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FUEL ASSEMBLY TESTS
UNDER SEVERE ACCIDENT CONDITIONS
(PARAMETER-SF test series)

Scientific and Research Final Report
on the work of ISTC Project No. 3194

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Abstract

In the final report the results of calculational-experimental studies are given. The studies were performed according to the Work Plan on ISTC Project No.3194 "Fuel assembly tests under severe accident conditions". The Project was executed jointly by three organizations: FSUE SRI SIA "LUCH", IBRAE RAS, FSUE EDO "GIDROPRESS" with participation of the leading specialists from A.A. Botchvar FSUE VNIINM, A.I. Leypunskiy SRC RF IPPE, RRC "Kurchatov Institute" and the methodical support of foreign collaborators (FZK, GRS, EdF, IRSN).

Within the framework of the Project two experiments were performed of the PARAMETER-SF test series with 19-fuel rod model FAs of VVER-1000 completed with the standard reactor structural materials.

The conditions of these experiments reproduce the conditions of the initial phase of severe accident of LB LOCA type when the core drying occurs, as well as its heating and reflooding with water in case of restoration of the emergency flooding systems. In the PARAMETER-SF1 experiment the behaviour of VVER-1000 assembly overheated to 2000ºC under top flooding conditions was studied. In another PARAMETER-SF2 experiment the efficiency of the combined top and bottom flooding was studied for the VVER-1000 assembly overheated to 1500ºC.

In the PARAMETER-SF2 experiment the initial stage of severe accident with large break LOCA was simulated with the core drying, its heating-up to ~ 1500°C and top and bottom water flooding.

The obtained results facilitate understanding of the phenomenology of the processes occurring under water flooding of the overheated core in the course of severe accident of LOCA type at NPPs with VVER and under similar situations at PWR, and can be used for verification of SFD computer codes.

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Introduction

In terms of consequences the most serious accident at NPP with VVER (PWR) is beyond design basis accident with loss of coolant (LOCA) – severe accident that could lead to core melting, damage of reactor installation vessel, release of hydrogen, fission radioactive products inside the containment and into the environment.

Special measures (accident management) must be taken to mitigate the consequences of severe accident. Elaboration of adequate technical solutions and measures on accident management is based on the knowledge of the laws of accident scenario and response of the reactor to interference (actions on accident management) into the natural course of accident process. One of the urgent measures on accident management is the reflooding of the overheated core. Thereupon, experimental and computational studies of physico-chemical processes occurring in the overheated core during quenching are relevant. And one of the most important tasks can be considered the integral experiments aimed at checking the effectiveness of accident management measures.

The main method of SA analysis is the numerical modeling using computer codes. The complexity, consistency of the physical processes and phenomena, accompanying the accident scenario including the stage of temperature escalation and the core reflooding require the comprehensive verification of the calculated models. Moreover, it is necessary to have a good representation of the core behavior control in the course of accident and of possible techniques of its cooling down to make the justified solutions on accident management and bringing the reactor into a safe state. In this respect, the study of the initial stage of BDBA with the investigation of a possibility of cooling down the overheated core as a possible way of accident management is of particular importance. Obtaining such information requires, in its turn, the detailed experimental studies of fuel rods assemblies, including studies at ex-reactor test facilities.

At present time the similar experiments are performed in Research Centre of Karlsruhe within the frame of the program of QUENCH experiments aimed at studying the mechanical and physical-and-chemical behaviour of claddings of the overheated fuel rods under intensive bottom cooling. However in these experiments the pellet simulators of zirconium dioxide are used instead of fuel pellets of uranium dioxide that not permit to reproduce the process of the high-temperature interaction of the core materials under severe accident conditions.

The ex-pile experiments on studying the severe fuel damage (CORA-W1, CORA-W2), performed in 1993 on simulators of VVER-1000 assemblies with the standard fuel pellets of uranium dioxide in Research Centre of Karlsruhe in close cooperation between

Russian and foreign organizations, proved absence of principal differences in the laws of development of severe damage of assemblies in VVER-1000 and PWR. However in the program of the mentioned experiments no studying of the mechanical and physico-chemical behaviour of fuel rods under bottom flooding conditions cooling was provided.

The issue concerning behaviour of the reactor core structural materials under severe accident conditions under top flooding, as well as under the combined top and bottom flooding is also studied insufficiently.

In this connection the aim of Project No. 3194 was studying of the behaviour of the two 19-fuel rod model assemblies of VVER-1000 completed with the standard reactor structural materials, and namely, with the fuel rod claddings of Zr+1%Nb alloy, fuel pellets of uranium dioxide, spacer grids and shroud of Zr+1%Nb alloy, under the conditions of the severe accident initial stage including the stage of low rate cooling at the top flooding an the high rate cooling at the top and bottom flooding.

The Project was executed jointly by three organizations:

- FSUE EDO “GIDROPRESS” – the leading organization of the Federal Atomic Energy Agency engaged in NPP safety justification;

- IBRAE RAS – the leading Institute of the Russian Academy of Sciences engaged in safety analysis of nuclear power engineering objects;

- FSUE SRI SIA “LUCH” – the organization possessing the unique test facility PARAMETER, the experience in performing the ex-pile tests of VVER-1000 model FAs with the following material studies.

The work under the Project was executed with participation of leading specialists from A.A. Botchvar FSUE VNIINM, A.I. Leypunskiy SRC RF IPPE, RRC “Kurchatov Institute” with the methodical support of foreign collaborators (FZK, GRS, EdF, IRSN).

According toe the Work Plan of ISTC Project No. 3194 the two experiments were performed of PARAMETER-SF test series.

In the PARAMETER-SF1 experiment the behaviour of VVER-1000 assembly overheated to 2000ºC under the top flooding conditions was studied. In PARAMETER-SF2 experiment the efficiency of the combined top and bottom flooding was studied for the VVER-1000 assembly overheated to 1500ºC.

The performed experiments allowed:

- to obtain the information on the behaviour of the FA overheated to ~ 2000°C under top flooding;
- to study the propagation of the cooling front for the assembly heated to the temperature of ~ 2000°C at the low rate top flooding of the model assembly;

- to obtain the information on the behaviour of the FA heated to $\sim$ 1500°C under combined top and bottom flooding;
- to study the propagation of the cooling front for the assembly heated to the temperature of $\sim$ 1500°C at the high rate combined top and bottom flooding of the model assembly;
- to obtain the information on physico-chemical processes and structural changes in fuel rod claddings and in fuel pellets;
- to extend the database for verification of computer codes.

Results of tests are intended for improving the understanding of the phenomenology of the processes in VVER core and under similar situations at PWR in case of the core heating under severe accident above 1200°C with the subsequent flooding and can be used for verification of computer codes.

1. SCENARIO OF EXPERIMENTS OF THE PARAMETER-SF TEST SERIES

One of the specific features of VVER reactor plant design is a possibility of the core flooding from top and bottom simultaneously. Under severe accident with the initiating event of LB LOCA and with superimposing of the additional failure, for instance, at complete failure of the ECCS active part with NPP blackout. The core drying occurs. As the initiating event the large coolant leak of Dnom 200 is considered, that corresponds approximately to the size of pipelines connected to the main coolant pipeline at the reactor inlet. In VVER-1000 there is possibility to restore the core cooling. One of the peculiarities of the given RP design is a possibility of the core reflooding from top and bottom: low-pressure pumps supply boron solution into 2 loops (into cold and hot legs), and then to the reactor collection (RCC) and pressure chamber (RPC) (Fig. 1). Water supply simultaneously from top and bottom allows to avoid the situation when the most part of the cooling water can be entrained into leak. Under top water supply the core cooling is started earlier than with bottom water supply only.

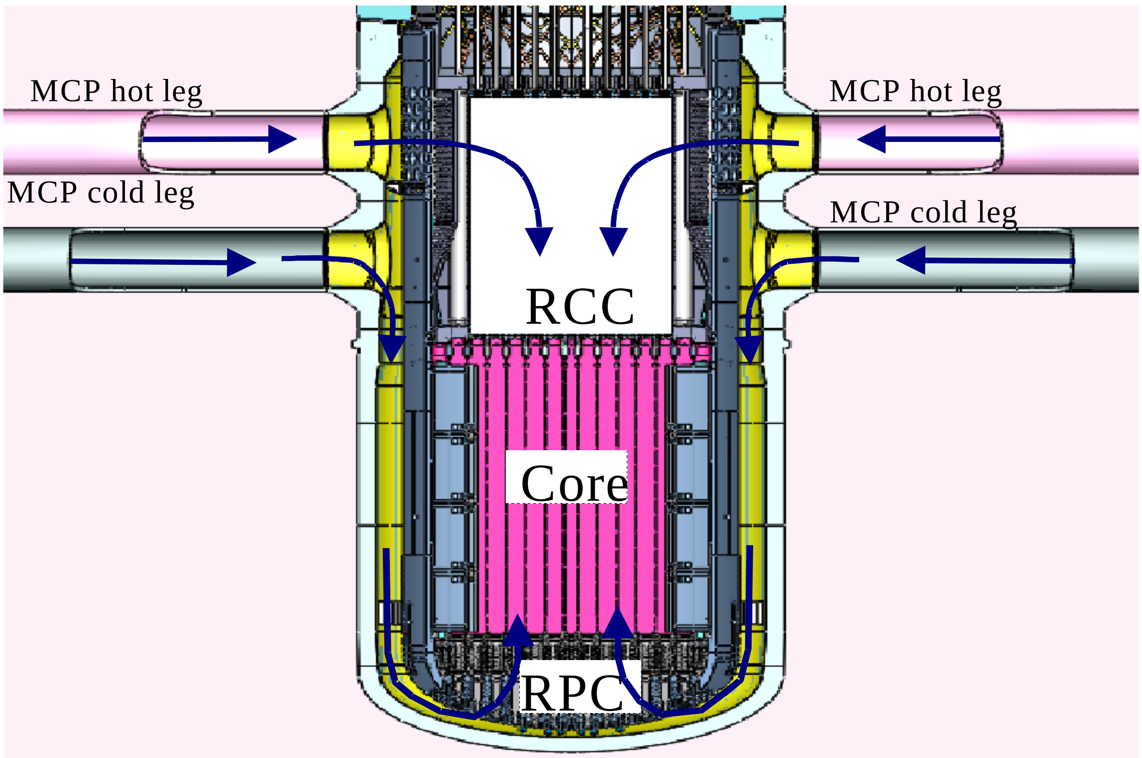


Fig. 1. Scheme of the core reflooding.

Water, supplied by the low-pressure pump in the course of flooding, is redistributed between the reactor collection and pressure chambers depending on the backpressure. Depending on the number of the connected low-pressure pumps the flooding water flow rate, entering the core, can be varied from 1.4 g/s to 19.7 g/s in terms of 1 fuel rod. These values have been obtained proceeding from the maximum capacity of the low-pressure pumps. The backpressure in the primary circuit can decrease the flow rate of this pump. The delay time for connection of low-pressure pump influences greatly the maximum temperature of fuel rod cladding.

Under top flooding of the core a part of water in the form of droplets and jets can pass through the core and partially enter the pressure chamber. A part of water as a result of boiling shall be entrained in the form of steam into the collection chamber entraining also a part of water coming from the top. Therefore, only a part of the top water goes to direct core cooling as a result of wetting. For description of the process of water entrainment with the opposite steam flow (the so called “flood”) the appropriate models are used in computer codes, however the phenomenology of this process is extremely difficult and not sufficiently studied in the experiments, some parameters of these models require exact determination, and the models themselves require careful verification. The core bottom flooding is more definite because after filling the reactor pressure chamber the water front comes upwards and boiling of water begins with steam generation, and the fuel rods are cooled with steam and then with water after wetting.

During first 30 seconds of operation of the low-pressure pumps a greater amount of water is supplied to the collection chamber, then, after appearance of water level in the core, the greater amount of water is supplied to the reactor pressure chamber. During these 30 seconds wee can expect a decrease in the temperature of fuel rod claddings in the core upper part which sizes are not determined. The calculations show that the steam from boiling of water coming from the bottom, results in displacement of the top flooding water, its subsequent cooling is started only after the lower flooding front passed a half of the core.

After flooding the core (the calculated hot channel) by 2/3 of the height (to the level of 2.5 m) the water supply from the top increases greatly. By this moment the core hot places have been already cooled to the saturation temperature.

Position of the phase interface boundary under flooding in the VVER core is essentially non-uniform both over the height, and radius. This phenomenon depends on many factors; distance from the nozzle through which water is supplied, the FA power, temperature filed, etc. The data on steam quality in the core show that in the course of the top and bottom flooding there are moments when the hot place (the calculated section in the hot channel) is surrounded with water on the top and bottom.

The core cooling is governed by two main processes:

- entering of water into the upper chamber, non-uniform distribution over the core section; water passing through the core and its cooling; filling the lower chamber with the rise in the coolant level;

- increase in the water level in the lower chamber and the core both due to bottom flooding, and due to entering a part of water under top flooding.

Positive role in the core cooing is attributed to the spacing grids wherein the droplets break occurs with increase in surface and the evaporation rate, as well as the steam generation in contact of droplets with the grid.

The following shall be considered to the first priority directions of studies of the core top and bottom flooding:

- determination of the water fronts motions under top and bottom flooding;

- determination of formation of the steam flow lifting motion;

- study of the efficiency of cooling the experimental fragment of FA (19-rod assembly) with non-equilibrium steam-water flow at the assigned values of steam flow and water droplets;

- determination of hydrogen release under top and bottom water flooding of the assembly.

- study of thermo-physical, thermo-mechanical and chemical processes in the fuel rod assembly under their flooding at high temperatures.

On the basis of the results of numerical analysis of the processes in the core, in case of leak with the equivalent diameter of 200 mm, using code TECH-M, the scenario of the experiment at the PARAMETER test facility can be constructed. Proceeding from the length of the model FA active part, the 1/3 part of the calculated hot channel located in the core upper part, or in the center, is modelled. Parameters of water, entering the model assembly, are determined according to the thermohydraulic characteristics of the calculated channels, as well as by the conditions in the collection and pressure chamber.

The measured parameters of the experiment:

- cladding temperature in the points over the assembly height and section;
- coolant temperature;
- amount of hydrogen;
- water flow rate for the “top” flooding;
- coolant inventory at the test section outlet;

The model FA parameters and characteristics studied after the tests:

- state of fuel rods and spacing grids;

- assembly flow area in different place over the height;
- degree of oxidization and hydrogenation of the cladding.

The experiments are implemented in the following way:

- the fuel rods are heated in the steam medium to the assigned cladding temperature;
- water is supplied to the test section upper part and the water droplet flow is formed from top to bottom;
- the fuel rods are cooled as a result of droplets motion, their break, interaction with claddings, with each other and with grids;
- the test section lower part is filled and with its filling the process of fuel rods cooling is intensified with steam flow (water evaporation in the section lower part);
- pressure in the test section is maintained at the level determined in the calculation.

After bench tests the material and mechanical tests are performed for the model assembly components.

2. PARAMETER-SF1 EXPERIMENT

2.1. Results of the PARAMETER-SF1experiment

The experiment PARAMETER-SF1 was performed on April 15, 2006, the full information on the results of the experiment is given in Annex 1.

The conditions of the experiment simulated the severe stage of LOCA type accident when the core, overheated to 2000^{o}C, is flooded from the top in case of restoration of the emergency flooding system.

For performing the PARAMETER-SF1 experiment the process systems of the PARAMETER test facility were prepared according to the functional diagram (Fig. 2).

