

## Current and Prospective Tests in Reactor MIR

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**Abstract.** The MIR reactor was purposely designed to perform loop tests of fuel rods and fuel assemblies of various types of reactors. Nowadays, there are seven loops used for testing (two loops are cooled with pressurized water, two loops are cooled with boiling water, two loops are cooled with overheated steam and one loop is gas-cooled). The current key activities are loop tests of VVER fuel rods to study their characteristics under the conditions simulating the normal operational ones, deviations from normal conditions and design-basis accidents as well as fission gas release from leaky fuel rods. Moreover, high-dense low-enriched fuel is tested under the RERTR program. The paper presents the current and prospective test programs for fuels and structural materials of various types of reactors, describes the experimental-methodical base of tests, and gives some test results.

### 1. Introduction

Regarding physical features, reactor MIR is a thermal heterogeneous reactor with a moderator and reflector made of metal beryllium (FIG.1). Regarding design features, it is a channel-type reactor immersed in water pool [1]. Such design allowed an advantageous combination of pool-type and channel-type reactor features. At present, MIR reactor is equipped with the following experimental facilities and devices:

- loop facilities that are the experimental base of the reactor and provide for its attractive capabilities;
- test devices where heat is removed from a tested item by primary circuit water or pool cooling water (used to test fuel and components of research reactors);
- critical assembly that is a reactor physical model;
- hot cells with related facilities;
- stand to inspect fuel assemblies and fuel rods in the storage pool.

High-temperature and boiling water loops provide for the desired coolant parameters to test fuels of both pressurized-water and boiling reactors (Table 1). To test fuel and structural materials of high-temperature gas-cooled reactors, there is a gas-cooled loop, where either nitrogen or helium or other inert gas mixtures can be used as coolant [2].



FIG. 1. Research reactor MIR.

TABLE I: Parameters of MIR loops.

Parameter	Loops						
	PV-1	PVK-1	PV-2	PVK-2	PVP-1	PVP-2	PG
Coolant	Water	Boiling water	Water	Boiling water	Boiling water, steam	Boiling water, steam	N <sub>2</sub> , He
Number of loop channels	2	2	2	2	1	1	1
Max channel capacity, kW	1500	1500	1500	1500	100	2000	160
Max coolant temperature, °C	350	350	350	365	500	550	500
Max temperature at the irradiation rig outlet, °C	350	350	350	365	1000	1200	1300
Max pressure, MPa	17,0	17,0	18,0	18,0	8,5	15,0	20,0
Max flow rate through the loop channel, t/h	16,0	16,0	13,0	13,0	0,6	10,0	

A large reactivity margin achieving  $-37\beta_{\text{eff}}$  and availability of 27 shim rods allow simultaneous testing in the loop channels differing greatly in the neutron flux density: the neighboring channels may differ in neutron flux values by three times while diametrically-opposite channels may differ by 20 times. A high neutron flux density (achieving  $5 \times 10^{14}$  1/cm<sup>2</sup>·s) allows maintaining the required linear heat rate (LHR) on the VVER standard high-burnup (up to 80 MWd/kgU) and experimental fuel [2].

A wide range of MIR experimental equipment and parameters allow the following tests and experiment to be carried out:

- loop and in-pile tests of VVER fuel rod characteristics under conditions simulating the normal operational ones, deviations from normal conditions and design-basis accidents as well as fission gas release from leaky fuel rods;
- tests of fuel rods and fuel assemblies of propulsion, low-power, floating, high-temperature gas-cooled and research reactors;
- tests of fuel rods and structural materials under simulated PWR conditions, including water chemistry.

Besides, the MIR reactor is used to accumulate radioisotopes and perform interim inspections and primary post-irradiation examinations of fuel rods, fuel assemblies and structural materials in the reactor storage pool and hot cells.

## 2. Equipment and Techniques to Test and Examine Fuels and Structural Materials

JSC “SSC RIAR” carries out a wide range of activities to perform reactor tests: design and manufacture of irradiation rigs (IR), gages for in-reactor measurements and some types of experimental fuel rods; pre-test evaluation of characteristics of full-size and refabricated fuel rods from either standard or experimental FAs from nuclear power reactors; equipment of fuel rods with gages; reactor tests; interim and post-irradiation examinations; waste disposal and temporary storage of spent fuel. FIG. 2 shows the full cycle of activities for MIR testing of VVER fuel rods spent at an NPP [2].

To test fuels, structural materials and experimental FAs of nuclear power reactors, different IRs were developed (FIG. 3) [2, 3, 4].

- dismountable IRs containing from 10 to 19 fuel rods and having an active part < 1000 mm long to test fuel rods of water-cooled reactors;
- dismountable IRs for comparative lifetime tests of shortened (< 250 mm) dummies of fuel rods (up to four IRs containing up to 144 fuel rods can be installed in the loop channel);



FIG. 2. Preparation and performance of a MIR reactor experiment for refabricated and full-size VVER fuel rods.

