Abstracts
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Computational Modeling of Kinetic Experiments on a Full-Scale RBMK Assembly Using Various Algorithms Based on the Monte Carlo Method

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The results of non-stationary calculations based on the Monte Carlo method, simulating experiments on the RBMK critical assembly, are presented. Experimental information was obtained on Assembly No. 1, which is a fragment of the full-scale initial reactor load with 191 fuel assemblies, 32 boron absorbers and 6 simulators of absorber rods. The software packages developed at the NRC “Kurchatov Institute” for calculating the kinetics of reactors using the Monte Carlo method were used:

— KIR — implements direct calculation of the kinetics of nuclear reactors without the use of approximations [1];

— KIR-P — implements calculation of kinetics of nuclear reactors using adiabatic or quasi-static approximations [2, 3].

Calculated results are compared with experimental ones, and on this basis the applicability of the used algorithms is assessed. Recommendations are given on the use of one or another algorithm for calculating reactor kinetics based on the Monte Carlo method for various transient processes.

Key Words: kinetics, reactor kinetics, nuclear reactor, adiabatic approximation, quasi-static approximation, Monte Carlo method, dynamics of nuclear reactors, dynamics, time dependent, time dependent Monte Carlo, uranium-graphite reactor.

EDN: ZKVBMM

UDC 539.125.523.43

CHARM: Parallel Three-dimensional Neutron-Physical Code Based on the Method of Characteristics

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The article describes the developed transport code based on the method of characteristics. The theoretical foundations of the methods and approximations included in the code are briefly described. The results of calculations of benchmark problems of reactor physics and radiation shielding problems are presented. The analysis of the obtained results is carried out.

Key Words: neutron transport, reactor physics, radiation shielding, computer program code.

EDN: GTTGWB

UDC 621.039.586

Numerical Study of the Efficiency of Re-flooding of the Heated Core of a VVER-1000 Reactor in the Beginning of Beyond Design Basis Accident with a Large Break LOCA

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The paper presents the results of a parametric study of the re-flooding efficiency on the example of a representative LB LOCA of a VVER-1000 reactor power unit. The study was performed using the SOCRAT/V3 code, considering two parameters: the temperature of the fuel rod cladding at the beginning of the coolant supply to the reactor downcomer and the cooling water flow rate. The level of radiation consequences according to the INES scale was considered as an efficiency criterion. As a result of the analysis, conclusions were formulated on the effectiveness of re-flooding as an accident management measure in accordance with the current state of knowledge.

Key Words: beyond design basis accident, VVER, LB LOCA, re-flooding, SOCRAT, modeling, hydrogen.

EDN: ONEEBE

UDC 621.039.52

Prediction of Relative Power Density Distribution in IRT-T Research Reactor Core by Machine Learning Methods

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Most of the research reactors use partial refueling pattern to refuel spent fuel that is the reason of heterogeneity of power density distribution (PDD) for every particular fuel cycle. The change of power density distribution in a core does affect negatively on maximum fuel burn-up thereby increasing its cost. In exceptional cases, high heterogeneity of PDD may be the reason for restriction operational conditions of reactor. Thus, dependency determination of PPD in core from one fuel cycle to another is important part in reactor operation. In this paper, application of machine learning methods for prediction of PDD in IRT-T reactor core is proposed. Statistical analysis methods have been used to find relationships between fuel burn up and PPD on more than 500 unique core loading patterns. It is shown that supervised learning with Gaussian process regression model fits test data with coefficient of determination value of 0.99 and can be used to predict PDD in a reactor core.

Key Words: power density distribution, machine learning, nuclear research reactor IRT-T, MCU-PTR.

EDN: OPVYRZ

UDC 621.039; 621.039.524.2.034.3

Simulation of Neutronic Characteristics of the HTGR Core During the Fuel Cycle Using the Precision Calculation Code MCU-HTR

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The article presents a methodological approach to the calculation of neutronic characteristics during the burnup of a block-type HTGR core using the MCU-HTR code based on the Monte Carlo method. A description of the developed full-scale three-dimensional model of the core, the approximations used, as well as some calculated results are given to demonstrate MCU-HTR capability to process and visually represent the output data.

Key Words: high-temperature gas-cooled reactor, microspherical fuel, neutron calculations, Monte-Carlo method.

