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Safety Assessment of Nuclear Reactors in Maneuvering Modes as a New Class of Computational Problems

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The article presents an overview of domestic and world practice of developing tools and methods aimed at safety assessment of nuclear reactors taking into account operation in maneuvering modes. Specific examples show that the historically established solution of such a problem was actually carried out for the condition of significant limitations and assumption in terms of the duration and speed of maneuvering, as well as the dynamic behavior of the energy release field. At the same time, a number of factors of fundamental importance for the safety of a nuclear reactor during power maneuvering are determined and substantiated. As a result of the review, a practically significant conclusion is made that the fulfillment of modern requirements for maneuvering modes forms a new class of problems in the field of safety assessment, relating to a higher level of computational complexity. The solution of such problems is performed for a significantly increased volume of initial data, which allows for a full consideration of the influence of maneuvering on safety.

Key Words: safety assessment, maneuvering modes, mathematical modeling, dynamic initial state, operator error, local protection, artificial intelligence.

EDN: YEDZZC

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Tests of the Coolant Natural Circulation Mode at Power Units with VVER-1200

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According to the results of tests on four power units with VVER-1200, the transition to the natural circulation mode with the termination of forced circulation of the coolant and the possibility of achieving the maximum allowable reactor power level for a given test, were confirmed. Changes have been made to the test success criteria to ensure a more correct assessment of the achievement of test objectives. The expediency of introducing the modeling methodology into the practice of physical and dynamic tests during the commissioning of NPP power units with VVER is shown. Data were obtained for validation of the computational model used in calculations to justify the safety of the AES-2006 project, recommendations were given for the introduction of algorithms into the in-core monitoring system that provide more representative control in modes at low power levels and in the mode of natural circulation.

Key Words: natural circulation, reactor, steam generator, reactor power, the cold leg, the hot leg, fuel assembly, in-core monitoring.

EDN: LHEYFD

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Method for the Estimation of Power Density Nonuniformity Coefficients of VVER RP in Dynamic Processes during Power Maneuvering

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The materials of the report relate to a series of works carried out at OKB Gidropress JSC as part of methods, algorithms and programs development for numerical modeling of processes at the VVER reactor, taking into account the features of a number of physical processes under conditions of maneuvering modes. This paper examines the problem of pin-by-pin irregularity factors calculation of the reactor energy field, the main feature of which is the correct transition from the model for calculating neutron-physical characteristics to the model for calculating dynamic processes. For coupled modeling of processes in VVER reactor plants, the KORSAR/GP software package is used. To calculate the neutronic characteristics of the core, the SAPFIR_95&RC program is used.

The work proposes a universal approach for estimating the pin-by-pin irregularity factors for the most heated fuel rods under the conditions of a dynamic process of power maneuvering. The dynamic calculation model is detail down to the level of the fuel assembly and the assessment of extreme parameters is carrying out using the “hot channel” model. The proposed approach is base on the construction of a multi-dimensional phase space in which the pin-by-pin irregularity factors presented as a function of the neutron power of the core and the position of the control rod groups. The basic nodes for constructing the space selected based on the results of a stationary neutronic calculation with detail down to the level of an individual fuel element. When carrying out a related dynamic calculation for the maximum value of the pin-by-pin irregularity factor, a phase trajectory constructed that describes its increment when the reactor power changes and the movement of control rod groups involved in the transition process.

Based on the results of the work, a number of practically significant conclusions are drawing regarding the most correct usage of the “hot channel” model for carrying out local energy release conservative estimation during maneuvering. Perspective directions for the development of the used approach for modern types of VVER reactors also analyzed.

Key Words: safety assessment, load following, mathematical model, pin-by-pin irregularity factor, multi-dimensional phase space.