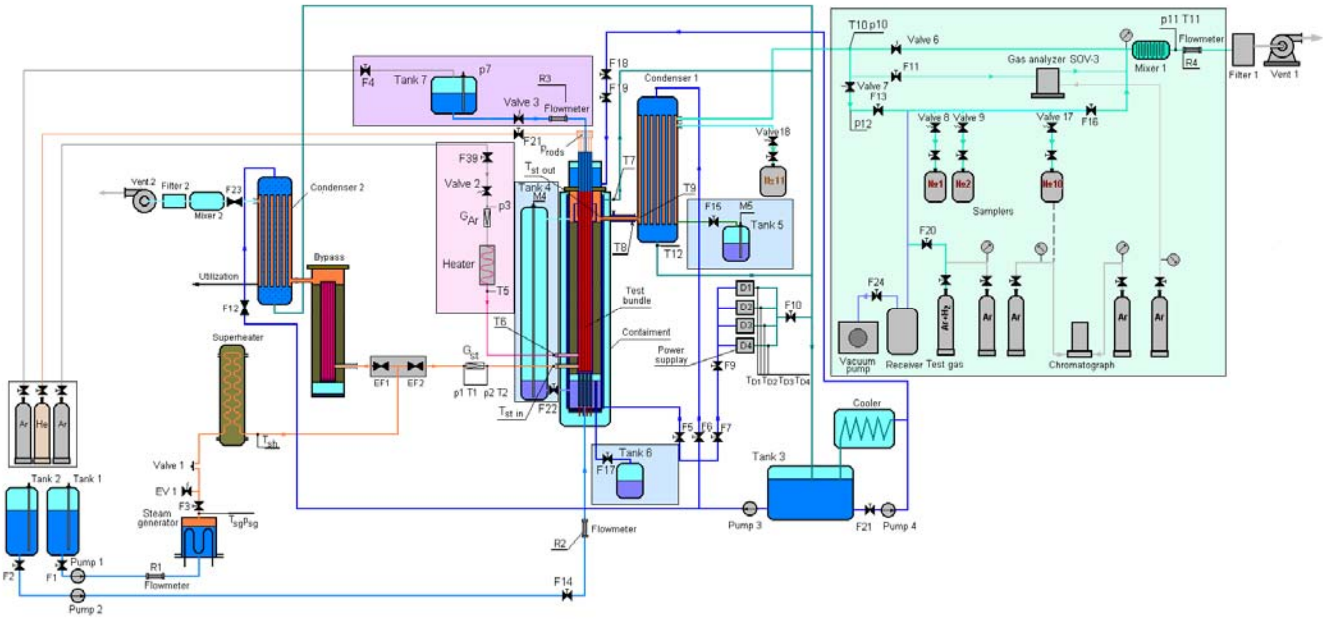


Fig. 2. Diagram of the PARAMETER test facility
(SF1 experiment).

The SF1 experiment included the preparatory stage with duration of about 7000 s, the stage of pre-oxidization of claddings (7000 – 14945 s) and the flooding stage. At the preparatory stage the model FA was heated in the argon medium to 500°C, initially at the low flow rate of argon, then, at the assigned flow rate of 2.0 g/s, with introducing after 5445 s of steam flow with the flow rate of 3.3 g/s (see Fig. 3).

Before ~10400 s the PARAMETER-SF1 experiment was performed according to the pre-test protocol of tests. All the measuring and process system functioned satisfactory.

Beginning from ~ 11000 s it was found out that the behaviour of some FA parameters deviates from the expected values. In particular, increase in electric power (from 4.6 to 6.4 kW) resulted in rise of FA temperature to higher level that was expected at the assigned convective heat removal with the steam flow of 3.3 g/s (see Fig. 4, 5).

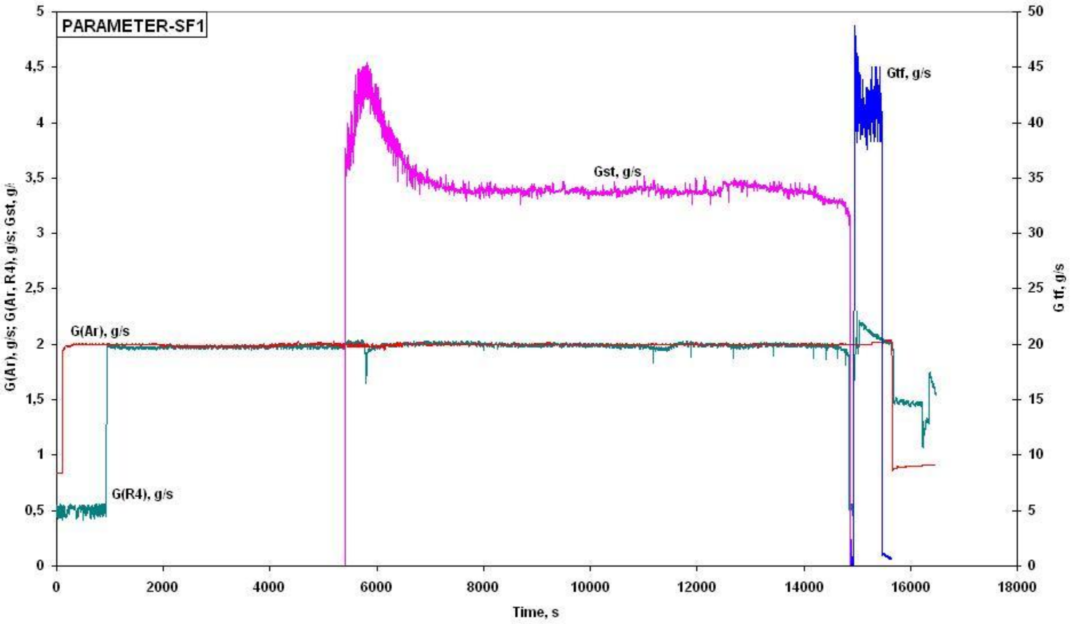


Fig. 3. Flow rates of argon (G_{Ar}, $G_{Ar}(R4)$), steam (G_{st}) and top flooding water (G_{tf}).

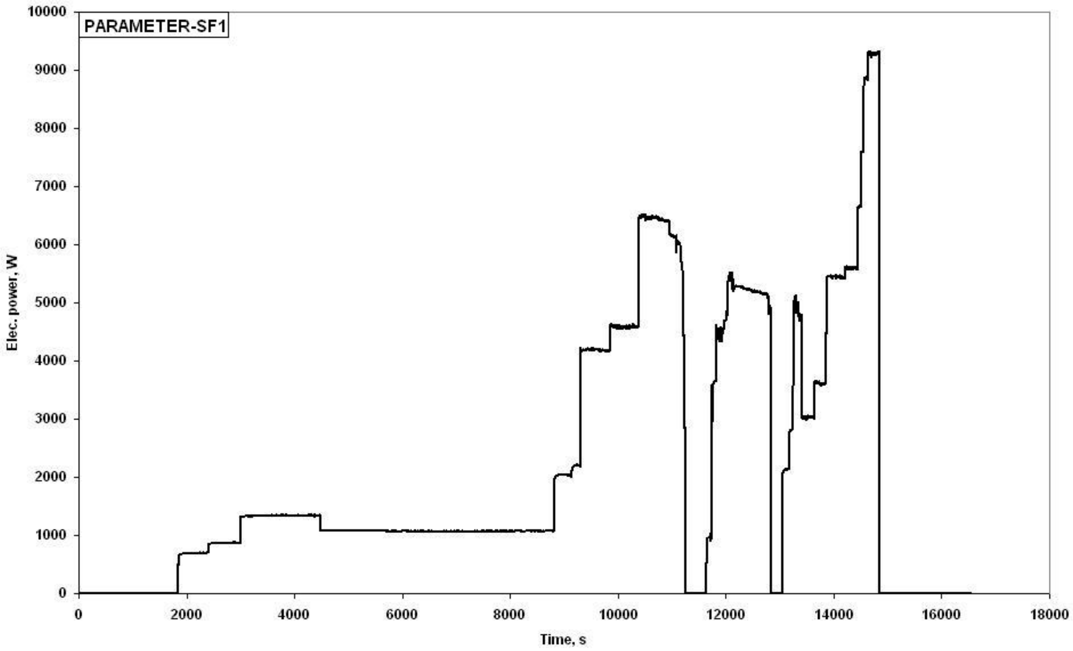
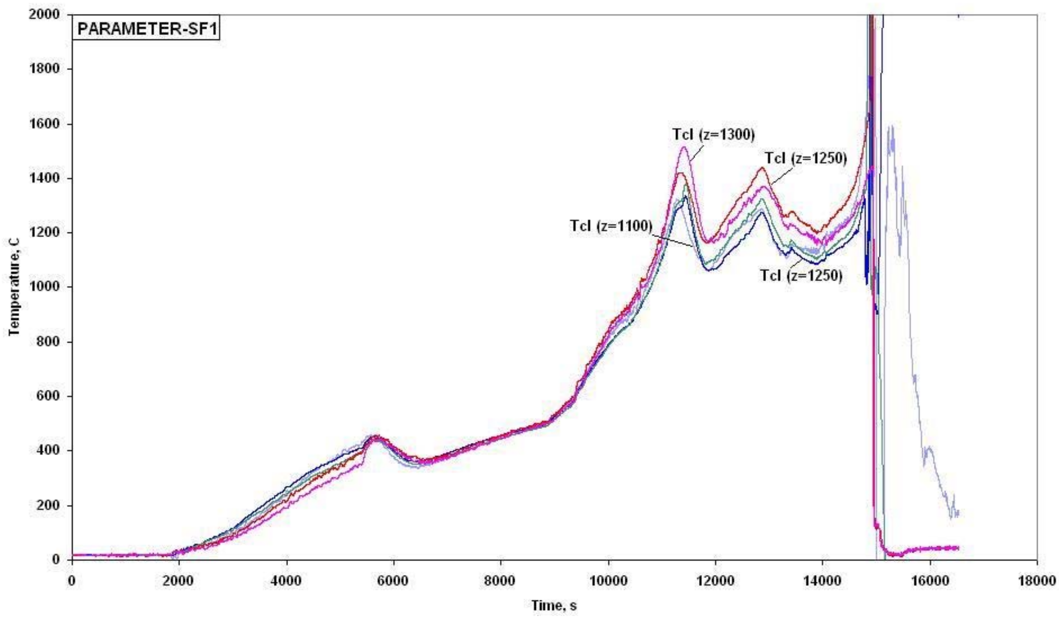


Fig. 4. FA electric power.

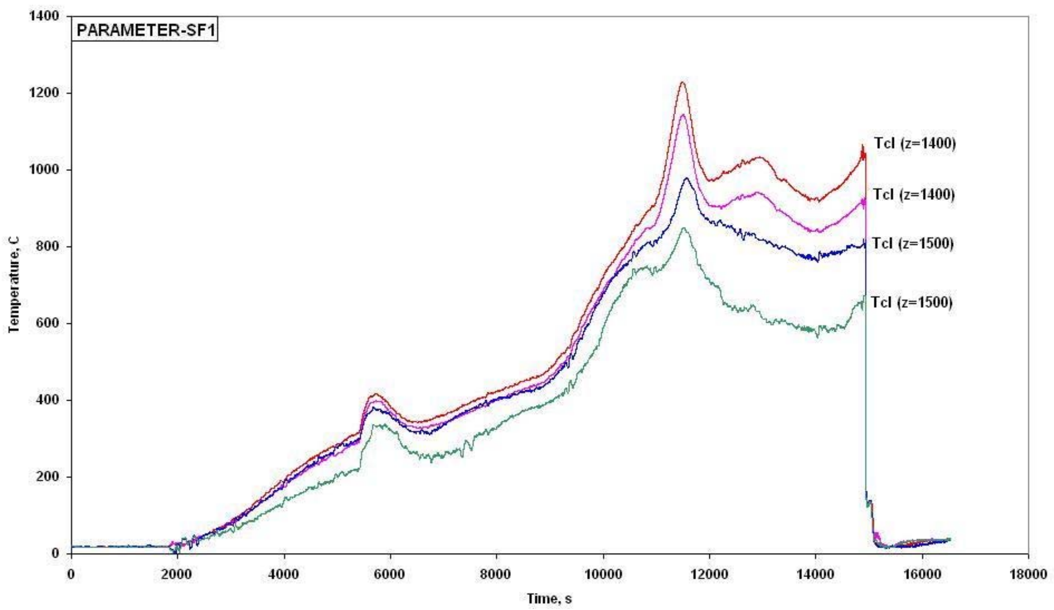
Starting from that moment the electric power was brought by the operator to the manual control only by indications of thermocouples. The assembly was held at temperature of ~1200°C during 3250 s (in the hottest zone at the levels of 1000 – 1300 mm), though the maximum temperature deviations in the hottest section at the level of 1300 mm reached ~330°C.

Post-test analysis of the data allowed to assumed that the higher experimental temperatures in the assembly are caused by less flow rate of steam through the assembly due to its bypassing and condensation on the test section cooled shell.

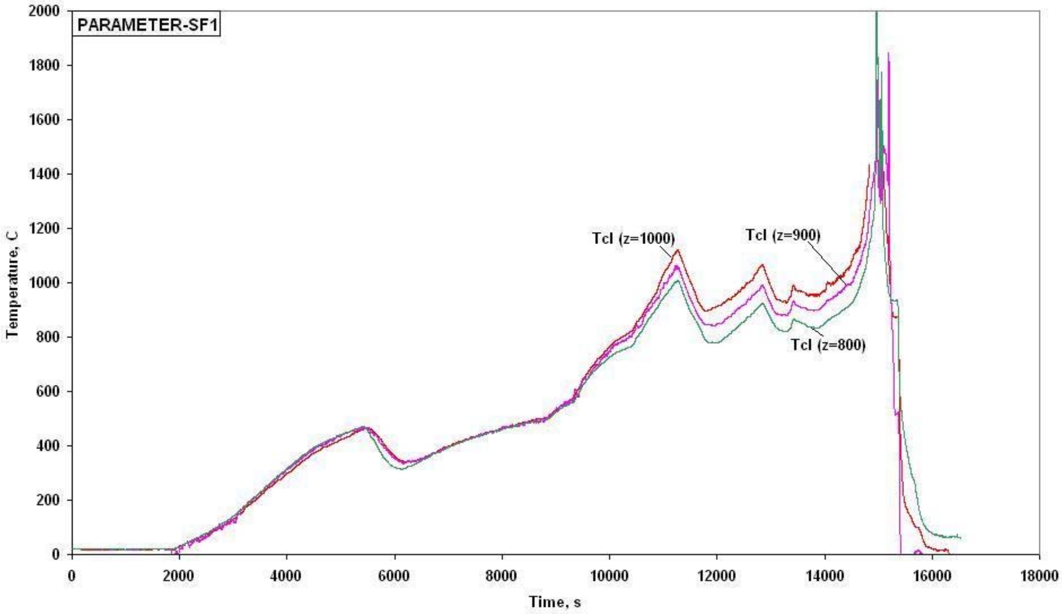
At ~14500 s the increase in electric power was started and the assembly was heated to temperatures of ~ 2000°C and higher (see Fig. 5). At 14945 s the top water supply was started with the flow rate of ~ 41 g/s.



a) indications of thermocouples in the hot zone (1100, 1250 and 1300 mm)



b) indications of thermocouples at the upper (1400, 1500 mm) assembly level



c) indications of thermocouples at the lower (800, 900, 1000 mm) assembly level

Fig. 5. Indications of thermocouples placed on fuel rod claddings at the levels of 800-1500 mm.

In the course of top flooding the assembly upper cooling was reached (at the level from 1250 to 1500 mm, see Fig. 6). At the same time the flooding of the assembly lower part (0 … 600 mm) took place considerably later (in 400-600 s).

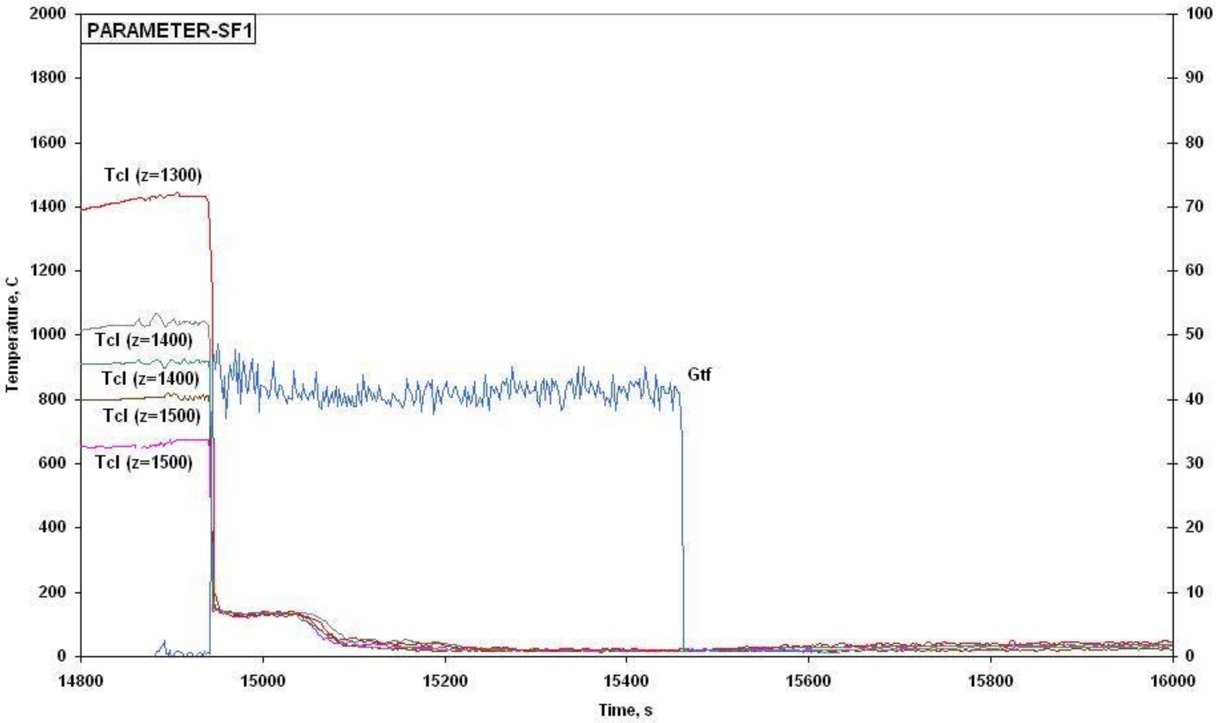


Fig. 6. Indications of thermocouples placed on fuel rod claddings at the level of 1300-1500 mm at the flooding stage.