EDN: PEBOCF

UDC 621.039.56

Estimation of Uncertainties of Modern ^{232}Th Nuclear Data Libraries in the Field of Thermonuclear Neutron Energy

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The article presents a comparative analysis of the ^{232}Th neutron cross sections of modern files of estimated nuclear data in the field of thermonuclear neutron energy (14.1 MeV). Different versions of the ENDF-B, JEFF, JENDL, and TENDL libraries, the Russian BROND and ROSFOND libraries, as well as the Chinese CENDL library are considered. The total number of assessed nuclear data file systems reviewed is 21. Specially prepared multigroup libraries of the UNK software package with a more detailed group description of cross sections in the fast energy range were used for computational analysis. (the (n, 2n), (n, 3n) reactions region). A special module for neutron slowing down calculation in an infinite medium has also been developed to solve the problem with a generational source. The results revealed significant discrepancies in the estimates of ^{232}Th cross sections in the energy range of 14.1 MeV. The differences in the evaluation of the cross sections of the reactions (n, 2n) and (n, 3n) are about 50%, elastic and inelastic scattering— about 9 and 95%, respectively, and the total cross section is — about 5%. The differences in the estimate number of secondary neutrons formed as a result of ^{232}Th fission are 3,5%, and in the fission cross section — 6%.
Key Words: estimated nuclear data, neutron cross sections, thermonuclear reactor, blanket, thermonuclear neutron.

EDN: QWHABB

UDC 621.039.531:620.186.1:539.374.1

Computational and Experimental Estimation of Strength Characteristics and Brittle Fracture Resistance of the VVER-SKD Pressure Vessel Made from Candidate Materials of Different Strength Categories

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The traditional steels of the VVER-1000/1200 reactor pressure vessels with a strength category SC-45 and the steel under development with high nickel content and a strength category SC-65 were considered from the perspective of their performance as materials for VVER-SKD reactor pressure vessels operating at temperatures of ~400 °C and pressures of ≥25 MPa. This included considering both the technical feasibility of manufacturing an elongated reactor core shell without a welded joint and the analysis leading to the preliminary selection of 10KhN5MFBA-A steel for the VVER-SKD pressure vessel. The choice of this material was justified by its increased short- and long-term strength, while maintaining a wall thickness of ~200 mm, which is within the capabilities of industrial production, and its superior brittle fracture resistance among candidate materials under operating conditions.
Key Words: high-nickel steel, VVER-SKD, reactor pressure vessel, strength category, candidate materials, short-term and long-term durability, resistance to brittle fracture, stress intensity factor.

EDN: SJJKFQ

UDC 621.039.531, 620.17

Methodology of Mechanical Testing for Fuel Rod Cladding Materials of Russian Nuclear Reactors

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The article provides an overview of various testing methods for fuel cladding materials of Russian nuclear reactors, and also identifies problematic issues in the domestic methodology for ring tension and compression tests. Possible ways to improve the methodology were identified based on the formation of a unified approach aimed at increasing the informativity, representativity and comparability of test results. The structure of this approach is proposed and the areas of necessary work for its formation are outlined.

Key Words: fuel rod cladding, mechanical tests, test methods, ring tension test, ring compression test, gauge length, short-term mechanical characteristics, residual plasticity.

EDN: UCARWG

UDC 544.3, 621.039.531

Silicon as an Effective Carbide Former in (U, Pu)N-fuel Containing Carbon and Oxygen Impurities

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According to the results of thermodynamic analysis, it was established that carbon in the (U, Pu)N-fuel can be in a bound state, in the form of carbides, or be dissolved in the fuel in an unbound form. (U, Pu)N fuel is a source of carbon, which can lead to corrosion damage to the steel cladding of fuel elements. One of the possible ways to reduce the influence of carbon on the strength characteristics is to block its penetration into the cladding. The binding of free carbon in the fuel by introducing effective carbide-forming elements into its composition reduces the amount of unbound carbon. This will suppress corrosion damage to the steel fuel cladding. In this work, special attention is paid to studying the influence of silicon, introduced into the composition of unburned fuel and analyzing its impact on the state of oxygen and carbon impurities.

Key Words: (U, Pu)N-fuel, thermodynamics, corrosion, carbon and oxygen impurities, silicon.

