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Opportunities and Prospects of Large-Scale Nuclear Energy

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The article presents an assessment of the current trends in the global energy development, taking into account the heightened perception of anthropogenic factors impact on the ongoing climate change. This review is carried out from the position of identifying the place and role of nuclear energy in the energy structure for the long term. The importance of this issue is gradually increasing both for solving current energy problems due to the expansion of restrictions on traditional energy sources using, and for the future, due to the increasing sensitivity of public opinion to environmental problems in various aspects of their manifestation. Despite the fact that a skeptical attitude towards the development of the nuclear energy complex remains on the world agenda, the use of nuclear energy that promises to minimize the negative impact on environmentally important natural processes. The development of the Russian nuclear energy complex is considered in more detail, taking into account the potentially negative factors of the organization of a closed fuel cycle, which is due to the fact that the problematic issues of its radiation safety have not yet been fully analyzed. The article presents a description of the current state of nuclear energy in Russia. The development of a nuclear energy system potential based on the coordinated use of fission reactors for energy production and hybrid fusion reactors for the production of artificial fuel for fission reactors from thorium raw materials, i.e. a new structural organization of fuel cycle closure of the nuclear energy complex is considered, capable of ensuring its acceptability for large-scale development.

Key Words: nuclear power, fission reactor, fusion reactor, hybrid fusion reactor, thorium, uranium, fuel cycle closure.

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Risk and Benefit Analyses of Thorium and ^{233}U Introduction in VVER-1200 Fuel Cycle in the System with Fusion-Fission Reactors

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The limitation of economically suitable supplies of natural uranium challenges us to improve the existing structure of the nuclear energy system and its fuel cycle. With the available supplies of natural uranium, it becomes practically impossible to significantly increase the capacity of nuclear power plants in the existing system, the basis of which are thermal reactors operating in the open nuclear fuel cycle (ONFC). To solve this problem, the paper considers the introduction of a fusion fission reactor (FFR). The FFR blanket produces ^{233}U, which is subsequently used in thermal reactors. Fuel compositions of VVER-1200 reactors with the addition of ^{233}U and thorium allowing the reduction of natural uranium consumption and/or the use of available raw materials in the form of depleted and regenerated uranium are investigated. Using the example of the Russian Nuclear Power System (NPP) development scenario in the SFC with VVER-1200 reactors for all fuel options the required operating amounts of ^{233}U and the corresponding FFR fraction were determined. The risks of introducing thorium into the nuclear fuel cycle were reviewed.

Key Words: fusion-fission reactor, thermal reactor, uranium, thorium, nuclear energy system, risk.

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Raw Material Effect on the Characteristics of the Fiction and Fusion Reactor System

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The most topical problems that are considered when development of nuclear energy in the long term is researching are limited resources of natural uranium and the high level of radioactivity in the fuel cycle. Basically, nuclear energy development strategies are based on an open fuel cycle with natural uranium consumption and on closed fuel cycle with fast reactors. In the latter case, the reprocessing of highly active spent fuel is required. Both of these problems can be solved through the implementation of hybrid fusion reactors, where ^{233}U is produced, and further this uranium is used in thermal reactors. Various options for fuel compositions of VVER-1000 reactors are considered in the paper. These compositions allow reducing the consumption of natural uranium and/or using existing raw materials in the form of depleted and regenerated uranium. The principal difference between the studies presented in this article is the consideration of options in which uranium-233 obtained from hybrid fusion reactors is used in thermal fission reactors together with depleted and regenerated uranium.

Key Words: uranium, thorium, nuclear power, closed fuel cycle, hybrid fusion reactor, regenerated uranium, system research.

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Study on Minor Actinide Incineration in Molten Salt Reactors with Uranium or Plutonium Loadings

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The paper is about use of uranium wastes that left after regenerated fuel purification from ^{232}U in molten salt reactor (MSR) in application of minor actinide burning. A comparison of the uranium and plutonium loadings in MSR were performed. The calculation results show that uranium MSR loading is the possible option that allows (1) to ensure the higher rate of MA burning during the initial period of MSR operation and (2) to increase the effective delayed neutrons fraction. The presented calculations are of an estimated nature, and first of all show the possibility of effective use of uranium from the waste stream of the enrichment cascade as a fissile material.

Key Words: closed nuclear fuel cycle, dual cascade, enrichment, isotope separation, minor actinides, molten salt reactor, separation cascade, uranium recycling, used nuclear fuel.

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Fuel Cycle of the LiF—BeF_2 Molten Salt Actinide Burning Reactor

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The general principles of the LiF–BeF_2–AnF_n (where An – Pu, Np, Am, Cm) fuel salt cleanup from fission products for the molten salt reactor are formulated: (1) fast (one month) recycle of actinides to the reactor circuit after cleaning from fission products, (2) removal of lanthanides (neutron poisons) from the fuel salt with a period not more than 1 year, (3) main operations include only molten salt and liquid metal with relatively low concentrations of fissile nuclides, (4) avoid operations with isolated pure fraction of fissile materials. The main R&D needs of the fuel LiF–BeF_2–AnF_n salt cleanup for molten salt reactor, are determined: the degree of the fuel salt purification from certain groups of fission products; limits on actinide losses to waste; safety limits on fissile materials loading into pyrochemical processing units with fuel salt;

regulations for the operation of taking part of the fuel salt for processing with simultaneous loading of the same amount of fuel salt purified from fission products.

