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Validation of Code TEMPA-SC Based on Experiments with Fuel Rod Bundles Cooled by Supercritical Pressure Water

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A short list of modern calculation programs for simulating thermal-hydraulic processes with supercritical pressure water is provided. Code TEMPA-SC and modernization performed by the authors of the article are described, which are associated with implementation of new closing relations for heat exchange and hydraulic friction resistance, as well as extension of the module of calculation of thermophysical properties, which made it possible to accelerate performance of the calculations. The code validation stages are described. The results of the “blind stage” calculations of two benchmark problems with cooling down of rod bundles of various geometries, performed within the framework of the International Forum Generation-IV, are provided.

Key Words: cell-by-cell computer program, supercritical pressure, heat exchange, rod bundles, validation.

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Development of Empirical Correlations for Assessment of Conversion Coefficient Value of Postulated Steam Explosion under Severe Accident at VVER Reactor Plant

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In the course of severe accidents (SA) with the core melting, such a phenomenon as the energy interaction of the melt with the coolant (steam explosion) may occur, which threatens the reactor vessel integrity and can be a trigger for hydrogen explosion. One of the main parameters determining the power of a steam explosion is the conversion ratio CR (another common designation is δ). CR is the melt energy fraction converted into mechanical work. In this study the results of more than 190 experiments simulating the interaction of melt with coolant are considered using the corium melt (TROI, FARO, KROTOS-KFC, ZREX, ANL) and the simulated melts (KROTOS Huhtiniemi, MISTEE, SUW, WUMT, MIXA, EXPO-FITS, FITS, ALPHA, WFCI, etc.). Based on the results of this study, the dependences of the maximum value of the conversion ratio on the following parameters were derived: the melt mass reduced to the coolant mass (M_m/M_f), the ratio of the reduced value of the coolant subcooling to the saturation temperature to the reduced value of the melt overheating above the melting temperature ($\overline{\Delta T_f}/\overline{\Delta T_m}$); the ratio of components in the corium (Zr, UO_2 and others). The obtained dependences are supposed to be used to specify existing semi-empirical procedures for assessment of the power of steam explosions in order to decrease their conservativeness in the framework of safety justification for operating NPP and those to be designed with VVER RP.

Key Words: severe accidents, VVER, steam explosion, conversion ratio, correlations, empirical dependences.

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Performance of Pretest Calculations and Development of Experimental Program of PSB-VVER Test Bench within the Framework of International Project "ETHARINUS" of OECD NEA

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The article describes the experimental program of PSB-VVER test bench within the framework of International Project "ETHARINUS" of OECD NEA. The interest of the experiments lies in extension of the calculational and experimental database in the field of phenomena associated with asymmetric single-phase natural circulation. The main stages of developing the experimental scenarios are described. The results of pretest calculations performed using software package KORSAR/GP are provided.

Key Words: OECD NEA "ETHARINUS", VVER, PSB-VVER, asymmetric single-phase natural circulation, heat removal loss, reactor plant cooling, experimental database, pretest modeling, KORSAR/GP, validation of calculation.

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Validation of Code KORSAR/GP by Results of Tests of Load-follow Conditions of VVER RP with Extended Control Range

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The report presents a part of the considerable work on validation of code KORSAR/GP in terms of calculation of VVER RP parameters in the load-follow conditions. The relevance of the work is due to the new requirements for NPP Units related to the flexible schedule of electric load and the feasibility of its operational change, which is stipulated for both Russian and foreign users of VVER technology. Over the past two years, a significant quantity of experimental results was accumulated obtained during tests of the daily-load curve at high-power VVER RP. A number of tests of ten daily cycles in succession were performed for the moments of the beginning and end of the fuel cycle. An important distinguishing feature was the application of an extended range of the secondary pressure control, which made it possible to use the coolant temperature variation as one of the main methods to influence reactivity. The first stage of validation work was performed for pretest calculations and allowed checking the qualitative coincidence of RP main parameters with the experimental data. This work is devoted to the validation of RP power and the temperature of coolant in the circulation loops with obtaining the quantitative error assessment. The work provides the results of validation for various curves of RP power variation and, in addition to this, reasonable conclusions were made about accuracy of the calculational model of code KORSAR/GP.

Key Words: power maneuvering, daily-load curve, code KORSAR/GP, validation, extended control range, integral parameters.

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Analysis of Uncertainties for SFP Conditions with Loss of Cooling

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The uncertainty analysis was performed for the key parameters studied in accident conditions with loss of the spent fuel pool cooling using improved estimate code SOKRAT/B1. Based on the analysis results, the parameters which have the largest impact on the key parameters of the calculation are determined, and it is also shown that the simultaneous deviation of the initial parameters first to the conservative side and then to the side for obtaining the most favourable result leads to a range of the results wider than the range of uncertainties. The results of the performed analysis show that this approach can be used for NPP safety justification in accident conditions with loss of spent fuel pool cooling in addition to the methods recommended in RB-166-20.

