Abstracts
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Modeling of Heat Exchanging Processes in Steam Generators of Reactor Facility with Coolant Pb-Bi in Intercircuits Leaks Accident

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Results of modeling heat-transfer processes in steam generators of reactor facility with Pb-Bi coolant are presented in this paper. Numerical experiments were calculated by TRIANA-6/Ver2.0 code with nonequilibrium one-dimensional approximation of two-component flow of Pb-Bi and water coolant.
Key Words: Pb-Bi, steam generator, heat-transfer, intercircuits leaks, TRIANA-6/Ver2.0.

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Influence Estimation of Dependent Control Rods Failures to the Probability of VVER RP SCRAM Emergency Shutdown Function Implementation

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This article deals with taking into account of dependent multiple control rods failures during a transient condition applied to VVER. The main purpose is influence analysis of failures to safety-critical parameters of the reactor plant. The problem of multiple control rods failures is considered by the example of accident with initiating event “Steam Line Break”. The evaluation is made by comparison results for two cases: failures are situated throughout the whole core with equal probability and compactly in the sector of the core that adjoins the emergency loop. The second case is possible if failures are depended.
Key Words: VVER, Reactor, Multiple Failures, Dependent Failures, Scram System, Steam Line Break.

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Statistic Simulation of Realistic Assessment of VVER-1000 Vessel Materials Radiation Embrittlement

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Realistic assessment of brittle fracture resistance of reactor vessel materials, based on testing results of surveillance specimens of VVER-1000 vessel base metal and weld metal, ensures the minimum level of conservatism T_K in the boundary of the data scatter band. An additive statistic model of approximating dose-time relationship is proposed $T_K = (T_{K0} + A_F \cdot F^n) + \delta T_K$ in the form of upper enveloping data scatter band. The proposed model realistically reflects the processes going in vessel materials metal during operation of VVER-1000 vessel made of steel 15Х2НМФА-А, in spite of application of shift criteria determining radiation embrittlement of metal.

The proposed statistical model of approximating dose-time relationship $\Delta T_K = A_F \bullet F^n + \delta T_K$ is not linear of argument F^n, since the radiation embrittlement factor A_F is not a constant and depends on n exponent of fluence (dose) power F(t), which, in turn, is a linear function of irradiation time during VVER-1000 vessels operation.

The conservatism levels of dose-time relationships parameters ΔT_K are analyzed with simulation of radiation embrittlement of VVER vessel materials with taking into account the following:

- power exponent m subject to T_{K0};

-correlation γ of normative and actual values of A_F;

-pessimistic δT_{KA} with regard to actual conservatism margins

Key words: Brittle fracture resistance, radiation embrittlement, surveillance specimens, VVER pressure vessel.

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Differential Reactivity Coefficients Estimation Method for VVER Reactor by KORSAR/GP and TRAP-KS Codes Application.

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This work is oriented to actual problem resolving. The problem is VVER reactor facility stationary conditions preparing for coupling physical and thermo-hydraulic dynamical processes calculation. Work contains fuel temperature and coolant density reactivity coefficients estimation approach. It is based on finite-difference approximation. Developed model is built on macroscopic library structure and feedback parameterization principles. Method is suitable for KORSAR/GP and TRAP-KS codes which are used physical data preparing by SAPFIR_95 code. Material contains some application results of introduced method.

Key Words: Reactivity Coefficients, TRAP-KS Code, KORSAR/GP Code, Feedback Parameters, Dynamical Processes.

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Development of Computer Code CORE_1 for Modeling the Thermomechanical Behavior of All Fuel Assemblies in the Core

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The article deals with the computer code CORE_1 is intended for modeling of thermomechanical behavior of all FAs in the core under normal operation conditions. The code CORE_1 enables to perform the calculation of stressed-strained state (calculation of FA bowing of longitudinal forces in fuel rods and GTs, moments in contact pairs “GT-SGr” and “fuel rod-SGr” and determination of responses in supports) for all jacket-free fuel assemblies making up the core, and to perform the calculation of inter-assembly gaps under the effect of force and temperature loads on FA group, typical for NOC and AOO with regard for radiation growth and creep of structural materials, as well as contact interaction of adjacent FAs between themselves and of periphery FAs with the core baffle.