EDN: YTCAVH

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Development of Virtual Sensor Models Using Convolutional Neural Networks

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The study is devoted to the creation of virtual sensors operating as a complex data approximator based on the instrumentation and control (I&C) system available at a nuclear power plant (NPP). The proposed approach allows simulating NPP parameters that are difficult or impossible to measure using existing technical means. In particular, the neural network models developed within the framework of this study are designed to determine such important parameters for accident management as the coolant level in the reactor (unlike the current I&C capabilities, which imply discrete measurement of the value, machine learning models allow obtaining its continuous distribution over time) and the maximum temperature of the fuel rod cladding (until now, there was no possibility of determining this parameter).

At this stage of the study, the neural network models of the two types under consideration were trained based on data obtained as a result of numerical modeling using the SOCRAT-B1/B2 integral thermal-hydraulic code. In the future, it is planned to further train the models on real data from the I&C.

Key Words: artificial neural networks, CNN, virtual sensors, instrumentation, I&C, VVER, SOCRAT.

EDN: TLRNVI

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Diagnostics of Instrumentation Sensors Based on Models of Auto-Associative Neural Networks

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The reliability of the results of any diagnostics largely depends on the data on which it is based. The principle "false input — false output" applies. In other words, if you try to analyze incorrect data (any sensor of the instrumentation and control system (I&C) has failed), the conclusions will be false. Therefore, during the operation, it is necessary to continuously verify the sensors and equipment based on data from the I&C.

To solve this problem, in addition to generally accepted methods, there are many approaches using machine learning (ML). As part of this study, a model for early diagnostics of I&C based on auto-associative neural networks (AANN) [1] was developed to search for anomalies in the operation of sensors. At the current stage of the study, the models were trained on data obtained as a result of numerical simulation using the star-ccm+ (thermohydraulic code in approximation of distributed parameters). In the future, it is planned to train the models on real data obtained using sensors.

The study is part of a series of works carried out at OKB Gidropress JSC within the framework of analysis of the possibilities and prospects for the application of machine learning methods to solve current problems of nuclear power plant management in emergency situations [2, 3].

Key Words: artificial neural networks, AANN, diagnostics, sensors, instrumentation, VVER, CFD.

EDN: MGLBZZ

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On the Methodical Error of the Jasper Radial Shield Experiment Calculation Based on DDL-schemes of the Discrete Ordinate Method for the Case of the Source Spherical Shape

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When solving the problem of validation of shielding codes, data from the JASPER experiment (IAEA SINBAD- 2010 database) can be effectively used. The JASPER experiment models the shielding of a fast neutron reactor (steel, boron carbide). But a groupwized source of neutrons with an angular distribution of flux is determined in a non-standard way — not volumetric, not point, but releasing from the emitting surface and at the same time releasing into the air (into a weakly scattering medium). When calculating the experiment in a three-dimensional formulation according to the FRIGATE program, a numerical effect was revealed consisting in the fact that the use of numerical DDL-schemes of the DS_N method of discrete ordinates to calculate the radiation transfer from such a surface source leads to an underestimation of the calculated results in the surface layers of the shielding composition to 50% for a spherical source. In this work, to reduce the methodical error of the numerical scheme, it is proposed to use the well-known technique of representing the surface source as the volumetric one in a small area. The proposed method has been tested on a model problem and has proven to be very effective.

Key Words: FRIGATE, DS_N method, surface source, TDMCC, testing, validation, protection experiment, numerical schemes, airborne radiation transfer, fast neutron reactor.

EDN: BZWKCE

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Updating the Scope of Application of the JARFR Software Package for Neutron Physics Calculations of Fast Sodium Reactors

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The results of verification of the JARFR code [1] are presented to update its scope for calculating the current state of the BN-800 reactor core using measurement results. The results of the measurements of the neutron-physical characteristics of the BN-800 core for 8—11 microcampaigns were used for code verification. It is shown that the deviations of the calculation results from the experimental ones do not exceed the passport error provided by the JARFR code in the field of acceptable application.

Key Words: BN-800, fast sodium reactors, simulation of reactor experiments, JARFR software package, distribution of energy release, calculation of ^{140}La activity.

EDN: DICGFC

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Features of Uncertainty Analysis for the Purposes of Probabilistic Safety Analysis

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Uncertainty analysis (UA) is one of the main tasks within the framework of probabilistic safety analysis (PSA) of level 2 [1, 2] solved taking into account the characteristics of the uncertainty of reliability parameters of the reactor plant elements: failures of elements and systems, events of severe accidents (SA) etc.

