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Development of a Method for Assessment of Loads on VVER Reactor Vessel Components in Energetic Melt-Coolant Interaction during Severe Accidents

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Safety of nuclear power is the key factor which determines its competitiveness with regard to other types of electric power generation. However, for the last 35 years three severe accidents (SA) have occurred at NPP in various countries. It shows that SA probability is not negligibly low and this type of accidents shall be comprehensively studied. There is such a phenomenon as energetic melt-coolant interaction (steam explosion) which takes place in the course of SA, it can endanger the reactor pressure vessel integrity and is the course of hydrogen explosion.

In the framework of this study a method was developed on the basis of international experience and considering the state-of-the-art knowledge in order to assess loads on the reactor vessel components of VVER RP in energetic melt-coolant interaction during SA which was simulated using code package SOCRAT/B1 and module VAPEX-M, the energetic interaction was considered conservatively with the use of semi-empirical correlations. This model was used as a basis for calculation of loads on the VVER-1200 reactor vessel components in a number of determining SA scenarios.

Key Words: Energetic interaction, melt, coolant, SA, RP, VVER, loads, SOCRAT/B1.

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Development and Implementation of Design of the Computational Methods for the Calculation Strength Analysis of Nuclear Reactor Core Internals

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The paper contains methodology used in the calculational strength analysis during designing the reactor core internals.

Key Words: Reactor core internals.

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Cross Verification of ROK2 Code in Benchmark Test of Loss of Spent Fuel Pool Cooling in VVER-1000 Reactor Unit

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During solving the engineering problems to justify the fuel assembly (FA) cooling in the spent fuel pool (SFP) in case of the initiating event of loss of SFP cooling (and in a possible combination with a SFP leak), a necessity to determine the temperatures and the time of reaching the limiting temperatures of the FA fuel rods located in the SFP is occurred. ROK2 code was developed for this purpose. Despite a wide application of thermal-hydraulic system codes in the engineering practice, a necessity for the application of the specialized programs that allow implementing a wide scope of case calculations occurs in the engineering

practice. The cross-verification of ROK2 code was performed by the results of the calculations performed using the certified system codes KORSAR/GP and RELAP5/mod.3.2 for the cases of loss of SFP cooling in the reactor plant (RP) VVER-1000.
Key words: ROK2, SFP, fuel assembly, VVER, cross-verification, heat transfer

UDC 539.4

Stress Intensity Factor Analytical Results for the Underclad Crack Model

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One of the basic emergency failure mechanisms of WWER pressure vessel is the unstable crack growth. Within the framework of the brittle fracture resistance justification for the cladded pressure vessel the stability of postulated underclad crack shaped flaws should be analyzed.

On the basis of the complex variable technique the singular integral equation has been derived for a two-dimensional problem on a finite crack reaching at a right angle the interface between materials with different elastic properties. The explicit expressions in elementary functions were developed for the stress intensity factor evaluation in dependence on the following parameters: elastic modulus ratio of the base or weld metal and the cladding, crack length, stress distribution along the crack line.
Key Words: Underclad Crack, Stress Intensity Factor, Brittle Fracture, Reactor Pressure Vessel.

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Conceptual Proposals on Supercritical Water-Cooled Reactor (Overview of Foreign and Russian SCWR Designs)

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This overview paper deals with conceptual designs of supercritical water reactors (SCWR) which have been developed or are being designed at present in various countries (Russia, USA, Japan, Germany, Canada, China etc.). Concepts of the vessel-type and channel reactors HPLWR and Canadian SCWR, which have been designed for the last five years, are considered here in more detail. The diagram of a power unit and its equipment is proposed to be simplified by use of a single loop layout for coolant circulation and by reduction in capital costs and operational expenses. With this approach and reactor outlet coolant temperature of 500-625 oC the construction capital costs of 1000-1200 MW power units are reduced by 20-40%. The considered reactors are characterized by thermal, fast and mixed neutron spectra in the core with various coolant circulation layouts.
Key words: LWR, supercritical parameters, RPV

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Sensitivity Analysis at Severe Accidents Modeling by Best Estimate Code

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Onrush of new technologies of computational safety analysis for nuclear power plants, capabilities of high performance computers and more stringent requirements of Rostehnadzor of Russia and foreign customers for safety analysis of operating VVER nuclear power plants necessitate development of domestic computational methods and tools of safety analysis at a new level.

