Experimental Study of the VVER-1000 Fuel Rods Behavior under the Design-basis RIA and LOCA in the MIR reactor

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Abstract. Since 2001 RIAR has been conducting irradiation tests in the MIR reactor under the design basis loss-of-coolant accident (LOCA) and reactivity-initiated accident conditions (RIA), which are targeted at obtaining experimental data on the VVER-1000 fuel performance under these conditions. Each experiment confined itself to examination of fuel, fuel-cladding interaction and analysis of gaseous fission products release from irradiated fuel. Several experiments were carried out under both the RIA and LOCA conditions with the use of the VVER-1000 fuel rods operated at nuclear power plants and attained a burnup of 40 to 70 MW·d/kgU. The irradiation experiments were followed by post-irradiation examinations. In order to conduct irradiation testing of fuel in the loop facilities of the MIR reactor under the VVER-1000 primary circuit conditions, it was necessary to develop appropriate test methods, manufacture fuel test rigs and related engineering equipment.

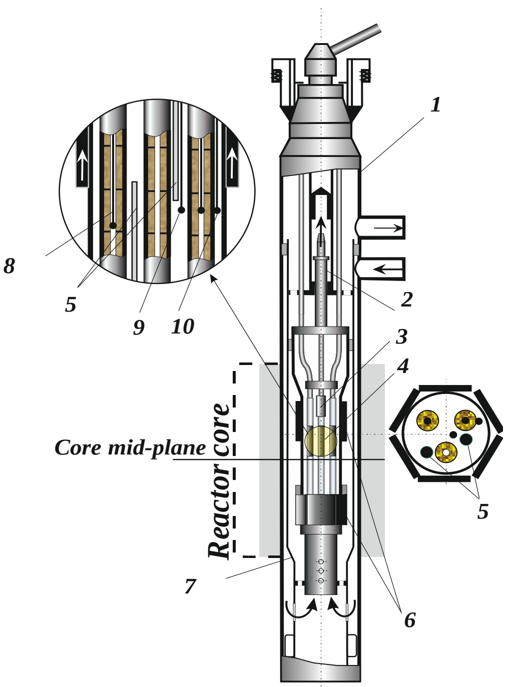
1. RIA tests: testing methodology and experimental data

The existing database for the VVER-1000 fuel performance was obtained during the experiments conducted under the power pulse conditions in pulse reactors IGR and BIGR. These experiments were carried out with narrow power pulses with a use of capsules with stagnant water as coolant at a room temperature and air pressure. Such operating conditions were consistent with the world’s best practices relevant to testing of light water reactor fuel under the RIA conditions. A peak radial average enthalpy (h_{MAX}) is used as a parameter for assessing safety criteria (cladding failure, fuel fragmentation and fuel melting) in such experiments. Table I summarizes the main operating parameters for the VVER fuel rods tested in RIA simulation experiments in power pulse reactors [1].
However, in view of the fact that the conditions of pulse reactor tests are substantially different from the actual initial parameters of fuel rods and the reactor coolant, the obtained experimental data could be excessively conservative on the one hand side, but on the other hand side they may not account fully a real situation in the event of RIA initiation. This fact gave an impetus for conducting RIA simulation experiments on fuel rods in water at a temperature and pressure which were almost identical to their actual ones.

TABLE I: Main parameters of RIA tests attained in power pulse reactors.

	Number of fuel rods under testing	Fuel burnup, MW day/kg U	Power pulse half-width, ms	Peak radial average enthalpy, 10^5J/kg
IGR	8	50	750 – 900	2.5 – 11.1
BIGR	8	50	2 - 4	4.8- 7.8
	4	60	2 - 4	5.2- 6.9

Recently, such irradiation tests have been performed on the PWR and BWR fuel rods in power pulse reactor NSRR with the use of test capsule containing high-temperature stagnant water as coolant[2, 3] and on the VVER fuel rods in the loop test facility of the MIR nuclear research reactor in the coolant flow.
To conduct irradiation tests on the VVER-1000 fuel rods in the MIR research reactor under the design-basis RIA conditions, a method of neutron pulse generation was developed relevant to a separate test channel in the reactor [4]. Feasibility of this neutron pulse generation method was demonstrated by means of experiments. Fuel rods are subjected to MIR reactor tests as a part of test assembly comprising three fuel rodlets. The test fuel assembly consisted of two irradiated fuel rodlets and one reference un-irradiated fuel rod. Triangular or trapezoid pulsed neutron flux is created by removing hafnium screen from the test channel that is intended for shielding test rodlets in the initial condition. Pulse transfer is stopped when the research reactor is shut down at a target time.
A screening device (see FIG.1) consist of absorber screen and compensator to prevent addition of positive reactivity into the reactor core during pulse generation.Neutron pulse width depends on the speed of screening device movement (from 0.3 to 0.5 m/s) by the hydraulic power drive. An array of fuel rodlets remains immovable so it can be instrumented to monitor the process parameters.
Shown in FIG. 2 are variations in the temperature of fuel stack and radial average enthalpy during the RIA simulation experiment in the MIR test channel relevant to different parameters of neutron pulse [4]. For purposes of comparison, calculated temperatures and enthalpy for the VVER-1000 fuel are given here in the event of one control rod ejection. FIG. 2 implies that a satisfactory variation of the main parameters can be achieved for the VVER-1000 fuel by selecting the appropriate pulse parameters during the pulse irradiation test in the MIR reactor.
The design-basis RIA simulation experiment was conducted at design parameters on the rodlets re-fabricated from mother fuel rods which had burn-up of 50, 60 and 70 MW·d/kg U. TABLE II gives main specifications of rodlets. Main parameters of RIA simulation experiments are summarized in TABLE III below.



