Abstracts
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Space-dependent Integral Model of Neutron Kinetics using the Monte Carlo Method

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The article describes a space-dependent kinetics model based on an integral representation of the Boltzmann equation. The method is based on the integration of fission matrices over neutron lifetime step with delayed neutron effect. The concept of criticality coefficient is introduced, which characterizes the properties of a fissile system in a transient process.

The space-dependent kinetics model is implemented as part of the SAPFIR program package. The spectral program GS63 is used to calculate 26-group macroscopic cross sections for the fragments of the nuclear system. The XT26 program is used to calculate the full-scale system by Monte Carlo method.

The calculation of the test problem is performed.

Key Words: space-dependent neutron kinetics model, Monte Carlo method, nuclear reactor.

EDN: EKUBGZ

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Calculations of Three-dimensional Spatial Kinetics Tests of the Benchmark C5G7-TD with Instantaneous Ejection of Groups of Rods Using the SUHAM-TD Code

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The C5G7-TD benchmark was developed within the framework of the NEA OECD project to verify codes solving the non-stationary neutron transport equation without feedback. Description of tests with positive introduced reactivity were added in last C5G7-TD benchmark specifications (v. 1.8, 1.9). The introduction of positive reactivity is achieved by removing groups of control rods from the core. Using the SUHAM-TD code, 3 three-dimensional tests were calculated with instantaneous ejection of three different groups of rods. For each test, the time dependencies of the following quantities were calculated: the total power of the calculated benchmark reactor model, the power of the fuel assemblies, reactivity.

The aim of the work is to draw attention to the possibility of comparing different methods and codes of three-dimension spatial kinetics calculations (with positive introduced reactivity) using the C5G7-TD benchmark and to provide the first calculation results for such a comparison.

Key Words: benchmark C5G7-TD, Surface Harmonics method, neutron transport equation, SUHAM-TD code, tests with instantaneous ejection of groups of rods.

EDN: IHSOQO

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The Channel Reactor Core for the Accumulation of Cobalt-60

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The concept of a uranium-graphite reactor for the production of cobalt-60 (^{60}Co) and other radioactive isotopes is presented, which makes it possible to significantly increase the production rate of ^{60}Co compared to RBMK-1000 by increasing the neutron flux and increasing the number of assemblies with cobalt. The raising in the energy intensity of the core is achieved by reducing the diameter of the fuel rods and placing assemblies with cobalt in the channels of the CPS circuit with a relatively low water temperature. Measures have been taken to prevent cracks in graphite blocks and channel curvature.

Key Words: radioactive cobalt, channel reactor, graphite, specific activity.

EDN: MBFKKG

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Enhancing Methodology for Processing Measurement Results of the RBMK Steam Reactivity Coefficient

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The steam reactivity coefficient is one of the most important neutron-physical characteristics of RBMK-1000 reactors, characterizing the safety of the reactor, largely determining decisions on the formation of the core loading. In this regard, high requirements are imposed on the control of the steam reactivity coefficient. Measurements to determine the steam reactivity coefficient are performed by the indirect method based on the measurements of the reactivity and the reactor parameters with disturbances in the feed water flow rate. Measurements are performed at the energy power level. The article considers an approach that reduces the uncertainty of estimating the steam reactivity coefficient due to the "expert" choice of values of technological parameters used for thermohydraulic calculation and calculation of the steam reactivity coefficient.

Key Words: RBMK-1000, reactivity, void reactivity coefficient, measurement uncertainty.

EDN: OBLTMB

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Temperature Regimes of the Dewatered RBMK Spent Fuel Pool

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The article analyzes the temperature change in the RBMK spent fuel holding pool in an accident with its complete dewatering. For this purpose, a two-dimensional model of the pool was developed, taking into account convection and radiation, implemented in the code TRANS.

When managing beyond design pool accident with loss of coolant, wall cooling with water is used with the connection of mobile equipment. However, under extreme conditions, the cooling system may be destroyed. Calculations have shown that in some cases, the heat sink due to air convection allows maintaining the integrity of the fuel rods shells, concrete and slotted floor beams. A necessary condition is that a sufficient amount of cold air enters the pool volume either through the hole at the bottom or from the reactor room when opening the covers of the slit floor.

Key Words: RBMK, spent fuel holding pool, beyond design accident, pool dewatering, convection, radiation.

EDN: ZDHQHI

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Simulation of Feedwater Deaerator Main Level Controller Functional Failure Mode for VVER-1200 Reactor Unit

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Mathematical simulation is widely applied to solve multiple problems in various branches of science and production including nuclear industry. The simulation tool used herein is the PRIM AES software package developed by National Research Center “Kurchatov Institute”. Designed to simulate normal operating conditions and anticipated operational occurrences at nuclear power plants, this software has wide-ranging capabilities as regards computational analysis of neutronic, thermohydraulic and computerized processes occurring in control systems of nuclear power plants with VVER reactors.

This paper describes computational simulation of a VVER-1200 transient mode caused by level controller failure in the feedwater deaerator, compares simulated data with archive ones, and relies on simulation results to conclude whether the software package operates correctly, and what are the prospects for its further use.

Key Words: mathematical simulation, thermohydraulic calculation, dynamic processes, power unit, control system, functional failure, safety systems.

