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Research Article

Results of validation and cross-verification of the ROK/B design code on the problem of loss of cooling in the spent fuel pool*

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Abstract

The procedures of validation and cross-verification of the newly developed computational code ROK/B are described. The main problem solved using the ROK/B code is the substantiation by calculation of the coolant density in the spent fuel pool (SFP) and the temperature regime of the fuel assemblies during a protracted shutdown of the cooling systems (break in the supply of cooling water). In addition to the above, it is possible to use the ROK/B code to carry out calculation of an accident with the discharge of the coolant from the SFP with simultaneous prolonged shutdown of the cooling systems.

The ROK/B code allows carrying out calculations for various types of designs of the fuel assemblies and VVER reactors, in particular, VVER-1000, VVER-1200 and VVER-440 power units with single- and two-tiered fuel assembly arrangement, with clad pipes in racks (for compacted assemblies storage) and pipes without cladding, with cased assemblies and caseless ones.

During fuel reloading, a high level of the coolant is maintained, which makes it possible to do "wet" transportation of the assemblies from the reactor to the SFP. The mathematical model for heat and mass transfer calculation, including the boiling coolant model, implemented in the ROK/B code, includes: the motion equation, equations for calculating the enthalpy along the height of the fuel section of a fuel assembly with natural circulation of coolant within the channel containing the fuel assembly (lifting section) and in the inter-channel space (lowering section), the equation of mass balance between the channels of the racks with assemblies and in the inter-assembly space and the amount of evaporated (and outflowed) water, the heat balance equation for a fuel rod in a steam environment. The system of equations is supplemented by closing relations for calculating the thermal physics properties of water and steam, fuel and cladding materials, as well as the coefficients of heat transfer from the wall to the steam, hydraulic resistance and density of the steam-water mixture in the channels, and the heat released in the reaction of steam with zirconium.

Validation of the computational code was carried out on the basis of the data of the ALADIN experiment performed by German specialists and the data of JSC OKB Gidropress. Cross-verification of the ROK/B code was carried out in comparison with the results of calculation using the KORSAR/GP and SOKRAT/B1 codes. Based on the results of the validation, it has been concluded that the deviation of the ROK/B results from the experimental data is not more than 2 to 10% (10% for the option with a fuel rod power of 20 W). Based on the results of cross-verification, it has been concluded that the discrepancy between the ROK/B results and the calculation results for the KORSAR/GP and SOKRAT/B1 codes is not more than 0.5% (for SOKRAT/V1) and less than 10% (for KORSAR/GP).

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Keywords

Spent fuel pond, ROK/B code, KORSAR/GP, SOKRAT/V1, loss of cooling, VVER, swelling, validation, fuel rod, fuel assembly

Introduction

In connection with the fact that spent fuel assemblies (SFA) continue to evolve decay heat for as long as many years after they are withdrawn from the reactor, the spent fuel storage systems in the NPP designs include spent fuel pools (SFP) used to cool fuel for several years before it is shipped for disposal and recycling. The key purpose of SFPs is to reduce the FA activity and to remove decay heat. We shall note that the SFP design includes an SFA cooling system (Safutin et al. 2005, Kritsky et al. 2008, Gagarinskiy 2014).

Beyond-design-basis accidents can be expected to involve an admissibly prolonged failure of all active normal-operation channels and safety systems, which may lead to a severe accident scenario in the event when the spent fuel pool cannot be provided with makeup water in a timely manner using alternative design-basis technology (mobile machinery) (Phenomena Identification and Ranking Table). In Japan in 2011, for example, most of the critical safety components at the Fukushima Daiichi site were lost or severely damaged as the result of the beyond-design-basis ocean wave impact (Wang et al. 2017).

The experimental studies used to verify Russian and foreign thermal-hydraulic codes with the coolant parameters typical of the SFP emergencies include experiments undertaken at a reflooding test facility at OKB Gidropress in the 1960s and the 1990s (Report 76-0-011 1962, Partmann et al. 2018) and later experiments at the ALADIN facility (Petkevich and Uvakin 2013). At the present time, this data provides a framework for verifying the SOKRAT/V1, KORSAR/GP, LOGOS (Pantyushin and Sorokin 2009, Petkevich and Uvakin 2013, Aleksandrova et al. 2014) and other codes.

The key issue addressed by the upgraded ROK/B code is to justify analytically the coolant density value in the SFP and the assembly temperature behavior during long-term failure of cooling systems (loss of cooling water). Besides, the ROK/B can be used for calculations for an accident with the SFP coolant escape involving a simultaneous long-term failure of the cooling systems.

