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REGULAR ARTICLE

A study of different cases of VVER reactor core flooding in a large break loss of coolant accident

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Abstract. The paper covers the results of VVER core reflooding studies in fuel assembly (FA) mockup of 126 fuel rod simulators with axial power peaking. The experiments were performed for two types of flooding. The first type is top flooding of the empty (steamed) FA mockup. The second type is bottom flooding of the FA mockup with level of boiling water. The test parameters are as follows: the range of the supplied power to the bundle is from 40 to 320 kW, the cooling water flow rate is from 0.04 to 1.1 kg/s, the maximum temperature of the fuel rod simulator is 800 °C and the linear heat flux is from 0.1 to 1.0 kW/m. The test results were used for computer code validation.

1 Introduction

Loss of primary coolant accidents in pressurized water reactors (VVER reactors in Russia and PWR reactors in the West) belong to the most severe cases in the spectrum of accidents in the nuclear power industry. The MCP guillotine break is considered to be the maximum design basis accident. Different thermal-hydraulic processes take place in the reactor in the course of such an accident, namely, a sharp drop of pressure followed by coolant boiling up and loss of primary coolant mass, which leads to partial reactor emptying. At this point the fuel rods heat rapidly to high temperature, due to a sharp decrease in heat removal efficiency. After the emergency core cooling system has been activated, the coolant mass is replenished and the partially-dried out core is reflooded. In later-designed American PWRs water is transported from the ECCS system into the cold leg of the MCP, and in VVER-type reactors the water is uniformly supplied to the upper and lower reactor plenums. Water supply into the upper plenum is connected with the problem of countercurrent flow of the water poured down into the core and the flow of steam released out of the FA. Steam flow is assumed to counteract water penetration into the core from the top and actually "seal" the water level above the core. The paper covers a brief review of reflooding studies performed in different countries and the relevant tests performed in OKB GIDROPRESS are discussed in more detail.

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2 Brief descriptions of experimental studies in western countries

This section contains a brief description of some experimental studies performed in western countries. The background of the studies of the structure of the steam-water flow in the rod bundle goes back to the investigations performed in the USA. In their papers, Lahey and Shiralkar presented the studies of General Electric in a heated 9-rod bundle [1,2]. The purpose of the studies was to determine the velocity fields and the distribution of the flow enthalpy across the rod bundle section. Pressure differentials and flow temperatures were measured in separate fuel rod bundle cells. The fuel rod simulators were not equipped with thermocouples.

At about the same time, investigations were performed on a full-scale 36-rod Marviken facility in Sweden and the results are covered in reference [3]. They were distinguished by the studies of the axial and radial distribution of the flow steam quality. The measurements were made with a gamma-transmission unit. References [1–3] describe studies dealing with boiling water reactors.

The first studies devoted to reflooding in pressurised water reactors date back to the mid-seventies. Reference [4] mentions the studies conducted under the FLECHT program in the USA. The main purpose of these experiments was to obtain data that could be useful for reflooding calculations during loss-of-coolant accidents in the USA. The experiments were performed with a 10×10 bundle with 91 heated fuel rod simulators and 9 simulators of guide tubes that housed the instrumentation. The bundle was placed inside a square housing with a 19.05 mm thick wall and then heated up on the outside during the tests. The fuel rod simulators have an outside diameter of 10.72 mm and were located in a square grid with 14.3 mm pitch, and their heated length was 3.66 m. Electrically-heated fuel rod simulators had a cosine power distribution with peak power of 1.66 due to the different pitch of the internal heater wiring. One of the peculiarities of these experiments was a wide variation of the flooding rate. Also, in these experiments the initial temperature of the fuel rod simulator claddings was relatively low. Thermocouples were installed inside the fuel rod simulators to measure the cladding temperature. The heat flux from the cladding surface was determined by calculation. The temperature of the control rod guide tube simulators was determined with the thermocouples installed inside the tubes. The housing temperature was measured at several points along the height. The flow rate of the flooded water was measured as well as its temperature and pressure at the bundle outlet. Several pressure drop gauges were installed along the column height to measure the water mass in the bundle. One of the specific features was the installation of thermocouples that were built into the wall on the internal surface of the housing at heights of 2.137 m, 3.048 m and 3.810 m. They were used as indications of continuous steam in the given section. In the event that the thermocouple showed the housing temperature to be above the saturation temperature, it was considered to be located inside the superheated steam.

A series of FLECHT experiments was the first to calculate the mass and power balance at the outlet of the testing facility, in order to determine the local conditions and to divide the heat transfer by irradiation between the droplets, steam and housing. In subsequent studies under the FLECHT program, experiments were performed with another heat release profile along the fuel bundle height [5], and with the bundle flow area partially blocked (simulation of fuel rod cladding ballooning during the accident) [6].

The most complete information on all of the issues that deal with reflooding is presented in reference [7]. It identifies and ranks the phenomena that are typical of different modes of the steam-water flow during reflooding. A description of the experiments conducted at 12 different facilities is given with rod bundles of different scales available in the USA and Europe. In addition, many single-tube experiments are considered. Such deep analysis was performed in order to develop the technical requirements to carry out the tests under the Rod Bundle Heat Transfer (RBHT) program. The scope of information that was required to develop mathematical models of reflooding and introduce them into computer codes was determined. Requirements were offered for the RBHT experimental facility unit, and equipment modeling and requirements for the FA instrumentation were defined.

A large cycle of work to study reflooding phenomena was performed in Germany under the FEBA and REBEKA [8,9] programs; this studied the effect of such factors as the presence of a gas gap, internal structure of the fuel rod simulator, heating-induced cladding deformation and the availability of spacer grids. The experiments have shown that the presence of the gas gap, cladding ballooning and rupture contribute to quicker cool down fuel rod.