At these levels the cooling front went upwards slowly from the test section bottom (see Fig. 7). At the level of 800 mm the assembly was cooled actually at the end of the experiment.

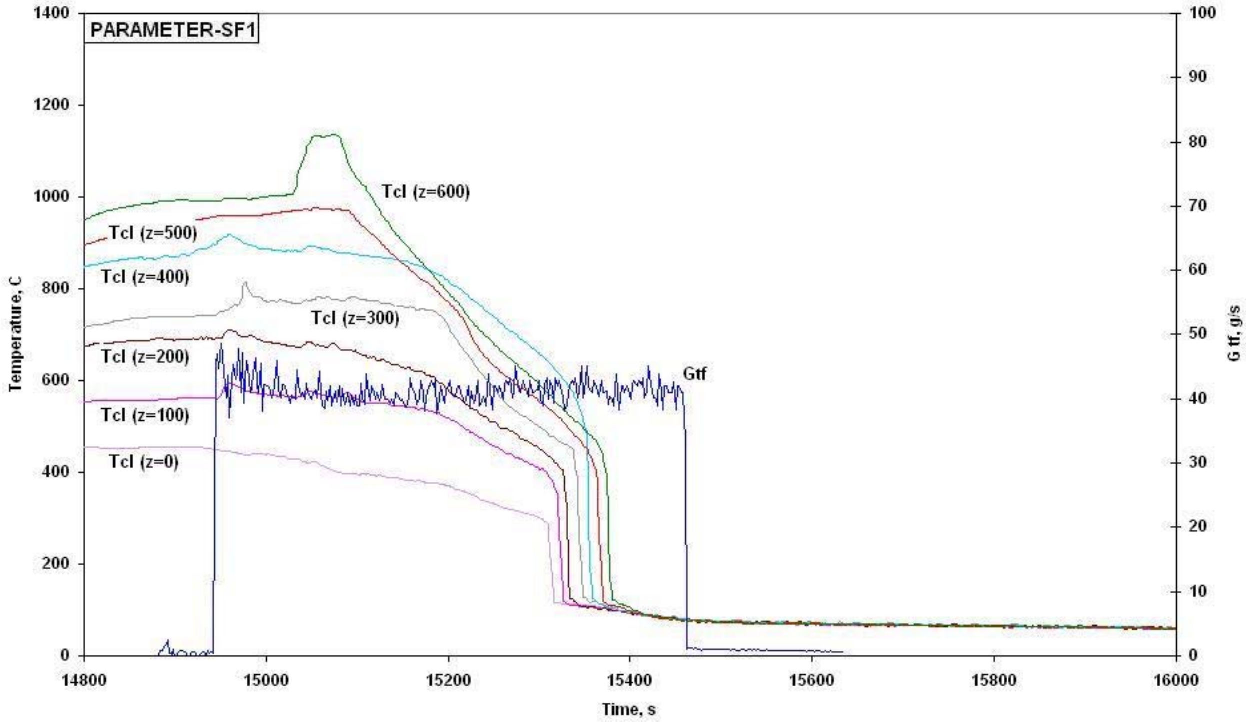


Fig. 7. Indications of thermocouples placed on fuel rod claddings at the level of 0-600 mm at the flooding stage.

In the course of the top flooding a small pressure jump occurred in the assembly (see Fig. 8).

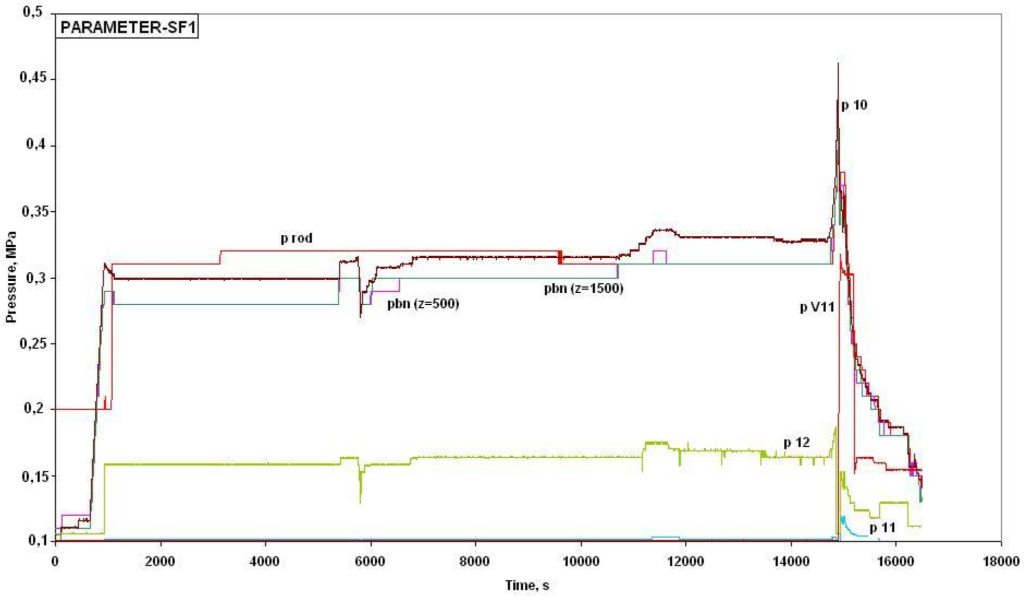


Fig. 8. Pressure in fuel rods (p_{rod}), model FA (p_{bn}) and in the facility gas path (p10, p11, p12, pV11).

Measurements of volumetric hydrogen concentration were carried out by the system SOV-3 at all stages of the experiment. Fig. 9 presents the variation of volumetric hydrogen concentration with regard for the sluggishness of the system SOV-3 (in this plot the time delay in hydrogen measurements is not considered). Fig. 9 presents also the results of test samples. Generally, a conclusion can be made that indications of SOV-3 and the results of test samples analysis are in close agreement.

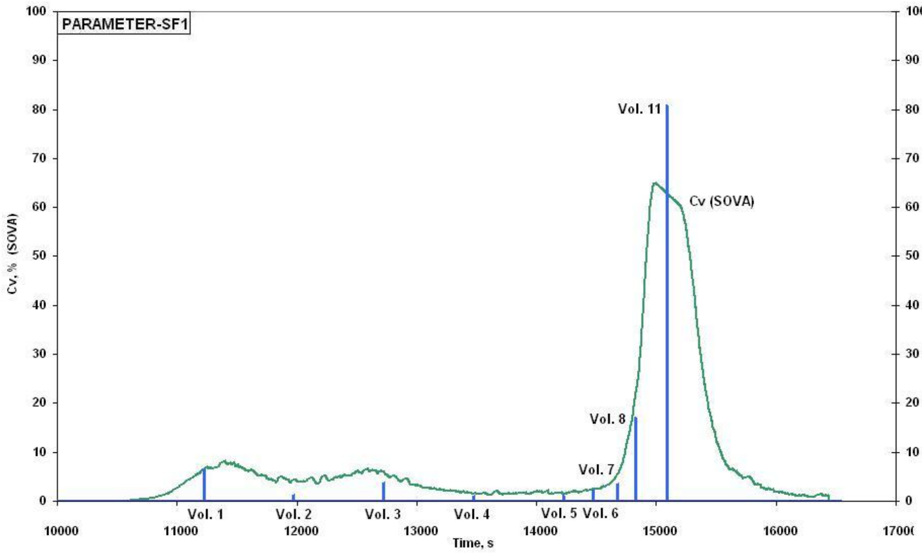


Fig. 9. Variation of volumetric hydrogen concentration versus time.
(Vol.1, …, Vol.8 – test samples (duration 8 s), Vol.11 - 300 s).

Maximum rate of hydrogen generation was ~ 0.19 g/s. Total amount of the generated hydrogen was ~ 91 g, from this amount about 60 % was produced during the flooding stage (see Fig. 10).

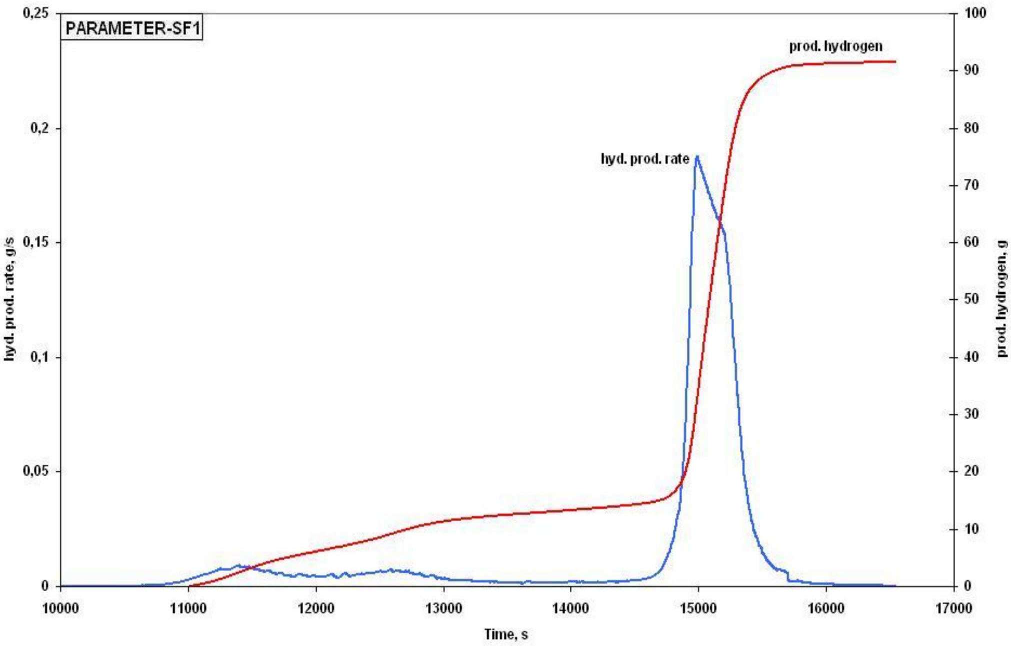


Fig. 10. Rate of hydrogen generation and mass of hydrogen released.

2.2. Post-test material analysis of the SF1 model assembly

On completion of the PARAMETER-SF1 experiment the preservation, cutting and material studies of the tested model assembly were performed. The results are presented in Annex 2.

Preservation of the model FA was made in vertical position with filling the epoxy resin ЭД-20 without cutting of thermal insulation jacket.

The cross cuts were made by cutting machine Delta-Abrasimet. External appearance of the typical cross-section is presented in Fig. 11.

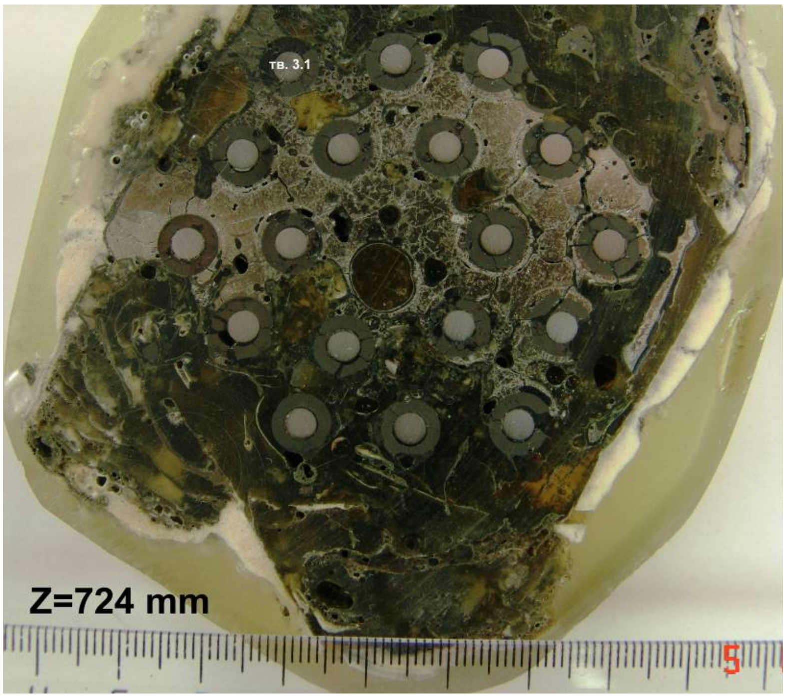


Fig. 11. Assembly cross-section at the level of 724 mm (top view).

Material studies were performed with the aim of:

- description of the stage of the model FA of VVER after the tests and the analysis of its components (degree of oxidization, structure, chemical and phase compositions);

- evaluation of the clear opening blockage distribution, the melt relocation and thinning of fuel pellets over the assembly height.

As to the degree and character of damage the assembly heated part was conventionally divided into 5 zones (Fig. 12).

In the first zone (Z ~ 0…540 mm) there are no considerable damages of the assembly. Thickness of oxide on surface of zirconium components does not exceed 10 μm.

In the second assembly zone (Z ~ 540…700 mm) there are solidified droplets and streams of the melt (U,Zr,O), relocated from the assembly upper part. Beginning from the level of 650 mm and above the zones of fuel-clad interaction are revealed. At the higher levels of this zone there was a partial melting of metal zirconium, fuel dissolution with the formed liquid mixture, dissolution of cladding both from the outside, and from the inside. Thickness of oxide on cladding surfaces in this zone increases 10 to 170 μm with increase in the height coordinate. The shroud is oxidized on both sides. Oxide thickness, with increase in height coordinate, increases within the range of 10…80 μm and 10…200 μm on the inner and outer surfaces, respectively. On the melt surface there is the oxide layer with the thickness not exceeding ~300 μm.

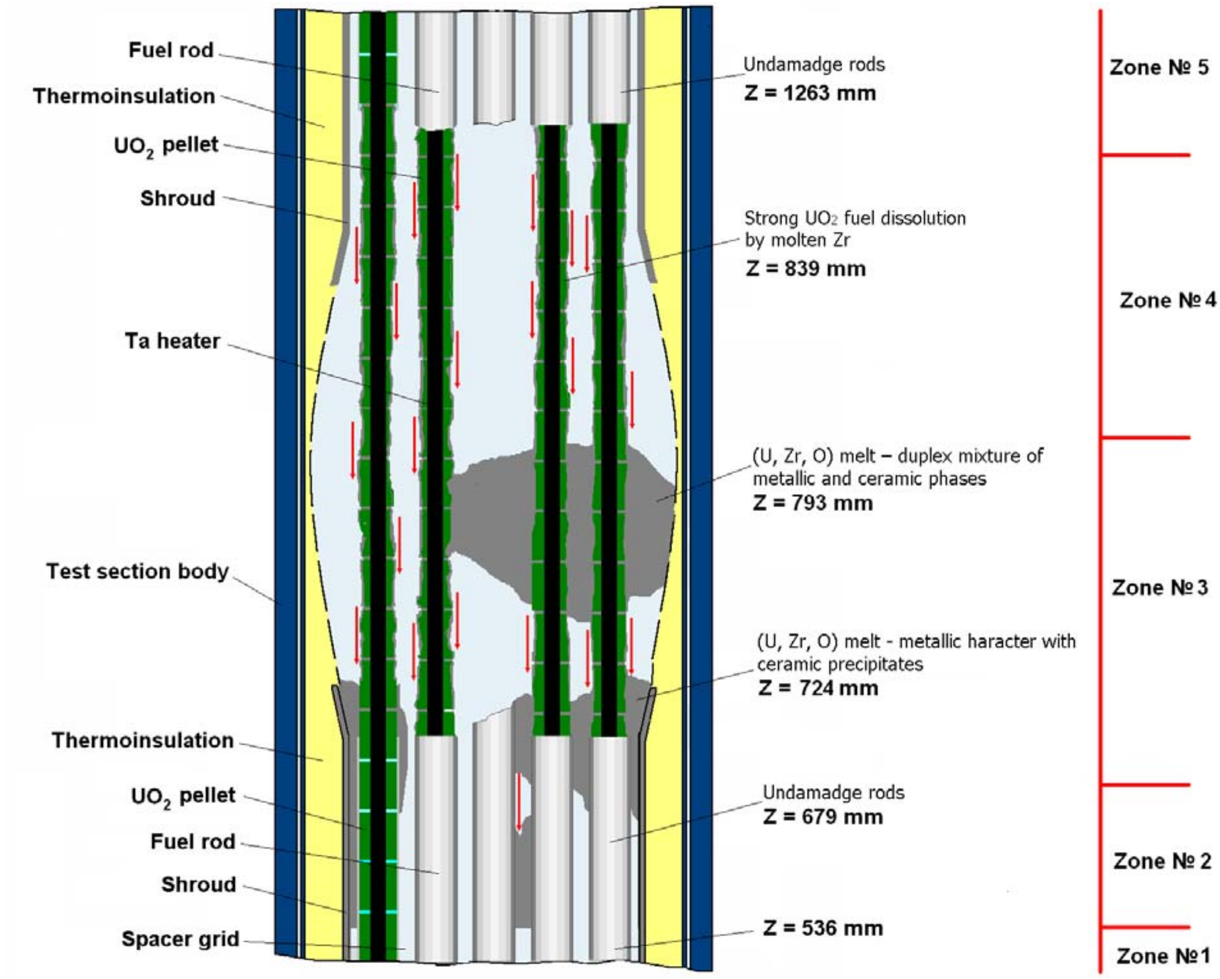


Fig. 12. Structural diagram of the assembly after the tests.