The ROK/B validation in this study was based on the above experimental results. We shall note that the use of dedicated programs is a common practice within design organizations and is dictated by the need to:

- reduce multiply the code's count time as compared with the system codes;
- increase the stability (no program aborts if the input has not been altogether correct) as compared with system codes;
- use rigid computational patterns embedded in dedicated singly certified codes;
- consider the same phenomena in the SFP as in system codes with a much smaller resource capacity;
- obtain the same range of results as with system codes with a much smaller resource capacity;
- investigate, in a rapid and reliable manner, some of the effects and phenomena (e.g., coolant swelling) using relatively simple procedures embedded in the dedicated code;
- simplify the code operation procedures;
- train design engineers in using dedicated codes for a shorter time than in the event of system codes.

Some examples of dedicated codes are STAR-1, TIGRSK, TIGRSP and the ROK/B code presented by the authors (developed in 1990, in the process of upgrading since 1995).

This study aimed to:

- validate the ROK/B program based on experimental data;
- cross verify the ROK/B against the KORSAR/GP and SOKRAT/V1 codes.

Physical and mathematical models of ROK/B CODE

We shall consider in brief the description of the ROK/B physical and mathematical models. The geometry of the region of interest is shown in Fig. 1 which presents the case of the SFA one-tier arrangement.

It is assumed that the SFP forced cooling system stops to operate at time $t = 0$ and cooldown is only through natural convection. The driving pressure drop is defined entirely by the temperature difference in the SFA lower and upper parts. The coolant boils up locally as it reaches the saturation temperature, this leading to the need to calculate the steam-water mixture parameters. We shall note another important circumstance imposed by the SFP design: the initial conditions for the second tier calculation with a two-tier SFA arrangement are the coolant and steam flow rates obtained in the first tier calculations.

The mathematical model to calculate the heat and mass exchange, including the ROK/B coolant boiling model, includes

– the motion equation

$$\Delta P_{fr} + \Delta P_{ac} + \int_0^{n_{ch}} \rho_{mix}(z) g dz = H_{ic} \rho' g \quad (1)$$

– the ratio to calculate the enthalpy at point z along the FA fuel part height with natural coolant circulation in the channel with FAs (riser) and in the interchannel space (downcomer)

$$i(z) = i_{in} + \frac{Q_{FA}}{H_{FA} G_c} \int_0^z q(z) dz \quad (2)$$

– the equation of balance between the mass variation in the assembly rack channels and in the space between assemblies and the amount of the water evaporated (and flown out) for time interval $\Delta\tau = \tau_i - \tau_{i-1}$

$$\sum_{k=1}^{m} \left[\int_0^{H_{ch}} \rho_{mix}(z)_{i-1} dz - \int_0^{H_{ch}} \rho_{mix}(z)_i dz \right]_k S_{ch} n_{ch} + (H_{ici} - H_{ici-1}) \rho' S_{ic} =$$

$$= \left[\sum_{k-1}^{m} \frac{N_{ch}(k) \cdot n_{ch}(k)}{r} + \sum_{k-1}^{m} G_{stin}^{ch}(k) \cdot n_{ch}(k) \right] \Delta\tau + G_{cl}; \quad (3)$$

– the heat balance equation for the fuel element in a steam environment (in the event of an exposed fuel element)

$$\frac{N_f}{H_f} q(z) dz + \Delta Q_{szr}(z) = \alpha_z \left(\frac{F_f}{H_f} \right) (t_f(z) - t_{st}(z)) + \left(\frac{(mC_p)_f}{H_f} \right) \frac{dt_f(z)}{d\tau}; \quad (4)$$

– the equation of heat exchange between the fuel element and the steam environment

$$\alpha_z (F_f / H_f) \cdot (t_f(z) - t_{st}(z)) dz = G_{st} (C_p)_{st} dt_{st}(z). \quad (5)$$

The system of equations is complemented by closing relations to calculate the thermophysical properties of water and water steam, the fuel and cladding materials, the coefficients of heat transfer from the wall to steam, the steam-water mixture hydraulic resistances and the density in channels, and the steam-zirconium reaction heat. Besides, they include formulas to calculate values ΔP_{fr} and ΔP_{ac} representing the local drag, friction and flow acceleration pressure loss for sections 1 and 2 respectively (Klemin et al. 1980):

$$\Delta P_{fr} = (\xi_l + \xi_r dz/D_h) \times (\rho' w_c^2/2) \times \Psi \{1 + x[(\rho'/\rho'') - 1]\},$$

$$\Delta P_{ac} = \rho_1 w_1^2 - \rho_2 w_2^2.$$

The density of the steam-water mixture in the channel is determined by the relation

$$\rho_{mix}(z) = \rho'(1 - \varphi(z)) + \rho''(z) \varphi(z).$$

More detailed information in the code's mathematical model, including the rationale for selecting the ratio to calculate the coefficient of heat transfer from the fuel surface to a single-phase medium (water) and from the fuel surface to a two-phase medium (steam-water mixture) is provided in (Sledkov and Stepanov 2017).