1 – test channel vessel;
2 – hydraulic power drive;
3 – pressure gage;
4 – fuel rods;
5 – in-reactor direct-charge detector;
6 – movable absorber screens;
7 – flow spreader;
8 – thermocouple attached in the center of fuel stack,
9 – thermocouple in the coolant;
10 – cladding attached thermocouple

FIG. 1. Fuel Test Rig.

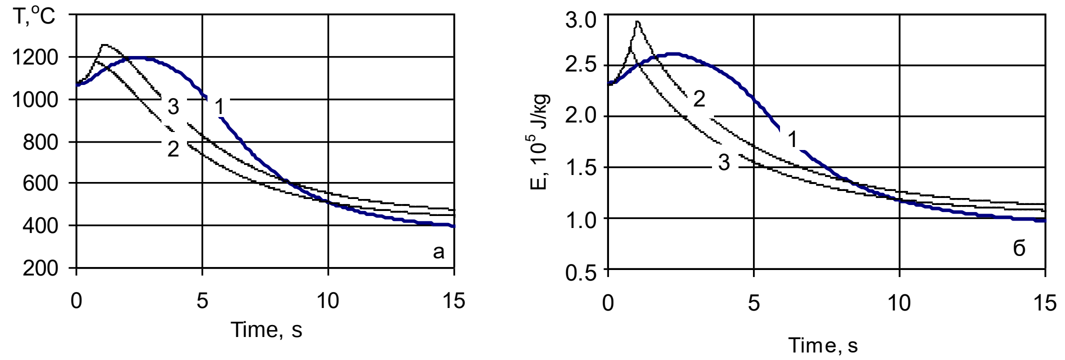


FIG. 2. Changes in the temperature in the center of fuel stack for irradiated fuel rodlet (a) and radial average enthalpy (b) of irradiated fuel as a function of time at different parameters of pulse: 1- calculated profiles for the VVER-1000 fuel; 2-3 – calculated profiles for pulse irradiation tests in the MIR reactor at a linear heat generation rate of 250 W/cm (initial value), pulse amplitude of 3.25, τ=0c(2); τ=0.5s(3).

The initial average heat generation rate of fuel and parameters of neutron-induced pulse and thus the peak temperature and fuel enthalpy, which were attained during the RIA simulation experiments in the MIR reactor, exceeded the predictive calculated values for the VVER-1000 fuel. That is why the obtained experimental data are conservative ones.

As threshold values of peak radial average fuel enthalpy responsible for cladding failure (~ $5.8 \cdot 10^5$ J/kg) that had been obtained in the course of previous experiments were substantially higher compared to values attained during the RIA simulation experiments in the MIR reactor, any significant changes in the state of cladding and fuel meat were not anticipated. This anticipation was confirmed with the results of post-irradiation examinations (PIE) of fuel rodlets. The only exception was fission gas release from high-burn-up fuel. In this case, it was important not only obtain an absolute value of fission gas release under the cladding but define dynamics of fission products release based on the readings of pressure gages attached to the irradiated fuel rods.

TABLE II: Main Specifications of Fuel Rodlets for the RIA Simulation Experiment.

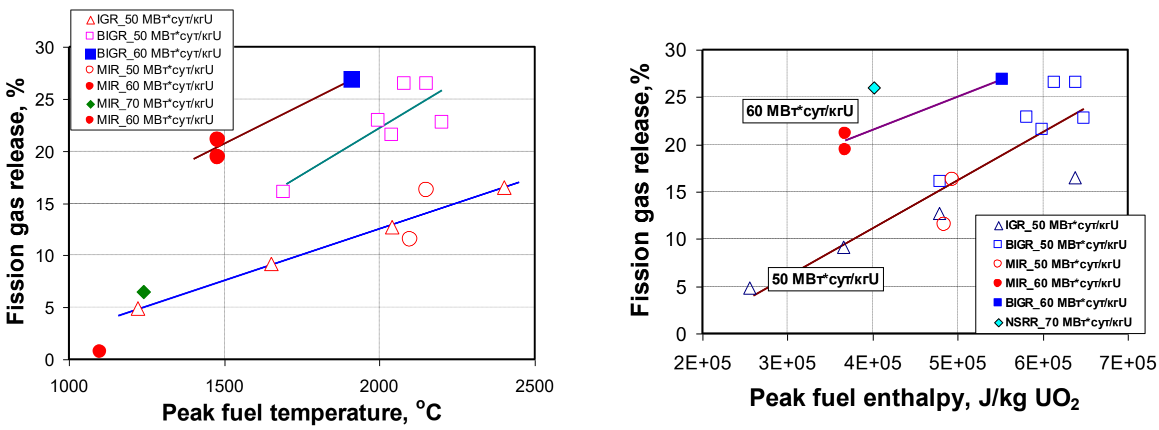
Parameters		**Test #1**	**Test #2**	**Test #3**	**Test #4**	**Test #5**
Bundle of fuel rodlets	Un-irradiated fuel rods	1	1	1	1	1
	Re-fabricated rodlets	2	2	2	2	2
	Burn-up of re-fabricated rodlets, MW·d/kgU	~60	~50	~60	~70	~60
Instrumented fuel bundle	Thermocouples exposed to coolant: - at the inlet of fuel bundle; - throughout the fuelled length of rod ; - at the outlet of fuel bundle	 1 1 1	 1 1 1	 1 1 1	 1 1 1	 1 1 1
	Thermocouple in the center of fuel stack (un-irradiated fuel)	1	1	1	1	1
	Thermocouple attached on the cladding of un-irradiated fuel rod	2	2	1	-	2
	Thermocouple in the center of fuel stack of the rodlet	2	2	1	1	2
	Direct-charge detector	1	1	2	2	1
	Gas pressure transducer inside the rodlet plenum	-	-	1	1	-