In the framework of this research a sensitivity analysis method for the key parameters of severe accidents (SA) at VVER NPP was elaborated using best estimate code SOCRAT/B1. Sensitivity analysis of the NPP blackout accident modelling

results was performed using the proposed method for VVER-1200 reactor plant. By the results of this sensitivity analysis we determined parameters producing the maximum effect on the key parameters of calculation.
Key words: Severe accident, RP, VVER, SOCRAT/B1, method, sensitivity analysis

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CFD-modeling of the Dynamics of Coolant in the Top Part of the Experimental Facility V-200

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In the paper results of CFD modeling of water coolant flow in V-200 experimental facility are presented. The experiments were conducted to investigate mixing process of water streams with different temperature in upper plenum of experimental pool-type reactor. Using the results of modeling and their comparison with the experimental data conclusions about the type of leak, temperature pulsations and their impact on components of the reactors have been made. The assessment of CFD modeling potential, resource consumption and code verification, as well as issues the phenomena of such type has been performed.
Key words: CFD-modeling, computational fluid dynamics, experimental unit V-200, BN, convection, mixing coolant, temperature ripple

UDC 620.17

Concerning the Application of Some Solutions of a Variational Problem in Research Thermostressed Conditions of Thin Shell under Action of Local Thermal Loadings

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This paper investigates the problem of temperature extrema in a thin cylindrical shell under local axially symmetrical heating when this problems is used to calculate the optimum parameters of three-dimensional shell structures. The relevant variation problem is defined and the equations which determine the extreme thermal load are received. A mathematical model is provided to study thermal stress at the intersection line of two connected shells
Keywords: local heating, temperature field, state of thermal stress, cylindrical shell, thermal stresses, welded joint

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Specification of the Concept of VK-50 Reactor Stability Boundary

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In the operating experience of vessel boiling reactors a concept of stability boundary is widely used. The meaning of this concept is determined by the fact that exceeding the stability boundary leads, as a rule, to scram by the power level or power increasing rate. The scram cause is a rise of neutron flow fluctuation amplitude up to inadmissible values. The more working pressure of vessel boiling reactor decreases the more the stability factor affects the selection of operating limit by the reactor power level. That is why, during development of power plants with lowered working power (for example, nuclear heating plants, nuclear cogeneration plants), it is necessary to have a reliable design toolset to predict stability margin of their operating conditions. The experimental study of VK-50 reactor stability as well as the experience of operation of this reactor plant with the lowered working power shows that in order to create the adequate calculation instrumentation it is necessary, first of all, to specify the ideas of stability margin and stability boundary. The ground for such revision is given in this article.

Key words: boiling water reactor, natural circulation, vessel, neutron flux density fluctuation, stability, instability, transfer function, resonance, autocorrelation function, engineering stability boundary.

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VVER-1000 Reactor Vessel Annealing Process Prototyping

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Nuclear power plant (NPP) life extension is a pressing issue in Russia and abroad. Units with WWER-1000 implemented at the beginning of the eighties are close to expiry of their design lifetime. It is related to Balakovo NPP Unit 1 and Kalinin NPP Unit 1 first of all. The main structural element of the Unit with nuclear energy plant defining its resource is reactor vessel. The reactor vessel annealing is a compensating measure for prolongation of its lifetime. Prolongation of lifetime of a great number of WWER-440 reactor vessels is provided by realization of recovery annealing of the irradiated weld of reactor vessel. The recovery annealing of the irradiated welds of reactor vessel shall be realized at Balakovo NPP Unit 1 to prolong the lifetime of its reactor vessel. Structural differences between reactor vessels of WWER-1000 and WWER-440, as well as the difference of chemistry of materials of the types of these vessels did not allow to apply the techlology and equipment for reactor vessel annealing developed for WWER-440. This article provides justification of necessity and structural aspects of full-scale benchmark for prototyping and development of the technology of WWER-1000 reactor vessel metal annealing.

Key words: prolongation of lifetime of NPP with WWER-1000, WWER-1000 reactor vessel annealing, full-scale benchmark of reactor vessel metal annealing.

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Experimental Research of Hydrodynamics of Coolant Flow Behind Spacer and Mixing Grids FA-12PLUS of WWER-1000

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Results of experimental investigations of local hydrodynamics and intercell mass transfer of coolant flow in a core of WWER with FA-12PLUS have been presented. The aim of work has been consisted in investigating distribution of local hydrodynamic and mass-transfer characteristics of flow in FA behind spacer and mixing grids. Investigations were carried out by the diffusion method (the tracer-gas method) with the help of pneumometric probes. Cells distribution of axial speed components, tracer concentration distribution in experimental model have been received on the base of conducted investigations results. The obtained results allowed to specificate and to visualize coolant flow pattern (picture) behind mixing and spacer grids of FA-12 PLUS. Investigation results have been taken for practical use by Africantov OKBM JSC when assessing heat engineering reliability of cores WWER-1000 with FA-12 PLUS.

Key words: FA (fuel assembly), spacer and mixing grids, hydrodynamics, heat mass-transfer.

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Results of Application of the Least Error Method for WWER Power Determination Using Data of Neutron Flux Monitoring System

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Results of modeling with application of the method of least error during definition of neutrom power of the reactor using out-of-pile detection units of neutron flux monitoring equipment (NFME) in emergency conditions (without using algorithms of correction of NFME data).
Key words: NFME, detection units, method of least error.

UDC 001.2

Improvement of geometric options of the SN-code PMSNSYS-II

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The new ODETTA's LD scheme for solving of the transport equation on tetrahedral meshes was realized in the SN code PMSNSYS-II. Numerical results are presented which demonstrate the correctness of realization of the scheme in PMSNSYS-II. The accuracy of this numerical approach was investigated.
Key words: SN-method, PMSNSYS-II, ODETTA, ATTILA, LD-scheme, arbitrary tetrahedral meshes.