In the third zone (Z ~ 700…800 mm) the melted and rundown masses formed practically complete blockage of the FA flow area. The specific feature of the tight blockage zone is the fact that it comprises two parts over the height (at levels Z = 724 mm and Z = 793 mm), and between them (Z = 745 mm) there is no melt practically in the space between fuel rods and fuel pellets are considerably dissolved.

The fourth zone (Z ~ 800 – 1070 mm) is characterized by damage of zirconium structural components and considerable dissolution of fuel pellets, however no substantial masses of the melt were revealed.

In the fifth zone (Z ~ 1050 – 1250 mm) the remains of oxide parts of fuel rod claddings and metal and ceramic melt were revealed, as well as remains of the shroud. Thickness of oxide of claddings at the level of ~ 1250 mm is ~ 200 μm. The shroud is oxidized on both sides, thickness of oxide on the outer and inner surfaces is ~ 150 μm. No facts confirming melting of metal zirconium at this level were revealed.

As a result of studies of cross-sections the considerable dissolution of fuel pellets was revealed.

2.3. Pre- and post-test calculational analysis of the SF1 experiment

Experiments at the electrically heated test facilities with the flooding of the assemblies overheated to high temperatures require thorough preparation including the preliminary numerical simulation of the planned experiment (see Annex 3). Such numerical analysis is necessary for estimation of a contribution of the physical processes at different stages of the experiment. This allows to justify the scenario of the experiment and to establish the main requirements for the measurement and monitoring systems.

Pre-test scenario of the experiment is given in Table 1.

Table 1

Pre-test scenario of the PARAMETER-SF1 experiment

No.	**Stage**	**Main parameters**			
		FA temperature, ^{o}C	**Environment**	**Heating/ heating rate**	**Time, s**
1	*FA heating with argon flow*	20-100	Argon flow at temperature to 160^{o}C (argon flow rate - 2 g/s)	-	0-500
2	*FA heating in the flow of steam-argon mixture*	100-500	Steam-argon mixture (argon/ steam flow rate 2/3 g/s)	0.1-0.3 ^{o}C/s	500-1600
3	*Stabilization of the main parameters*	~ 500	Steam-argon mixture (argon/ steam flow rate 2/3 g/s)	-	1600-2500
4	FA heating to 1200^{o}C	500-1200	Steam-argon mixture (argon/ steam flow rate 2/3 g/s)	~0.25 ^{o}C/s	2500-5000
5	FA pre-oxidization	~ 1200	Steam-argon mixture (argon/ steam flow rate 2/3 g/s)	-	5000-9000
6	*Quick heating of the assembly*	1200-1850	Steam-argon mixture (argon/ steam flow rate 2/3 g/s)		Determined experimentally when the assembly reached the required temperature
7	Assembly top flooding *(in 10 s after FA reached Tmax=1850^{o}C)*	To complete assembly cooling	Water (flow rate of 40 g/s per assembly		~300 (duration)

The pre-test calculations of the experiment have been performed using the computer codes SOKRAT/B1 (RATEG/SVECHA), PARAM-TG, RELAP/ MOD3.3, MELCOR, ICARE/CATHAR, MAAP4.04d4, ATHLET-CD.

Most of these codes have been developed for the detailed numerical simulation of the physical processes and phenomena on the in-pile phase of severe beyond design basis accidents at NPP with VVER or PWR.

On the whole, the codes predict close values of maximum temperatures of fuel rod claddings at the phases of preheating, pre-oxidization and quick heating. A comparative analysis of the results of numerical simulation, made with different codes, allows to conclude that the proposed mode of electric power supply allows to implement the main stages of the experiment, however, taking into account the available differences in the values of full electric power, the change in electric power should be made smoothly following the indications of thermocouples.

For assessment of reliability and self-consistency of the obtained experimental data the post-test calculations have been performed using the computer codes SOKRAT/B1 (RATEG/SVECHA), PARAM-TG, RELAP5/MOD3.3, ICARE/CATHAR, MAAP4.04d4.

On the whole, all the codes give a satisfactory description of the temperature behaviour of fuel rod claddings in the first 10000 seconds of the running calculation time. The calculated data are in rather good agreement with each other as ell as with the indications of thermocouples. Beginning from 10000 second, when the temperature of fuel rod claddings in the hottest sections of the assembly begin to increase above 730°C, the results of the calculation of cladding temperature behaviour at the levels of 1100-1300 mm become noticeably different. It can be supposed that causes of the difference are the simplified modelling of the condensation and bypassing processes (some users assigned the approximations for the steam flow rate through the assembly instead of direct modelling of these processes). The difference becomes still greater at the end of heating phase (immediately before flooding) when a number of codes (RATEG/SVECHA, ICARE) show that during this time period the processes of melting of metal zirconium take place in the assembly – the calculated temperatures exceed 2000°C. The codes simulating the processes of zirconium melting and melt yield on the surface of fuel rods, show a greater hydrogen release because a considerable contribution into the common mass is made by the melt oxidization.

For assessment of the degree of self-consistency of the results of the SF1 experiment, on the basis of the developed nodalization of the test section at the PARAMETER test facility a number of test calculations were made for assessment of the effect of uncertainties of possible ranges of parameters variation of the plant itself (insulation heat conduction, coolant leakages bypassing the core, cladding temperatures reached in the sections where there are no reliable indications of thermocouples, etc.) on the results of calculations; the through modelling was made with the use of the code SOKRAT/B1 for all phases of the experiment including the top flooding phase.

In spite of the uncertainty of experimental data (coolant flow rate through the assembly under the conditions with damage of fuel rods and the surrounding structures; damage of the shroud and thermal insulation leading to reduction in cooling the melt nucleus, mass failure of thermocouples at high temperatures), typical for such kind of high-temperature experiments, it was managed to reproduce a set of experimental data.

For example, in the course of the experiment the gradual displacement of the assembly hottest zone was observed from the level of ~1250 mm downwards to the level of ~1100 mm. The numerical analysis confirmed that such a displacement is caused by decrease in the steam flow rate through the assembly at the expense of the condensation processes and possible steam leakages in the process channels. It was demonstrated that the evolution of the temperature profile, observed in the experiment, is described with the sufficient accuracy. The appropriate modelling of the spatial-time temperature distribution allowed, in its turn, to give rather accurate description for the degree of fuel rod cladding oxidization: a good agreement was obtained as to thickness of the oxide film in the assembly upper and lower parts, and in the middle part – complete damage of fuel rod claddings.

In the assembly middle part a complete damage of fuel rod claddings and partial dissolution of fuel pellets was observed. It is known that the amount of uranium dioxide dissolved with liquid zirconium depends greatly on the degree of zirconium pre-oxidization, the moment of damage of the protective oxide layer and the intensity of the melt relocation over the fuel rods. In the analysis of results of the experiment SF1 it was shown that the most intensive dissolution of UO_2 takes place in the assembly middle part (of the order of 20%), the code gives rather accurate prediction for the degree of fuel dissolution at the levels of 800÷1100 mm.

So, the qualitative and quantitative agreement of the calculated and experimental data has been obtained on the assembly temperature behaviour, distribution of oxide layers, dissolution of uranium dioxide. These results are indicative of the fact that the data of the experiment SF1 are inter-consistent and present a valuable verification material in the part concerning:

- studying the thermo-mechanical behaviour and cooling processes of 19-rod model FA of VVER-1000 under simulated conditions of severe accident including the stage of low rate cooling with top flooding;

- studying the thermo-mechanical behaviour of the assembly structural components (fuel pellets and claddings, shroud, spacing grids) under the conditions of top flooding of the model assembly overheated to high temperatures;
- studying the degree of oxidization of the assembly structural components;
- studying the interaction and structural-phase transformations in the model assembly materials (fuel pellets and claddings);
- studying the time history of hydrogen release including the stage of low rate cooling with top flooding.

3. PARAMETER-SF2 EXPERIMENT

3.1. Results of pre-test calculations of the PARAMETER-SF2 experiment

Results of pre-test calculations of the PARAMETER-SF2 experiment and their analysis are presented in Annex 3.

The scenario of the PARAMETER-SF2 experiment (see Table 2) was planned in such a way that by the results of the experiment we could obtain the valuable verification material for the SFD codes. With this aim the analysis was performed preliminary for the thermo-hydraulic and physical-chemical models applied in the codes SOKRAT and ICARE. By the results of the analysis the main tasks were formulated – to check the adequacy of modelling the flooding and oxidization processes by the codes and to obtain the missing data for improving these models, and namely:

- to determine the heat transfer coefficients in POST-CHF mode by the time history of the fuel rod simulators temperature variation;
- to evaluate the post-deformation of the components and possible limitation of surface wetting at the expense of change in clear opening;
- to evaluate the engineering coefficients of redundancy, required in designing the VVER equipment, a portion of water, ejected from the test section at the stage of flooding (CCFL), shall be measured;
- to specify the parameters of POST-CHF heat transfer and “flood” and to justify their applicability to calculation of the processes in the conditions similar to those observed in the SF2 experiment.

Table 2

Pre-test scenario of PARAMETER-SF2 experiment

No.	Stage	Main parameters			
		FA temperature, °C	**Environment**	**Heating rate, °C/s**	**Time, s**
1	***Heating of the assembly within argon***	20-100	Argon flow at temperature to 400°C (argon flow rate is 2 g/s)	–	0-1000
2	***Heating of the assembly within the steam and argon flow***	100-500	Steam-argon mixture (argon/steam flow rate is 2/3.3 g/s)	_	1000-4000

3	***Heating of the assembly up to 1200℃***	500-1200	Steam-argon mixture (argon/steam flow rate is 2/3.3 g/s)	~0.3	4000-6000
4	***Pre-oxidation of the assembly***	~ 1200	Steam-argon mixture (argon/steam flow rate is 2/3.3 g/s)	–	6000-10000
5	***Heating of the assembly up to 1500℃***	1200-1500	Steam-argon mixture (argon/steam flow rate is 2/3.3 g/s)	~0.3	10000-11000
6	***Top flooding of the assembly (when the assembly reaches the temperature T_{max}=1500^{o}C)***	Till complete cooling of the assembly	Water (flow rate of 40 g/s per assembly)	–	Duration of flooding ~ 150 s
7	***Bottom flooding of the assembly (in 50 s after the beginning of the top flooding or when the assembly temperature decreases less ≈630^{o}C at the level of 1250 mm)***	Till complete cooling of the assembly	Water (flow rate of 100 g/s per assembly)	–	Duration of flooding ~ 100 s

In the course of the experiment SF1 the fuel rod simulator claddings were considerably damaged (up to 40%). A large number of the inter-related processes, including zirconium melting, resulted in the assembly damage, make the scope of possible verification material narrow (for instance, on verifying the oxidization models). Therefore a relatively low maximum temperature of fuel rod claddings before flooding was chosen (to exclude the process of zirconium melting), corresponding to the value of 1500°C. Nevertheless, this temperature provides for rather long-term staying of claddings under POST-CHF mode and allows to evaluate the time history of the fuel rod simulators heating/cooling with fronts moving, with regard for the frequency of data inquiry on thermocouples temperature. The rate of the top and bottom flooding corresponds to the prototype (as to water mass per one fuel rod) under the conditions of severe LOCA at VVER RP.

The multi-variant calculations were performed to specify the experiment scenario with the assigned parameters (steam flow rate – 3.3 g/s, argon flow rate – 2 g/s, temperature of fuel rods at the pre-oxidization phase – 1200°C, thickness of oxide layer on fuel rod claddings on completion of the pre-oxidization phase –150-200 μm, rate of temperature rise – 0.3 °C/s, maximum temperature of fuel rod claddings at the moment of flooding – 1500°C, water flow rate of top flooding – 40 g/s, of bottom flooding – 100 g/s).

By the results of pre-test calculations using codes SOKRAT, ICARE and PARAM-TG, carried out with regard for new data on the plant design and the data on results of SF1 experiment, the required level of power was determined, supplied to the model assembly depending on the required level of temperatures in the assembly. With this, a rather good agreement was obtained in calculations of power released in the assembly at the expense of electric heating and calculated with the use of various computer codes.

To choose the scheme of performing the combined flooding (of sequence and time delays in switching the top and bottom flooding) the multi-variant calculations were performed. The numerical analysis shows that when the bottom flooding water reaches the level of -300 mm, where the temperature of fuel rod claddings and shroud is high enough and amounts to ~480°C, the intensive water boiling is started, and the claddings at the level of 1250 mm, having been already cooled with the top flooding water to the temperature of ~130°C, begin to be heated-up again. Evidently, this phenomenon is caused by displacing the top flooding water with the steam, generated as a result of the bottom flooding, i.e. the “flood” phenomenon occurs.

Taking into account the fact that one of the main tasks of the experiment is verification of CCFL models, and that the calculated data on the assembly cooling are preliminary, it is reasonable to develop such a scheme of the combined flooding that would allow to obtain assuredly the qualitative experimental data not permitting the double interpretation. It is expedient to start the bottom flooding after the top flooding water has passed into the assembly to the level of 1250 mm (to the level of the upper spacing grid) and cooled the fuel rod claddings to the temperature of 630°C or below (close to the threshold of wetting).

Pre-test calculations with the use of the code SOKRAT show that the scheme of the combined flooding with a delay in the beginning of bottom flooding of ~50 s meets the tasks of the experiment. Numerical analysis predicts that this time is sufficient for cooling the fuel rod claddings in the hottest section of 1250 mm with the top flooding water to the temperature of 630°C. The average calculated rate of steam generation is ~14 g/s throughout the flooding phase, the total mass of the generated hydrogen is 23 s, and 3 g of them – during the flooding phase.

3.2. Results of the PARAMETER-SF2 experiment

The PARAMETER-SF2 experiment was conducted on April 3, 2007, the full information on the results of the experiment is presented in the Protocol of PARAMETER-SF2 experiment results (see Annex 4).

The conditions of the experiment simulated the severe stage of LOCA type accident when the core, overheated to 1500°C, was cooled with the combined top and bottom flooding in case of restoration of the emergency flooding system.

For performing the PARAMETER-SF2 experiment the process systems of the PARAMETER test facility were prepared according to the functional diagram (Fig. 13).

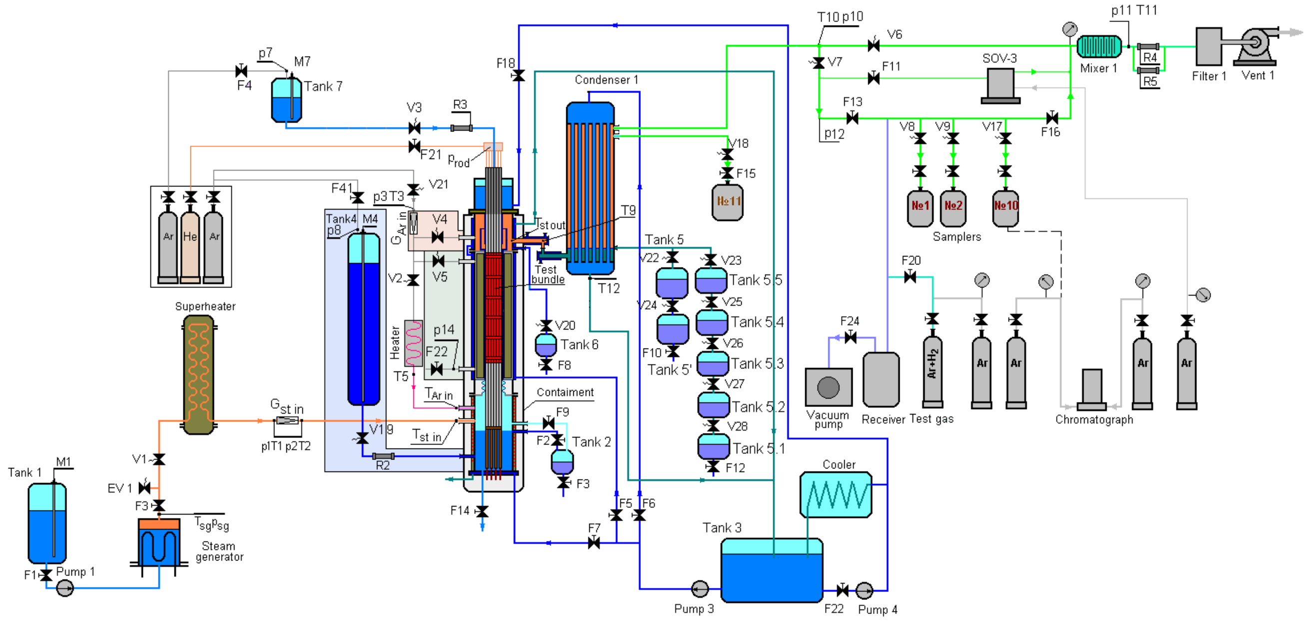


Fig. 13. Diagram of the PARAMETER test facility (SF2 experiment).