TABLE III: Main Parameters of the RIA Simulation Experiment

Parameters		**Measure-ment units**	**Test #2**	**Test #3**	**Test #4**
Burn-up of re-fabricated rodlets		MW·d/kgU	48	59	67
Initial average linear heat generation rate throughout the length	Un-irradiated fuel rod	W/cm	270	210	175
	Re-fabricated rodlets		230	205	140
Pulse amplitude at the level of thermocouple attachment	Un-irradiated fuel rod	-	3.32	3.36	3.23
	Re-fabricated rodlets	-	3.32	3.14	3.23
Pulse half-width		c	1.75	1.58	2.9
Time of screen movement (time of pulse rise)		c	2.0	1.2	0.4
Peak temperature in the center of fuel stack at the place of thermocouple attachment	Un-irradiated fuel rod	°C	1670	1318	150 8
	Re-fabricated fuel rodlet #1		1458	1406	117 3
	Re-fabricated fuel rodlet #2		1468	-	-
Calculated h_{MAX} of fuel stack	Un-irradiated fuel rod	10^5 J/kg	5.3	4.1	4.0
	Re-fabricated fuel rodlet #1		4.9	3.9	2.8
	Re-fabricated fuel rodlet #2		4.8	-	-
Enthalpy increment of fuel stack in pulse	Un-irradiated fuel rod	10^5 J/kg	2.0	1.6	1.7
	Re-fabricated fuel rodlet #1		2.0	1.5	1.1
	Re-fabricated fuel rodlet #2		2.0	-	-

In this case, it was possible to define dynamics of stresses in the cladding under dynamic variation of parameters in RIA simulation experiments. Gas pressure under the cladding was measured online with a measurement error of 1%. These data were recorded with a frequency of 100Hz. Pressure gain of fission gas and its total yield from fuel under irradiation testing were obtained based on processed measurement data. Integrated yield of fission gas was also verified with the use of PIE data.

FIG. 3 demonstrates fission gas measurement data for fuel with burn-up of 50, 60 and ~70 MW·d/kg U compared with the data that have been obtained in the course of RIA simulation experiments in the BIGR and IGR reactors.



a) b)

FIG.3. Fission gas release as a function of the peak temperature (a) and peak fuel enthalpy (b).

It is obvious that as fuel burn-up increases, fission gas release becomes higher at a power pulse. The experimental data on fission gas release for the VVER-1000 fuel rods obtained during the RIA simulation experiments in the MIR reactor channel for fuel rodlets with a burn-up of 50 MW·d/kgU fit well into the fission gas release curves obtained in the power pulse reactors. The experimental data of fuel rodlets with a burn-up of 50 MW·d/kgU along with the experimental data obtained during the RIA simulation tests in the BIGR reactor make it possible to define the trend for fission gas release in relation to the temperature and peak fuel enthalpy at this burn-up.The experiments conducted in the MIR reactor channel indicate that the peak pressure under the cladding coincides in time with the peak temperature of fuel meat but fission gas releases into the plenum after generating the power pulse.

2. LOCA tests: testing methodology and experimental data

The loss-of-coolant accident (LOCA) is classified as an accident which has the most severe consequences and leads to failure of fuel rods in a great number and activity release in the primary circuit. There are three phases in the course of LOCA accident: the first phase that lasts for ~ 30 s when the coolant pressure drops and fuel claddings heats up to ~1100°C for a short time; the second phase when fuel claddings heat slowly up to 800 – 900 °C in steam due to residual heat release in the fuel; the third phase when fuel claddings cool down very fast owing to reflooding.

In-pile LOCA experiments targeted at simulation of its 2nd and the 3rd phases have their own independent value because fuel is exposed to extreme conditions in the core within a long period of time at this phase. It is also anticipated that the cladding will experience the maximum strain and the coolant flow area will reduce.

In the first stage of LOCA simulation experiments, the top target was to reveal the nature of fuel cladding straining in case of bundle test with non-uniform radial heat generation rate, define conditions for core cooling maintenance, and study degradation phenomena of cladding material due to the temperature impact typical for the second and third phases of LOCA.