Key Words: actinide fluorides, fission products, fuel salt, lithium and beryllium fluorides, molten salt reactor, processing reductive extraction.

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HTGR Neutronic Characteristics Calculation for the Software Package MCU-HTR Verification

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This work is devoted to the preparation of materials for the certification of the MCU-HTR software package, which implements the Monte Carlo method, as applied to the calculation of the neutronic characteristics of high-temperature gas-cooled reactors and critical assemblies with fuel based on microparticles with a multilayer coating. A verification matrix has been developed for certification of the MCU-HTR software package. All calculations were carried out using a special methodology that allows taking into account the double heterogeneity of fuel placement in the core, and a detailed full-scale description of the geometry of the calculated objects. An obtained result set is unique in terms of completeness and content for the practice of calculating such systems. The results of the calculation of reaction rate distributions, control rods efficiency, neutron-physics functionals for start-up experiments of the HTTR reactor with prismatic fuel blocks and the pebble bed HTR-10 reactor, as well as experimental configurations implemented on the VHTRC critical assembly are presented. Discrepancy between the calculated values and experimental data for the indicated characteristics were estimated.

Key Words*:* high temperature gas-cooled reactor, software package, neutron calculations, experimental data, verification matrix, Monte-Carlo method, verification, MCU-HTR.

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Computational Analysis of the Possible Location of Ionization Chambers in Fast Reactors

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Algorithms and codes have been developed to help select the locations of ionization chambers that record reactor power changes. For each time step, the non-stationary neutron transport equation is solved and the reactivity changes are found using the inverse solution of the kinetics equation in the point approximation. Each step determines a set of the best points for moment in time where the reactivity deviations from the reference results are minimal. At the end of the whole calculation such points with maximum total occurrence are selected. Then studies with different scenarios of motion of the control rods are carried out with the selected variants of the ionization chamber location taking into account design constraints. The work is based on the assumption that the equations of point kinetics are used in the processing of ionization chamber readings at the operating NPP. The limiting factors of using this methodology are the inapplicability of point kinetics at high reactivities and the potential impossibility of reconciling the contradictory results obtained when modeling different scenarios of control rod motion. Compromise decisions must be made by experts. The algorithms and codes developed and verified on the BN-600 test model can be applied both to BN-1200 and BREST reactors and to reactors of other types.

Key Words: ionization chamber locations, point kinetics, reactivity, control rods, delayed neutrons, fast reactors, diffusion approximation, ShIPR Intellectual Code System.

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Research of the Neutron Source Effect on the Results of CEFR Start-Up Tests

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This paper presents the results of the calculation analysis of the activation measurements at the research fast reactor CEFR. The possibility of the influence of the experimental configuration on the obtained results is investigated. It was shown that a neutron source located in the core can make a certain perturbation in the experimental results. The magnitude of this effect depends on the degree of subcriticality of the reactor.

Key Words: reactor CEFR, code Serpent, project CRP, fast reactors, physical start-up, simulation of reactor experiments, neutron source, reaction rate.

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Calculatons of Axial Distributions of the Reaction Rates in the Framework of the IAEA Project on the Analysis of the Startup Experiments at the CEFR Research Reactor

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The computational analysis of the axial distributions of the fission reaction rates of ^{235}U, ^{238}U, ^{237}Np, as well as the neutron capture reaction rates for ^{197}Au, is performed. Calculations are made using JARFR diffusion code with the BNAB-93 nuclear data and the Monte Carlo code SERPENT with JEF3.3 nuclear data. Their comparison with experimental data is presented. The work was carried out within the framework of the IAEA project on the analysis of launch experiments at the Chinese fast research reactor CEFR.

Key Words: fast neutron research reactor CEFR, software package, neutron physics calculations, experimental data, verification matrix, Monte Carlo method, verification, JARFR, SERPENT.

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Use of Eutectic Na–Tl Coolant in Small Fast Reactor

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One of the main safety problems of fast reactors with sodium coolant is its high chemical activity in interaction with water and air. This requires complex safety systems, fire extinguishing, diagnostics and complicates the design of steam generators and pipelines. The use of Na—Tl eutectic as a coolant for the first and intermediate circuits of a modular fast neutron reactor can alleviate this problem. Eutectic sodium – thallium (92.9%Na—7.1% Tl) has a much lower chemical activity and prevents or extinguishes ignition formed by the surface layer. Also, this coolant has a higher boiling point and a lower melting point in comparison with sodium. Thallium has not previously been used as a core material, and its introduction into the coolant can change the neutron spectrum in the core and the neutron-physical characteristics of the reactor. In this paper the effect of replacement of sodium coolant on the eutectic sodium – thallium on the neutron-physical characteristics of a modular fast reactor with metal fuel is studied. In addition, the influence of possible changes in the isotopic composition of thallium is studied. The activation of thallium isotopes in the coolant is simulated and their contribution to the radiation source is investigated.