The PARAMETER-SF2 experiment included four stages:

- *preparatory stage* – stabilization of the assigned flow rates of argon and steam, check of the state of the assembly and process systems at FA temperature $T_{FA} \approx 700^oC$, heating-up the assembly to temperature of ≈ 1200°C in the hottest zone;

- *pre-oxidation* – FA holding at temperature of ~ 1200°C in the hottest zone;

- *transient* – rise of FA temperature in the hottest section to ~ 1500°C;

- *flooding:*

  - top flooding – top water flooding of the assembly with flow rate of $G_{tf} \approx 40$ g/s;

  - bottom flooding – bottom water flooding of the assembly with the flow rate of $G_{bf} \approx 130$ g/s.

At the preparatory stage the assigned parameters were assured for the steam and argon at the test section inlet: steam flow rate ($G_{st\ in}$) – ~ 3.5±0,1 g/s at temperature ($T_{st\ in}$) ~ 500±25°C, argon flow rate ($G_{Ar\ in}$) – ~ 2.0±0,1 g/s at temperature ($T_{Ar\ in}$) ~ 400±20°C (see Fig. 14).

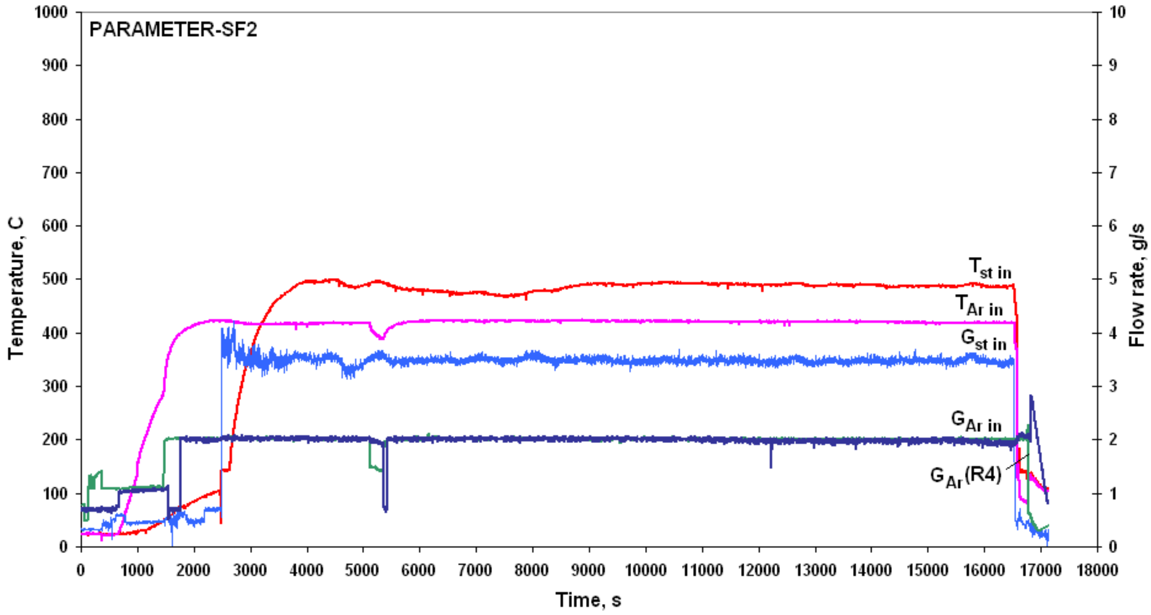


Fig. 14. Parameters of steam ($G_{st\ in}$, $T_{st\ in}$) and argon ($G_{Ar\ in}$, $T_{ar\ in}$) at the test section inlet and argon flow rate (G_{Ar}(R4)) at the outlet to special ventilation.

At the pre-oxidation stage, on stabilization of temperature of fuel rod claddings at the level of ~ 700°C, the rise in the assembly electric power (P) was made according to the test specification to ~ 8.2 kW (see Fig. 15), corresponding to the increase in the assembly temperature (by indications of thermocouples) to ~1200-1250°C in the hottest zone of the assembly at the level of 1250 mm (see Fig. 16).

At the stage of the assembly transient, after holding at the stage of pre-oxidation during ~ 3400 s, the assembly electric power was increased to ~ 10.5 kW providing for the rise in the assembly temperature (by indications of thermocouples) to ~ 1500°C at a rate of ~ 0.2°C/s (see Fig. 16).

At the stage of flooding, on reaching the assigned temperature of fuel rod claddings of ~1500°C in the hottest zone, the assembly power (P) and steam supply were switched off and the system of the top flooding was switched on with the average water flow rate of

R3 – Gtf(R3) ~ 41 g/s by indications of flow meter. On decreasing the temperature of fuel rod claddings below ~ 700°C at the level of Z = 1250 mm (in ~ 26 s) the bottom flooding system was switched on with the average water flow rate of R2 – Gbf(R2) ~ 130 g/s by indications of flow meter (see Fig. 17).

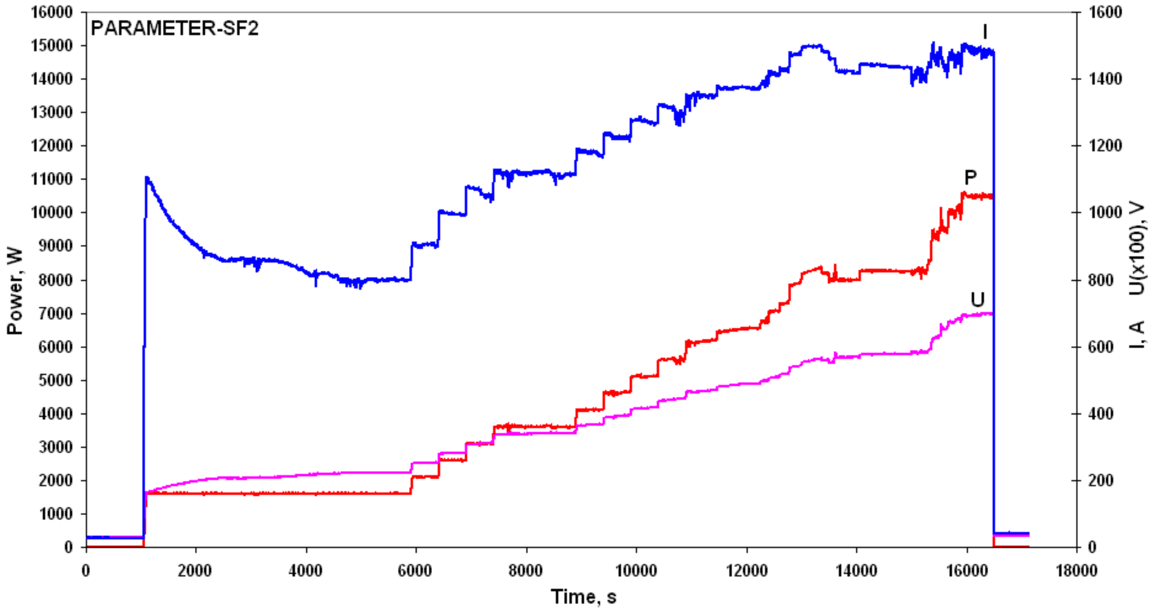


Fig. 15. FA electric parameters.

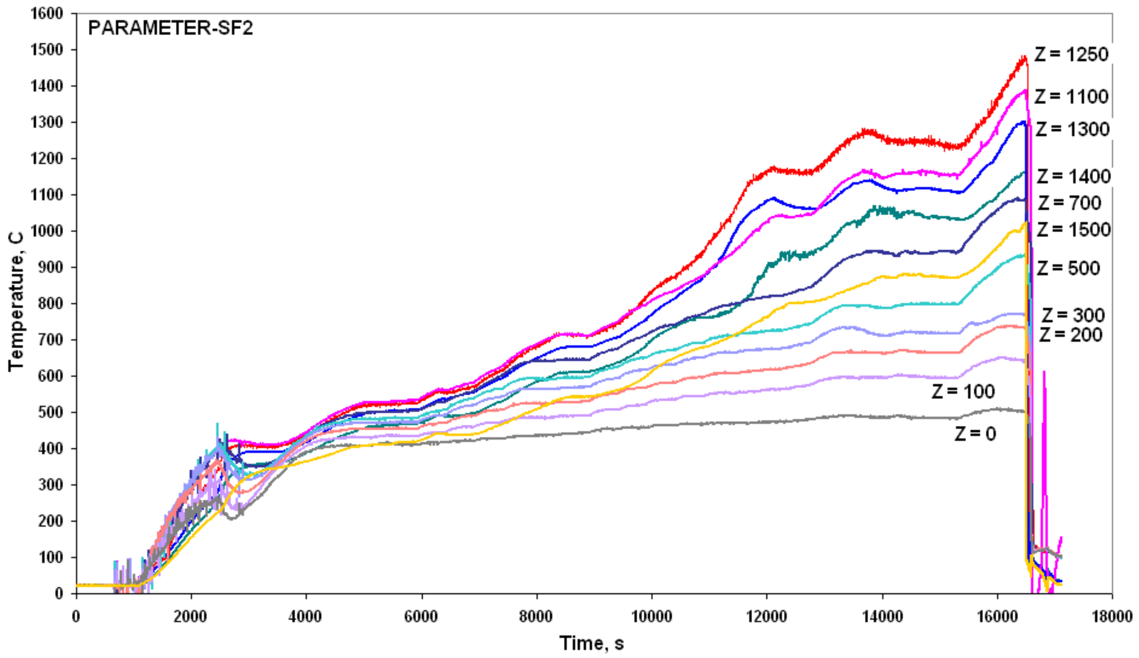


Fig. 16. Indications of thermocouples located over the height of fuel rod claddings.

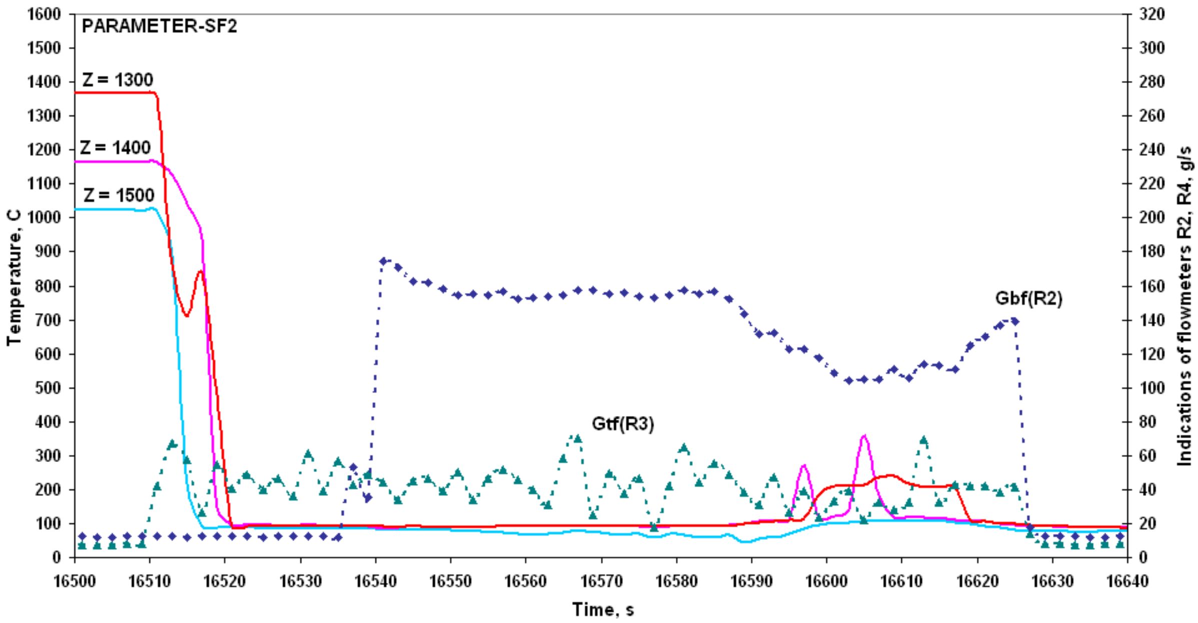
Water supply into the assembly under top flooding resulted in the following:

- quenching (during ~ 10 s after the beginning of flooding) of claddings at the levels of Z = 1500 – 1300 mm to temperature of ~100°C (see Fig. 17a);

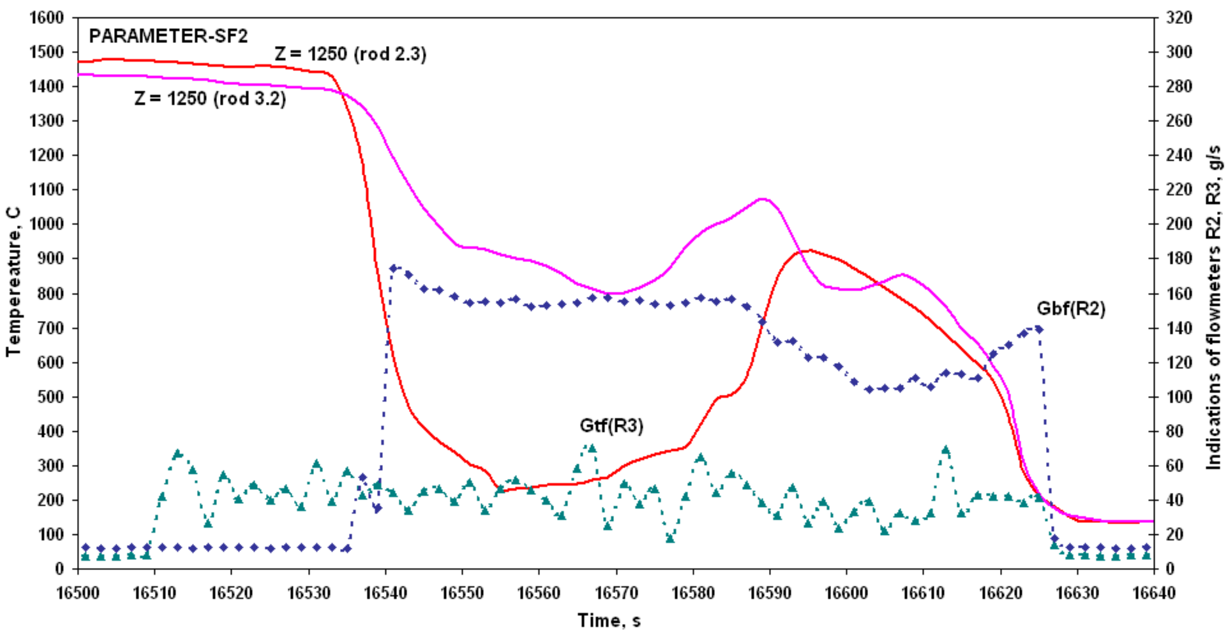
- cooling the fuel rod claddings at the level of Z = 1250 mm to temperature of ~ 200 – 700°C during ~ 40 s (see Fig. 17b).

The effect of top flooding on temperature of claddings of the fuel rods of the second and third rows at the level of Z = 1100 mm was not recorded by thermocouples (see Fig. 17c).