Key words: computer code, thermomechanical behavior of FAs, inter assembly gaps.

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Cross-Verification of TIGRSP Code Fuel Assembly Cells Model Using CFD-Code on Example of 7-Rods Assembly

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Coolant thermohydraulics in 7-rods assembly of VVER-1000 at nominal parameters of coolant was considered. CFD-model was tested and validated by experimental data and by empirical correlations of hydraulic resistance, turbulent cross-mixing and fluid-wall heat transfer. Cross-verification of TIGRSP code by CFD calculation of mass velocity distribution in 7-rods assembly flow area is carried out. The results could be used for cross-verification of 1D- thermohydraulics codes and for development of VVER reactors fuel assemblies.

Key words: TIGRSP, CFD, fuel assembly, VVER, nuclear reactor, hydraulic resistance, heat transfer.

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Vibration Analysis of VVER-1000/1200 Reactor CPS Drives

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The control and protection system (CPS) of VVER-1000/1200 reactor plant (RP) is intended to control the reactor during its commissioning and power operation, as well as during planned and emergency shutdowns. A complete set comprising of a displacement unit and an electromagnet unit SHEM-3, a pitch position indicator (PPI), an extension shaft, housing, CPS absorbing rods (AR) and guiding channels (GCh) of fuel assemblies (FA) makes a reactivity control channel. Assurance of this channel operability and functional reliability is the most important factor that determines VVER-1000/1200 reactor plant safety. To support the analysis of CPS drive vibration behavior the modal studies of CPS drive SHEM-3 have been performed and the data on operational vibration loading have been achieved followed by diagnosing and simulating various vibration states of the object.

Key words: drive, kinematics, diagnosing, vibration dynamics, vibration state, reliability, safety.

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Peculiarities of Neutronics Characteristics of Integral Reactor WWER of Small Capacity

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The neutronic characteristics of the stationary fuel loading of the core of integral reactor WWER with a small power 600 MW (th.) and about 200 MW (el.) calculated with Russian code BIPR-7 are analyzed. Design solutions by the core and FAs, and also the technological parameters are basically analogous to serial reactor WWER-1000; however, there are essential differences in comparison with the WWER-1000:

- in the core are placed 85 FAs (163 in WWER-1000) with an active length 250 cm (355 cm in WWER-1000);
- it is used the reactor campaign of 24 months length and the fuel campaign of 48 months (in WWER - 1000 is used 12-18 months the reactor campaign and 36-48 months the fuel campaign);
- for reduction in the neutron leakage on the core periphery are placed burnt out FAs with the smallest multiplication properties, and also FAs have axial blankets with the smaller enrichment (in WWER-1000 they are used to a lesser degree or they are not used);
- there are 18 Guiding Tubes placed in FAs for reactor control and EP provision for the arrangement in them the Black and Grey CRs CPS (in WWER-1000 they are used only Black CRs CPS for EP provision);

- for compensating of the reactivity margin for fuel burn-up and for subcriticality provision of reactor in shut-down conditions are widely used the IFBA in FAs – 18-30 FRs with 8% of natural Gd (tvegs), and also Grey and Black CRs CPS and is not used the dissolved boron in the coolant (in WWER-1000 in essence it is used dissolved boron in the coolant and to a lesser degree the tvegs);

- under the conditions for boron-free control, for guaranteeing the minimum power peaking factors in the core it is fitted the optimum axial enrichment profiling of FAs and axial profiling of concentration of BA in tvegs, and also the axial profiling of concentration of absorber in CRs CPS (in WWER-1000 it is not used).

The neutronic characteristics were investigated in the process of burning out of fuel in the base operating mode at the nominal power for variant of boron-free control. They are compared with analogous characteristics for usual variant of boron control. The mode of the daily manoeuvring in a wide range of power change 100-30-100% of nominal power is also analyzed for variant of boron-free control and with use of additional regulation by various primary coolant temperatures (modes "P2=const", "tin=const", "tav=const"). The positive results were obtained, which make it possible to make a conclusion about the relatively simple feasibility of the integral small-power reactor with WWER technology in the neutronic aspect.

