Abstracts
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The Role of Reactivity Effects in the Chernobyl Accident

I.A. Bazulin, A.M. Fedosov,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

This paper investigates the influence of positive reactivity effects on the progression of the accident at the 4th power unit of the Chernobyl Nuclear Power Plant, specifically examining the reactivity effects involved with: control rod insertion, steam, control rod circuit dehydration, graphite temperature, and ^{135}Xe burnout.

The analysis hereof demonstrates that only the combined impact of these positive reactivity effects can account for the catastrophic scale of the accident, nuclear fuel evaporation, and graphite dispersion.

Key Words: Chernobyl accident, RBMK, reactivity effects.

EDN: AQDVEB

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Study of Pulse Energy Fluctuations in IBR-2M Reactor using the Basic Element Method

Yu.N. Pepelyshev, N.D. Dikusar, Ts. Tsogtsaikhan, D. Sumkhuu,
Joint Institute for Nuclear Research, 6, Joliot-Curie st., Dubna, Moscow region, 141980

Pulse energy noise spectra structure dynamics in IBR-2M reactor has been studied with the resonant baseline component singled out using the basic element method. The behavior of resonant noise component depending on the reactor power and operating time demonstrates good agreement with the results of parallel reactor dynamic simulations. The baseline resonant noise component with a peak at frequency of 0.1 Hz is shown to coincide with IBR-2M amplitude frequency response resonance. This confirms that the dynamics model and reactor noise analysis methods described herein provide complete and accurate assessment of reactor dynamic properties, as well as that the dynamics model and basic element method apply to reactor status analyses and behavior forecasts.

Key Words: IBR-2M, pulse energy noise, basic element method, baseline, BEM-polynomials.

EDN: DINQJE

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Computer Simulation of HTGR First Criticality Procedures

A.V. Grol, A.S. Lapin, V.A. Nevinitsa, P.A. Fomichenko, V.F. Shikalov,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182,
M.V. Shchurovskaya,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182,
NRNU MEPhI, 31, Kashirskoye sh., Moscow, 115409.
V.P. Alferov,
NRNU MEPhI, 31, Kashirskoye sh., Moscow, 115409

This paper describes calculations performed to underlie HTGR first criticality procedures. Core fueling sequence is selected, and the minimum critical load is identified. Neutron detector signals are estimated for different fueling stages. Control rod worths are calculated for both the first critical configuration and the completely fueled core. Parameter selection is validated for neutron sources and detectors used in first criticality procedures. The calculations rely on full-scale 3D core model for the Monte Carlo MCU-HTR code.

Key Words: high-temperature gas-cooled reactor, neutronic calculations, Monte-Carlo method, first criticality.

EDN: ENYUTR

UDC 621.039.46

Issues in Validating HTGR Neutronic Computing Software Based on HTTR and VHTRC Experimental Benchmark Calculations

Ya.A. Kotov, V.A. Nevinitsa, V.F. Boyarinov, A.V. Grol, P.A. Fomichenko,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182**,**
S.E. Sorokin, G.S. Filippov,
JSC "Afrikantov OKBM", 15, Burnakovskiy proezd, Nizhny Novgorod, 603074

To validate the JAR-HTGR design code developed by NRC "Kurchatov Institute" to validate neutronic parameters of modular-core HTGR design developed by Afrikantov OKBM, a computer simulation of startup experiments was performed for HTTR high-temperature research reactor. These experiments identify HTTR parameters such as criticality, reactivity margin, CPS control rod worth, axial distribution of fission reaction rate, and temperature reactivity effects. Analysis of measured temperature reactivity coefficients shows that descriptions of benchmark experiments do not account for CPS CR elevation, so that only two of six available measurements can be interpreted as reactivity variations caused by reactor temperature variations, compensated by respective CPS CR movement, and attributed to the corresponding temperature difference. In other cases, benchmark authors introduce artificial amendments to CPS CR worth, which are comparable to measured worth values.

Results calculated for HTTR as regards temperature reactivity coefficient simulation experiments are supplemented by analysis of experiments with VHTRC critical facility.

Key Words: HTGR, HTTR, VHTRC critical facility, axial distribution, three-dimensional computing models, neutronic calculation, code validation.

EDN: FBODCJ

UDC 621.039.514

Transient Calculation with Feedback using SAPFIR Program Package

A.A. Panchenko, G.A. Avramenko,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

The article presents the calculation results of the transient problem TMI-1 Mini core using the SAPFIR program package and the KORSAR code developed to thermal-hydraulics modeling. The results are compared with the TRIPOLI-4 results.

Key Words: space-dependent neutron kinetics model, Monte Carlo method, thermal-hydraulics feedback, TMI-1 Mini core.

