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Artificial Intelligence in the Field of Atomic Energy Usage — Existing Possibilities and Perspectives

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The article presents an overview of the existing domestic and foreign practices of using artificial intelligence technologies at designing, safety assessment and operation of nuclear facilities. The concept of artificial intelligence is interpreted in a general way, covering a whole range of information technologies and software and computational methods.

Today, there is a growing interest around the world in the use of artificial intelligence (AI) technology in almost all technological areas. Nuclear energy, as an extremely science-intensive industry, has its own characteristics compared to the areas of mass application of AI (medicine, economics and finance, marketing, design, logistics, traffic analysis, etc.). The correct setting of tasks for the use of AI in the nuclear industry requires a clear definition of the possibilities and limitations of the use of AI. In this study, the authors analyze various aspects of the use of AI for designing, safety assessment and operation of nuclear reactors. The main attention is paid to the opportunities for the development of the industry and improving the efficiency of technological processes.

Examples of the development and testing of methods based on AI in the field of activity of OKB Gidropress JSC are given. Conclusions are drawn about the promising areas of using AI as a modern information technology, and as a development direction for the long term.

Key Words: artificial intelligence, machine learning, nuclear energy perspectives.

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Machine Learning Method for Calculation Model Validation of Non-stationary Xenon Processes in the VVER Reactor Based on the Variables Separation Algorithm

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Work presents a method for performing validation of the KORSAR/GP code in part of the mathematical model of non-stationary xenon processes in the VVER reactor, based on the separation of spatial and time variables. The data obtained from different NPPs with high-power VVER during experiments to study the spatial power distribution under conditions of non-stationary poisoning of the reactor caused by the action of various controllers are used.

The model is based on the classification of means of influencing reactivity according to the type of the influencing to power variable, which undergoes significant changes for the process as a result of this impact. Non-stationary processes of xenon poisoning are considered, in which CPS control rods are involved, water exchange operations with a change in the concentration of boric acid, as well as both of the above methods, both in the presence of a change in the neutron power of the reactor, and while maintaining its constant value.

A machine learning method based on regression analysis was develop, which makes it possible to estimate the error in calculating the energy release field parameters under conditions of spatial, time, and both feedback on the xenon concentration and controllers. Based on the processed experimental data, a training array was form, which was use for machine learning of this model. As a result of the work of the created algorithm, an error estimate is made for the calculation code model, taking into account the separate effect of various means of changing reactivity in a given calculation.

Key Words: mathematical model, machine learning, regression analysis, non-stationary poisoning, reactivity change, space-time dependence of power distribution, code KORSAR/GP, validation.

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MATADOR Code for Calculating Local Inhomogeneous Heat and Mass Transfer Processes in Rod Bundles.

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The article provides a brief description of the MATADOR code. The code is based on a sub-channel (cell-by-cell) procedure for analyzing thermal-hydraulic processes in fuel rod bundles cooled by single-phase coolant in the presence of local inhomogeneities. The mathematical model of the code is based on three-dimensional equations of mass, en-thalpy and momentum transfer writ-ten in integral form. The transfer equations are solved by a numerical method based on the SIMPLE algorithm.

Testing of the code on test problems is presented and a comparison of the calculation results for the MATADOR and TEMPA-1F codes is given. The initial stage of validation of the MATADOR code is performed and the calculation results of rod bundles cooling for various geometries are presented. The estimation of the computer time spent on solving the test problem using the MATADOR and TEMPA-1F codes was made.

Key Words: sub-channel (cell-by-cell) computer code, heat and mass transfer, rod bundles, code testing, validation.

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Experimental Investigation of Condensation Water Hammers in Emergency Injection Pipeline of Pressurizer

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The article presents brief description of experimental facility, methods of experiments, and also results of experimental studies of condensation water hammers in emergency injection pipeline of pressurizer in different conditions. The experiments were performed at OKB Gidropress JSC using “Water Hammer” test facility with models of emergency injection pipeline for different NPPs in scale 1:1. Experimental results present that obtained values of condensation water hammers of low intensity do not affect the operability of emergency injection pipelines.