To achieve the aforesaid objective, the design-basis LOCA simulation experiments is performed in the test channel of the MIR reactor with shrouded test fuel assembly comprising 16 un-irradiated fuel rods and 3 rodlets re-fabricated from mother rods. Fuel rods length is from 1300 to1350mm. Fuel stack length is 1m. The test assembly represents itself a fragment of the VVER-1000 fuel assembly to model all the types of feedthrough cells in the spacer grid. In this way deformation of claddings can be reveled for fuel rods inserted in different parts of fuel assembly. Shown in FIG. 4 is schematic arrangement of fuel bundle and its instrumentation. The test assembly is inserted in the test channel provided with thermal insulation of the upward and downward coolant flows.

In addition to thermocouples and sensors attached to fuel rodlets, the coolant temperature and power density are measured. The reactor coolant temperature is measured at different locations throughout the test channel height. Power peaking factor is in the range of 1.05 to 1.1 at the cross-section of fuel bundle. So all nineteen fuel rodlets can be heated up to a target temperature despite the fact that cooling is not uniform to a certain extent. FIG. 5 represents a temperature scenario of the LOCA simulation experiment.

The LOCA experiment road map (FIG.5) provides for irradiation testing of fuel rodlets at a nominal power of the VVER-1000 reactor (phase I), water expulsion from the raising part of the test channel, (phase II), heating of fuel rodlets in steam at a predetermined rate of heating (phase III) and re-flooding of test assembly when the reactor is brought to the sub-critical state with shutoff rods (phase IV). When the LOCA simulation experiment is underway, a pressure of 1.2 MPa is maintained in the reactor primary circuit.

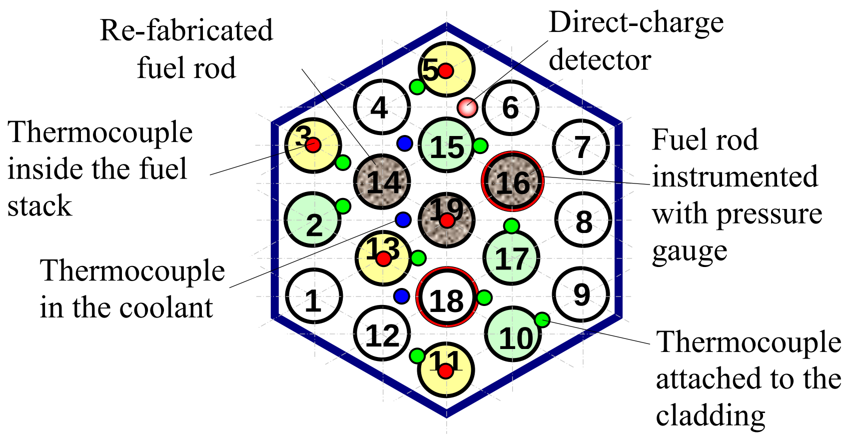


FIG. 4. Schematic arrangement of fuel rodlets, thermocouples and sensors in the test assembly.

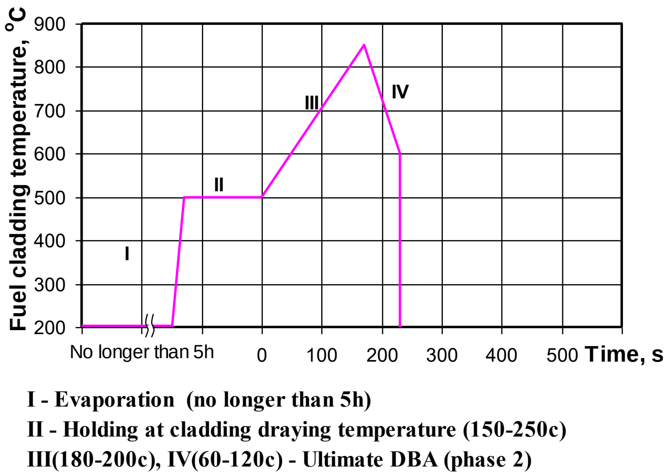


FIG.5. Temperature scenario of the LOCA simulation experiment.

The pressure drop on the cladding depends upon the gas pressure under the fuel cladding that is crated at the stage of fuel rod fabrication. Phases II and III of the LOCA simulation experiment are performed at a power of fuel rodlets almost the same as residual heat generation rate at a lower coolant flow rate (10-15 g/s).

Two tests were conducted (tests BT-2 and BT-3) were conducted within the framework of the project on VVER-1000 high burn-up fuel performance under the simulated LOCA conditions in the test channel of the MIR reactor. TABLE IV provides the main specifications of these tests.