Key Words: spent fuel pool, NPP, loss of cooling.

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Experimental Studies of Local Fields of Coolant Velocities in the Reactor Downcomer

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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 reactor downcomer in various conditions of flow. The experiments were performed in OKB Gidropress JSC using four-loop hydraulic plant with VVER reactor model on 1:5 scale. Results of the experiments are intended for verification of CFD-codes.

Key Words: coolant local parameters, coolant mixing, reactor downcomer, RCP set, ECCS.

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Improvement of the Safety Parameter Display System Taking into Account Modern Standards and Requirements of Regulatory Documentation

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The article considers the issues of creating the safety parameter display system, taking into account the inspection of NPP states at all levels of DID and the new requirement in NP-001-15 on adoption of measures to exclude threshold effects in NPP design. According to RB-152-18, "the main directions for reducing the negative impact of the threshold effect on safety are justification and application of design margins, as well as implementation of full-fledged DID". These issues are systematically considered in this article.

Key Words: safety functions, operating personnel support system, defense-in-depth, safety parameters, threshold effect, VVER RP.

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On Application of Fluorocarbon and Hydrocarbon Working Media in Thermal Power Cycles of NPP (Review Material)

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Publications on operation of the reactor plant with organic coolant ARBUS and experience in application of hydrocarbons as NPP coolants are considered. The prospects of materials with a higher application temperature (~500 °C) — fluorocarbons and their possible application as the secondary coolant of BREST-type plants are shown.

Key Words: organic coolant, hydrocarbon, fluorocarbon working media.

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Proposals for the Creation of Tolerant Fuel by Modifying the METMET Fuel Composition

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The METMET dispersion fuel composition developed at SC "VNIINM" is fuel granules of increased density uni-formly distributed in the core made of U—Mo, U—Nb—Zr, U_3Si alloys, metallurgically bonded to each other and to the fuel rod cladding by zirconium-based matrix alloys. The possibility of using METMET fuel as ATF for VVER reactors is considered and justified, primarily due to its high uranium content, high thermal conductivity and heat resistance, as well as the compatibility of fuel rod components during manufacture and emergency situations. The possibility of modifying the existing METMET fuel, aimed at increasing its melting point and radiation resistance, is considered and justified.

Key Words: dispersion fuel element, zirconium alloy, irradiation, uranium capacity, ATF, resistance to emergency situations.

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Study of Dependence of Main Equipment Dimensions of Reactor Plant with Natural Circulation of Primary Circuit on Type of Coolant

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The article provides a review of the reactor plant (RP) with natural circulation (NC) of coolant and contains an analysis of four coolants for applicability as a primary coolant for a low-power nuclear power plant with NC of coolant in the primary circuit, which was not previously described in one article. The operating temperature ranges of the considered primary coolants and the achievable parameters of the steam generated by the reactor plant are shown. The dependence of the main equipment dimensions of RP with NC on the type of coolant was studied. The possible dimensions and masses of the RP vessels are given depending on the coolant used. As a result of comparison of coolants, conclusions were made that water and lead-bismuth coolants (LBC) are the most effective for the low-power nuclear power plant with NC of coolant. It is noted that RP with LBC in the primary circuit is superior to RP with water coolant in terms of safety and refers to the fourth generation. In addition to this, the reactor vessel and steam generator for the low-power nuclear power plant with LBC NC are easier and cheaper to manufacture in comparison with the nuclear power plant with water coolant in the primary circuit. Taking into account the thermophysical properties of liquid metal LBC and the available experience in the technology of this coolant in low-power reactors on ground-based prototype test benches and in transport facilities, eutectic lead-bismuth alloy is very promising as a coolant for the low-power nuclear power plant with NC of coolant in the primary circuit.

Key Words: coolant, low-power nuclear power plant, lead, eutectic alloy, bismuth, lead-bismuth, sodium, water, pressure head, natural circulation, safety.

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Evaluation of Possibility of Using 100% MOX Fuel in WWER-600 Reactor

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The possibility of using 100% MOX fuel in a WWER-600 reactor with nominal capacity of 1600 MW(t) is analyzed. The WWER-600 core consists of 163 fuel assemblies similar in design to the fuel assemblies WWER-TOI. The MOX fuel consists of dump uranium (99,8% ^{238}U and 0,2% ^{235}U) and a mixture of plutonium isotopes. A campaign with Uranium Oxide (UOX) fuel is modeled also for comparison of neutronics with MOX variant. The effectiveness of fuel utilization is analyzed with an increase of Water/Fuel ratio for both variants – MOX and UOX in comparison with the basic variants. W/F increases in three ways. Software package KASSETA-BIPR-7 is used for calculated estimates. Some neutronic characteristics for WWER-TOI (100% MOX fuel, 3300 MW(t)) were presented for possibility to compare with calculations using other independent codes.

Key Words: 100% MOX, plutonium, WWER-600, WWER-TOI, increase of W/F ratio, profit in fuel utilization, efficiency of neutron absorbers.