EDN: GPBICH

UDC 621.039.51...12+621.039.51...17

Algorithms for Modeling Non-Stationary Functionals and their Uncertainties by the Monte Carlo Method

V.I. Belousov, M.I. Gurevich, V.D. Davidenko, I.I. Dyachkov, M.V. Ioannisian, K.F. Raskach.
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents a description of the algorithms implemented in the KIR-S program for calculating nonstationary functionals and their variances in Monte Carlo simulations of neutron transport. The program implements the classical Monte Carlo method for simulating a random process of particle trajectory formation in the form of a Markov chain with multiplication and branching. To verify the developed methods, a model of a homogeneous infinite medium was used, in which the criticality after the insertion of a point source of prompt neutrons is calculated. The solution obtained by the point kinetics method was used as a reference one. The results obtained using the KIR-S program are consistent with the reference ones. The convergence of the functionals and their statistical errors with an increasing number of histories is analyzed. In agreement with the

definition of the root-mean-square error, the calculations showed that as the number of neutron histories increases by a factor of n, the mean deviations decrease by a factor of $\sqrt{n}$.
Key Words: neutron transport, KIR-S program, Monte Carlo method, modeling of stationary and non-stationary processes, error, non-stationary functional, neutron kinetics.

EDN: JYJEPH

UDC 539.125.523.43

Program for Multiparameter Approximation of Nuclear Constants with Quasi-Random Distribution of Grid Nodes and Automatic Selection of Coefficients

A.S. Myazin, I.A. Bazulin,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182***;***
National Research Nuclear University MEPhI, 31, Kashirskoe highway, Moscow, 115409.
V.I. Savander,
National Research Nuclear University MEPhI, 31, Kashirskoe highway, Moscow, 115409

This paper describes CINDER software package developed to generate libraries of group nuclear constants, as well as the methodological framework underlying each software module thereof. The module that describes functional dependencies based on Python tools performs sequential approximation and regression procedures. Various methods are implemented to define the grid for multidimensional space of model status parameters. A polycell model is developed for channel-type uranium-graphite reactor intended for radioactive isotopes production. Test calculations performed using the CHARM program confirm high quality of the neutronic group constants library generated by CINDER by comparing respective reactivity coefficients and neutron multiplication factors K_∞.
Key Words: Sobol sequence, library of nuclear constants, OpenMC, CHARM, computer programs.

EDN: KDNOGA

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Methodical Analysis of Experiments on a Single Channel of Critical Facility

A.M. Degtyarev, Yu.I. Zorin, A.A. Ivanov, M.A. Kalugin,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents computational and methodological analysis of results yielded by model experiments on a single channel of critical facility with RBMK-type fueling. Experimental results are shown to have significant dependence on the critical facility's specific fueling pattern. Practical interpretation of relevant experimental data requires calculations with precision codes.
Key Words: critical facility, single channel, calculation, interpretation of results.

EDN: KESLEN

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High Burnup of Rhodium SPNDs

A.Yu. Kurchenkov, A.M. Musikhin, M.A. Raschetnova
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents calculated results for the functions that account for burnup of rhodium SPNDs within ICIS in VVER reactors, and briefly describes the history of this issue, which is important for ICIS. It also determines the key parameters for these functions to enable future SPND use up to high burnup, and emphasizes that computations should be confirmed by experiments with both highly burnt-up and fresh SPNDs with known charge flown therethrough in similar radiation conditions.
Key Words: self-powered neutron detector (SPND), SPND burnup, in-core instrumentation system (ICIS), VVER, communication line, background current, SPND individual technical characteristics (ITC).

EDN: LHMTGC

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Estimation of Neutron Fluence Rate Calculation Error due to Source Setting Uncertainty

L.V. Kravchuk, V.N. Kochkin, P.P. Panferov, A-r A. Reshetnikov, A.M. Timofeev, D.Yu. Makhotin
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents the results for estimated fluence rate (FR) calculation error due to uncertain neutron source settings in VVER reactor cores. Systematic and random components of this error were estimated by comparing calculated results with experimental data. Neutron field parameters were calculated using the KATRIN program. The FR calculation error due to source setting uncertainty is shown to have an insignificant systematic component. Among quantitative estimates of FR calculation errors obtained for various spatial regions of VVER-1000 reactor pressure vessel, the maximum one is 8%, which is below the currently used value of 15%.

Key Words: neutron transport calculation, neutron fluence, neutron fluence rate, calculation error, neutron source uncertainty, VVER, reactor pressure vessel, neutron activation measurements, KATRIN.