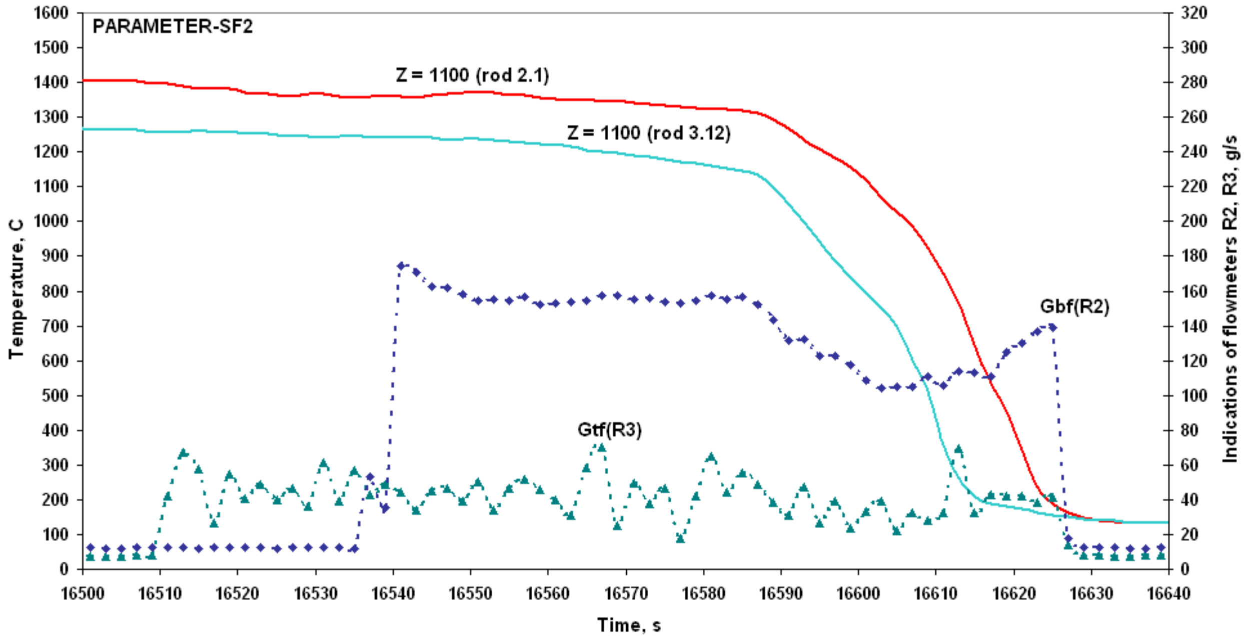
Water supply into the assembly under bottom flooding resulted in successive cooling of claddings at the levels of Z = 0 – 1100 mm. The front of cooling to temperature of ~150°C moved from the level of Z = 0 mm to the level of Z = 700 mm at a rate of ~ 90 mm/s, and from the level of Z = 700 mm to the level of Z = 900 mm – at a rate of ~ 5 mm/s (see Fig. 17d).



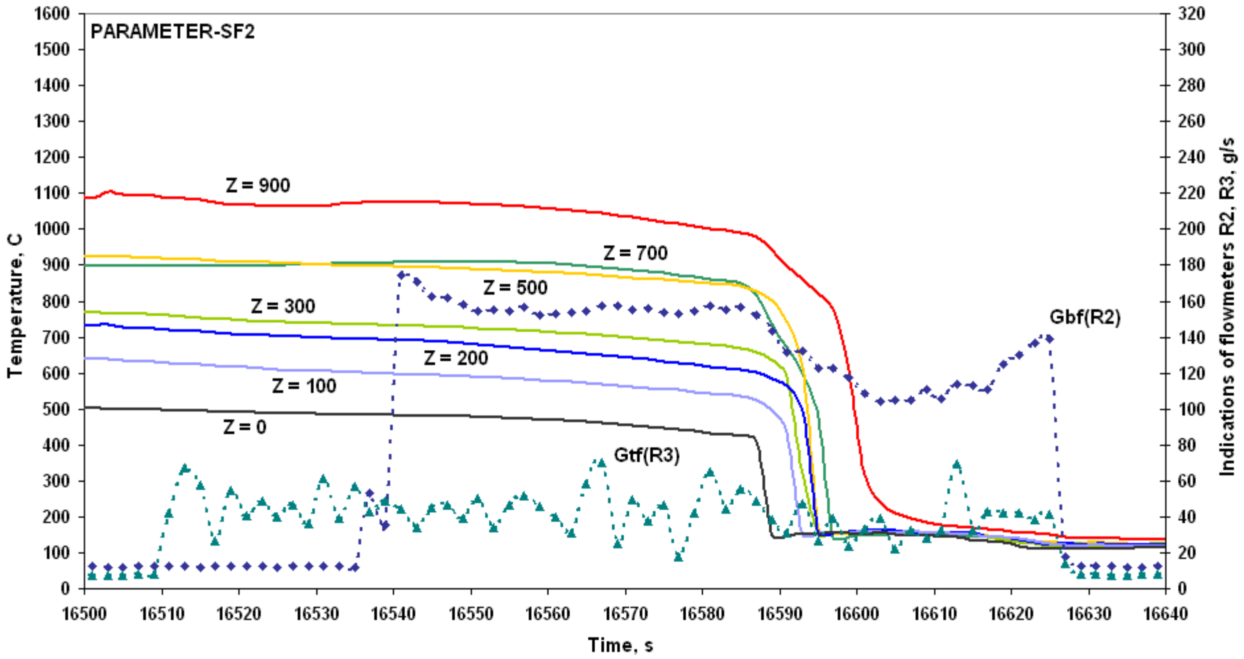
a) Z = 1500 – 1300 mm



b) Z = 1250 mm



c) Z = 1100 mm



d) Z = 0 – 900 mm

Fig. 17. Temperature of fuel rod claddings measured with the thermocouples fixed on the surface over the FA height during the flooding stage.

In the course of the experiment the pressure jumps were registered in the assembly (P_{bnl}), caused by evaporation process (see Fig. 18):

- to ~ 0.4 MPa during ~ 20 s in ~ 2 s after starting the top flooding;

- to ~ 0.6 MPa during ~ 50 s in ~ 50 s after starting the bottom flooding.

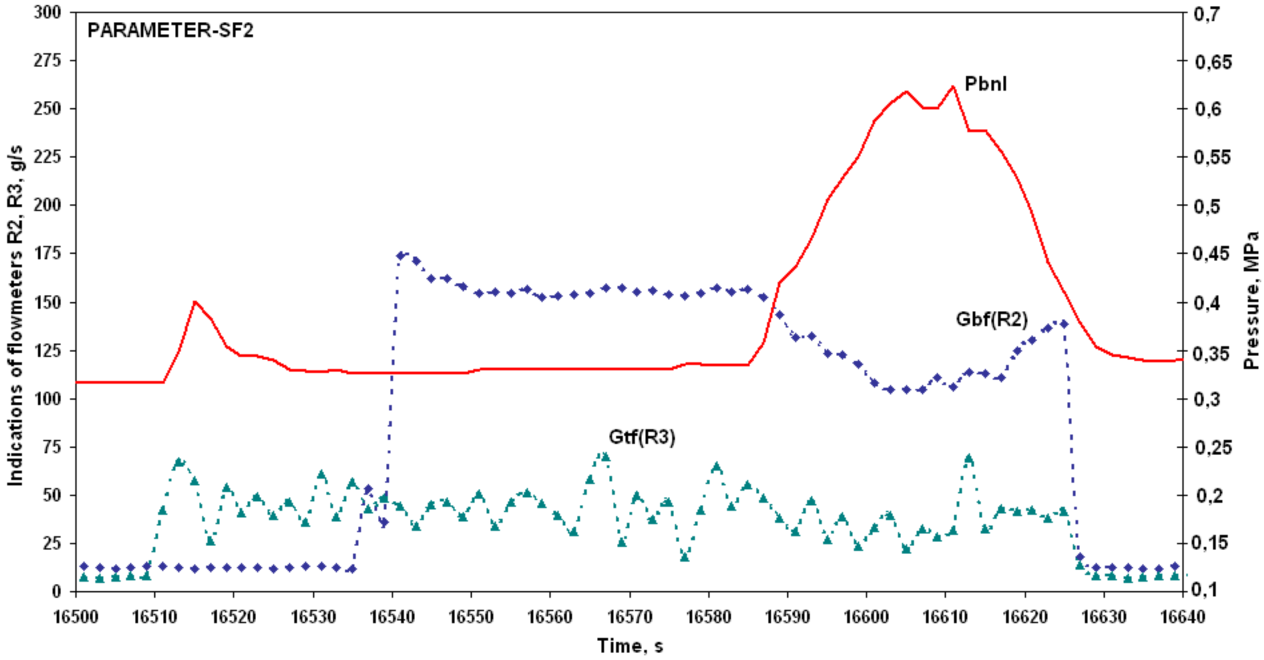
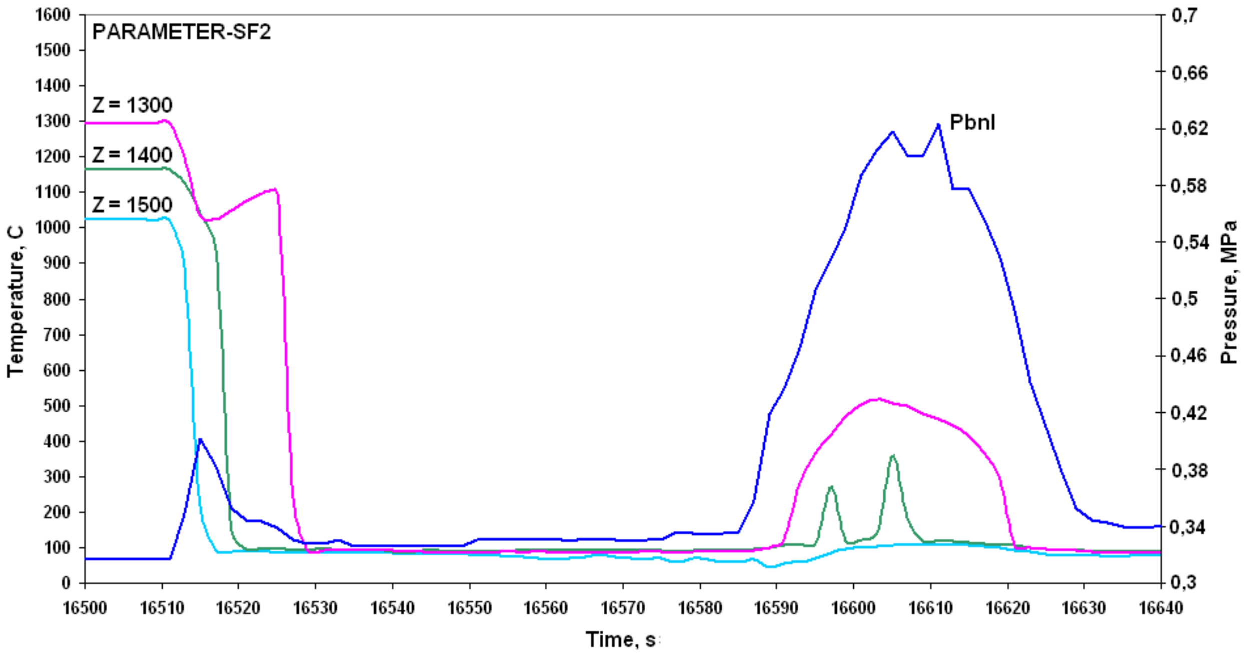


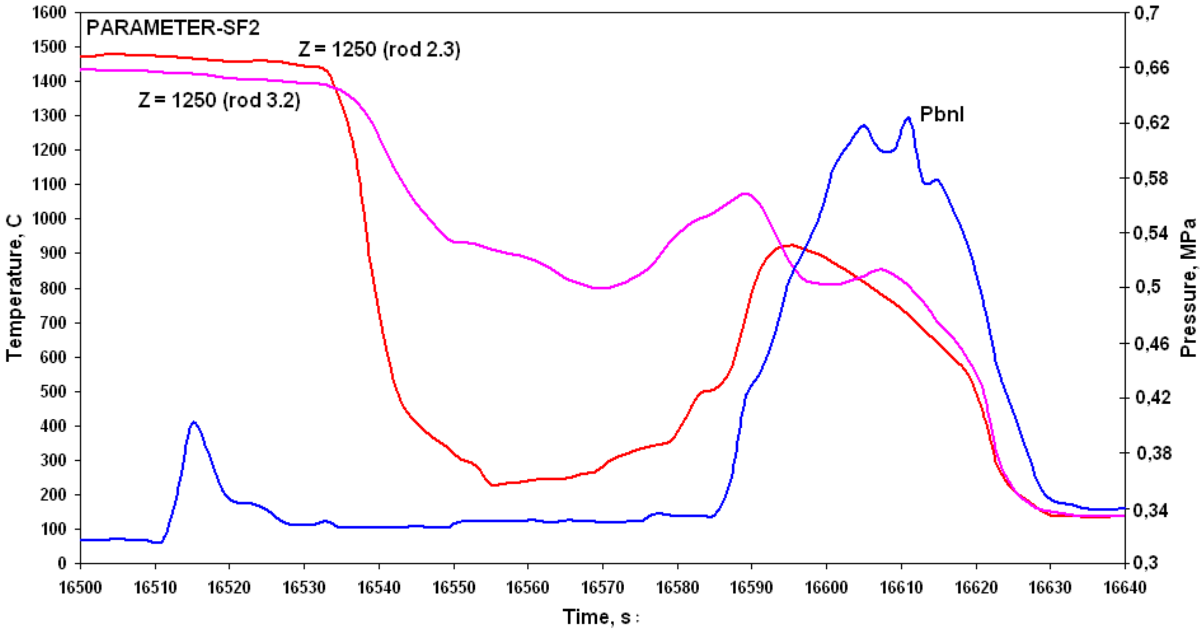
Fig. 18. Pressure changes in the assembly (P_{bnl}) at the flooding stage.

Bottom flooding resulted in intensive evaporation in the assembly lower part and in displacing the top flooding water from the assembly that, in its turn, resulted in repeated rise of temperature in the assembly upper part:

- to ~500°C at the level of Z = 1300 mm (see Fig. 19a);
- to ~1000°C at the level of Z = 1250 mm (see Fig. 19b).



a) Z = 1500 - 1300 mm



b) Z = 1250 mm

Fig. 19. Indications of thermocouples fixed on fuel rod cladding surfaces in the assembly upper part under flooding stage.

In the course of the experiment the two systems of hydrogen registration (see Fig. 20) were used: continuous – SOV-3 and discrete – sampling system (8 test samples), that recorded hydrogen generation at the phase of pre-oxidization (6 test samples) and temperature rise (2 test samples).

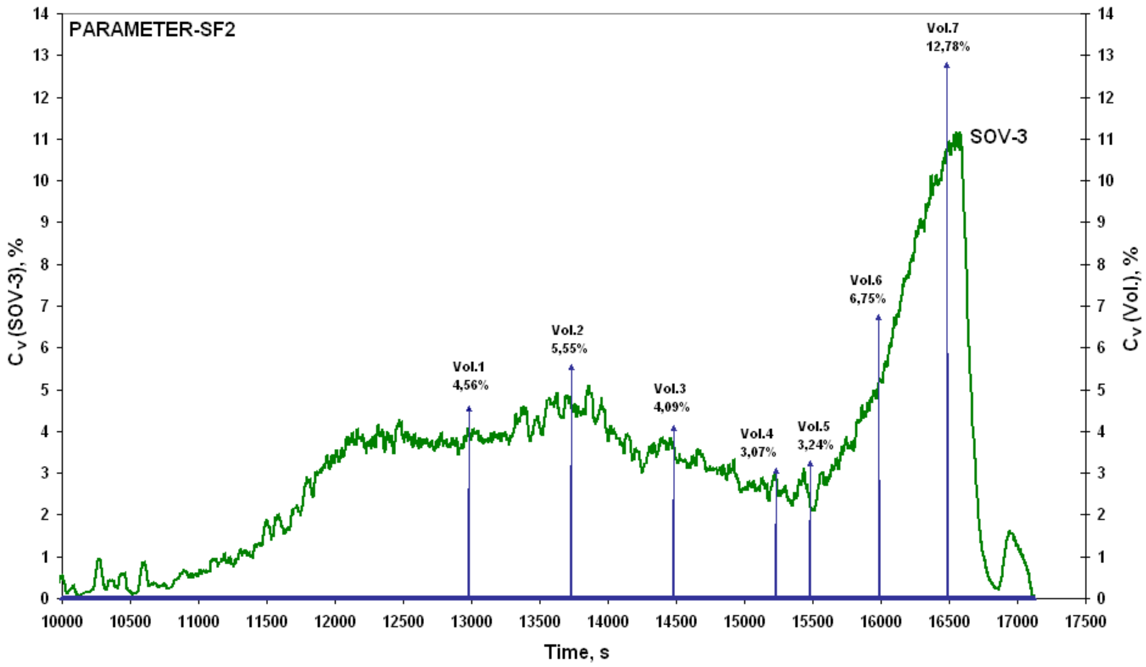


Fig. 20. Variation of volumetric hydrogen concentration versus time (Vol1, …, Vol7 – gas sampling (duration 8 s)).