Many reactor system failures depend on SA scenario. The assessment of such probabilistic indicators of emergency conditions as the probability of dependent failures is not possible without additional calculated estimates. Therefore, PSA of level 2 is usually accompanied by a series of deterministic supporting calculations with corresponding UA [3].

In this study, a methodology for assessing the probability of dependent failures of reactor plant systems has been developed based on recommendations for assessing uncertainties of safety analysis calculations, taking into account the tasks of PSA of level 2.

Key Words: PSA, uncertainty analysis, statistic methods, VVER.

EDN: BTIVVI

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Analysis of Uncertainties in Justification of VVER RP Safety in Terms of Mathematical Optimization

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The article proposes a new method for uncertainty analysis, which consists in using optimization to find the extreme value of the uncertainty boundary under study. The proposed method was also proved in considering an accident with the unintentional closure of a main steam isolation valve on a steam generator pipeline. Optimization was carried out using pSeven program tools.

Key Words: uncertainty analysis, GRS method, KORSAR/GP, deterministic safety analysis, VVER, mathematical optimization, pSeven.

EDN: KYYUAY

UDC 621.039

Sensitivity Analysis for Beyond Design Basis Accidents of the Power Unit with VVER-1200 Reactor

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In accordance with the current requirements of Russian and foreign regulatory authorities, modeling of beyond design basis accidents should be accompanied by analysis of the sensitivity of the calculation results. The result of the sensitivity analysis is the determination of the degree of influence of each of the uncertainties on the studied parameters.

As part of the work done, an approach was proposed to conduct sensitivity analysis using the SUSA program and the TRAP-KS software package on the example of beyond design basis accidents “Disconnection of various numbers of main circulation pump sets with common cause failure of programmable automation systems”, “Loss of vacuum in the condenser or other cases leading to turbine shutdown with common cause failure of programmable automation systems”, “Loss of non-emergency AC power supply to auxiliary power plant equipment (NPP blackout) with common cause failure of programmable automation systems”. The considered modes were selected based on the condition of reaching the maximum values in them up to the acceptance criteria: the maximum temperature of claddings, the maximum pressure of the primary and secondary circuits, respectively.

The results of the sensitivity analysis presented in the paper are as follows:

— values characterizing the spread of criterion parameters (maximum temperature of claddings, maximum pressure of the primary and secondary circuits);

— Spearman rank correlation coefficients;

— density plots and distribution plots of calculation results.

The approach considered in the article makes it possible to take into account the current requirements of international and Russian supervisory authorities, and in the future, it can be extended to all NPP projects with VVER reactor plants. Also, this approach can be further applied for beyond design basis accidents with emergency protection failure (ATWS) and common cause failure of an intermediate circuit.

Key Words: sensitivity analysis, design extension conditions, VVER, SUSA, uncertainty, acceptance criteria.

EDN: KZKJTN

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Analysis of Thermal Hydraulics of Various Designs of Fuel Assemblies in a Two-Way Version of the VVER-SKD Reactor

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The paper presents a brief overview of the developed fuel assemblies of foreign and domestic projects of reactor plants with supercritical parameters of the coolant. The results of the analysis of thermal hydraulics of various designs of fuel assemblies in the two-way core of a single-circuit VVER-SKD installation using the TEMPA-SC program are presented. Preliminary calculations revealed the existence of an irregularity in the distribution of the coolant temperature across the cross-section of the fuel assemblies, resulting due to different hydraulic diameters and through sections of the central and peripheral cells. To eliminate this effect, options for changing the design of the fuel assembly casing were considered. The necessity is shown of conducting coupled neutron-physical and thermohydraulic, as well as strength calculations of fuel assemblies to confirm the detected conformities and the choice of a feasible configuration.

Key Words: VVER-SKD, fuel assemblies, TEMPA-SC, supercritical pressure, thermohydraulic characteristics.