Equations (1) through (5), Fig. 1 and additional relations use the notations the meanings and measurement units for which are shown in Table 1.

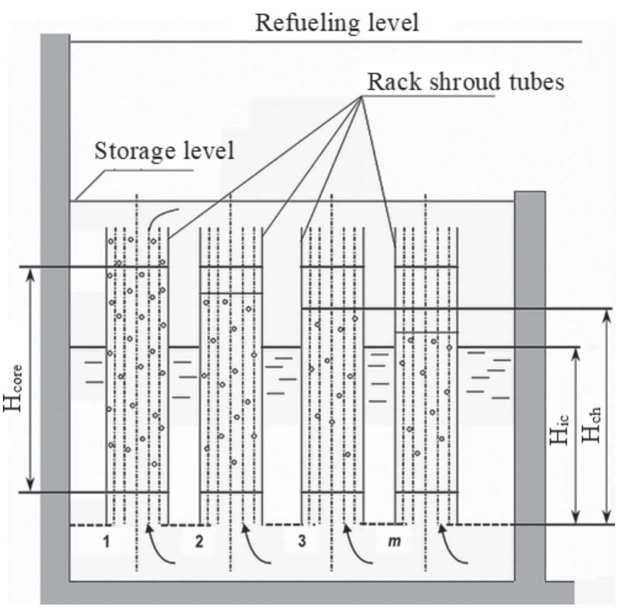

Figure 1. Overall view of the computational region with a single-tier SFA arrangement; 1, 2, 3,..., m – numbers of groups of channels with assemblies.

The steam flow rate distribution obtained at the bottom rack outlet is used as the initial condition for calculating the heat exchange in the top rack. This steam flow rate is determined from the bottom rack SFA power.

The ROK/B code makes it possible to calculate different types of the assembly and SFP designs, specifically, for the VVER-1000, VVER-1000 and VVER-1000 units with single- and two-tier assembly arrangement, with shrouded tubes in racks (for compacted storage of assemblies), with bare and jacketed assemblies, and single- and multi-compartment SFP layouts (for VVER-1000).

Computational patterns for the KORSAR/GP, SOKRAT/V1 and ROK/B CODES

The computational pattern for the KORSAR/GP and SOKRAT/V1 codes (with no fuel part detailing) is as follows. The SFA group is simulated by a group of parallel channels. Axially, each channel is broken down into 12 sections. The ten inner sections along the channel height simulate the FA's main (heated) part, the upper section simulates the FA's unheated upper part, and the lower section simulates the unheated lower part.

The computational pattern for the ROK/B code is similar to that for the KORSAR/GP code with the only difference that the channels are grouped depending on the FA power, and each channel is broken down into 50 to 100 axial sections.

One of the most complicated issues in calculations of heat exchange for two-phase flows is calculation of void fraction, φ. A comparative analysis in (Sledkov and Stepanov 2017) has shown that the best-suited correlation was that from OKB Gidropress obtained in 1962 at a water-air test bench with a full-scale unheated VVER-1000 FA mockup with air fed to the assembly inlet instead of steam (Report 76-0-011 1962). The correlation from OKB Gidropress is presented by the relation

$$\Delta w = 0.13 \times \exp(4.9 \times \varphi),$$

where $\Delta w = w'' - w'$ is the relative steam velocity, m/s; $w'' = w_{п}/\varphi$ is the true steam velocity, m/s; $w' = w_w/(1-\varphi)$ is the true water velocity, m/s; $w_{st} = Q_{FA}/(\rho'' \times r \times S_{ch})$ is the steam velocity reduced to (cross-section S_{ch}), m/s; and $w_w = G_c/(\rho_w \times S_{ch})$ is the reduced water velocity, m/s. At the present time, the OKB Gidropress correlation is used in the ROK/B code for the void fraction determination.