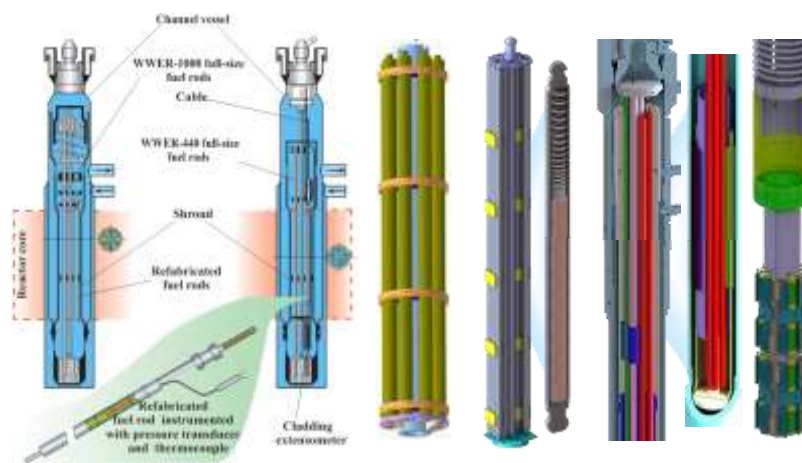


FIG. 3. IRs to test fuels, structural materials and experimental FAs.

- dismantlable IRs to test non-instrumented refabricated fuel rods (< 1000 mm long) and full-size fuel rods (< 3500 mm long) from FAs spent at NPPs;
- dismantlable IRs to test fuel rods under power ramping and cycling conditions by moving either screens or fuel rods;
- instrumented IRs to test single fuel rods and FA fragments under simulated LOCA conditions;
- instrumented IRs to perform tests under simulated design-basis RIA conditions;
- IRs to examine the behavior of leaky fuel rods;
- IRs to test structural materials and FAs components of water-cooled reactors.

New reactor testing techniques have been developed and verified to comply with current requirements:

- testing of full-size VVER fuel rods under power ramping conditions, the fuel rod axial strain being measured;
- testing of a single instrumented refabricated fuel rod under the simulated LOCA conditions;
- irradiation of stressed samples of structural materials and FAs components in the set water chemistry with an interim inspection of their state and measurement of relaxation.

To evaluate the characteristics and conditions of fuel rods testing, different in-reactor gages have been developed for on-line parameters control during the experiment. Table 2 presents the specifications of gages applied (FIG. 4) [2, 3].

TABLE II: Types and specifications of in-reactor gages installed in the IRs and fuel rods.

Parameter to measure	Type	Measurement range	Error	Size, mm	
				diameter	length
Coolant and cladding temperature	Chromel-alumel thermocouple of cable type	Up to 1100 °C	0,75%	0,5	
Fuel temperature	Chromel-alumel thermocouple of cable type	Up to 1100 °C	0,75%	1 – 1,5	
Fuel temperature	Tungsten-Rhenium 5/20 thermocouple, Duct Mo + BeO	Up to 2300 °C (Up to 1750 °C*)	~ 1,5%	1,2 - 2	
Cladding elongation	Differential transformer	(0 - 5) mm	±30·µm	16	80
Cladding diameter change	Differential transformer	(0 - 200)·µm	±2 µm	16	80
Change of gas pressure under the cladding	Bellows + Differential transformer	(0 - 20) MPa	~ 1,5 %	16	80
Neutron flux density (rel.unit)	Self Powered Neutron Detector (Rh, V, Hf)	10 <sup>15</sup> - 10 <sup>19</sup> 1/m <sup>2</sup> ·s	~ 1%	2 - 4	50-100

\* - experimental data for high-burnup fuel rods



FIG. 4. Differential transformer of a high-temperature gage to measure gas pressure under the cladding.

Equipment has been developed and manufactured to test fuel rods, FAs components and other irradiated items in the reactor storage pool and to purify them from deposit, if necessary. The equipment includes an inspection stand and ultrasonic purification facility (FIG.5, 6). The inspection stand is intended for a visual inspection of items by means of color radiation-resistant TV cameras and measurement of cladding diameter and oxide film thickness on the cladding surface. Besides, it is possible to measure the geometry of experimental fuel rods and FAs [5].

To examine the stress relaxation of stressed samples of structural materials under irradiation, a technique and equipment have been developed. Flat samples are measured in the hot cell by a relaxometer under a four-point bend (FIG. 7) [6].