Since 1976, VTT Energy and the Lappeenranta University of Technology have cooperated in researching nuclear reactor thermal-hydraulics. During these years they have built a series of experimental test facilities (REWET-II, REWET-III and PACTEL). The REWET-II and REWET-III facilities were designed for investigation of the reflooding phase of a LOCA [10]. The main design principle was the accurate simulation of the rod bundle geometry and the primary system elevations. The rod bundle consists of 19 indirectly-electrically-heated simulator rods. The heated length, the outer diameter and the lattice pitch of the fuel rod simulators as well as the number (=10) and construction of the rod bundle spacers are the same as in the reference reactor VVER-440. The aim of the tests was to improve the understanding of the basic phenomena of accident situations and to provide experimental data for the development and verification of the LOCA and SBLOCA codes aimed for analysing pressurised water reactors in use in Finland.

3 Descriptions of experimental studies in Russia

The study of the reflooding processes in Russia began in 1974 in OKB GIDROPRESS, with the investigations using single-rod and 7-rod bundles. The purpose of the studies was to investigate the effect of different kinds of cooling water supply on the cladding temperature. Experiments on the heated 7-rod bundle at the OKB GIDROPRESS test facility were started in 1975. These tests are described in reference [11]. The test facility is a two-loop installation that schematically models the VVER-440 reactor. The facility had a reactor model, one simplified loop with a rupture device to simulate a MCP leak and one large loop with a circulation pump that models the remaining five operating loops. The facility was used to simulate circulation pipeline guillotine break, and also to simulate reflooding of the heated bundle with the cooling water from the ECCS. The above bundle consists of 7 fuel rod simulators 9.1 mm in diameter and the heated length of 2.13 m.

The experiments were implemented according to the procedure below: steam was supplied to the test section and simultaneously the bundle power was smoothly increased. The bundle heat-up was confined to the central rod simulator cladding with the temperature not above 600 °C. The steam was discharged from the circuit via the damaged loop. After the steady state was established, steam supply to the model was quickly interrupted, drainage was stopped, the instrumentation system measurement devices were switched on and the test section was fed with water at 40 °C. After the rod bundle was cooled down to a temperature below 200 °C, the flooding stopped and the power supply to the bundle was interrupted. All-in-all, the experiments covered 11 tests with different versions of flooding. The plots in Figures 1 and 2 show the thermocouple readings in the tests with the core flooding from the top and the bottom.

The plots show that for the 7-rod bundle with flooding from the bottom, the bundle cool down takes place far more quickly and without significant temperature pulses. At this point, the cooling down front goes from the bottom to the top.

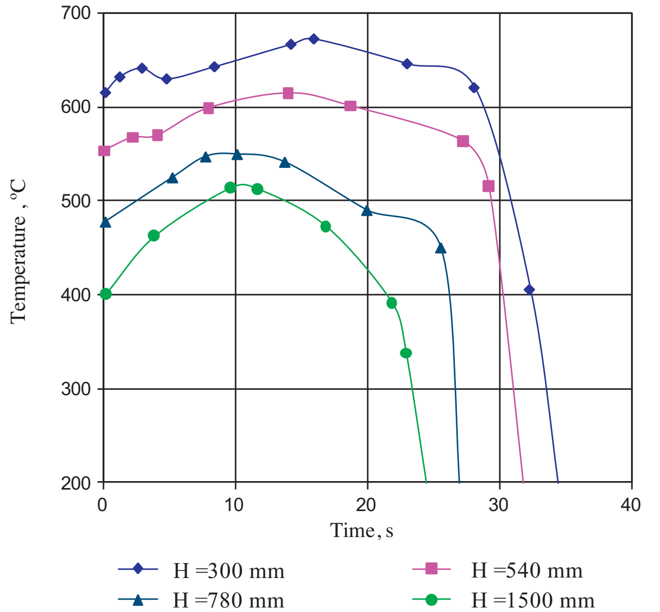

Fig. 1. Variation of cladding temperature in a 7-rod bundle in the case of flooding from the bottom.

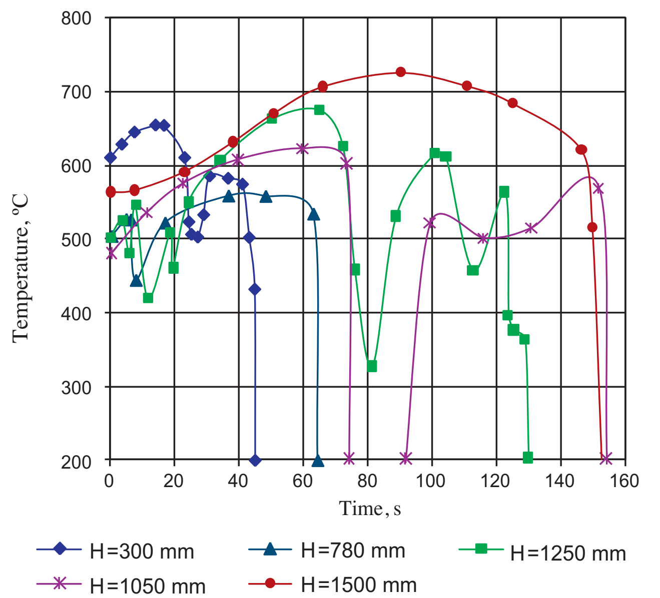

Fig. 2. Variation of cladding temperature in a 7-rod bundle in the case of flooding from the top.

In the case of core flooding from the top, the cooldown time increases significantly. The nature of the cooling front movement is also changed. The area where the outermost upper thermocouple is installed is the first to be cooled down. The thermocouples located below are cooled down later and with considerable fluctuations. This means that it is difficult for water to go inside the narrow bundle. The steam that is leaving the bundle impedes the water flow, i.e. the effect of the countercurrent flow of steam and water is quite significant.