Key Words: eutectic coolant, fast reactors, reactivity effects, SMR, void reactivity effect, coolant activation.

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Peculiarities of Changes in the Isotopic Composition of Pilot Fuel Elements of a VVER-SKD Reactor under Successive Irradiation in the Fast and Thermal Neutron Spectrum

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The paper considers the methodological aspects of experiments planning on the irradiation of experimental fuel rods of a VVER-SKD reactor and analyzes the features of the concept of sequential irradiation in fast and thermal reactors. It is shown that the ^{149}Sm accumulated at the irradiation in BOR-60 plays the role of a burnable absorber at the initial stage of irradiation in the reflector cell of the IR-8 reactor. This absorber is quickly burns out due to large absorption cross section and its concentration gradually comes to equilibrium. Burnup of ^{149}Sm leads to a gradual increase in the linear load of the irradiated experimental fuel rod, followed by a decrease and reaching the equilibrium irradiation regime. This effect must be considered when planning the irradiation regime for experimental fuel elements of the VVER-SKD reactor.

***Key Words*:** BOR-60 fast neutron research reactor, VVER-SKD, IR-8 reactor, samarium-149, irradiation.

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Effective Fraction of Delayed Neutrons in Molten Salt Reactor with Circulating Fuel

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In molten salt reactors (MSR) with circulating liquid fuel, delayed neutron precursor can escape from the core and be distributed throughout the entire reactor circuit. Taking into account the decrease in the effective fraction of delayed neutrons associated with the motion of the emitters is important for further analysis of the reactor kinetics. A new approach is proposed to account fuel circulation in the fuel circuit based on a combined calculation by the Monte Carlo and finite volume methods, which was implemented using the interface between the codes SERPENT and OpenFoam as OFSI code. Calculations of the effective delayed neutrons fraction for the 10 MWt MSR reactor circuit with a cavity type cylindrical core and the fuel Li, Be, Pu/F salt mixture are performed. On the basis of the results obtained, conclusions were drawn about the applicability of the technique and the ways of its further development were outlined.

Key Words: molten salt reactor, calculation methods, molten salt fluorides, circulating liquid fuel, effective fraction of delayed neutrons.

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Radiolysis of the Coolant During the Decomposition of a Hydrogen Solution Near the Saturation Temperature in VVER Reactors

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The calculations of the radiolysis of the VVER coolant during the release of hydrogen from the liquid phase into gas-vapor bubbles are performed. It was found that the decrease in the concentration of hydrogen in the liquid phase occurs until the flow of hydrogen into the bubbles is comparable to the production of hydrogen due to radiolysis. In this case, a jump in the hydrogen concentration leads to a sharp restructuring of the entire picture of the time dependences of the radiolysis products concentrations. This is especially true for the behavior of the O_2 and H_2O_2 concentrations, i.e. main oxidants of zirconium cladding of fuel elements. It is shown that the local oxygen concentration relative to the permissible value when the reactor operates at a power >50% N_{nom} can increase more than 1000 times. Hydrogen peroxide H_2O_2 demonstrates a similar behavior. However, the increase of its relative concentration is somewhat more than one order of magnitude. A significant local increase of the concentration of oxidizing products of radiolysis O_2 and H_2O_2 in the coolant, apparently, is the main reason for the appearance of a white deposit on the cladding of VVER fuel elements near 12-th spacing grid.

Key Words: radiolysis, corrosion, fuel rod cladding, coolant, water chemistry mode.

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Modeling of Tritium Behavior in Molten Salt Reactor

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Liquid fuel reactor based on lithium and beryllium fluorides produces significant amount of tritium as a result of nuclear reactions of fuel salt components with neutrons. At high operating temperatures (600—750 ^{o}C**)** tritium can permeate through metal walls of primary circuit and effect ecological situation. In this paper methodology is developed and verified to determine tritium distribution for this type of reactor. Method is based on differential equations of tritium mass balance and considers redistribution of tritium chemical compounds, caused by redox potential. Verification of method was performed on experimental data from 10 MW(t) LI, Be, Zr, U/F Molten Salt Reactor Experiment (MSRE) operation (ORNL, USA).

Key Words: molten salt reactor, lithium and beryllium fluorides, primary circuit, fuel salt, tritium.

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Influence of Construction Delay Risks on the Cost-Effectiveness of NPP Projects of Various Capacities

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The risk of delaying of the NPP construction and the associated increase in costs are becoming a typical phenomenon for the global nuclear power industry. The calculated results of accounting for such risky situations during the construction of NPP units of various capacities are presented in the form of changes in the performance indicators of the project — the payback period and income from the implementation of the project. The greatest impact of financial and investment risk caused by bank interest on the loan was revealed. Doubling the construction period leads to unprofitability of projects.

Key Words: NPP of small, medium and large power; risks, schedule of NPP construction; project efficiency, systems approach.