Abstracts
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Use of Experience in Creation and Operation of Single-Circuit Reactors with Coolant Boiling and Nuclear Superheating for Designing Reactors with Coolant Supercritical Parameters

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Experience in creation and operation of single-circuit plants with boiling coolant and steam nuclear superheating and direct supply of superheated steam to the turbine (reactors VK-50, RBMK, AMB-100 and AMB-200) was applied to its use when creating a water-cooled reactor with coolant supercritical parameters (SCWR). Water chemistry of Units in thermal power engineering and single-circuit reactor plants was also considered, which ensures comparatively small deposits on the turbine flow part surface. In addition to this, issues on nuclear environment and circuit decontamination were addressed.

Key Words: water-cooled reactor, supercritical parameters, steam superheating in the reactor, operation, water

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Development and Testing of Interpolation Procedure Keeping Order of Accuracy of Finite-Element LD-scheme at the "Not–Node-to-Node" Coupling of the Adjacent Tetrahedrons

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The results of development and testing of interpolation procedure to be used in calculations with LD-scheme for tetrahedrons, joined not at the vertices, are presented.

Key Words: LD-scheme, tetrahedrons, interpolation procedure, adjacent are not "node to node".

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Experimental Studies of Local Fields of Coolant Velocities in the Fuel Rod Bundle

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The article provides brief description of the experimental facility, procedure of performance of experiments as well as results of experimental studies of local fields of coolant velocities in the fuel rod bundle with single-phase flow. The experiments were performed at OKB Gidropress JSC using full-scale fuel assembly model. Results of the experiments are intended for verification of CFD-codes.

Key Words: local parameters of coolant, fuel rod bundle, fuel assembly.

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Analysis of Fulfilment of “Controlled Operation” Concept for MCP and Surge Line at Implementation of “Leak Before Break” (LBB) Concept at VVER-1200 Units

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The “controlled operation” concept for the main coolant pipeline (MCP) and surge line (SL) in accordance with GOST R 58328-2018 requires monitoring of their operation conditions in terms of parameters of operation, mechanical (due to displacement) and general thermal loads. In terms of diagnostic systems, it is necessary to ensure on-line monitoring of thermal pulsations, thermal shocks of coolant at MCP and SL nozzles, coolant stratification in MCP and SL, loads on MCP and SL due to equipment displacements as well as general temperature stresses. In terms of the calculation of life characteristics, it is necessary to ensure off-line calculation of low- and high-cycle fatigue and fatigue growth of defects during operation.

Key Words: cumulative damage, stratification, equipment and pipelines, reactor plant, equipment displacement, hydraulic snubber monitoring system, ageing management.

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Validation of Code KORSAR/GP by the Results of Tests of Operating VVER RP Unit in Daily Loadfollow Curve

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Calculations of load-follow conditions were performed to the maximum possible extent in accordance with the load-follow procedure and approved program of tests at the stage of safety justification for operating large-power VVER Unit in the daily load-follow curve. Code KORSAR/GP with three-dimensional model of neutron kinetics was used in the calculations. Successful performance of tests of the daily load-follow curve at the end of fuel cycle allowed using obtained experimental data for preparation for validation based on pre-test calculations in order to improve accuracy when using code KORSAR/GP for solution of such problems in the future. The given results are the first part of the work on code KORSAR/GP application for VVER RP safety justification, taking into account load-follow conditions. The validation stage is an urgent task and it will be supplemented as more results are obtained.

Key Words: daily load curve, power load-follow, KORSAR/GP, VVER RP, code validation, calculational model accuracy improvement

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Cross-Verification of Code KORSAR/GP and CFD-code for Conditions of Two-section At-Reactor Spent Fuel Pool

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Testing, verification and validation of CFD-model of heat exchange and convection in a single-phase area were performed using empirical correlations and experimental data. Results of calculations by CFD-model of two-section spent fuel pool and code KORSAR/GP were compared for normal operating conditions and an accident with loss of spent fuel pool cooling – from the beginning of an accident to the beginning of boiling. Both qualitative and good quantitative agreement of the results of the calculations was shown for different time moments. The results of the work are intended for use with additional verification of code KORSAR/GP in terms of peculiarities of distribution of parameters in the spent fuel pool, the number of available experimental data on which is extremely limited to date.

Key Words: spent fuel pool, CFD, loss of cooling, safety systems, VVER, natural convection, heat exchange.

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Development of Complete List of Beyond Design Basis Accidents for Balakovo NPP Unit 4

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NP-001-15 “General provisions for safety assurance of nuclear power plants” establish that for determination of measures for management of beyond design basis accidents, an analysis shall be performed for scenarios included in the complete list of beyond design basis accidents, elaborated for each individual Unit of the nuclear power plant (NPP). The above list shall meet a number of requirements and, first and foremost, the requirement of representativeness, i.e. it shall include scenarios, analysis of which is sufficient for determination of strategies on management of each possible accident, no matter how unlikely it might be. In 2018 Rostekhnadzor introduced Safety Guide RB-150-18, which contains recommendations and a rough algorithm of formation of a complete list of BDBAs1. Based on approaches given in the specified Guide, such a list was elaborated for Balakovo NPP Unit 4 (VVER-1000/V-320) and it was the first experience of practical application of RB-150-18. This article provides results of this work.

Key Words: beyond design basis accidents, severe accidents, representativeness

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Calculational Study of Thermal-Hydraulic Characteristics of Block-Containers for ^{99}Mo Production

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The work provides a calculational study of thermal-hydraulic characteristics of targets for 99Mo production, namely, numerical simulation of circulation water flow in the experimental channel of VVR-ts reactor when arranging different models of block-containers in it with uranium-bearing material. Four versions of block-containers were considered in the calculation. The mode of operation of the experimental channel for all models was assumed proceeding from the standard cycle of VVR-ts reactor. Values of fields of the circulation water velocity, temperature and pressure as well as temperature of walls of the containers and uraniumbearing mixture matrix were obtained for all models under consideration as a result of the thermal-hydraulic calculation. The conclusions on the target structure were made, which is optimum from the viewpoint of heathydraulics.

Key Words: VVR-ts reactor, block-container, thermal-hydraulic characteristics, experimental channel.

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Calculational Justification of Service Life Extension Feasibility for Steam Generator Modules of Beloyarsk NPP Unit 3

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The article provides an overview of studies performed by OKB Gidropress JSC for justification of service life extension feasibility for modules of steam generator PGN 200M of BN-600 Unit for the operation periods of 2010–2025 and 2025–2040.

Keywords: service life extension, module, operating time, steam generator, life, studies of structural materials.

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Risk-Informed Approach in Methodology of Service Life Management

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The purpose of the risk-informed approach (RIA as per NP-089-15) is to consider and manage risks at the level of pipeline components, which allows optimizing metal inspection programs in order to reduce costs with maintaining and even increasing the safety level. In addition to this, both the component significance for the plant safety assurance and determination of risk of the further application of the component, depending on its technical state, are assessed in RIA procedures. Stochasticity sources are considered when describing processes of damage which determine the service life. Examples of determination of risk classes are given and assessment of frequencies of failures is provided for pipelines under conditions of flow accelerated corrosion (FAC) and for the steam generator heat exchanging tubes (SG HXT).

Key Words: degradation mechanism, inspection program, probabilistic model, distribution function, risk matrix, pipeline, defect detection probability, failure frequency.