TABLE IV: Main Specifications of the LOCA Simulation Tests

Test	**Fuel, number of rodlets in the test assembly**		**Primary pressure, MPa**	**Temperature range, °C**	**Dewatering time, min**	**State of fuel rodlets**	
	Un-irradiated fuel rods	**High-burn-up fuel rodlets (burn-up, W·d/kgU)**				**Intact**	**Failed**
BT-2	16	3(50)	1.7	500-940	40		+
BT-3	16	3(58)	1.2	500-820	10	+	+

The results of the BT-2 test were fairly fully presented [5-6]. Below are explained the main results of BT-3 test that was conducted with three fuel rodlets attained burn-up of 60 MW·d/kgU. Its temperature scenario is shown on FIG.5.
FIG. 6 and 7 demonstrate changes in the cladding temperature for fuel rodlets of the first and second circle of the fuel bundle at the place of thermocouple attachment under fuel heating.
The peak temperature of the hottest fuel rodlet was 820°C. A rate of temperature rise was in a range of 1.6 up to 2.0 K/s in different periods of time. The temperature in the center of fuel stack exceeded the cladding temperature by 50-60ºC Pressure drop was equal to 4.4 and 5.2MPa on the claddings of un-irradiated and irradiated fuel rodlets, respectively (FIG.8).
As evidenced by the results of X-ray diffraction examination, fragments of fuel pellets were displaced towards ballooning in the failed rodlets and local change in cladding diameter of the intact fuel rodlets. The maximum circumferential strain of claddings in the test assembly is shown as a diagram on FIG.10-a. Fuel was subjected to post-irradiation examinations to study cladding deformation, fuel fragmentation, fuel displacement in the axial direction and its possible expel into the coolant through the cladding rupture. Four un-irradiated fuel rodlets failed as their cladding temperature was beyond 800ºC (FIG.9).

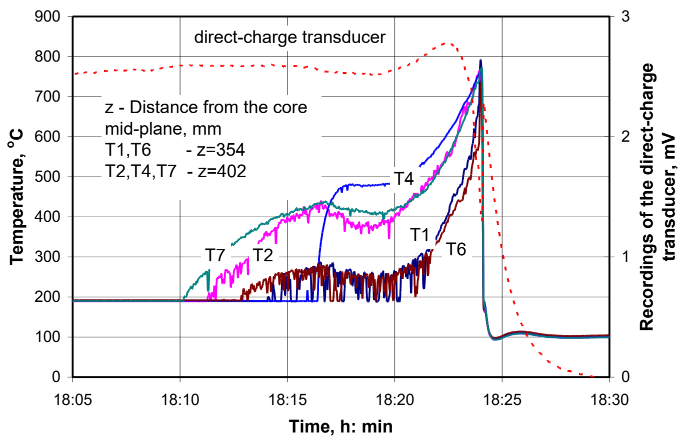


FIG. 6. Readings of the direct-charge detector. Data recorded by thermocouples attached on the claddings of fuel rodlets from the outer circle.

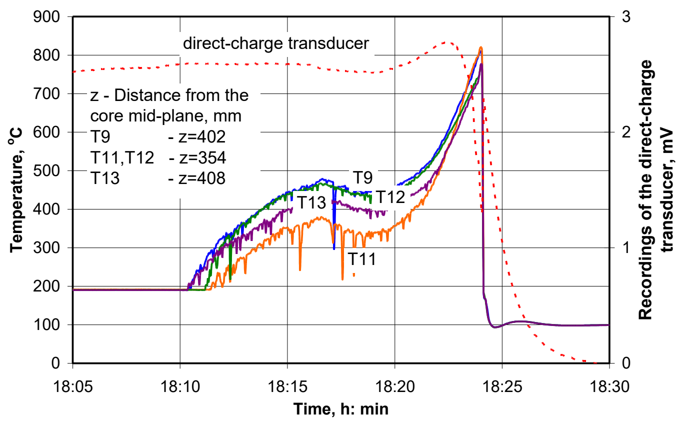


FIG. 7. Readings of the direct-charge detector. Data recorded by thermocouples attached on the claddings of fuel rodlets from the middle circle.