Table 1. Notations used for the ROK/B mathematical model

Value	Meaning	Dimension
1	2	3
H_{core}	Fuel rod heating height	m
H_{ic}	Interchannel space water level height	m
H_{ch}	Channel swell level height	m
ρ_{mix}	Steam-water mixture density	kg/m^3
Z	FA fuel part axial point coordinate	m
G	Gravity acceleration	m/s^2
ΔP_{fr}	Friction pressure loss	Pa
ΔP_{ac}	Flow acceleration pressure loss	Pa
ρ'	Saturated water density	kg/m^3
i_{in}	SFA channel inlet enthalpy	kJ/kg
Q_{FA}	SFA power	kV
H_{FA}	SFA height	m
q(z)	Power density value at axial point z	J/m
m	Number of groups of channels with assemblies	
S_{ch}	Channel flow area	m^2
n_{ch}	Number of channels	
S_{ia}	Interassembly (interchannel) space flow area	m^2
N_{ch}	Channel power	W
r	Specific heat of evaporation	J/kg
$G_{st\ in}^{ch}$	Top rack channel inlet steam flow rate	kg/s
$\Delta\tau$	Time interval	s
G_{cl}	SFP coolant leak rate	kg/s
N_f	Fuel rod power	W
H_f	Fuel rod heating height	m
ΔQ_{szr}	Steam-zirconium reaction heat	W
a_z	Fuel-to-steam heat-transfer coefficient	W/(m^2×K)
F_f	Fuel heat-exchange surface	m^2
t_f	Fuel rod temperature	°C
t_{st}	Steam temperature	°C
C_p	Specific heat capacity of water	J/(kg×K)
ΔP_d	Drag pressure loss	Pa
ξ_l	Local drag coefficient	–
ξ_r	Friction resistance coefficient	–
D_h	Hydraulic diameter	m
w_c	Coolant circulation rate	m/s
Ψ	Two-phase flow inhomogeneity correction factor	–
x	Mass void fraction	–
ρ''	Saturated steam density	kg/m^3
φ(z)	Channel void fraction	–
G_c	Coolant natural circulation rate	kg/s

ROK/B validation and cross verification results

The ALADIN experiment (Partmann et al. 2018) was undertaken by German experts on an FA dummy with simulation of adjoining FAs by peripheral fuel rod rows. The initial water level was relatively low and the pressure fitted the barometric pressure. The FA dummy rod temperature values were determined throughout the FA height and the assembly coolant level was also determined. This helped determine the time to the onset of the rod heating part uncovering. The collapsed level was measured in the channel with FAs in the process of the experiment. The time to the onset of the fuel heating part uncovering was determined from the fuel upper portion temperature increase. The swell level in the FA channel is assumed to reach the top of the fuel heating part at the given instant. The collapsed level in the channel with FAs and in the interassembly space was measured at the time and was found to be below the swell level. It was in the same way that the difference in the FA channel swell and collapsed levels was determined. We shall note that the above difference was determined also in an experimental study (Research Work Report 1985).

The importance of measuring this parameter is obvious as the difference in levels is the initial parameter for the nuclear safety analysis. The experiment was undertaken for the following fuel power values: 20 W (the power for an assembly of 96×20 = 2.88 kW), 50 W (the assembly power is 7.20 kW), 70 W (the assembly power is 10.08 kW), and 100 W (the assembly power is 14.40 kW). These powers are comparable with the decay heat power for an FA

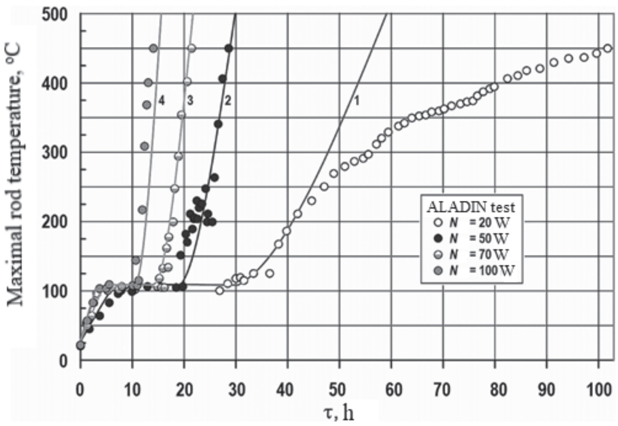

Figure 2. Experimental dependences of the maximum fuel temperature variation on time compared against the ROK/B calculations for different fuel rod powers: 1 – N_f = 20 W; 2 – N_f = 50 W; 3 – N_f = 70 W; 4 – N_f = 100 W.