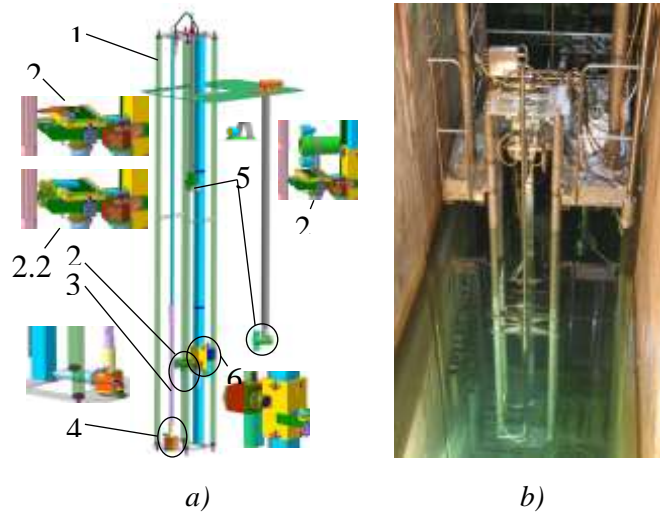


FIG. 5. Layout (a) and appearance (b) of the inspection stand in the MIR storage pool  
 a) 1 – skeleton; 2 – 2D table; 2.1 – 2D table with a diameter measuring module;  
 2.2 – 2D table with an oxide film thickness measuring module; 2.3 – 2D table with a TV camera;  
 3 - EFA; 4 – drive connection; 5- TV cameras; 6 – carriage with a drive.

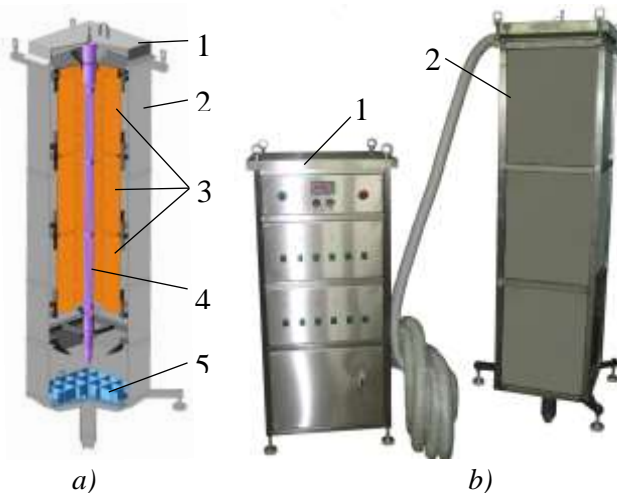


FIG. 6. Layout (a) and appearance (b) of the ultrasonic purification facility: a) 1 – cover, 2 – case, 3 – immersed piezoelectric converters, 4 – EFA, 5 – filter;  
 b) 1 – control unit, 2 – ultrasonic module.

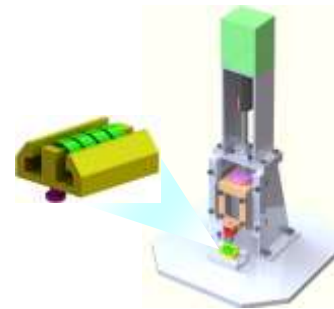


FIG. 7. Relaxometer to measure stress relaxation of structural materials samples.

### 3. Fuel Tests

The following programs and techniques have been developed and implemented to test water-cooled reactors fuels tests in the MIR reactor [3, 4, 7]:

- tests under simulated normal conditions, including re-irradiation of refabricated and full-size fuel rods;
- tests under transient and accidental conditions: RAMP, FGR, power cycling, design-basis LOCA and RIA;
- tests of leaky fuel rods with artificial defects;
- tests of research reactors fuels.

#### 3.1. Lifetime Tests and Re-irradiation of Fuel Rods to Higher Burnups

Purpose of testing:

- justification of the fuel rods performance and check of new design solutions, revealing of

peculiarities of changes in the fuel rods conditions depending on their design and fabrication technology;

- generation of experimental data on the fuel and cladding materials performance under conditions close to standard ones, including power, temperature and water chemistry;
- evaluation of changes in the fuel rods conditions under rising burnup and at the set power level as well as preparation of high-burnup fuel rods for experiments RAMP, LOCA, and RIA.

FIG. 8 presents the change in the experimental VVER type fuel rods LHR during their tests in the MIR reactor under the simulated normal conditions [7, 8].

### 3.2. Power Ramping

Power ramping means its significant increase at a certain rate, the reactor being operated at lowered power for a long time. The purpose of test is to evaluate the influence of the following parameters on the fuel rods performance: fuel burnup, initial LHR, amplitude of LHR and its increase rate, time of exposure at the maximal LHR.

The increase in the LHR of fuel rods in question is done by re-distributing heat rate in the core by changing the location of control rods. If necessary, the reactor capacity is increased [7, 9].