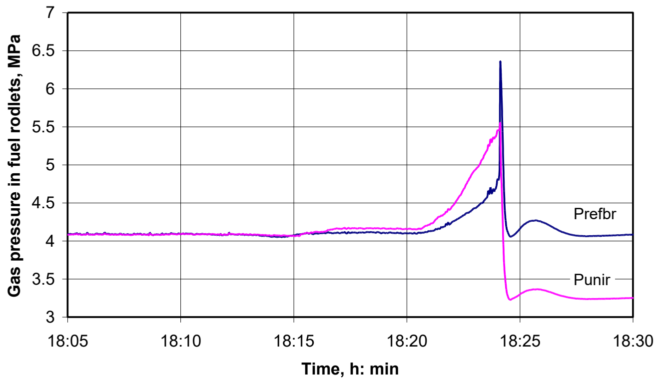
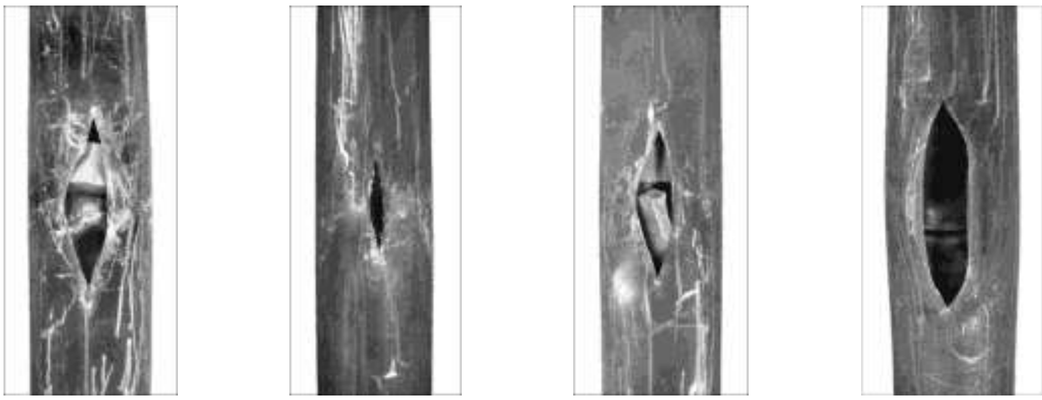
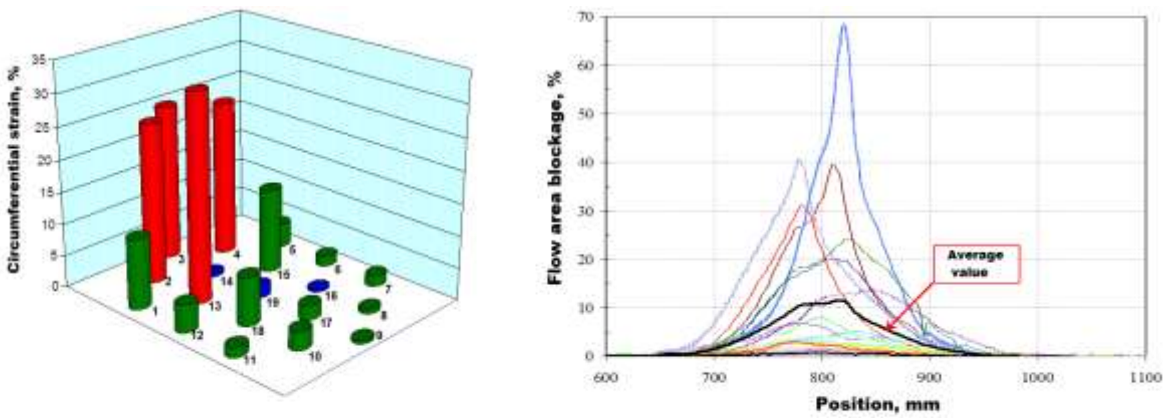


FIG. 8. Gas pressure in un-irradiated (P_{unir}) and re-fabricated rodlets (P_{refbr}).



Fuel rodlet #2 Fuel rodlet #3 Fuel rodlet #4 Fuel rodlet #13

FIG. 9. Outer appearance of the claddings at the place of fuel failure.



a) b)

FIG. 10. Maximum circumferential strain of claddings in the test fuel assembly (a).
Changes in the cross-sectional flow area of the coolant (b).

The regions of localized straining were distributed within an area of 100mm long. It indicates that a rupture of one cladding in the fuel bundle did not cause local overheating of neighboring fuel rodlets. FIG.10(b) represents changes in the cross-sectional flow area of the test assembly as result of cladding deformation. The maximum reduction of flow area occurred on a short length that did not decrease the rate of rodlet cooling.

As there was no additional fuel fragmentation, it was one of the major achievements compared to its state before the LOCA simulation experiment and it was the reason for low fission gas release and a very little probability of its dispersion in the primary circuit through the rupture in the cladding.

As the result, the BT-3 test and subsequent post-irradiation examinations made it possible to obtain such properties of high-burnup fuel, which are responsible for its safe performance in the event of the main coolant circuit break.

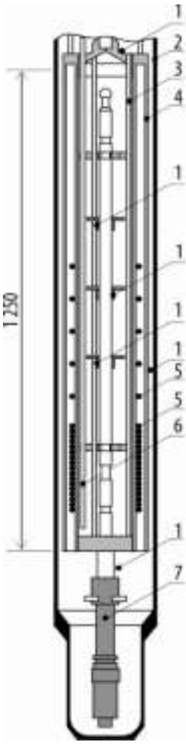
The top target of the second phase was to obtain the parameters of fuel failure and reveal the behavior of fuel stack (fuel fragmentation, fuel expulsion outside the cladding) in the case of rupture formation in the cladding. The object under study was a single fuel rodlet that tended to be an advantage for its test conditions.

Each test was carried out with one fuel rodlet that was inserted along the centerline of the test rig thus enabling a uniform heat generation rate and temperature distribution around the perimeter of the cladding. The fuel rod under test was fixed in the spacer grids arranged for every 200 mm (FIG.11).