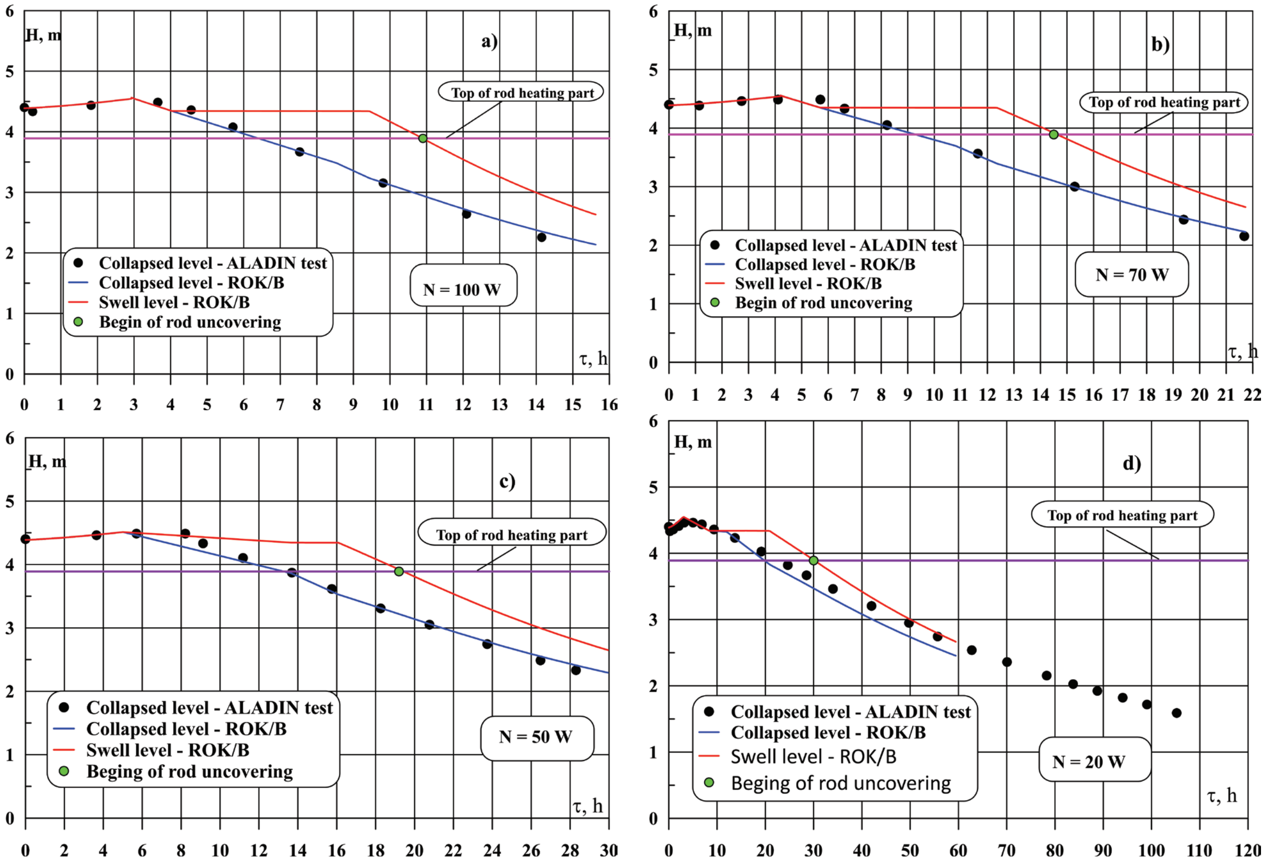

Figure 3. Compared experimental and calculated data of the coolant level dependence on time for different fuel rod powers.

after month-long (or longer) cooling in the VVER SFP. In our opinion, it is exactly where the major experiment drawback lies as applied to the VVER SFPs (Partmann et al. 2018), where no powers of fuel rods in the freshly withdrawn FAs are considered (~ 320 W).

Fig. 2 presents the results of the ROK/B code validation with the ALADIN data. It can be clearly seen from the figure that the ROK/B code shows a good fit with the experiment data, except small-power fuel rods of 20 W (corresponds to the power of the SFA fuel rod after long-term cooling in the SFP), this being explained by the fact that the experimental FA power becomes comparable with the test facility heat loss value (the fuel heat-up rate in this case differs from the estimate).

Fig. 3 presents variations in the coolant swell level, H_{phys}, and collapsed level, $H_{mass,}$ (collapsed level) in the SFP obtained using the ROK/B code and the experimental points from (Research Work Report 1985, Partmann et al. 2018). The ROK/B coolant level is seen to fit the experiment. The time for the ROK/B swell level to be reached for the fuel rod heating part corresponds to the instant the fuel heated part becomes uncovered in the ALADIN experiment. This also indirectly confirms that the selected Gidropress correlation is correct for the void fraction determination.