In 1976, construction of the OKB GIDROPRESS test facility began [12]. The venture schematically modeled the primary circuit of theVVER-440 reactor, with a full-scale FA mockup as the core simulator. As preparations for the work were underway, fuel rod simulators were designed, manufactured and tested, and their indirect heating up and thermal-physical characteristics were found to be close to a full-scale actual fuel rod. Such simulators were incorporated into a full-scale mockup FA for VVER-440 containing 126 heated rods 2.5 m long with uniform axial heating. The simulators were axially spaced with cell-type spacer grids 10 mm in height. One unheated rod was placed in the centre of the mockup. The grids were installed with a separation of 240 mm. The test facility was equipped with a large number of thermocouples to measure the cladding temperature, with a probe to measure the swell level along the fuel assembly height in the middle part of the bundle and the pressure values in the central part of the fuel assembly. A schematic diagram of the test facility is shown in Figure 3 and the FA mockup cross-section and the fuel rod simulator location pattern is given in Figure 4.

The procedure for the fuel assembly mockup testing is as follows. The valves in the damaged and operating loops were opened. Steam was supplied to the lower chamber of the test section at 0.3 MPa pressure.

The power supplied to the FA mockup kept increasing until the temperature of the most heat-powered simulator had reached 600 °C. Steam supply increased as the power increased. After the steady state was established, the power was increased spasmodically until it reached the assigned level. Simultaneously all of the recorders were switched on, the steam feed to the test section was stopped and the valve

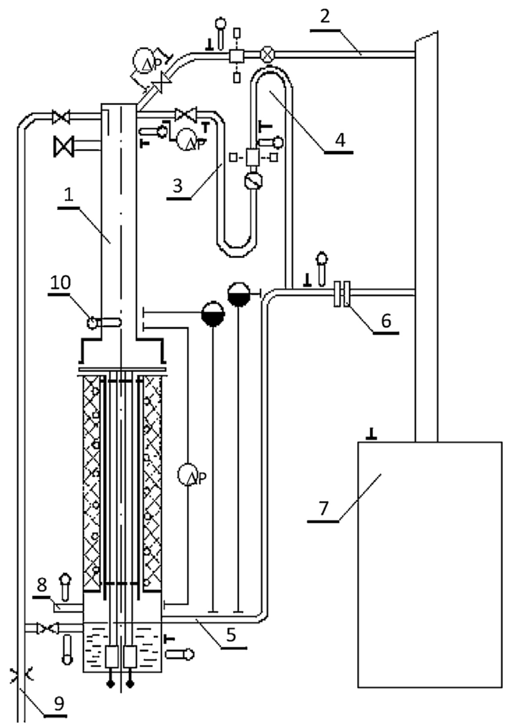

Fig. 3. Diagram of test facility with full-scale mockup of the VVER-440 FA. 1: test section; 2: tube of emergency leg; 3: tube of operable leg; 4: loop seal; 5: downcomer; 6: flowmeter; 7: discharge tank; 8: steam pipe; 9: tube of flooding water; 10: water-steam probe.

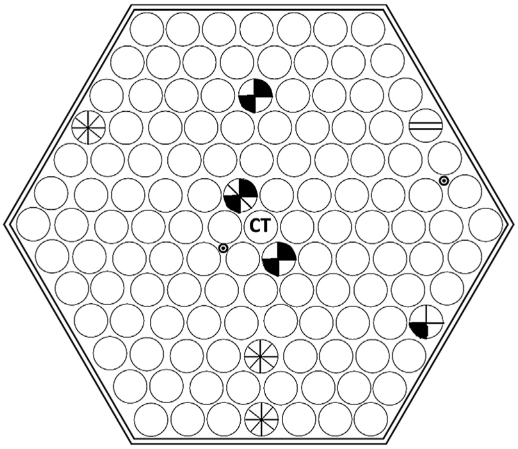

Fig. 4. Diagram showing the layout of imitators in mockup and their equipment by measuring sensors. CT: central tube with 5 submerged thermocouples; ⊕: simulator with 5 thermocouples; ◕: simulator with 4 thermocouples; ◕: simulator with 6 thermocouples; ◔: simulator with 3 thermocouples; ⊜: tube with 4 pressure taps; ⊙: the physical level sensor.

was opened on the cooling water supply line to the test section. In the course of the test, the assigned water flow rate was maintained.

The test was deemed to be over when the FA mockup was completely cooled down (to a temperature below 200 °C). The steam flowrate was limited by pressure increases in the test section not exceeding 0.5 MPa. The plots of the fuel rod simulator cladding temperature variations in two tests with different types of flooding are shown in Figure 5. The thermocouple location points were indicated by their distance from the upper boundary of the simulator heating up.

Parameters associated with the most typical tests of the FA mockup are listed in Table 1, where G_w and t_w are flowrate and temperature of the flooding water respectively, q_{max} is the maximal heat flux, and t_{cl} is the fuel rod simulator cladding temperature.