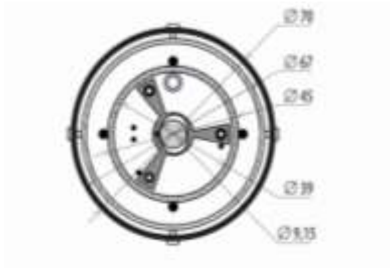
As the design of the test rig provided for two electric heaters, it was possible to simulate thermal effect of surrounding fuel rods inside the VVER-1000 fuel assembly and ensure initial flow rate of water vapor. The test rig incorporated spacers so that the thermocouple intended for temperature recording was attached with its measuring junction to the cladding of the fuel rodlet under testing, provided that the fuel rodlet was inserted in the test rig under the remote controlled conditions. The impact of grids on the axial profile of cladding temperatures and thus corrections to the data recorded with thermocouples were obtained with the use of laboratory-scale set-up and a dummy fuel rod provided with electrical heating from inside. The following parameters were recorded during the test: coolant temperature, cladding temperature at three elevation levels throughout the height, specific heat generation rate, and gas pressure under the cladding. The test was conducted in steam-argon environment.

Geometry of fuel rodlet under test:

- Fuel stack with a diameter of 7.8 mm without a central hole,
- Cladding with an outer diameter of 9.1 mm, wall was 0.585 mm thick.



a)



b)

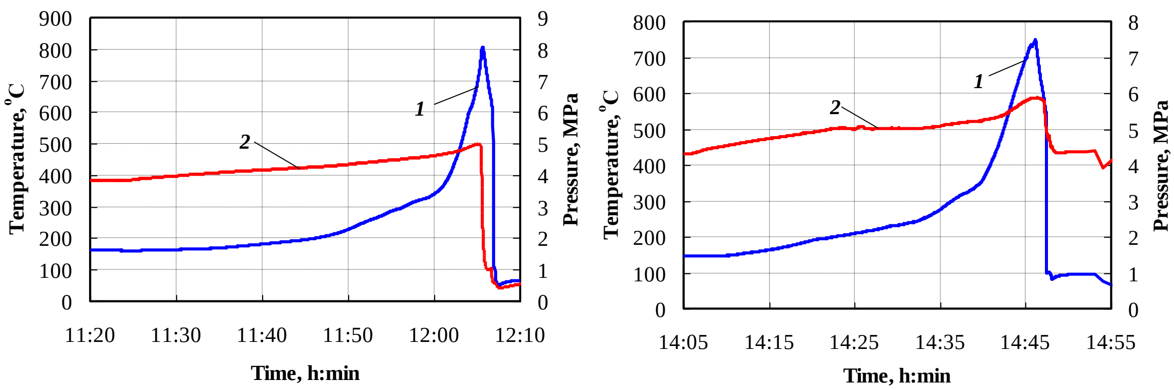
c)

FIG. 11. Schematic representation of design (a), cross-section (b) and fixing of fuel rodlet (c) in the test rig: 1 – thermocouple; 2 – shroud; 3 – basket; 4 – insulator; 5 – heater; 6 – water supply pipe of the test rig; 7 – pressure gage.

Design of fuel rodlet intended for the LOCA simulation test in the MIR test channel is shown schematically on FIG.12. There was gas plenum in the lower part of fuel rod as it was necessary to install a pressure gage.
Fuel rodlet had been pressurized with helium up to a design pressure prior to its sealing that was measured with the pressure gage after sealing.

So far two tests have been carried out (1 and 2) [7] in the MIR reactor channel with the use of the test rig (FIG. 11). The object under study was the VVER-1000 fuel rod with higher uranium content (thinned cladding, fuel stack without the central hole). Fuel burn-up was 45 MWd/kgU and 60 MWd/kgU, respectively. FIG.13 and TABLE V below show the main test parameters and experimental data obtained in the course of LOCA simulation test in the test channel of the reactor. Failure of fuel rodlet and the greatest change in the cladding diameter occurred during test #1 in the regions where the cladding temperature was the highest. FIG.14 shows photographs of these regions.

FIG.12. Fuel rodlet intended for irradiation testing in the MIR reactor channel: 1 – ferromagnetic core; 2 –; 3 – lower gas plenum; 4 – fuel stack; 5 – upper gas plenum.



a) b)

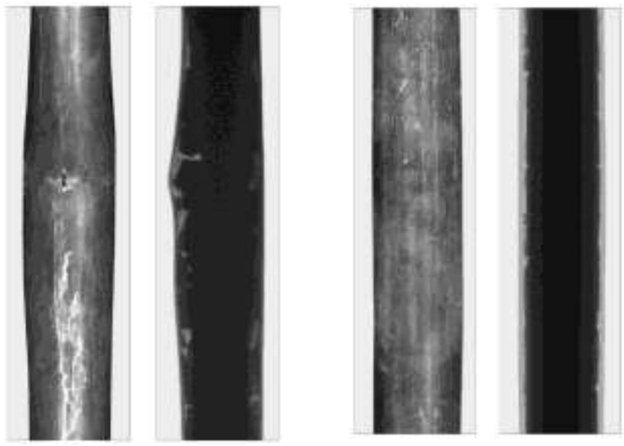
FIG.13. Cladding temperature variation with time (1) at 10 to 20 mm above the middle spacer grid and time history of gas pressure (2) in the lower gas plenum during tests 1 (a) and 2 (b)

TABLE V: Main Specifications of the LOCA simulation tests with the use of single fuel rods