The steam-water mixture weight is equal to

$$\rho_{mix} \times F \times H_{phys} = \rho' \times F \times H_{mass}$$

The collapsed level, H_{mass}, is the coolant level as it could be with the parameters on the saturation line, m; and F is the area, m^2. The collapsed level is occasionally referred to as the weight level. Since r_{mix} is an unknown value, no actual (swell) level can be determined. The instrument-measured pressure drop is converted to the collapsed level as follows:

$$H_{mass} = P/(\rho' \times F).$$

Fig. 4 presents the difference between the SFA swell and collapsed levels, DH_{PM}, obtained in experiments (Research Work Report 1985, Partmann et al. 2018) (round markers) and using the ROK/B code (curve). The figure demonstrates that the ROK/B describes the experiments very well in terms of the level difference parameter.

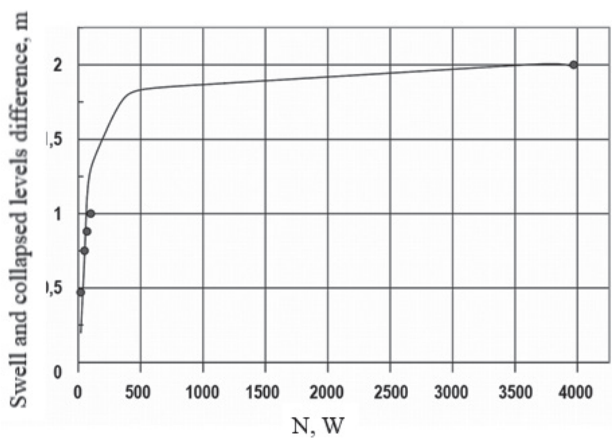

Figure 4. Calculation results as a function of the experimental data for ΔHpm with different fuel rod powers.