EDN: NTXNBO

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Assessment of Using Helicoidally-oriented Cooling Channels in VVER-1200 Core Baffle Design

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The article presents a further elaboration of the assessment of influence of helicoidally-oriented cooling channels usage in VVER-1200 core baffle design on the metal temperature decrease for an extension of service life of this reactor internals component.

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

EDN: FGVBVG

UDC 620.193.4+519.216.3

Evaluation of Reliability Indexes of Reactor Coolant Circuit Elements for the Damage Accumulation Model

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Operational monitoring data based on continuous improvement of monitoring of aging, degradation and damage processes of the reactor plant equipment and pipelines indicate real rather than planned loading, therefore the residual resource prediction becomes more realistic. Taking this fact into account, including errors in the methods for calculating loading parameters, allows us to assess the technical state of an equipment element as probabilistic reliability characteristics in compliance with the principle of conservatism. The paper considers issues of assessing the reliability indexes of reactor coolant circuit (RCC) elements, including residual lifetime index, as the basis for implementing aging management of these elements. The methodology developed by the authors is applied to the elements of two power units of the reactor plant with VVER-1000, operating under conditions of fatigue damage accumulation. Loading characteristics and damage values are obtained based on the readings of the APCS sensors used by SAKOR. The proposed methodology complements the established computational traditions in this area (calculation methods) by calculating probabilistic reliability indicators: probability of failure-free operation (PFO), intensity and frequency of failures, average residual life.

Key Words: fatigue, damage accumulation process, reactor coolant circuit, residual lifetime, probability of failure-free operation, failure rate.

EDN: FJXPJT

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Determination of the Thermal State of VVER RP Pipelines using a Neural Network

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The article considers the main aspects of neural network application to predict temperature condition of pipelines and tie-in sections of pipelines in VVER RP. Obtained results show good agreement between predicted temperatures and the actual ones. Examples of processing temperature sensors are shown separately, where physically justified prediction results are obtained. The work is aimed at developing the predictive ability of pipeline resource monitoring systems and expanding its functionality. Also, the developed approach can be applied during processing outer thermocouples' readings obtained during commissioning tests, which can be used in future calculation justification of the equipment strength.

Key Words: VVER, pipelines, temperature field, inverse heat conduction problem, neural network.

EDN: WOWHMS

UDC 539.3

An Efficient Numerical and Analytical Method for Calculation of Near-Boundary Crack Tip Stress Intensity Factors for Probabilistic Analysis of the Equipment and Pipelines Fracture

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Justification of resistance to brittle fracture and probabilistic analysis of the NPP equipment and pipelines fracture require the calculation of fracture mechanics parameters under multifactorial conditions of normal operation and emergency processes. To address this need, novel numerical and analytical approaches have been developed for the calculation of stress intensity coefficients. These approaches are based on the numerical solution of exact integral equations for two-dimensional elastic crack problems, incorporating analytical treatment of stress state disturbances induced by the proximity of the crack front to the body boundary or material interface. Comparison with the exact solution for one fracture mechanics problem has been performed; the accuracy and effectiveness of the proposed approaches have been demonstrated for other fracture mechanics problems with modeling calculation cracks near the surface of equipment or pipelines, as well as in the region of the interface between base metal and welding on.

Key Words: brittle fracture, stress intensity factor, integral equation.

EDN: TFIFNO

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Mathematical Modeling of Contact Interaction in the Roller Bearing of the Steam Generator PGV-1200MR in the Software “LOGOS”

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In nuclear power plants with light water-based VVER reactors, an important element of the primary circuit of the reactor plant (RP) is a steam generator (SG). The steam generator is installed in a box on supporting structures, trusses.

The PGV-1000M support structure is used on many steam generators of the VVER type. The study of the load distribution in the PGV-1000M roller bearing for normal operating conditions, taking into account the flexibility of the support structure, showed that the load between the rolling elements is unevenly distributed: the greatest load falls on the extreme rows of rollers.

Key Words: support structure, PGV-1000M, roller support.

EDN: XOPMBH