Abstracts
Journal “Problems of Nuclear Science and Engineering. Series: Physics of Nuclear Reactors”, issue No.3, 2018

UDC 621.039.5
RBMK’s Modernization as an Alternative to Decommissioning

V.F. Tsibulskiy, E.A. Andrianova,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182.

Economic and reputation problems that may appear in the developing Russian nuclear energy in the nearest term are analyzed in the paper. Earnings from the electricity sale, the new building investments and decommissioning cost are estimated. There will not be financial resources for nuclear energy development after decommissioning of 11 GW RBMKs and termination of state support. RBMK’s modernization in order to continue their operation is suggested to the consideration in the paper. One of the possible ways is replacement of graphite moderator blocks by graphite balls for increasing the safety characteristics. This step and replacement of used equipment will allow RBMKs to operate further, build new NPPs and develop the Russia’s economics.
Key Words: RBMK, decommissioning, continued operation, modernization.

UDC 621.039.55
Optimization Study of the IBR-2 Reactor

V.D. Ananiev, Yu.N. Pepelyshev, A.D. Rogov,
Joint Institute for Nuclear Research, 6, Joliot Curie st., Dubna, Moscow Region, 141980.

The neutron-physical aspect of optimization of the IBR-2 reactor is considered in respect of whether is it possible in principle to design and construct an IBR-2-type reactor with a higher neutron flux density in the beams then exists now $0{,}5\cdot10^{13}$ n/(cm$^2\cdot$s). The calculations have shown that the thermal neutron flux density can theoretically be increased up to $(2{,}0—2{,}5)\cdot10^{13}$ n/(cm$^2\cdot$s) only if the reactor design is changed completely (decrease of the core volume, replacement of the fuel type with a denser one, and change of the system of beam extraction from the radial to tangential one). The technical implementation of these requirements poses a significant problem.
Key Words: high-flux pulsed neutron source, reactor IBR-2, neutron beams, thermal neutrons, cold neutrons.

UDC 621.039.5
Methodical Aspects of Creation and Calculation of the Space Kinetics Benchmarks

M.N. Zizin,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182.

The problems arising at creation and calculation of the space kinetics benchmarks are considered by three tasks. The connection of non-stationary diffusion equation is considered by inverse solution of kinetics equation (ISKE) with the different methods of preparation and presentation of kinetic parameters. The following benchmarks are considered: non-diffusion Small LWR benchmark from series of Takeda’s tasks, complemented by modeling the movement of control rods; Ferguson’s benchmark, enlarged by calculation of the reactivity change with ISKE; TWIGL – 2D LWR model with inserting the positive reactivity.
Key Words: Ferguson benchmark, Takeda benchmark, TWIGL benchmark, space kinetics, inverse solution of kinetics equation, delayed neutrons, diffusion approximation, ShIPR intellectual code system.

UDC 621.039

Nonstationary Version of the Neutronic Code CORNER

V.P. Bereznev, I.S. Chernova,
IBRAE RAS, 52, Bolshaya Tulskaya st., Moscow, 115191.

The code CORNER potential was extended for analysis of non-stationary processes. There are different approaches for solving the time-dependent problems. The direct approach is most "expensive". The use of this approach with the code CORNER, which is based on the Sn method, is ineffective taking into account the calculation time. Improved quasi-static method with a modified procedure of the reactivity determining was selected for solving the spatial kinetics problems. The algorithm of solution and the basic formulas are described. The developed non-stationary calculation module has been tested on the test problem. The results obtained with the TIMER code that uses the direct method for solving the non-stationary problem are presented for comparison.

Key Words: neutronic calculations, transient process, neutron transport, the method of discrete ordinates.

UDC 621.039.5

Some Results of Verification of Code ODETTA for Shielding Calculations

V.I. Belousov, N.A. Grushin, E.P. Sychugova, E.F. Seleznev,
IBRAE RAS, 52, Bolshaya Tulskaya st., Moscow, 115191.

A brief description of the transport code ODETTA is given. It is intended for fast breeder reactors shielding calculations. The Linear Discontinuous Finite Element Method on unstructured tetrahedral meshes is used for solving the neutron and gamma multigroup Discrete Ordinates transport equations in the *X*—*Y*—*Z* geometry. The code is written on FORTRAN-90 using OpenMP. There are presented the results of the verification of the code ODETTA using experiments on radiation shielding (ASPIS and EURACOS shielding facilities of SINBAD data base). The results are comparing with the experimental data and the previously published results of calculations by other codes (DORT, MCBEND, KATRIN) using different nuclear data libraries.

Key Words: neutron and gamma transport calculations, transport codes, finite element method, discrete ordinates method.

UDC 621.039.54

Modeling of Fission Products Release from Microfuel Considering the Effects of the Trapped Fraction and Concentration Jumps at the Phase Boundaries

A.S. Ivanov, A.A. Rusinkevich,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182,
M.D. Taran,
SRC RF "Troitsk Institute for Innovation & Fusion Research",
pos. 12., Pushkovykh st., Troitsk, Moscow, 142190.

A modification of the FP Kinetics code [1] was performed to calculate the release of fission products from HTGR microfuel elements, allowing to take into account chemical binding, limited solubility effects, and concentration jumps of the components at the interfaces of the coating layers. A comparison was made of the release curves of Cs from microfuel calculated using the FP Kinetics and PARFUME [2] codes. It is shown that taking into account the concentration jumps at the interfaces of the silicon carbide layer makes it possible to give a consistent explanation of the experimental data on the Cs release obtained in the post-reactor thermal tests. The need for experiments to measure the solubility limits in coating materials was noted.

Key Words: microfuel, fission products, diffusion, solubility.