Key Words: experimental test facility, pipelines, pressurizer, condensation water hammer.

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Application of Machine Learning for Early Diagnostics of Accidents at NPP with VVER. Concept of Interactive Guideline on Accident Management

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The concept of using machine learning models for diagnostics of accidents was proposed more than 20 years ago. However, due to a number of factors, until recently its application and development were of low profit and in some cases an impossible task. The improvement of computer hardware and software, and also targeted funding of research projects on the use of artificial intelligence (AI) caused an exponential growth in the number of research projects with application of machine learning (ML) in the nuclear industry in the last 2—3 years. Concepts of ML in the nuclear industry began the transition from fundamental scientific research to industrial application. In this regard, there are high hopes for ML application for early diagnostics of accident sequences using ML models.

Within the framework of the research, the concept of virtual guidelines on management of an beyond design basis accident (BDBA) has been developed, which allows performing early diagnostics and giving recommendations to the operating personnel on accident management actions in the real-time mode. The concept is based on a system of interconnected ML models, taking into account data from the nuclear power plant (NPP) on equipment operability and thermal-hydraulic parameters available at the considered stage of an accident. According to the concept, a number of artificial neural network models have been developed, for instance, to determine the type of initiating events of the accident, location and diameter of the leakage and the time of reaching the maximum design limit of the fuel rod damage. The sets of the initial data obtained on the basis of numerical studies for VVER-1000 using computational code SOCRAT-V1/V2 were used for training models.

Key Words: machine learning, artificial neural networks, diagnostics, guidelines, accidents, VVER, SOCRAT.

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Water-Chemical Regime of VVER-SKD and the Main Systems of its Provision

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For the operation of an innovative nuclear power unit with a direct-flow water-cooled power reactor for supercritical parameters of VVER-SKD steam, a non-reactive neutral water-chemical regime (VCR) is proposed. To justify the choice, we used the analysis of the results of long-term operation of more than 120 direct-flow supercritical pressure power units in the thermal power industry of Russia, the experience of operating nuclear installations AMB-100 and AMB-200 with nuclear steam overheating on weakly alkaline and neutral-free VCR, as well as VCR single-circuit nuclear power plants on subcritical parameters of the coolant. The values of feed water quality indicators for the reactor, as well as indicators of the quality of make-up water, are recommended. The work of the main systems for providing a given water-chemical regime is considered. The main scientific and technical problems requiring additional research are identified.

Key Words: VVER-SKD direct-flow power reactor, water-chemical regime, water-chemical regime support systems.

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The Analysis of Fuel Rods Cooling Conditions in Elementary Cells of Different Design VVER-SKD Fuel Assemblies

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The article sets the task of computational analysis of fuel rods cooling conditions in the elementary cells of the VVER-SKD fuel assemblies. A simplified calculation technique is described which allows rapid analysis of fuel element cooling conditions in fuel assemblies. The proposed method was tested using TEMPA-SC computer code. The results of a computational analysis of fuel rods cooling conditions in elementary cells of various design fuel assemblies for a single-circuit VVER-SKD nuclear power plant are presented.

Key Words: VVER-SKD, fuel assembly, fuel rods cooling, water at supercritical pressure, TEMPA-SC code.

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Assessment of the Proposed Core Baffle Design Changes in Order to Decrease Maximum Temperature of the Metal and Increase its Operating Life

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The article provides an assessment of proposed core baffle design changes in order to decrease maximum temperature of the metal and increase operating life of this component of the internals. The possible types of changes or additions were proposed as the ways of improving the current design of the core baffle and increasing heat removal in its problem zones. In this work calculations of energy power and temperature fields in core baffle cross-section and for various core baffle designs with basic equilibrium fuel loading for conditions of operation at nominal power were conducted. Results of the calculations for the proposed ways of design change showed a significant decrease of the core baffle metal temperature and, consequently, decrease of void swelling rate, which, in its turn, leads to increase of service life.