The results of the FA simulator tests show that in the lower chamber flooding tests, gradual simulator cooldown from the bottom upwards can be observed. As the flowrate of the supplied water decreased, the cooldown time increased. In the tests with combined flooding and flooding from the top, significant pulsations of cladding temperature, especially in the upper and the middle parts of the FA, were observed. In the case of flooding from the top, there was no significant increase in the cooldown time, contrary to the phenomena observed in the 7-rod bundle. The only observation is that the middle part of the FA (the thermocouple is located at a height of 1510 mm) is actually cooled down simultaneously with the upper part of the FA. In addition, some temperature pulsations are observed in the middle part. It is likely that the water poured from the top goes down through the least heated parts of the mockup

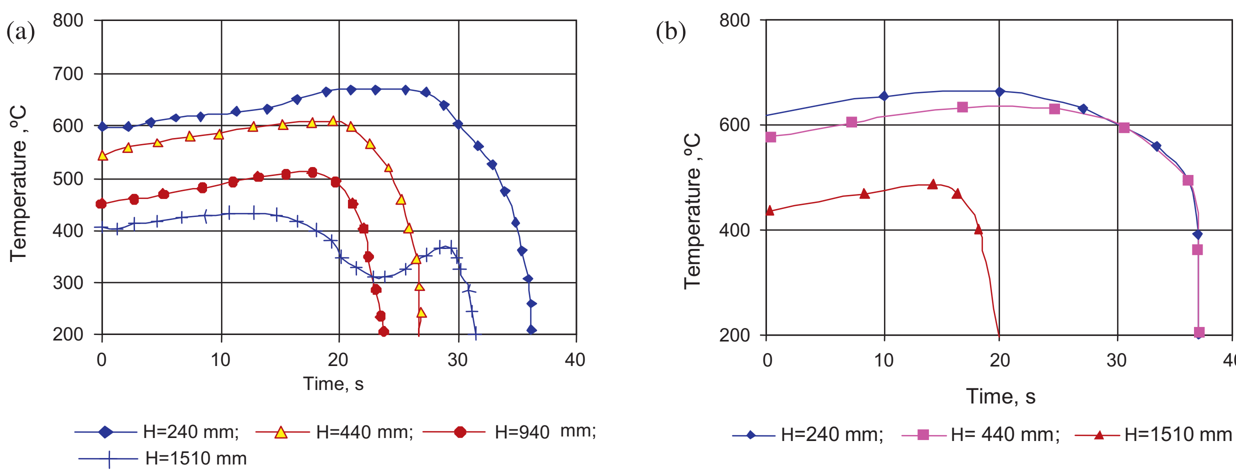

Fig. 5. Variation of cladding temperature depending on the kind of a reflood: (a) reflood from the top; (b) reflood from the bottom.

Table 1. Initial parameters of reflooding tests of the FA mockup.

Test No.	Flooding pattern	G_w, kg/s	t_w, °C	q_{max}, kW/m^2	Max. t_{cl}, °C
1	To the lower chamber	1.4	40	30	570
2	To the lower chamber	2.4	46	24	610
3	To both chambers	1.4/2.4	41	31.2	630
4	To the upper chamber	2.06	46	28.5	640
5	To the upper chamber	1.9	46	31.4	690
6	To the upper chamber	1.9	40	29.2	630

Table 2. Comparison of the parameters of the tests of the 7-rod bundle and the mockup FA for VVER-440.

Parameter	7-rod bundle	FA mockup
Pressure, MPa	0.12	0.12
Specific power per one simulator, kW	2.65	2.24
Fuel rod cladding temperature before flooding, °C	720	690
Amount of flooding water per one simulator, kg/h	128	57

(over the edge of the periphery fuel rods, close to the central tube, along the hexahedral housing surface) and then the flooding proceeds from the bottom.

A comparison was made of the appropriate tests in the FA mockup and 7-rod bundle to investigate the effect of the scale factor on the process of water penetration in the lower chamber when the water is poured from the top. The parameters of the two comparable tests with flooding from the top are listed in Table 2.

It can be seen that for approximately the same mode parameters, even when there is greater water supply to a 7-rod bundle, it takes much longer for the small-scale bundle to be cooled down than to cool down the FA mockup. It is one demonstration of the fact that the efficiency of top flooding is influenced by the scale factor.

At the end of the nineties, SRC IPPE began investigating reflooding processes [13,14]. Testing facilities were created that modeled the primary circuit that contained the testing facilities with bundles of 7- and 37-rod simulators that simulated the geometry of VVER-1000 FAs. Axial heat flux profiling with a power peaking factor of 1.62 was envisaged for the mockups. The 7-rod bundle contained two unheated rods and in the 37-rod bundle the power of the central simulator was 10% higher than that of the others. Flooding from the bottom and combined top-and-bottom flooding were modeled in the experiments.

The experiments were carried out as follows:

– initial state – lower plenum of the test section and the lower part of the rods (up to the level where the heated area begins) are filled up with water, the remaining part of the bundle and the upper chamber are filled up with saturated steam;
– power increased to the assigned level;
– rods heated up to the starting temperature;
– when the assigned temperature is reached, the power begins decreasing under a set law and the cooling water flow is switched on;
– the experiment stops when the rod temperature decreases to the boiling temperature.

Four standard problems were arranged from the numerous tests of these bundles, of which we are going to consider only two. The results obtained in these tests were used to verify the system computer codes KANAL-97 as a part of the computer code TRAP developed in OKB GIDROPRESS and KORSAR/V1 developed in the NITI Research Institute [15]. The description of the structure of the TRAP and KORSAR codes is given in reference [16]. Table 3 summarises the input parameters of the tests used in the calculations.

The time required for complete cooldown of the 7-rod bundle was about 700 s, and that of the 37-rod bundle was 300 s.

It is worth mentioning that both the experiment and the calculations were made in a one-dimensional statement. Therefore, the difference in the parameter behaviour in both bundles can hardly be attributed to the difference in the quantity of the rods. In a 37-rod bundle, the heat flux was smaller and the flooding rate was greater than in the 7-rod bundle.

A factor that accelerates the movement of the hot fuel rod wetting front in the case of flooding from the bottom appears in a multi-rod assembly and moreover, in the core. It occurs due to the fact that in the cold areas of the core, the level was increasing more rapidly which created additional motive water head for the "hot" areas.