UDC 621.039

Development and Verification of a Module of Chemical Kinetics of Iodine and Cesium Compounds. Part 1. Mathematical Tests

A.A. Kovalishin, M.N. Laletin, O.Yu. Poveshchenko,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182.

The important issue at an NPP accident consequences evaluation is the determination of physicochemical composition of iodine and cesium compounds, their activity, behavior, deposition in different places and atmospheric release. The article presents the description of a model of chemical kinetics of iodine and cesium compounds in steam-hydrogen atmosphere, during fission product release in VVER-type reactors. There are given values calculated using the presented model, as well as comparison of these results with calculated data by means of other programs.
Key Words: iodine, cesium, radioactive iodine release, iodine and cesium species, chemical kinetics, numerical modeling, chemical reaction coefficient, chemical reaction equation.

UDC 621.039.531

VVER-1000 Reactor Pressure Vessel Lifetime Assessment According to Criteria of Brittle Fracture Using Results of Surveillance Specimens Tests

L.N. Nedelchev,
Nuclear Power Plant "Kozloduy" PLC, 3321 Kozloduy, Bulgaria,
D.Yu. Erak, D.A. Zhurko, M.A. Skundin, V.B. Papina,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182.

The reactor pressure vessel (RPV) is of primary importance for residual lifetime determination of VVER-1000 type reactor, as RPV is irreplaceable. To control the condition of certain VVER-1000 reactor pressure vessel the periodic assessment of steel properties variation degree is carried out using surveillance specimens. This paper presents the results of work, the purpose of which was to substantiate the possibility of 40-year design lifetime extension for VVER-1000 RPV units № 5 and № 6 of "Kozloduy" NPP. Substantiation was carried out according to criteria of RPV brittle fracture under thermal shock in case of design accident with loss of coolant. Degree of steel embrittlement was determined according to the requirements provided by Russian regulatory standards considering testing results of reactor units № 5, № 6 "Kozloduy" NPP surveillance specimens. In forecasting ductile-to-brittle transition temperature shift (ΔT_k) under the impact of operational factors for an extended period a comparison between analytical and experimental method was carried out. Possibility of exploitation for "Kozloduy" NPP units № 5 and № 6 RPV beyond the design lifetime up to 60 years was substantiated.
Key Words: WWER-1000, reactor pressure vessel, surveillance specimens, lifetime extension, ductile-to-brittle transition temperature, brittle fracture criteria.

UDC 621.039

Strategies of Corium Localization under Severe Accidents with Core Melting for New NPP Projects with VVER

Yu.A. Zvonarev, Yu.M. Semchenkov,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182.

The computational analysis results of the in-vessel and ex-vessel melt retention strategies application for VVER of different capacity is presented. The choice in favor of in-vessel melt retention strategy for VVER-600 and ex-vessel core catcher for VVER-1200 is proved. It is shown that the ex-vessel core catcher effectively performs its functions on severe accident management and reliably ensures the melt localization

and cooling for high power reactors. The calculations of corium localization in the core catcher were carried out with the help of the HEFEST-ULR code developed at the NRC "Kurchatov Institute".
Key Words: severe accident, corium, molten pool, core catcher, code development, HEFEST-ULR code.

UDC 621.039.531

Method of Yield Stress and Elasticity Modulus Change Predicting of the Core Shroud of Nuclear Propulsion System

P.V. Alekseev, A.T. Alekseev, E.E. Alekseev, O.D. Loskutov, A.A. Tutnov,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123098.

This article contains description of the method of estimation of the yield stress and elasticity modulus of alloy TSM-7,especially during exploitation, which would be used for core shroud of nuclear propulsion system for space purposes. Probable conditions of operation and basic assumptions of the method are given. Approach for kinetics modeling of radiation-induced defects composition and subsequent annealing is proposed. Formula, based on the developed model, for estimating the decrease of yield stress and elasticity modulus of TSM-7 alloy are given.
Key Words: modeling of radiation embrittlement, nuclear propulsion system, alloy TSM-7, kinetics of radiation-induced defects, yield stress, elastic modulus, cluster formation energy.

UDC 621.039.531

Calculated Prediction of Yield Stress and Elasticity Modulus Change of Molybdenum Alloy TSM-7 when used in Nuclear Propulsion System

P.V. Alekseev, A.T. Alekseev, E.E. Alekseev, O.D. Loskutov, A.A. Tutnov,
NRC "Kurchatovskiy Institute", 1, Akademika Kurchatova sq., Moscow, 123098.

The article contains of the computational modeling of the yield stress and the elasticity modulus change of alloy TSM-7 under exploitation the core shroud of nuclear propulsion system. The results of modeling of the clusters kinetics at different temperatures are shown graphically. The elasticity modulus and yield stress change are given as function of temperature and operating time.
Key Words: modeling of radiation embrittlement, nuclear propulsion system, alloy TSM-7, kinetics of radiation-induced defects, yield stress, elastic modulus, cluster formation energy.

UDC 621.039

Tests to Evaluate the Survivability of In-Core Thermocouples in Case of Beyond Design Basis Accident

A.S. Timonin,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182,
V.I. Donetskiy, S.L. Petrunin,
NIKIET, 2/8, Malaja Krasnosel'skaya st., Moscow, 107140.

The results of the experiment to assess the survivability of the chromel-alumel (type K) cable in-core thermocouples used in WWER in beyond design basis accident conditions (1400 °C or more) are considered. The results obtained by the preservation of the integrity of thermoelectric circuits of thermocouples allow to consider them as elements of the Accident Monitoring System.
Key Words: VVER, thermocouple, beyond design basis accident, reliability, survivability.