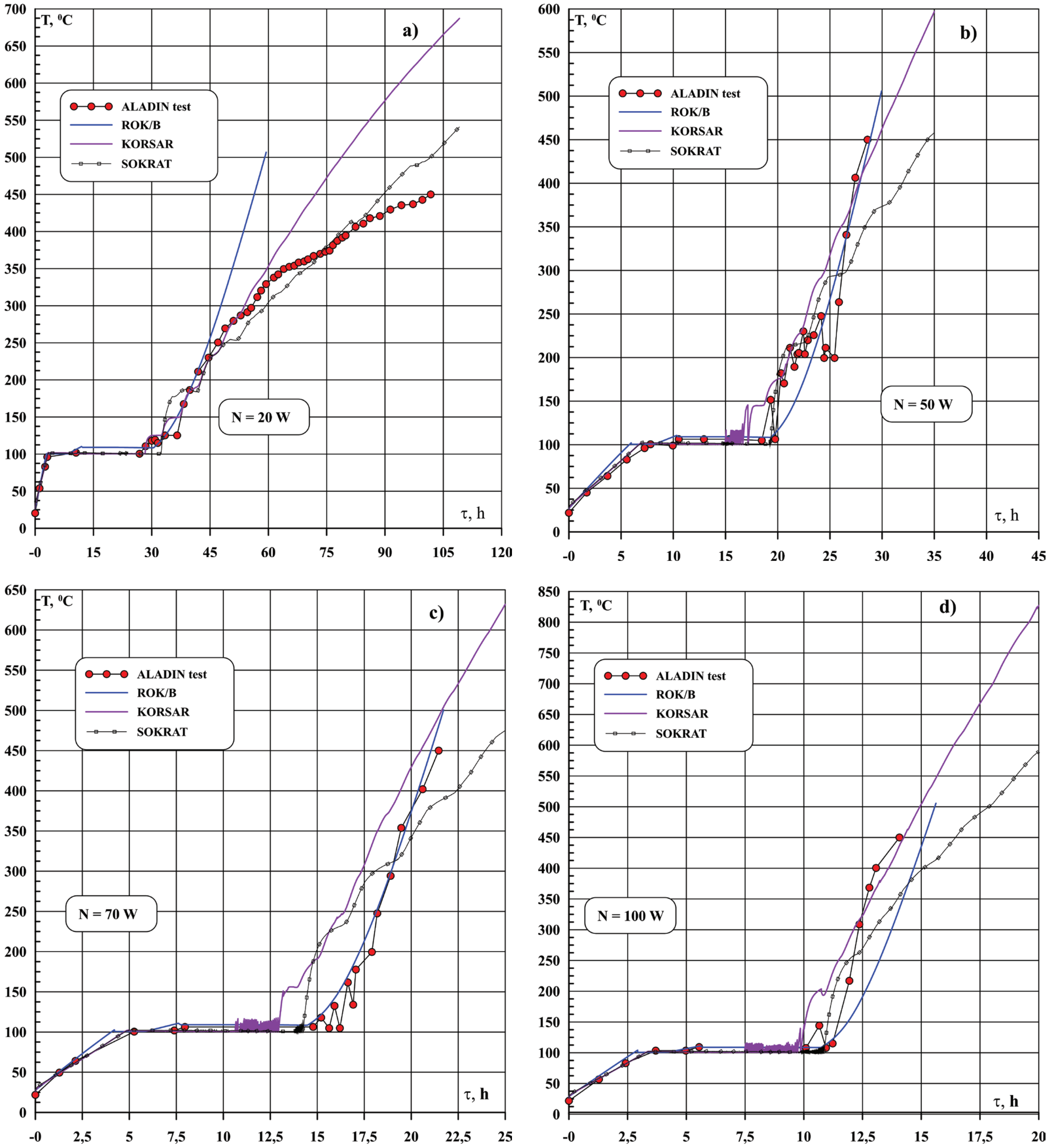

Figure 5. Cross verification results **a.** rod power 20 W; **b.** rod power 50 W; **c.** rod power 70 W; **d.** rod power 100 W (Partmann et al. 2018).

Experiment (Research Work Report 1985) was undertaken for a 126-rod FA of 500 kW (fuel rod power of 4000 W) and at a pressure of 0.35 MPa. Experiment (Partmann et al. 2018) was undertaken for a 144-rod FA of up to 14.4 kW (fuel rod power of 100 W). It can be seen from the figure that the higher is the fuel rod (FA) power, the greater is the difference in the swell and collapsed levels (the lower is the collapsed level).

The cross verification results are presented in Fig. 5 as comparative dependences of the fuel temperature dynamics obtained using the ROK/B, KORSAR/GP and SOKRAT/V1 codes in relation to the ALADIN experiments (Partmann et al. 2018) for different power levels. It can be seen that the codes, including the ROK/B, gives a good fit with the experiments for temperatures of up to 450 °C.

Fig. 6 presents the dependence of the time to the onset of the fuel surface uncovering on the fuel rod powers found with the use of different codes. It can be seen that the KORSAR/GP fuel uncovering time is somewhat smaller than in the experiment.

The time to the fuel uncovering onset obtained using the ROK/B code agrees well with the experimental data with the fuel rod power being over 50 W. The best experiment description is provided by the SOKRAT/V1 code. It should be however noted that no fuel power typical of an SFA after three-day cooling (~300 W) was considered in the experiment.

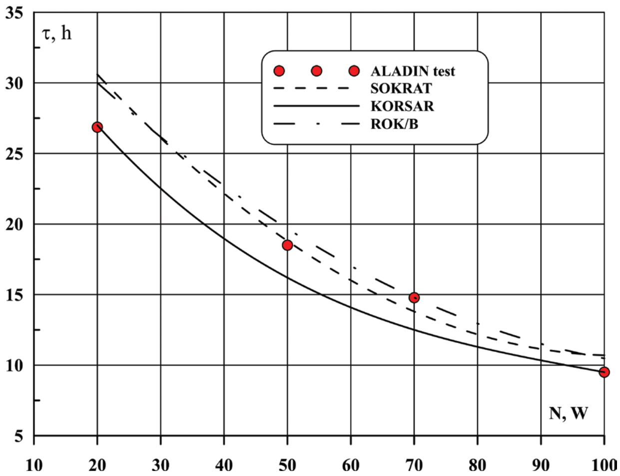

Figure 6. Results of the fuel surface uncovering onset time calculation (Partmann et al. 2018).

Conclusion

The ROK/B code was validated for the key parameters (coolant density, time to the onset of the fuel rod uncovering, swell and collapsed levels, fuel rod temperature). A conclusion has been made that the deviation of the ROK/B calculation results from the experimental data is not more than 0.5% for options with the fuel rod power of over 20 to 50 W, this corresponding to the VVER-1000 FA residual power of 6200 to 15600 W (i.e. the power of an FA after more than one year of cooling).

The cross verification was undertaken by comparing the ROK/B calculation results with the available experimental data and the results of the calculations based on the KORSAR/GP and SOKRAT/V1 system codes. A conclusion has been made that the deviation of the ROK/B calculation results from the SOKRAT/V1 results is not more than 0.5%, and that from the KORSAR/GP results is not more than 10% to 15%.

The validation and cross verification results allow a conclusion that the ROK/B code can be used in thermal-hydraulic calculations for the SFP fuel cooling in conditions with the loss of cooling and leak formation. Validation can be complete after experiments at test benches in the form of SFP models with several FAs. At the present time, such experiment is prepared by French experts (DENOPI test facility) (Research Program 2021).