EDN: MARZLA

UDC 621.039

Development of a Model of a Medium-Power NPP Power Unit to Study the Power Unit Maneuvering Characteristics

O.Yu. Kavun, M.A. Uvakin,
OKB Gidropress JSC, 21, Ordzhonikidze str., Podolsk, Moscow region, 142103,
V.V. Semishin,
Bauman Moscow State Technical University, 5, 2-d Baumanskaya str., Moscow, 105005

The article presents a model of the dynamics of a medium-power unit, created on the basis of the RAINBOW-NPP software package and designed to study the power maneuvering modes of the power unit. An example of modeling a deep maneuver mode with power is given.

Key Words: medium power NPP unit, VVER, deep power variation of NPP.

EDN QKNUTK

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Irradiation Device Design Development for Accelerated In-Core Testing of VVER-SKD Structural Materials in SM-3 Reactor

A.L. Izhutov, V.S. Moiseev, N.K. Kalinina, N.Yu. Marikhin, D.S. Moiseev,
Research Institute of Atomic Reactors (RIAR),
9, Zapadnoye sh., Dimitrovgrad, Ulyanovsk obl., 433510.

In developing new-generation supercritical water-moderated power reactors (VVER-SKD), a challenging task is to study the behavior of candidate in-core structural materials under neutron irradiation. Increasing safety requirements for nuclear power plants (NPPs) necessitate accelerated in-core testing to assess changes in material strength and plasticity, as well as radiation-induced swelling processes. Respective experiments rely on SM-3 research reactor that has high neutron flux density, as well as the equipment as necessary to monitor temperature, chemical conditions, and irradiation parameters. This paper presents the designs of both the flat specimen made of candidate material, and the irradiation device to be placed in SM-3 reactor core. Computational studies show that, during in-core testing, the temperature of flat specimens and the fast neutron flux density on surfaces thereof range within 478 ± 76 °C and $(5.90–17.8)10^{14}$ $cm^{-2}\cdot s^{-1}$, respectively. Accelerated in-reactor testing of flat specimens in the suggested irradiation device will provide comprehensive data on the behavior of VVER-SKD materials under intense neutron irradiation, which will contribute to improving the reliability and safety of new-generation reactors.

Key Words: irradiation device, thermophysical calculations, SM-3 reactor, VVER-SKD, in-core tests.

EDN: TUDSMV

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Experience in Creating and Implementing Modern Acoustic Leak Monitoring Systems for Coolant Pipelines and Equipment of VVER Reactor Facilities

P.A. Dvornikov, A.A. Kudryaev, A.N. Molyavkin, V.N. Zamiusskiy, A.A. Savinov, A.A. Budarin, S.S. Shutov, P.S. Shutov, A.V. Burlyaev, V.V. Sheshukova,
JSC Scientific-technical Center “DIAPROM” (JSC “STCD”),
9A, building 3, Lefortovo municipal district, 2-ya Sinichkina str., Moscow, 111020,
I.A. Chusov,
OKB Gidropress JSC, 21, Ordzhonikidze str., Podolsk, Moscow region, 142103

This paper discusses creation, quality assurance and implementation of modern acoustic leak monitoring systems (ALMS) for coolant pipelines and equipment of VVER reactor facilities. It shows that the VVER ALMS hardware complex based on the latest engineering solutions complies with present-day requirements of the leak before destruction concept, quality assurance during manufacture and implementation, as well as functionality under external impacts characteristic for NPPs, which enables safe and reliable NPP operation.

Key Words: coolant leak monitoring, pipeline, equipment, acoustic sensor, hardware complex, leak-before-break concept, amplifier-converter modules, system, reactor facilities.

EDN: UOUNIL

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Overview of Domestic and Foreign Studies on VVER Internals Dynamics During Pipeline Ruptures

A.M. Baisov,
OKB Gidropress JSC, 21, Ordzhonikidze str., Podolsk, Moscow region, 142103,
NRNU MEPhI, 31, Kashirskoe sh., Moscow, 115409,
D.A. Posysaev,
OKB Gidropress JSC, 21, Ordzhonikidze str., Podolsk, Moscow region, 142103,
N.N. Paputin,
NRNU MEPhI, 31, Kashirskoe sh., Moscow, 115409

This paper presents the results of domestic and foreign experimental studies of hydrodynamic effects on reactor internals performed on large-scale models of pressurized water reactors assuming pipeline ruptures. The paper outlines key approaches to accounting for coolant interactions with structural walls when evaluating the internals’ strength based on calculated thermohydraulic data. The paper also suggests a validation matrix, and identifies the need for similar experiments to cover the current pressure and temperature ranges typical for modern water-cooled reactor designs.

Key words: hydrodynamic effects, pipeline rupture, critical leakage, reactor internals, VVER, experimental test facility.

EDN: WGXZKK