPP, RAINBOW-TPP.

UDC 621.039.5

Nuclear Power Unit with Molten Salt Fuel for Arctic

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A flowsheet for the very small reactor unit based on molten salt fuel is considered. Principal possibility of the non-service facility with up to 500 kWe power and 5–10 or more years of operation time is demonstrated.

Keywords: molten salt reactor, fuel salt, campaign, criticality, fuel salt.

UDC 621.039.536.2

Elaboration of Technology of the VVER-1000 Reactor Vessel Recovery Annealing in the Full-Scale Testing Facility

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This report presents experimental proof of the possibility of safe application of the technology for recovery annealing of the reactor vessels at the operating NPPs with VVER-1000/320 reactors to ensure justification of the design lifetime extension, which is the strategic mission of the Russian nuclear industry. Part of the reactor vessel located opposite the core, in particular, metal of welds #3 and #4, as well as the base metal between them is exposed to the recovery annealing, i.e. heat metal treatment in the hold mode at the temperature of 565±15°C for no less than 100 hours. Set of equipment was designed and manufactured for the recovery annealing. It includes:

– heating device;
– fixture for assembly of heating device;
– complex for mounting external thermal insulation of the reactor vessel;
– heat control system.

Full-scale experimental testing facility applying real VVER-1000 reactor vessel, support plate and support frame was designed and constructed at the site of JSC “AEM technology” “Atommasch”, Volgodonsk for experimental verification of the equipment. The testing facility is equipped with system for temperature monitoring of the reactor vessel and support structures, as well as for movement registering of some components of the reactor barrel.

Equipment of the testing facility was mounted and commissioned. It was tested for the recovery annealing of the VVER-1000 reactor vessels and for a number of experimental heatings, which showed that:
– equipment ensures the necessary heat treatment temperature for the whole amount of the annealed vessel metal;
– the maximum temperature for the serpentine concrete and structural concrete of the support frame in the process of annealing did not exceed the allowable values.

Key words: reactor vessel, recovery annealing, support frame, serpentine concrete, structural concrete, heating device, full-scale testing facility, experimental heating