On the basis of the analysis of results of hydrogen measurement by SOV-3 system it was found out that at the preoxidizing stage ~ 15 g of hydrogen was generated, and at the stage of temperature rise - ~ 9 g of hydrogen. According to the results of the analysis of the discrete system samples at these stages ~ 17 g and ~ 11 g of hydrogen was released, respectively. The total amount of the generated hydrogen did not exceed ~ 28 g (see Fig. 21).

Maximum rate of hydrogen generation was ~ 0.012 and ~ 0.015 g/s, respectively, (Fig. 21).

The total mass of the condensed steam-water mixture, collected in the course of the flooding stage, was ~ 6506 g.

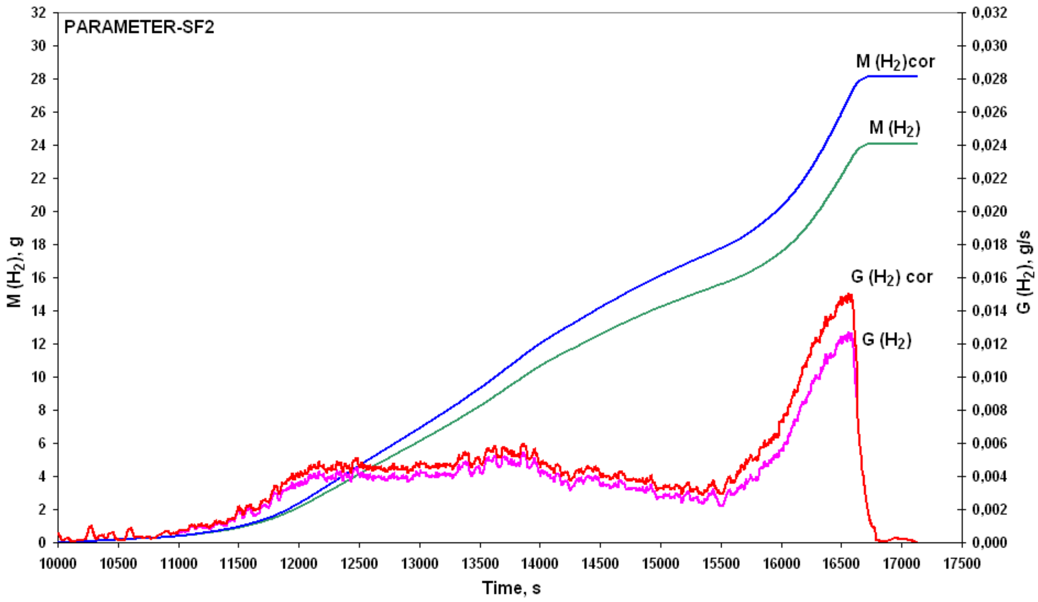


Fig. 21. Regeneration rate and mass of evolved hydrogen.

3.3. Results of cutting the SF2 model assembly

After the tests the dismantling, preservation and cutting of the model assembly were performed.

Analysis of the shroud state showed (see Fig. 22) that the shroud kept its integrity and leak tightness, there is a thin oxide layer on its outer surface, and its diameter at the level of 900 – 1300 mm increased by ~ 2 mm.

Visual examination of the state of the assembly structural components at the levels of 1100 – 1300 mm showed (Fig. 23 – 25) that the fuel rod claddings were oxidized considerably and partially fragmented, there is no melt in the space between fuel rods.

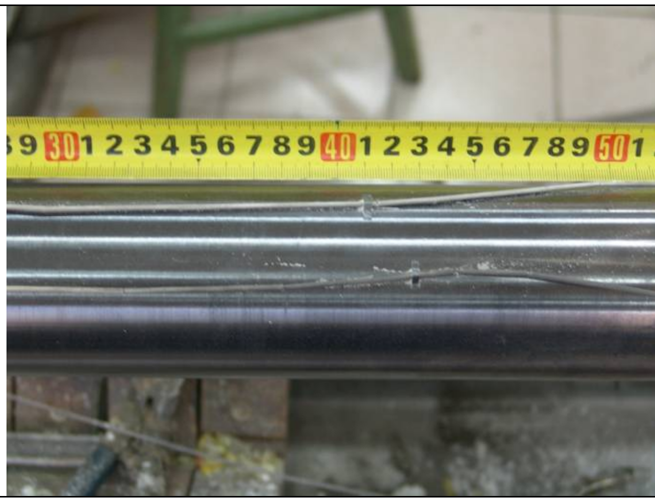
	
1300 – 1100 mm;	1200 – 1000 mm;
	
1000 – 900 mm;	800 – 700 mm;
	
600 – 500 mm;	500 – 300 mm;

Fig. 22. External appearance of the shroud after the tests.

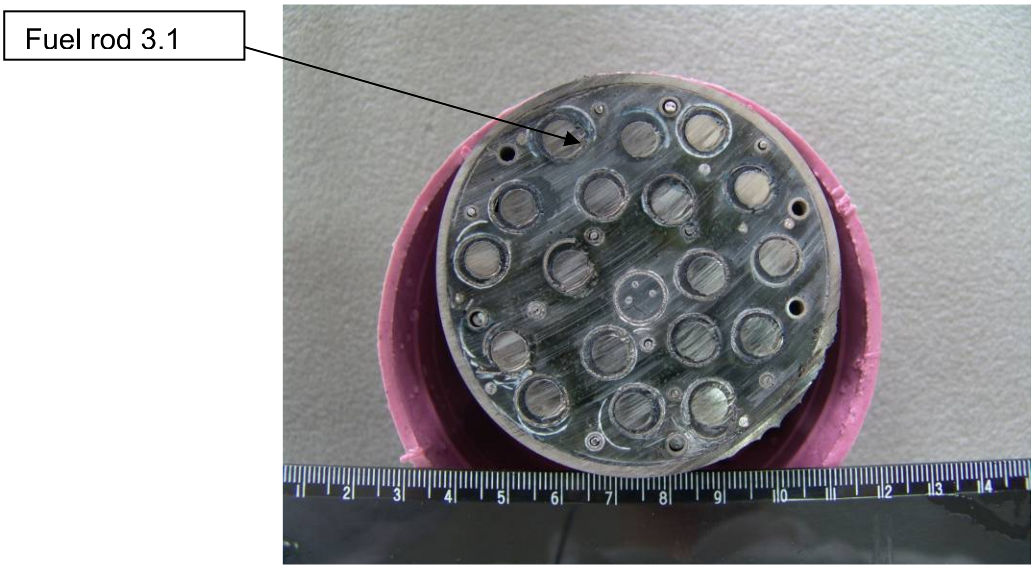


Fig. 23 The assembly cross-section at the level of ~ 1300 mm.

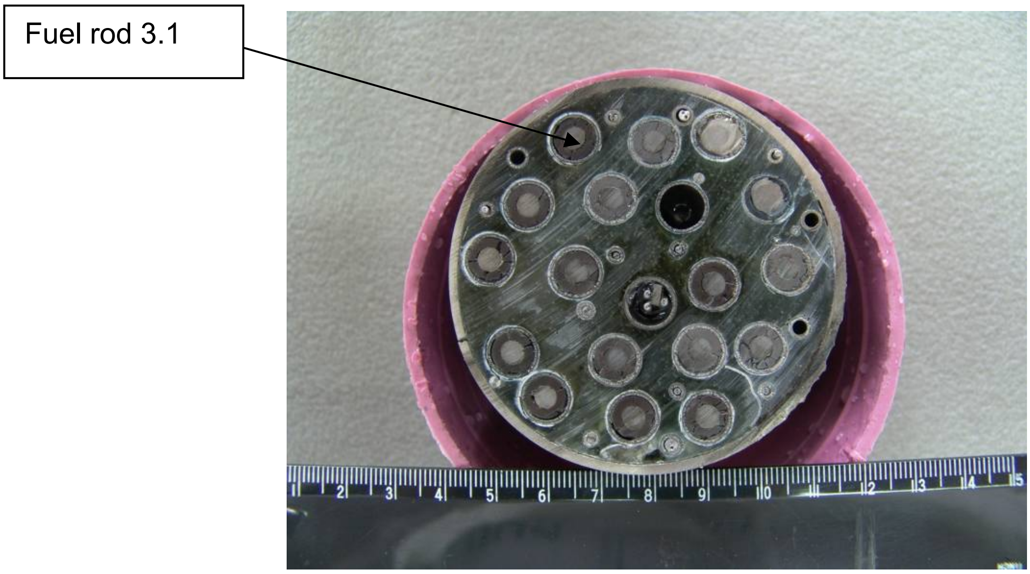


Fig. 24. The assembly cross-section at the level of ~ 1250 mm.

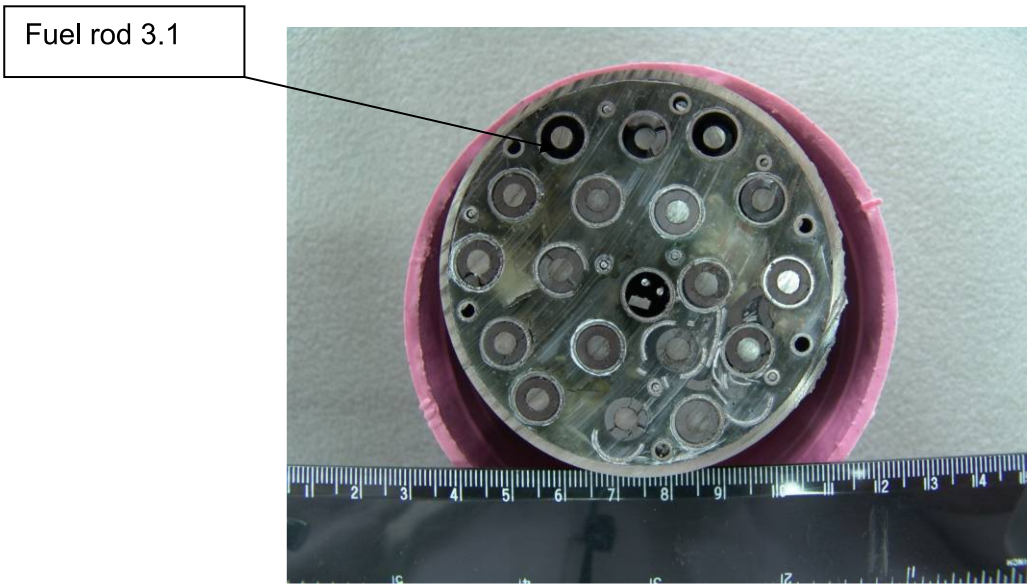


Fig. 25. The assembly cross-section at the level of ~ 1100 mm.

4. COMPARATIVE ANALYSIS OF THE OBTAINED EXPERIMENTAL DATA AND THE RESULTS OF QUENCH-06 TESTS

The experiments of PARAMETER-SF series, discussed in the given report, and the experiment QUENCH-06 [1], performed earlier in Research Centre of Karlsruhe, Germany, are aimed at studying the behaviour of the overheated RP assemblies under flooding conditions.

The objects of studying in the experiments were the assemblies of fuel rods for different reactor installation types. As, for instance, in the QUENCH-06 experiment the bundle simulated the PWR assembly, in the PARAMETER-SF experiments – the VVER-1000 fuel assembly. Fuel rod claddings are made of standard alloys (in QUENCH-06 – of Zircaloy-4, in SF - of Zr1%Nb), diameters of claddings and pellets are also different and correspond to those used in PWR and VVER. The assembly was heated-up by the electric heaters located inside the pellets. In QUENCH-06 the ZrO_2 simulators are used instead of UO_2 fuel pellets. The main design parameters of assemblies are given in Table 3.

The experiments CORA, performed earlier, showed that the behaviour of the overheated assemblies of PWR and VVER does not differ in principal. However the conditions of these experiments reproduced the severe accident conditions only partially not covering the conditions of reflooding.

Table 3

Main design parameters of model FAs

Parameters of model assemblies	**Experiment**		
	Quench-06	**PARAMETER-SF1**	**PARAMETER-SF2**
Model FA			
Type of model assembly	PWR	VVER-1000	VVER-1000
Number of fuel rods heated/un-heated	21/20/1	19/18/1	19/18/1
FA spacer grid, mm	14,3	12,75	12,75
Fuel rod cladding diameter outside/inside, mm	10.75/9.3	9.13/7.73	9.13/7.73
Cladding material	Zircaloy-4	Zr-1%Nb	Zr-1%Nb
Height of heated fuel rods, mm	2480	3120	3120
Height of non-heated fuel rod, mm	2842	2950	2950
Heater material	Tungsten	Tantalum	Tantalum
Heater sizes: diameter/height, mm:	6/1024	4/1275	4/1275
Coordinates, mm	from 0 to 1024	from 0 to 1275	from 0 to 1275
Fuel pellets			
Heated fuel rods: material, outside diameter/central hole, mm	ZrO_2, 9.15/6.15	UO_2, 7.6/4.2	UO_2, 7.6/4.2
Un-heated fuel rods: material, outside diameter/central hole, mm	ZrO_2, 9.15/2.5	-	-

Spacer grids			
Material	Zircaloy-4/ Inconel 718	Zr-1%Nb	Zr-1%Nb
Height, mm	42/38	20	20
Coordinates, mm	50, 550, 1050, 1410/-200	30, 285, 540, 795, 1050, 1305	30, 285, 540, 795, 1050, 1305
FA shroud			
Shape	Cylinder	Hexahedron	Cylinder
Material	Zircaloy-4	Zr-1%Nb	Zr-1%Nb
Outside diameter/wall thickness/height, mm	84.76/2.38/1600	62.5/1.5/1650	70/2/1490
FA shroud thermal insulation			
Material	porous ZrO_2	ZrO_2 ZYFB-3	ZrO_2 ZYFB-3
Thickness/height, mm	36.7/1324	19.8/1500	23/1490

The SF test series experiments and QUENCH-06 allow studying the processes occurring in the overheated assemblies at the flooding stage. The scenarios of these experiments are rather similar; the phase of the assembly preheating was followed by the pre-oxidization phase to grow the protective film of ZrO_2, then the assembly was fast heating-up to high temperatures, and at the end of the experiment the assembly was flooded with water. All phases of the experiments, except for the flooding phase, were performed in steam-gas environment.

The main typical parameters of the experiments were the steam flow rate, maximum temperatures of the assembly just before flooding, hydrogen release. These data are presented in Table 4.

Figs. 26a, 26b the electric power history is presented for QUENCH-06 and SF experiments. In plots the experimental data of SF are shifted in time to make the coincidence of the beginning of the transient stage for QUENCH-06 and SF experiments. It is seen that for heating-up the assemblies to the same temperature (see, for instance, the pre-oxidization phase at temperature of 1200°C for the hottest elevations in Figs. 26c, 26d) the different value of the supplied electric power is required. This derived from not only the difference in the length of the heated zones (in QUENCH-06 the length of the heated zone is 1024 mm, in SF – 1275 mm) and total length of fuel rods (in QUENCH-06 the length of the heated rods is 2480 mm, in SF – 3120 mm), but rather heat losses through the shroud and to coolant. The temperatures in the assembly during the stages of the assembly heating-up, pre-oxidization and at the beginning of the transient phase are controlled by the Joule heating and determined by heat power transferred with steam-gas mixture, heat losses into thermal insulation, heat capacity, radiative heat exchange (at temperatures of 1000°C and higher), and at temperature of 1200°C and higher the heat due to steam-zirconium reaction is essential. Different thicknesses of the used thermal insulation as well as different type of the test section cooling lead to different power values of heat losses into thermal insulation. Different types of spacer grids (triangular grid in SF and square grid in QUENCH-06) determine different radiative fluxes in the system including the emissive flows, absorbed or emitted by rods and shroud, and those absorbed by steam-gas mixture. Finally, a distance between the adjacent rods, the number of rods and their radii for these tests are also considerably different that leads both to different thermohydraulic characteristics of the channel, and to different surface of oxidation (comparison of some parameters of these test section is given in Tables 3 and 5). The lower value of electric power in the SF1 experiment is caused by the phenomena of bypassing and steam condensation in the facility lower part that resulted in decrease in the steam flow rate through the assembly heated part.

In comparison of maximum temperatures before flooding in QUENCH-06 and SF1, as well as by the results of the assembly cutting, it is seen that in the SF1 assembly the melting processes were intensive. These processes started just at the end of the quick heating-up stage and, as shown by the numerical analysis, immediately before the flooding the protective oxide layer at the levels of 1100-1200 mm lost the strength properties and the melt appeared on the surfaces of fuel rods. Formation of considerable amount of the melt was the governing factor for the increased hydrogen release in the SF1 experiment (see Table 3).