In 2003, a new FA simulator was installed in the reflooding test facility that modeled FA VVER-1000. The number of fuel rod simulators was the same as that in the previous mockup (126 pieces). The axial heat in the new mockup exhibited a cosine power distribution ($K_z = 1.345$) and the length of the fuel rod simulator was increased to 3.5 m. The mockups were axially spaced, with the cell-type spacer grids 20 mm thick with a 255-mm pitch.

The FA model was equipped with instrumentation far better than the previous FA VVER-440. Of the fuel rod simulators, 20 were equipped with thermocouples to measure the cladding temperature. In addition, 6 thermocouples were installed inside the cladding of the instrumented simulator. The thermocouples were located at 10 levels along the simulator height beginning with the bundle bottom.

The initial experiments were performed with cold water supply to the upper chamber of the test section (top

Table 3. Parameters assigned in the calculations.

Bundle	Pressure, MPa	Maximum heat flux, kW/m	Flooding rate, cm/s
7 rods	0.278	2.94	2.0
37 rods	0.246	1.77	4.9

flooding). The subsequent tests were experiments with water boiling down (natural level decrease) and with subsequent cold water supply to the reactor downcomer (bottom flooding) at different flowrates and power supplied to the FA.

Methodologically, the experiments were performed in the following way. In the top flooding experiments, a small steam flow (up to 45 kg/h) was supplied from the steam generator within 10–15 minutes through the test section bottom inlet for the sake of heat-up. At the same time, power was supplied to the FA simulator where the maximum fuel rod simulator wall temperature did not exceed 400 °C. After this temperature was reached, the steam flow was quickly arrested, the reactor model was fed with water and within 2–3 seconds the power in the test section rose to the assigned value.

In the tests with water boiling down, the FAs were initially filled up with water, then the power was increased to the assigned level, then water boiled up, evaporated and the FA got uncovered. When the temperature of the fuel rod simulator claddings reached 650 °C, water was fed at the assigned flow rate to the reactor downcomer model. At this point, the upper part of the downcomer model was connected with the discharge line and the water excess streamed down to the discharge tank, i.e. the FA mockup makeup was realized with the free level. Several experiments were performed for each power value, with flood water flow rate varied to get the minimum flow rate at which the FA was cooled down.

During the experiment, the following parameters were recorded:

- the temperature of the fuel rod simulator claddings at all the points;
- flow rate of the supplied water;
- temperature of the flood water;
- pressure in the test section;
- water level in the FA channel and in the reactor downcomer model.

The experiment is complete once the FA has been completely cooled down or if the temperature of the fuel rod simulator cladding exceeds 800 °C.

Four tests were undertaken with the top of the test section flooding. The pressure was equal to 0.15 MPa. The

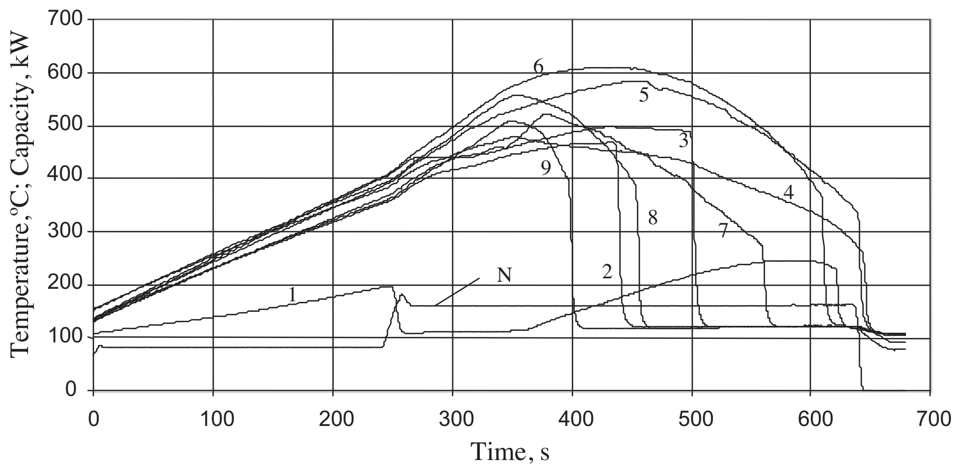

Fig. 6. Distribution of the cladding temperatures on the FA height in test No. 1. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

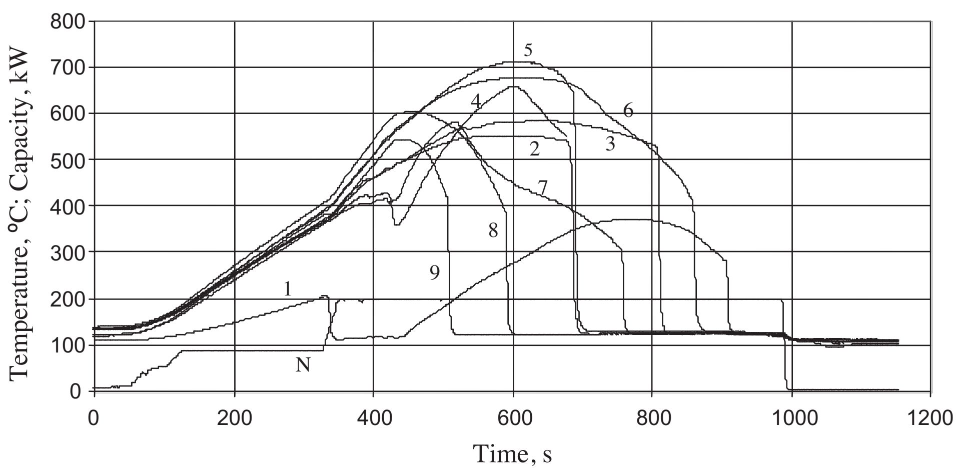

Fig. 7. Distribution of the cladding temperatures on the FA height in test No. 2. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