EDN: FIVWFZ

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Simulation of Radiation Heat Transfer in Relation to the Melt Localization Process in the Core Catcher of VVER-TOI

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This article presents the results of simulation of radiation heat transfer at the ex-vessel stage of the hypothetical severe accident during the localization of the melt in the core catcher of VVER-TOI project. The calculation of radiation heat transfer was carried out by the THERA code of the TSAR application package developed at the National Research Centre "Kurchatov Institute". The article presents a description of the calculation procedure implemented in THERA code, which is based on zonal method taking into account absorption and emission properties of vapour-gas mixture in the calculation domain volume using the SLWSGG model. The paper presents the results of simulation in the form distribution of average densities of net radiation heat fluxes over the particular surfaces of the computational.

Key words: VVER, severe accident, core catcher, radiation heat transfer, melt, corium.

EDN: GBGVEP

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Application of Neural Networks to Obtain Quick Estimates of Accident Consequences in Radiation Emergency Response Problems

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The article is devoted to the implementation analysis of an approach to the application of neural networks for preliminary prediction of the results of accidents calculations. The description of the developed approach is presented, as well as the results of application of the neural network model created and trained within the framework of this approach are demonstrated on the example of an accident "Blackout during the rated power operation at NPP with WWER-1000".

Key Words: emergency response, neural network.

EDN: PKJPMQ

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Application of Neural Network for State Selection in Nuclear Safety Analysis of LOCA

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This article discusses applicability of artificial neural networks for optimization of nuclear safety analysis. Regulatory documents require nuclear safety analysis for beyond-design-basis accidents at nuclear power plants to rely on realistic approach, which in most cases implies expert assessment. However, such assessment is difficult to apply to beyond-design-basis accident with coolant boiling in a VVER-1000 reactor being refueled. This article suggests a new methodology with elements of artificial intelligence, which allows nuclear safety analysis to rely on realistic approach without using expert assessments.

Key Words: nuclear safety analysis, VVER, beyond-design-basis accident, recriticality, Monte Carlo method, neural networks.

EDN: QZLFPL

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Development of Sensitivity and Uncertainty Assessment Methodology for the Analysis of Severe Accidents at VVER Reactors

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In the Russian Federation, uncertainty and/or sensitivity analyses including deterministic calculations of severe accidents (SA) are obligatory for safety assessment under NP-001-15 and RB-166-20. Despite the successful development of sensitivity and uncertainty analysis methodology, some issues of severe accident analysis are still unresolved.

This study presents a Monte-Carlo sensitivity and uncertainty analysis methodology (including test results thereof) developed for in-vessel stage of severe accident at VVER reactor plant. The list of input parameters with respective uncertainties thereof relies on design and reference documentation data. The number of calculations required for uncertainty analysis is based on the Wilkes formula and finalized by assessing the convergence of both average values of all significant parameters and normal deviations thereof. Spearman's, Pearson's and Kendall's correlation coefficients are calculated to determine whether significant parameters and variable input parameters do correlate. This paper also demonstrates the main principles of proposed methodology application for the severe accident scenario involving complete blackout of VVER-1000 NPP coupled with failure of all diesel generators.

Key Words: severe accidents, analysis, uncertainty, sensitivity, methodology, VVER, SOCRAT.

EDN: TJGGLJ

UDC 621.039.531, 620.186.1

Regularities of Radiation-induced Phase Formation in VVER Reactor Steels

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Based on atom probe studies of the operating and advanced materials of reactor pressure vessels (RPV) and pressure vessel internal (RVI) after neutron irradiation under various conditions, general regularities of phase formation were analyzed and revealed. It was supposed that in RPV and RVI steels, depending on the composition and irradiation conditions, two types of precipitates are formed: Cu-based in RPV and Ni—Al—Cu—Ti-based (γ'-phase) in RVI, that nucleate in displacement cascades; and Ni—Si—Mn-based in RPV and Ni—Si—Ti-based (G-phase) in RVI, that form as a result of radiation-induced segregation on various sinks. The rate of precipitate accumulation decreases with increasing accumulated dose, which indicates a decrease in the contribution of phase formation into the RPV’s and RVI’s properties change during the extended service life.

Key Words: reactor pressure vessel, reactor vessel internals, pearlitic steels, austenitic steel, neutron irradiation, radiation-induced segregation, phase formation, atom probe tomography.

EDN: UAARDD

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Development of a Dry Deposition Model in the ROM Code and its Application on Long-Term Meteorological Data for the Territory Around Ten NPPs

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The article details the development of a resistive model for dry aerosol deposition, integrated into an updated version of the ROM, which has been developed from the NOSTRADAMUS code. The models were validated using experimental data, considering the initial data provided by the model. The ROM code was utilized to determine the rates of dry deposition of non-hygroscopic aerosol at ten Russian nuclear power plants as of 2021—2023. The aerosol size in the atmosphere was correlated with a potential emergency release. The findings indicated that the absolute deposition rates vary by region, with vegetating features differing by as much as nine-fold and non-vegetating features by up to threefold. The relationship between the deposition rates remains consistent and aligns with the correction factors utilized in international models.

Key Words: NPP, emergency release, atmospheric dispersion, dry deposition, resistive model.

EDN: XAYTJF