EDN: INUSOG

UDC 621.039

Estimation of the Xenon Diffusion Coefficient in Uranium Dioxide Microfuel Kerns

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The diffusion coefficient of xenon in uranium dioxide kerns at a temperature of 1100 °C was assessed basing on measurements of the gaseous fission products release. The obtained results are compared with the results of other experimental studies of diffusion processes in uranium dioxide. The time of xenon release from UO_2 kern at a temperature of 1100 °C was estimated. It is noted that in order to obtain data on the activation energy of diffusion and the frequency coefficient, allowing one to calculate the temperature dependence of the diffusion coefficient within the framework of the Arrhenius model, it is necessary to conduct experiments at two temperatures. Based on the assessment results, it was concluded that it is necessary to take into account diffusion transfer when calculating the amount of xenon released from the fuel kernel into the buffer layer of the microfuel element.

Key Words: xenon, diffusion coefficient, uranium dioxide, radioactive decay.

EDN: LIALWZ

UDC 621; 544.63; 544.65

Electrochemical Approach to the Studies of the Oxidation Kinetics of Zirconium Reactor Alloys

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This paper presents a sequential analysis of the electrochemical model of oxidation of zirconium alloys. Based on the modified McDonald model, an algorithm has been developed for estimating the rate of zirconium oxidation for various initial concentrations of the components H_2, O_2, H_2O_2 of the VVER primary coolant. The kinetic parameters of the Butler—Volmer equation were determined. These values allowed obtaining anodic polarization curves for various concentrations of H_2, O_2 and H_2O_2 in the coolant. Based on the intersection points of the corresponding cathodic and anodic curves, the corrosion potential and corrosion current were determined at given concentrations of oxidizing components. The concentration dependence of zirconium oxidation rates from O_2 and H_2O_2 concentrations were estimated.

Key Words: corrosion, zirconium, oxygen, hydrogen peroxide, polarization curve, electrochemistry.

EDN: MURZHE

UDC 621.039.58

Criteria for Evaluating the Combustion Modes of Mixtures Containing H_2 and CO in NPP Containments. Part I. Flammability

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The article presents results of the analysis and generalization of Russian and foreign experimental programmes for the study of the flammability limits in hydrogen containing mixtures as applied to hydrogen issues at nuclear power plants (NPPs) with VVER during severe accidents. Based on the experience of modeling severe accidents for various VVER NPP projects (VVER-1000/1200/TOI), the typical range of values of parameters of hydrogen mixtures in the containment atmosphere is estimated. Recommendations are given for estimating the upper and lower flammability limits for mixtures containing hydrogen and carbon monoxide, including oxygen-lean mixtures. The results of this work are relevant for the assessment of hydrogen hazards at NPPs, including the development, validation and application of dedicated computer codes.

Key Words: flammability limits, hydrogen hazards, NPP, severe accidents.

EDN: OBUTGX

UDC 621.039.58

Criteria for Evaluating the Combustion Modes of Mixtures Containing H_2 and CO in NPP Containments. Part II. Flame Acceleration and Detonation

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The article presents the results of the analysis and generalization of Russian and foreign experimental programmes for the study of combustion modes of hydrogen-containing mixtures as applied to hydrogen issues at nuclear power plants (NPPs) with VVER during severe accidents. An approach to criteria-based estimates of flame acceleration modes and deflagration-to-detonation transition for mixtures containing hydrogen and carbon monoxide is formulated. The effects of water droplets in the gas phase on the flammability and combustion modes of hydrogen mixtures are discussed. The results of this work are relevant for the assessment of hydrogen hazards at NPPs, including the development, validation and application of dedicated computer codes.

Key Words: flammability limits, combustion modes, hydrogen hazards, NPP, severe accidents.

EDN: BGZSZY

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The Need to Improve the Domestic Regulatory Framework for Decommissioning of NPP Units

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The article presents the results of an analysis of the main provisions, requirements and recommendations, including terms and definitions used in the fundamental regulatory documents and safety manuals for the use of atomic energy, which form the domestic regulatory framework for decommissioning of NPP and regulate activities related to planning, preparation for decommissioning and decommissioning of NPP units with VVER reactor installations. Their comparative analysis was carried out with similar recommendations available in the documents of the IAEA, as well as the Nuclear Energy Agency of the member countries of the Organization for Economic Cooperation and Development (NEA OECD). Proposals have been formulated to improve the domestic regulatory framework for decommissioning NPP units.

Key Words: NPP units, VVER reactor installations, decommissioning, decommissioning concept, decommissioning option, end state, rules and regulations, safety manuals, terms and definitions, digital decommissioning support.

EDN: CZMTTP