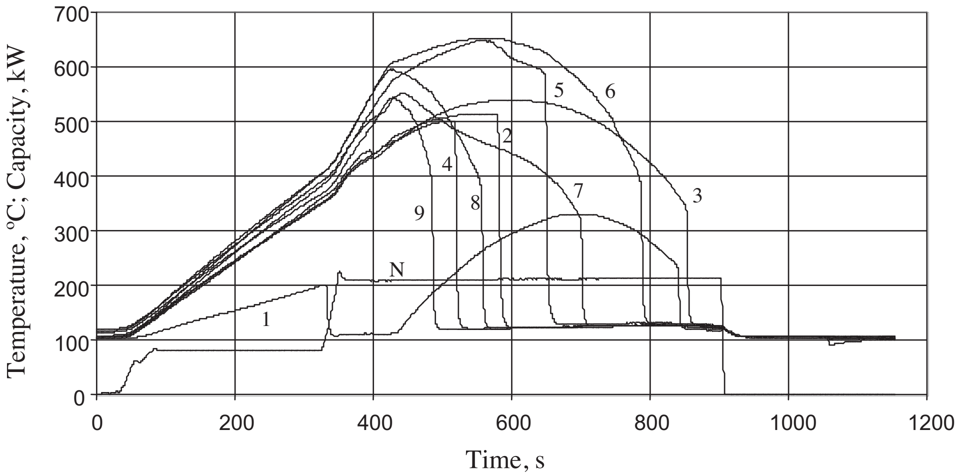

Fig. 8. Distribution of the cladding temperatures on the FA height in test No. 3. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

test results are given in Figures 6–8. The numbering of the cross-sections of the thermocouples takes place simultaneously with the lower boundary of the simulator heating. Detailed arrangement of thermocouples is listed in Table 4.

Cooling down of the FA takes place simultaneously from the FA top and bottom. The central axial parts of the fuel rod simulators remain hot for a longer time.

Full cooldown for top flooding is only realised after 400 s. Test parameters are listed in Table 5.

The maximum heat is released in the middle part of the fuel rod bundle, and therefore the maximum temperatures are also in the middle part of the FA mockup. It is worth mentioning that the flow rate of the fed water in the last experiments was twice as small as in the tests in the FA VVER-440, and far less than the flow rate in the VVER-1000 reactor. This is why it takes longer for the FA mockup to be cooled down. There was no level generation observed in the upper plenum. Thus, the upper flooding can be considered as efficient enough and all the supplied water quickly penetrates in the central part of the FA and cools down. Five tests were performed with level boiling down. The parameters of the boiling down tests are listed in Table 6.

Table 4. Coordinates of an arrangement of thermocouples on height of FA.

Number of section where installed thermocouples	Distance from the bottom of bundle, m
1	0.291
2	0.885
3	1.334
4	1.813
5	2.168
6	2.487
7	2.584
8	3.105
9	3.403

The results of the boiling down tests are given in Figures 9–11. It can be seen that at first, due to water boiling down, there was a level decrease in the mockup and after partial drying out and heat-up of the upper part of the fuel rod simulator cooling water supply into the lower chamber of the experimental model began. Due to water supply, the level in the model increased and the FA mockup was cooled down.

The model cooldown time depended on the flowrate of the cooling water and the value of the supplied power. Each test was repeated several times over in order to get the minimum value of water flow rate at which the FAs were cooled down.

4 Comparison with the system computer codes

As was mentioned above, SRC IPPE [14] has organised some standard problems for verification of the Russian codes. One of experiments has been simulated with the use of a code known as TRAP. The code package TRAP is intended for analysis of the variation of thermal and hydraulic parameters in the primary and secondary circuits and the core of NPP with VVER under conditions incorporating disturbances in operation of the primary and secondary equipment, such as accident conditions including LOCA. It is applied in the analysis of design basis accidents and beyond design basis accidents in substantiation of operability and safety of NPP with VVER and experimental facilities. The assumptions, common for the considered mathematical model, are given below:

- the equations to determine coolant parameters are put down as a one-dimensional approximation, not taking into account the power dissipation or metalwork strain;
- the process of surge of coolant boiling is assumed to be equilibrium from the point of view of thermodynamics;
- coolant movement in pipelines and in steam generator tubes is considered as an approximation to equilibrium steam-water mixture;
- the axial effect of thermal conduction in coolant and metalwork is not taken into account;

Table 5. The main results of the tests with top flooding.

Test No.	Supplied power in % from nominal heat transfer	Water flowrate, kg/s	Temperature of poured water, °C	Linear heat flux per one fuel rod, kW/m	Maximum temperature, °C	Cooldown time, s
1	2.55	1.10–0.9	88	0.46	610	400
2	2.7	1.10–0.9	80	0.61	700	600
3	2.95	1.1–0.9	87	0.67	650	500
4	3.1	1.0	75	0.70	870	Assembly was not cooled down

Table 6. The main results of the tests with level boiling down.

Test No.	Supplied power, kW	Water flowrate, kg/s	Temperature of flood water, °C	Heat flux density, kW/m	Maximum temperature, °C	Cooldown time, s
5	40	0.04	60	0.12	730	1200
6	80	0.07	60	0.24	800	1000
7	160	0.12	67	0.48	700	300
8	230	0.13	64	0.70	750	500
9	320	0.56	64	0.96	700	330

– the primary and secondary circuits of the plant represented in the model as a set of elementary cells (density, specific internal energy, etc.) are determined as average integrated per cells.

A comparison of the plots of cooling down the 37-rod bundle obtained from experiment and from the computations using the TRAP code is presented in Figure 12.