References

- Aleksandrova OL, Barabanov RA, Dyanov DYu, Kosarim SS, Naumov AO, Spiridonov VF, Filimonkin YeA, Tsiberev KV (2014) LOGOS software package. A finite element method for calculating problems of the static strength of structures taking into account the effects of physical and geometrical nonlinearity. VANT. Ser: Matematicheskoye modelirovaniye fizicheskikh protsessov 3: 3–17. [in Russian]
- Gagarinskiy AYu (2014) RW handling in the nuclear power strategy of Russia. Energiya: ekonomika, tekhnika, ekologiya 7: 2–9. [in Russian]
- Klemin AI, Polyanin LN, Strigulin MM (1980) Thermal-Hydraulic Calculation and Thermal Reliability of Nuclear Reactors. Atomizdat Publishing, Moscow, 327 pp. [in Russian]
- Kritsky VG, Berezina IG, Kalinkin VI, Tikhonov NS, Kozlov YuV, Razmashkin NV, Shafrova NP (2008) The “nuclear renaissance” and prospects of SNF treatment. Bezopasnost okruzhayushchey sredy 1: 68–71. [in Russian]
- Pantyushin SI, Sorokin YuS (2009) Comparison of the SOKRAT/V1, KORSAR/GP and TECH-M-97 codes in analyzing the initial stage of a severe accident at the VVER-1200 reactor plant. Proc. of the 6th International Scientific and Technical Conference on Ensuring the Safety of NPPs with VVER. Podolsk, May 26–29. OKB Gidropress Publishing, Podolsk, 71–79. [in Russian]
- Partmann C, Schuster C, Hurtado A (2018) Experimental Investigation of the Thermal Hydraulics of a Spent Fuel Pool under Loss of Active Heat Removal Conditions. Nuclear Engineering and Design 330: 480–487. https://doi.org/10.1016/j.nucengdes.2018.02.023
- Petkevich II, Uvakin MA (2013) Application of the LINQUAD program for analyzing uncertainties in calculations of a mode with a steam line rupture at the AES-2006 unit using the KORSAR/GP code. VANT. Ser: Fizika yadernykh reaktorov 2: 51–60. [in Russian]
- Phenomena Identification and Ranking Table (2012) Priorities for Loss-of-Cooling and Loss-of-Coolant Ac- cidents in Spent Nuclear Fuel Pools. Nuclear Safety and Regulation. OECD 2018 NEA No. 7443. https://www.oecd-nea.org/upload/docs/application/pdf/2019-12/7443-pheno_id_rank_table.pdf [accessed Jun. 10, 2012]
- Report 76-0-011 (1962) Hydrodynamic Characteristics of the Atmospheric Pressure Air-water Mixture Movement in a Vertical Beam at High Weighted Flow Rates and Elevated Temperature. OKB Gidropress. OKB Gidropress Publishing, Podolsk, 72 pp. [in Russian]
- Research Program (2021) DENOPI Project. Spent Fuel Pool Loss-of-Cooling and Loss-of-Cooling Accident. https://www.irsn.fr [accessed Jun. 10, 2021]
- Research Work Report [final version 341-O-044] (1985) Improving a serial reactor plant with VVER-1000 to ensure operational safety and reliability. Investigating the temperature modes for an assembly of fuel rod dummies at the reflooding stage. GKAE OKB Gidropress. GKAE OKB Gidropress Publishing, Podolsk, 112 pp. [in Russian]
- Safutin VD, Simanovsky VM, Tikhonov NS (2005) Transportation and storage of spent fuel assemblies. In Encyclopedia of Machine Building. Vol. IV, Machine-Building of Nuclear Engineering, Book 2. Mashinostroyeniye Publishing, Moscow, 280–300. [in Russian]
- Sledkov RM, Stepanov OYe (2017) Cross verification of the ROK2 code for the problem with the loss of cooling in the VVER-1000 spent fuel pool. VANT. Ser: Fizika yadernykh reaktorov 1: 32–36. [in Russian]
- Sledkov RM, Stepanov OYe, Galkin IYu, Strebnev NA (2017) An analysis of the void fraction correlations for problems with the loss of the spent fuel pool cooling. Teploenergetika 1: 20–24. [in Russian]
- Wang D, Gauld IC, Yoder GI, Ott IJ, Flanagan GF, Francls MW, Popov EL, Carbajo JJ, Jaln PK, Wagner JC, Gehin JC (2017) Study of Fukushima Daiichi Nuclear Power Station Unit 4 Spent Fuel Pool. Nuclear Technology 180: 205–215. https://doi.org/10.13182/NT12-A14634