Key Words: VVER, reactor internals, core baffle, void swelling, temperature distribution over a cross-section, cooling channels, displacers, additive technologies.

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Post-Test Calculations of Experiments at Reflooding Test Bench for Conditions with Loss of Spent Fuel Pool Cooling Using Code KORSAR/GP

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During this research, the set of experiments, which can be applied to the conditions with the long-term loss of the spent fuel pool cooling, were performed on the integral modernized experimental facility “Reflooding test bench”. The obtained experimental data were used in post-test calculations. The results of these calculations cover major simulated thermohydraulic processes of cooling of fuel rods in the spent fuel pool. The close quantity agreement between the calculation results and experimental data was obtained for the following phenomena: the beginning of coolant boiling in the column, the beginning of fuel rods uncovering (temperature increase), as well as reaching the maximum temperature by fuel rods.

Key Words: NPP, spent fuel pool (SFP), VVER, SF, FA, KORSAR/GP.

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Estimation of the Residual Life Time as a Measure of the Preventive Actions Effectiveness in the Problems of Aging Management

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Modern lifetime management methodologies [1, 2] are based on the principle of preventing the limit states of equipment during the assigned resource by organizing control and carrying out preventive actions to mitigate operating conditions and reduce the intensity of degradation processes in the equipment material. The resource management of existing equipment is associated with measures to improve operating conditions, to change the state of the element (cleaning, repair, replacement, ...), with the frequency of control and its volume, and the cost of ongoing actions. The effectiveness of lifetime management in terms of the considered aspects is closely related to the evaluation of the effectiveness of possible actions. A technique for evaluating the effectiveness of some aging management measures based on the value of the residual lifetime is proposed. As an example, a SG heat-exchange assembly is considered. The issues of probabilistic assessment of the residual life are considered, taking into account the uncertainties associated with the incompleteness of control, changes in operating parameters, etc. The methods are based on the use of the Weibull distribution for the relative number of muted heat-exchanged tubes (HETs). When substantiating the admissibility of using the Weibull model, the technique for constructing confidence intervals for statistical estimates of muted HETs was used. The techniques developed by the authors were tested on the data of the 1PG-1 of the Kalinin NPP.

Key Words: lifetime management, aging processes, residual resource, effectiveness of measures taken, steam generator heat exchange tubes, confidence intervals.

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Statistical and Parametric Analysis of Ultrasonic Testing Results in Order to Obtain Input Data for Probabilistic Fracture Analysis of VVER Pressure Vessel

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The article gives results of the statistical treatment of data on ultrasonic testing of WWER 440 reactor pressure vessels, including conservative schematization of the inspection results and construction of the Weibull distribution of depths for schematized defects. Based on the two-parameter probability of detection curve, a mathematical model has been developed for the transition from the detected depths distribution and defects density to the corresponding predicted quantities. A series of parametric calculations made it possible to isolate the maximum of fracture probability in the area of realistic values of the probability of detection curve, thereby allowing the construction of a conservative model for deficiency of welds and base metal of the WWER-440 reactor pressure vessel.

Key Words: reactor pressure vessel, ultrasonic inspection, statistical analysis of flaws, fracture probability.

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Post-Test Calculation Results of the Experiment “Shut-off Valves Closing During Natural Circulation” at the PSB-VVER Facility

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The report presents the scenario of the experiment “Shut-off valves closing during natural circulation” at the PSB-VVER facility, as well as the results of post-test calculation at the KORSAR/GP software package. The influence of heat and hydraulic losses on the results of the experiment is estimated.

Key Words: ETHARINUS, VVER, PSB-VVER, asymmetric single-phase natural circulation, heat removal loss, reactor plant cooling, experimental database, pretest modeling, KORSAR/GP, validation of calculation codes.