From the figure it can be seen that the peak of the cladding temperature in the experiment is a little more than in the calculation. However, the cooldown time of the bundle coincides for the experiment and for the calculation.

On the experiments OKB "GIDROPRESS" with initial evaporation of water from the test section and subsequent bottom reflooding, calculations with the use of code KORSAR/V1 [15] have been executed. The KORSAR is a code for the analysis of the non-stationary processes in NPP systems. It deals with water-cooled water-moderated reactor systems in stationary, transient and accident regimes, as well. The modeling of the thermal-hydraulic processes in RK KORSAR is performed on the basis of a non-equilibrium two-fluid model in one-dimensional approximation. The neutron kinetics calculation is performed in a quasi-three-dimensional approximation on the basis of the point kinetics model of the reactor.

Test No. 8 from Table 6 has been chosen to represent the results of these calculations. The initial and boundary conditions were set so that to comply fully with the scenario

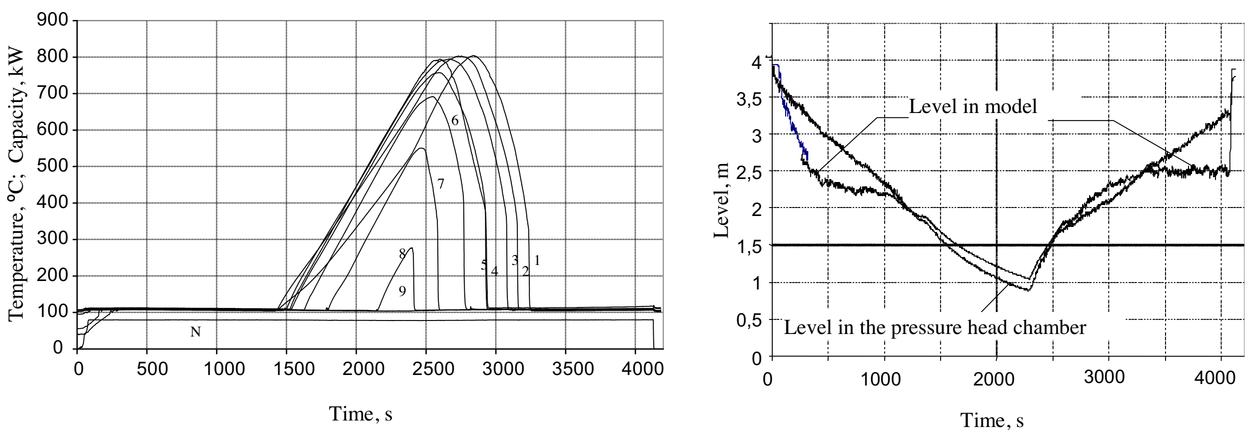

Fig. 9. Test 6. Distribution of the cladding temperature on the FA height. Variation of the level in the FA and the pressure head chamber. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

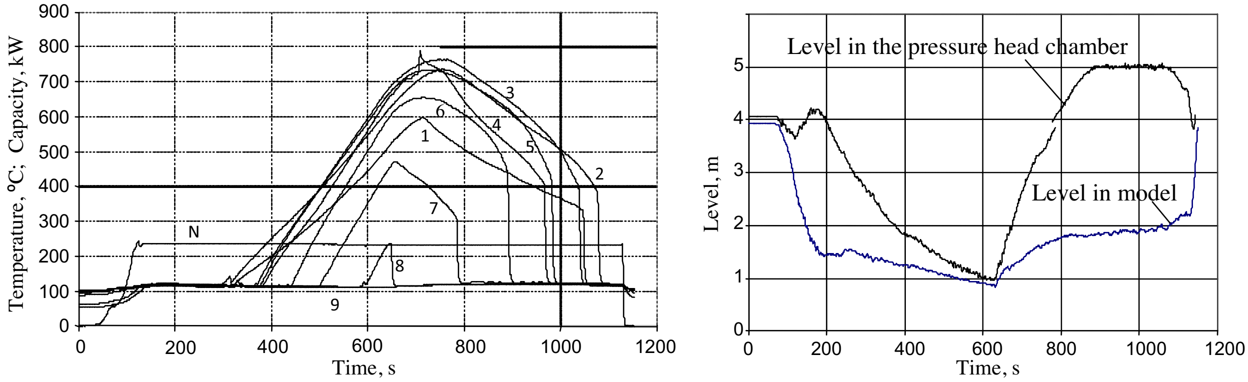

Fig. 10. Test 8. Distribution of the cladding temperature on the FA height. Variation of the level in the FA and the pressure head chamber. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

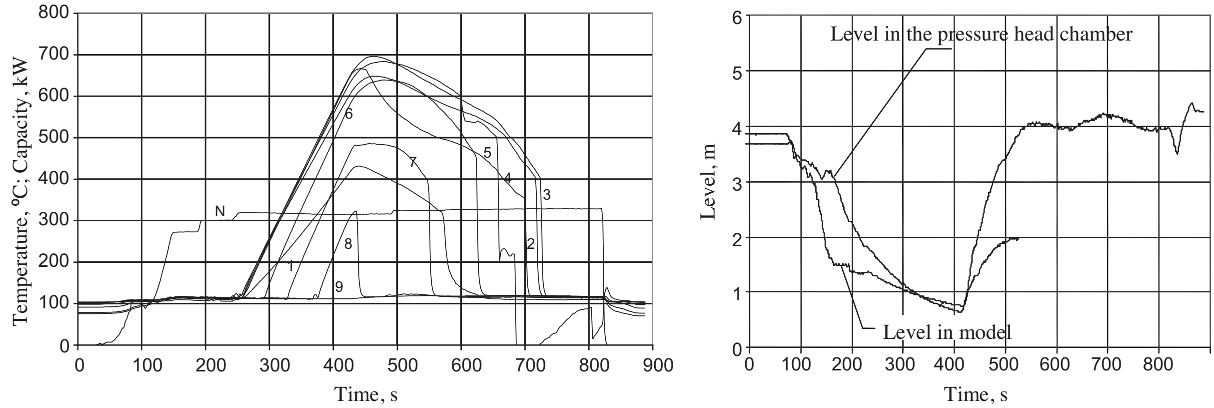

Fig. 11. Test 9. Distribution of the cladding temperature on the FA height. Variation of the level in the FA and the pressure head chamber. 1–9: numbers of sections of the thermocouples arrangement on the FA; N: capacity.