Key Words: Neutronics, Boron and Boron-free control, temperature control, WWER technology, Base and Maneuvering modes.

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Modeling of Thermohydrodynamic Processes in the Core of Water-Cooled Reactor

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The article discusses the results of mathematical modeling of thermohydrodynamic processes in the core of water-cooled reactor. Here we consider the three-field two-fluid subchannel mathematical model and the results of its application for investigation of thermohydrodynamic processes in the fuel rod bundles. Such a model is suitable for studying the processes of thermohydrodynamic in fuel assemblies in all possible modes of operation. For the numerical implementation of the proposed mathematical subchannel model used semi-implicit numerical scheme. For discretization in space (axial) variable used “chess” grid. In the report presented the results of computer simulation. Is proposed the variant of solution to the problem of integration of system thermalhydraulic code and of subchannel thermalhydraulic code.

Keywords: thermohydraulic analysis of rod assemblies; subchannel approach; three-field two-fluid mathematical model.

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Status of the In Vessel Melt Retention Strategy (IVMR) for VVER-1000 Existing Units

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Presented paper provides information on existing status of our research targeted on confirmation of the In Vessel Melt Retention Strategy (IVMR) for VVER 1000 existing units. First we would like to describe development of the small scale testing facility and status of design work on large scale test facility THS – 15. Results on small scale facility provides us Critical Heat Flux (CHF) curve for clean surface of the RPV steel and for surface with porous type of coating applied with the “cold spray” technology. Similar tests will be also performed on the large scale test facility fully representing the VVER 1000 configuration.

Key words: In Vessel Melt Retention Strategy (IVMR), Reactor Pressure Vessel (RPV) Cooling, Small scale experimental facility, Large Scale Experimental Facility (THS-15), Explosive welding, Porous coating

with Cold Spray technology, Critical Heat Flux (CHF) Curve, Results of analytical calculations. JRC Benchmark calculation. HORIZON 2020 , EU Project on IVMR for existing and future NPPs.

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Experimental Investigations of Influence of Mixing Grid "Run Row Upon Row"-type on Hydrodynamics of Coolant Flow in VBER Fuel Assembly.

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The article deals with the results of experimental studies of the coolant local hydrodynamics behind the mixing grid in VBER fuel assembly TVSA. The studies have been performed using simulation of the flow in TVSA on an aerodynamic bench by means of tracer diffusion with the help of pneumometric probes. The aim of studies was to determine the mixing grid influence on hydrodynamic properties of the flow in the area of standard cells and in the zone of guiding channel. The fulfilled studies have resulted in distribution of speed local fields behind the mixing grid deflectors, distribution of the coolant flow rates and concentration of tracer in experimental model cells. The obtained data have allowed to determine regularities and to discover peculiarities of coolant flow behind the TVSA mixing grid.

Key words: fuel assembly, spacing grid, mixing grid, guiding channel, hydrodynamics, heat-mass transfer.

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Calculation Simulation of Experiments on Single-Pipe Model of the Once-Through Steam Generator for BN Reactor Plant

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The article deals with a model of thermohydraulic calculation for single-pipe model of large-moduled steam generator with the help of computer code "KORSAR/GP" with connected software module "PGN-2K" for calculation of the coolant path. Besides, the article shows results of calculation simulation of conditions for bench tests of single-pipe model of large-moduled steam generator and their comparison with experimental data of single-pipe model for the purpose of verification of the thermohydraulic calculation code for steam generator with liquid-metal coolant.

Key words: single-pipe model, KORSAR/GP, PGN-2K, steam generator with sodium coolant, temperature distribution, thermohydraulic computer code.

UDC 001.2

Application of DDL-Schemes of GQ3D-Metod for Calculations of Ionizing Radiation Transport on Arbitrary Hexahedral Grids with Non-Planar Cells' Facets

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The approach of applying of the DDL-schemes GQ3D-method for DSN-equations on 3-D unstructured hexahedral meshes with non-planar cells′facets are presented. Numerical results are presented which demonstrate the accuracy and efficiency of this approach.

Key words: DSN-method, DDL-schemes, GQ3D-method, arbitrary hexahedrals grids, hexahedrons with non-planar facets.