In the SF2 experiment the total mass of the released hydrogen is somewhat less than in QUENCH-06, this is explained by lower level of the temperatures reached. It is seen from Fig. 27 that the behaviour of the hydrogen release curves is similar and, on the whole, repeat the phases of the experiments. In both cases the hydrogen release at the flooding stage was relatively small.

QUENCH-06 showed that with “quick flooding”, intended for flooding the lower chamber volume (~7180 s of the experiment time), a part of water was “pressed out” into the test section due to boiling (see Fig. 28), that resulted in partial cooing of the simulators (see Fig. 29). Then the repeat heating followed and in ~40 s after the beginning of the “main” flooding water supply the cooling was started. At the phase of the “main” flooding the gradual rise in water level in the assembly was observed. Complete cooling down is observed at the collapsed level of about 30% in the heated part.

In the course of the top flooding in SF1 the upper levels (1250 - 1500 mm) were cooled to ~ 130°C within 3-5 s (see Fig. 6). “Blockage” of the supplied water flow by the opposite flow of steam-gas mixture prevented water getting inside the assembly. This process was made worse by the strong blockage of the clear opening in the assembly due to formation of melt. Further cooling of the assembly took place at the expense of water flowing over the periphery, with this, the cooled front lifted slowly upwards form the test section bottom (see Fig. 7) the same way as under bottom flooding in QUENCH-06.

The combined flooding in the SF2 experiment was made with a delay in the bottom flooding of 26 s. During ~ 10 s, when water was supplied into the assembly only from the top, the claddings at the levels of 1500 - 1300 mm were cooled to temperature of ~100°C; the claddings at the level of 1250 mm began to be cooled (see Figs. 17a and 17b). Boling of the bottom flooding water on reaching the hot levels resulted in the intensive evaporation and displacing of the top flooding water: thermocouples indicate the repeat rise in temperature at the levels of 1300 and 1250 mm. Levels 0 - 1100 mm were cooled sequentially from bottom to top, the same way as under bottom flooding in QUENCH-06 (see Figs. 17c and 17d).

So, the experiments QUENCH-06 and SF1, SF2 were performed under similar conditions at the stages of preheating, pre-oxidization and temperature escalation. The main differences can be referred to the specific features of the facility designs, the method of flooding (top, bottom or combined) and to physical-chemical state of the assemblies at the moment of flooding.

Table 4

Typical parameters of the experiments QUENCH-06 and PARAMETER-SF

Experiment	Steam/argon flow rate (g/s)	Pre-oxidization temperature, °C	Maximum temperature before flooding, °C [c]	Flooding type	Total hydrogen release/during flooding phase
QUENCH-06	3/3	1200	≈1790	bottom	36/4
PARAMETER-SF1	3.3/2[a]	1200[b]	>2000	top	91/54
PARAMETER-SF2	3.5/2	1200	≈1500	combined (bottom and top)	28[d]

a) Measured at the assembly inlet
b) Average temperature
c) By indications of thermocouples
d) No considerable hydrogen release at the flooding stage was revealed

Table 5

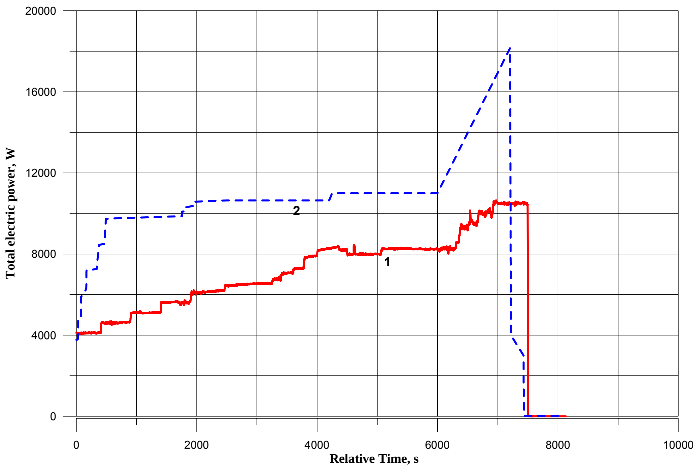
Comparison of geometrical, hydraulic and weight characteristics of FAs of QUENCH-06 и SF

Parameters	QUENCH-06	SF1	SF2
	S=14.6 мм; N_{PT}=4; d_{PT} =6 мм; TC=19(∅2)	S=12.75 мм; N_{PT} =0; TC_1=10(∅3); TC_2=15(∅1.5)	S=12.75 мм; N_{PT} =6; d_{PT}=4/2.8; TC_1=10(∅3); TC_2=10(∅1.6)
Area occupied with fuel rods (F_{rod}) and thermocouples (F_{TC}) and pressure tubes (F_{PT}), mm^2	2093	1340	1406
Total section area of the model assembly (F_{tot}), mm^2	5026	3068	3426
Coolant channel area (F cl), mm^2	2933	1736	2020
Average porosity in the model assembly (ε_{av})	0.586	0.563	0.59
Heated surface of fuel rod claddings (S_{heat}), m^2	0.675	0.656	0.658
Total surface of all zirconium components, (S_{totZr}), on the heated zone length, m^2	1.184	1.133	1.220
Ratio of coolant channel areas $F_{cool}(i)/F_{cool}(1)$	1	0.59	0.686
Ratio of heated surfaces of fuel rod claddings $S_{heat}(i)/S_{heat}(1)$	1	0.97	0.95
Ratio of total zirconium surfaces of all components, $S_{totZr}(i)/S_{totZr}(1)$	1	0.956	1.03
Mass of zirconium in the heated zone, M_{heatZr}, g	3042	2769	2769
Total mass of zirconium along the length of the heated zone, M_{totZr}, g	8235	5447	7936
Ratio of zirconium heated masses, $M_{heatZr}(i)/M_{heatZr}(1)$	1	0.91	0.91
Ratios of total zirconium masses along the core length, $M_{totZr}(i)/M_{totZr}(1)$	1	0.685	0.96

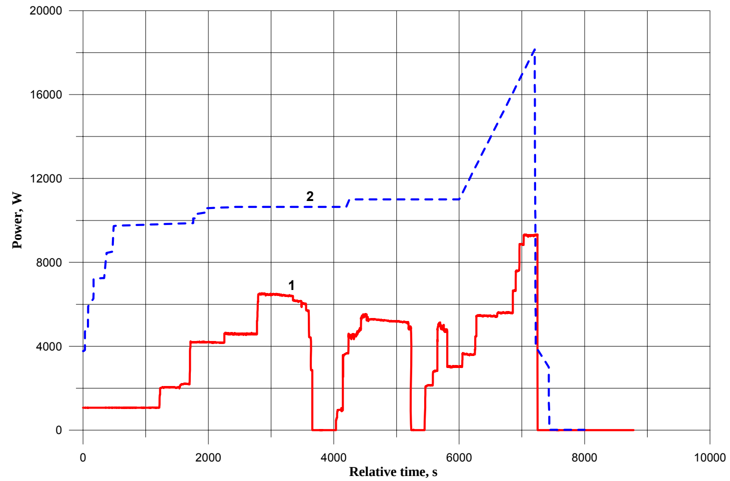
Designations: S – grid spacing, mm; N_{PT} – number of pressure tubes; d_{PT} – pressure tube diameter, mm; TC_1=10(∅3): 10 – number of TC, (∅3) –TC diameter, мм.

References

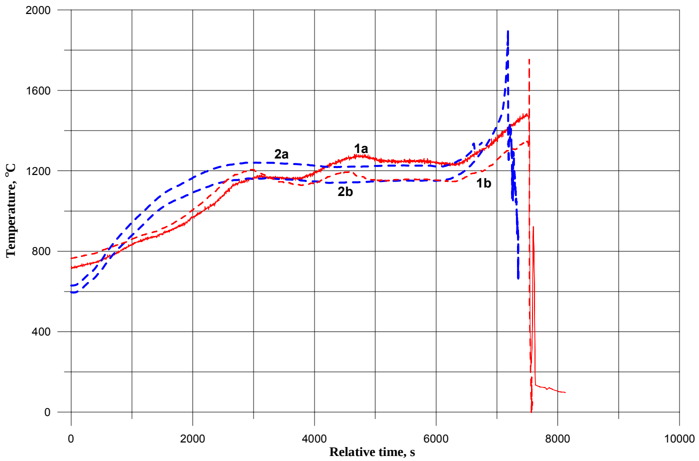
1. L.Sepold, W.Hering, C.Homann, A.Miassoedov, G.Schanz, U.Stegmaier, M.Steinbruck, H.Steiner, J.Stuckert. “Experimental and Computational Results of the QUENCH-06 Test (OECD ISP-45)”, Forschungszentrum Karlsruhe GmbH, Karlsruhe, 2004.



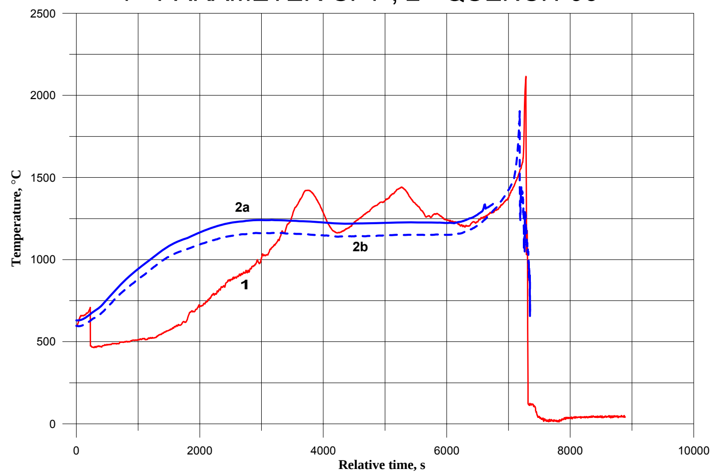
a) Total electric power in the experiments PARAMETER-SF2 and QUENCH-06:
1 - PARAMETER-SF2*, 2 - QUENCH-06



b) Total electric power in the experiments PARAMETER-SF1 and QUENCH-06:
1 - PARAMETER-SF1*, 2 - QUENCH-06



c) Cladding temperature at the hottest levels in the experiments PARAMETER-SF2 and QUENCH-06:
1a - PARAMETER-SF2* - level 1250 mm (T_2312,5)
1b - PARAMETER-SF2* - level 1250 mm (T_2612,5)
2a - QUENCH-06 – level 950 mm (TFS_3/13)
2b - QUENCH-06 – level 950 mm (TFS_4/13)



d) Cladding temperature in the heated zone upper part in the experiments PARAMETER-SF1 and QUENCH-06:
1 - PARAMETER-SF1* - level 1250 mm (T_3512,5)
2a - QUENCH-06 – level 950 mm (TFS_3/13)
2b - QUENCH-06 – level 950 mm (TFS_4/13)

Fig. 26. Main parameters of the experiments QUENCH-06, PARAMETER-SF1 and PARAMETER-SF2.

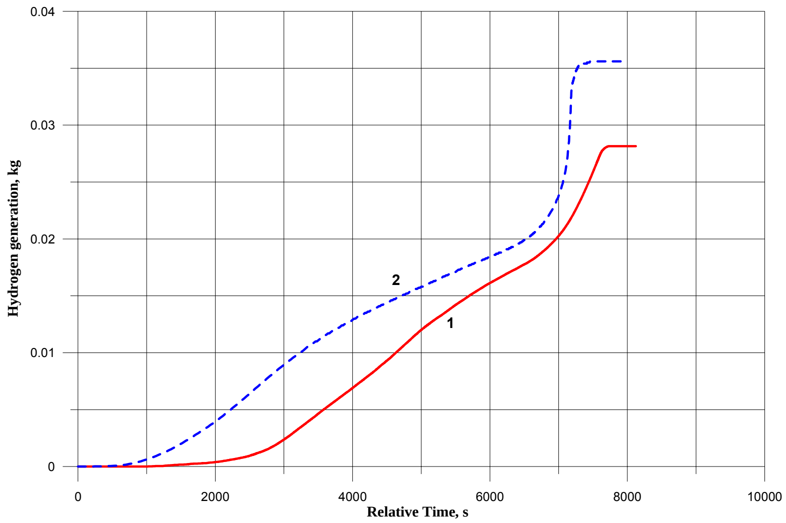


Fig. 27. Total hydrogen generation in the experiments PARAMETER-SF2 and QUENCH-06. (1 - PARAMETER-SF2, 2 - QUENCH-06)

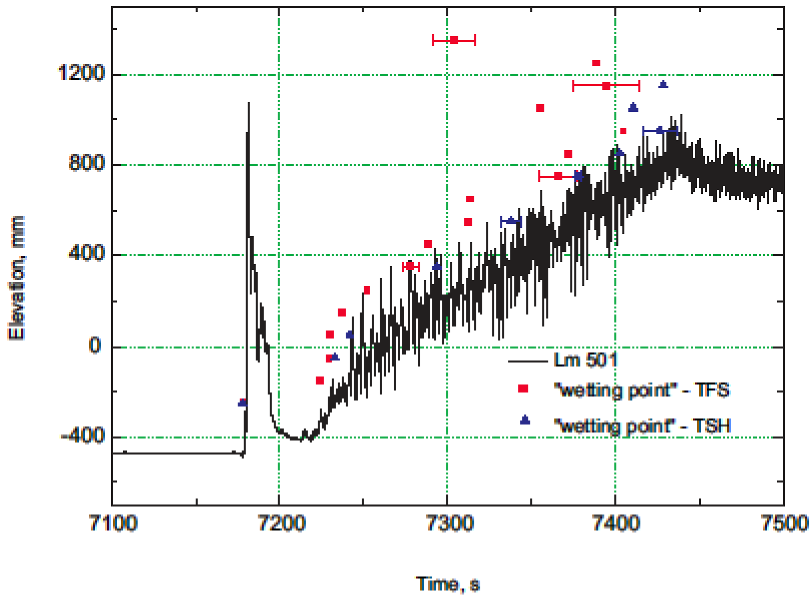


Fig. 28. QUENCH-06 – Indications of collapsed water level in the test section (Lm 501) together with the time point of complete wetting based on claddings (TFS) and shroud (TSH) thermocouples.

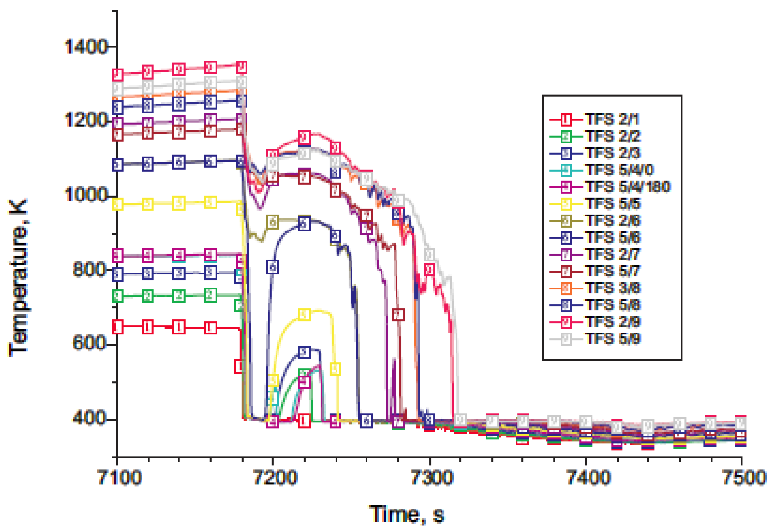


Fig. 29. QUENCH-06 – Temperature response of the cladding thermocouples up to 550 mm elevation during the quenching phase

Conclusions

1. The tests of two 19-fuel rod model FAs of VVER-1000, completed with the standard reactor structural materials, have been performed under the conditions of the initial stage of severe accident including the stage of the low rate cooling with the top flooding (PARAMETER-SF1 experiment) and of the high rate cooling with top and bottom flooding (PARAMETER-SF2 experiment).

2. The valuable experimental data on the behaviour of the overheated FAs of VVER-1000 under the conditions of the top and combined (top and bottom) reflooding have been obtained.

3. Results of material studies of the model FA after the SF1 experiment showed that the top flooding of the model assembly overheated to ~2000°C, with the fuel rod claddings of alloy Э110 and fuel pellets of UO_2, resulted in formation of a large amount of melt (U, Zr,O), melt relocation, fuel dissolution and blockage with the melt.

4. Comparative analysis of the calculated and experimental data on the assembly temperature behaviour, distribution of oxide layers, dissolution of uranium dioxide, showed that the data of the SF1 experiment are self-consistent and present a valuable verification data for the computer codes.

5. Results of the primary processing of the experimental data on temperature behaviour of SF2 show that the bottom flooding results in decrease of the assembly cooling efficiency by the top flooding water.

6. The results obtained on the experiments SF1 and SF2 present the valuable data for improving the models in severe accident codes and their verification.