of the experiment. During the initial moment, the pressure in the test section was equal to 0.1 MPa, FA mockup was filled with water and then the capacity was switched on to the rod bundle. The boundary conditions used in the calculations were the given changes in time of values of capacity of simulators, the flowrate, the enthalpy of coolant at the inlet in the test section, and the pressure in the inlet and outlet of a bundle. The comparison of the calculation with the experiment is presented in Figure 13.

The temperature curves are shown for the heat-intensity part of the FA mockup.

It can be seen that the results of the calculation are in reasonable agreement with the experimental data. The peak

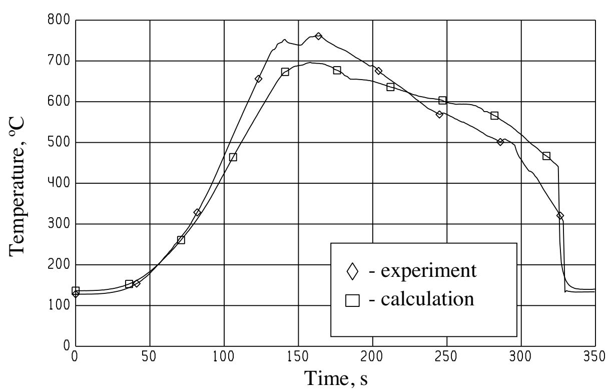

Fig. 12. Comparison of calculations using the TRAP code with experiment from SRC IPPE.

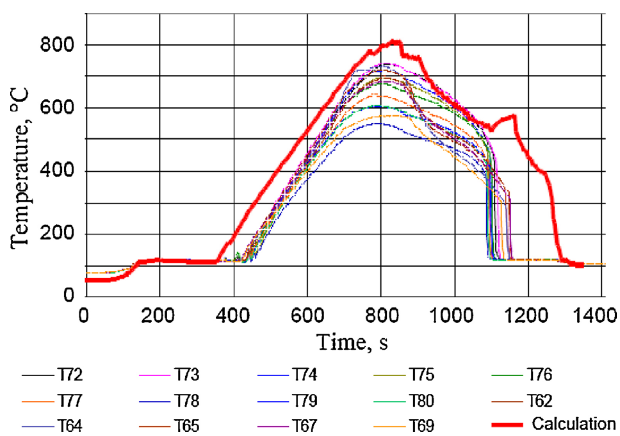

Fig. 13. Comparison calculation with experiment No. 8 from Table 6. Thermocouples located 2.487–2.584 m from the bottom of the bundle.

of the cladding temperature in the calculation is a little more than in the experiment. The code conservatively predicts the cooling down process of the experimental model.

5 Conclusions

The majority of the studies performed in OKB GIDROPRESS were devoted to top flooding. The reason is clear, as the flooding applied in the VVER is made immediately into the upper and the lower chambers of the reactor. The tests have shown that scale factor, i.e. the number of rods in the FA mockup influences the effectiveness of coolant supply from the top.

The experiments of OKB GIDROPRESS show that as the transverse dimension of the FA mockup increases, the flow choking of the water supplied from the top by the steam flow significantly decreases. This agrees well with the conclusions of the experiments in the UPTF facility [17,18] in Germany, where no flow choking was observed.

The experiments in the bundles with lower number of rods were performed at the end of the nineties in SRC IPPE. From the results of these experiments, several standard problems were solved and the Russian codes TRAP and KORSAR were verified.

It is worth mentioning that all of the experiments in OKB GIDROPRESS were performed in the rod bundles with strain-free fuel rod simulators equipped with cell-type spacer grids of small height. These grids, apart from the contemporary spacer grids applied in the VVER had small pressure loss coefficient and did not actually interfere with the cooling front movement. In the new designs mixing grids were additionally introduced to the spacer grids of increased height, which have considerable hydraulic resistance. Foreign researchers [19] have observed in experiments with bundles equipped with spacer grids of greater height with deflectors that the cooling front is passing, flood water accumulation is sometimes observed upstream of the grids and the cladding temperature increases downstream of the grids. However, this was not observed at low flooding rates; it only happened at high water flow rates with water supplied from the bottom.

The results of the studies in the RBHT facility in the USA [20] show that in case of bottom flooding, the spacer grids located along the bundle height quickly get wetted with the droplets of water flying in the steam flow, and have a temperature far lower than the cladding temperature in the same cross-section. No temperature increase of simulator claddings was observed in the places where the spacer grids were installed.

The introduction of the mixing grids into the new RP VVER designs and also for the operating NPPs with power increased to 107% requires experimental studies of the effect of the mixing grids for core reflooding in loss of primary coolant accidents.

Nomenclature

VVER	water-cooled and water-moderated power reactor
SRC IPPE	State Research Centre RF "Institute for Physics and Power Engineering"
NITI	Technology research institute
MCP	main circulation pipeline
PWR	pressurised water reactor
RP	reactor plant
ECCS	emergency core cooling system
FA	fuel assembly

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