Parameter	**Test 1**	**Test 2**
Outer / inner diameter of standard fuel rod selected for refabrication, mm: cladding fuel stack	9.1/7.93 7.8/0	9.1/7.93 7.8/0
Maximum fuel burn-up in the fuel rod under test, MW·day/kgU	45	60
Peak cladding temperature, °C	807	750
State of fuel rodlet after testing	failed	intact
Cladding temperature during cladding failure, °C	770-780	-
Rate of temperature increase during failure, °C/s	3.6	1.2*
Pressure drop on the cladding during fuel failure, MPa	5.0	5.8*

*Note: * pressure drop and rate of temperature increase for the intact fuel rodlet are given at the maximum temperature of 750°C achieved during test #2.*

Changes in the cladding diameter throughout the length of fuel rods under test revealed a significant impact of spacers on the cladding deformation which look like local minima on the profile diagram of fuel rod (test #2) and knees of the cladding diameter curve (test #1) (FIG.15)
The cladding ruptured during test #1 below the middle spacer grid at a distance of 435 mm from the beginning of fuel stack. The resultant rupture was 1.6 mm long and rupture opening was 0.4 mm. The cladding rupture was ductile in its nature with a 100% thinning of the cladding prior to its rupturing. Oxide films both on the outer and inner surfaces of the cladding as well as the remaining fuel fragments on it did not make any noticeable effect on its deformation (FIG.16).
As evidenced by gamma scanning and X-ray radiography, there was no significant axial transport of fuel. Fuel fragments were redistributed in the radial direction in the regions of major cladding deformation. Once the test fuel rodlets had been cut out in the region of the maximum deformation, fuel was extracted. The extracted fuel fragments of 2.5 mm were distributed into fractions (FIG.17). A proportion of fine fraction with a particle size comparable to the maximal opening of the cladding in test #1 was rather low in both cases.



a) b)

FIG. 14. Surface appearance of the cladding and X-ray images taken at place of cladding failure during test #1 (a) and the maximum deformation during test #2(b)*
*Note: * - X-ray image is rotated 90° relative to the photograph of surface appearance.*

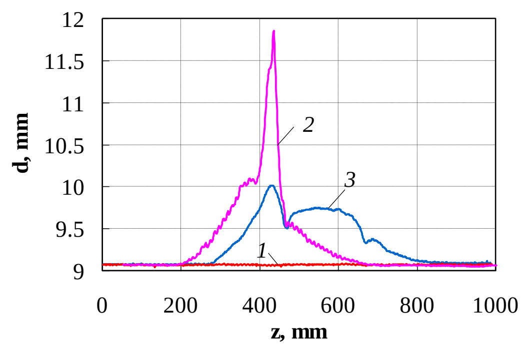


FIG. 15. Profile diagrams of the cladding and fuel rodlets before (1) and after test #1 (2) and #2(3).

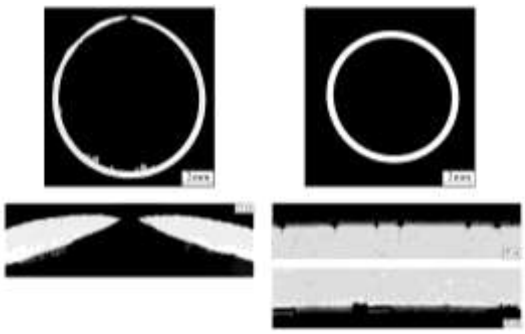


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FIG.16. Fuel cladding structure after test #1 (a) at the rupture cross-section and after test #2 (b) at the cross-section of its maximum deformation.

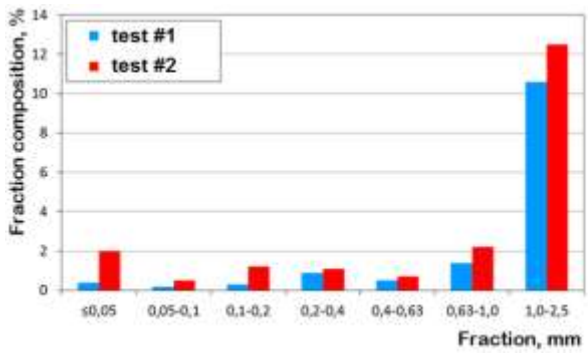

FIG.17. Grain particle size of extracted fuel.

The results of qualification tests for both original fuel rods and re-fabricated rodlets as well as the results of post-irradiation examinations were used as a basis to reveal the factors with possible impact on the cladding deformation under the LOCA conditions:

- high hydraulic resistance of fuel stack comprising pellets without a central hole could limit gas influx towards the hot region of fuel rodlet when the burn-up is sufficient for closing fuel-to cladding gap and thus the cladding deformation slows down due to the pressure of the filling gas;
- tight fuel-cladding contact that exhibits itself as remaining fragments of fuel on the inner surface of severely deformed cladding could make separation of cladding from fuel difficult.

3. Conclusion

The available physics-based simulation principles of parameters specific to transient processes in the event of design-basis accidents LOCA (loss-of-coolant accident) and RIA (unauthorized insertion of positive reactivity) in the VVER-1000 reactor as well as development of test equipment for the MIR reactor made it possible to conduct irradiation testing of high-burnup fuel under the design-basis conditions of each accident. Therefore, the simulation tests contributed to obtaining experimental data on fuel behavior under the extreme conditions.

4. References

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