About 100 VVER fuel rods were tested, both full-size and refabricated, with a burnup ranging from 10 MWd/kgU to 70 MWd/kgU as well as more than 40 experimental fuel rods of different modifications pre-irradiated in the MIR reactor to the desired burnup. As a rule, the initial LHR of fuel rods was set the same as during the last stage of their operation in a power reactor. Its maximal value exceeded the admissible level set by the operation specifications. FIG. 9 shows the change in the linear heat rate vs. fuel rods burnup [7, 9].

Analysis of the presented data shows the test parameters to change in a wider range. At that, neither of refabricated or full-size fuel rods filed during testing.

In addition, VVER instrumented fuel rods undergo LHR stepwise power increase tests (FGR experiments) to obtain data on the dependence between fission gas release and LHR and burnup of fuel rods [7, 9].

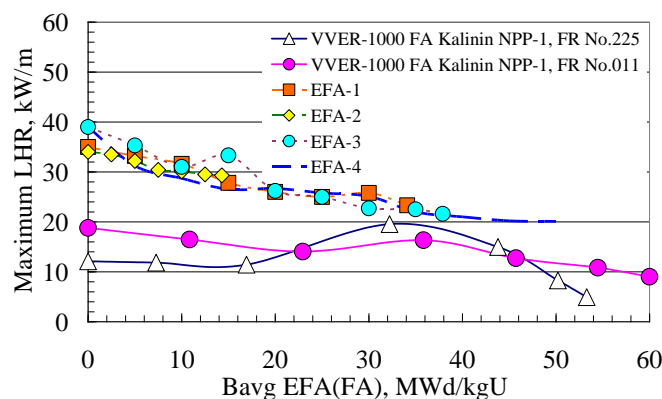


FIG. 8. Change in the maximal LHR of experimental VVER type fuel rods during MIR tests and full-size VVER-1000 fuel rods under normal operation depending on the average burnup through the FA (EFA).

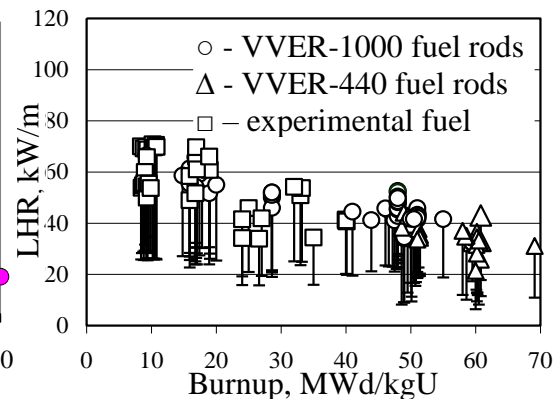


FIG. 9. LHR change vs. fuel rods burnup during the RAMP experiments in the MIR reactor.

### 3.3. Power Cycling

The implementation of operating modes, which follow the line load, requires experiments to prove the performance of fuel rods under the power cycling conditions. The test scenario provides for a periodical decrease in the fuel rods LHR by  $\sim 1.5$  times and its increase up to the same level. The transient process is to take 20-30 minutes, the exposure under the stationary conditions being for 6-12 hours [7, 9, 10].

An IR was purposely developed for the experiments to install four refabricated fuel rods equipped with detectors as well as four movable absorbing screens providing a heightwise sequential screening of either one pair of fuel rods or the other one (FIG.10).

The MIR reactor was successfully used to test VVER-440 and VVER-1000 fuel rods under the power cycling conditions. FIG. 11 and 12 show a change in the fuel temperature during testing and dependence between the fuel rod elongation and fuel temperature. No fuel rod leakage was recorded during the tests [7, 9, 10].

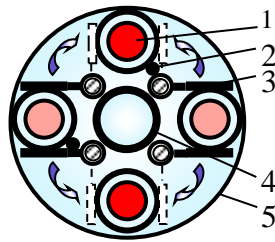


FIG. 10. IR cross-section:

1 – fuel rod; 2 – neutron detector; 3 – absorbing screen; 4 – displacer; 5 – channel body.

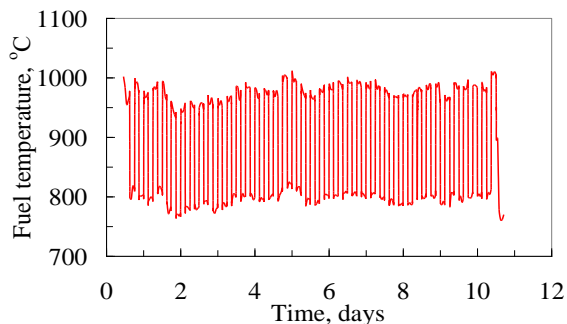


FIG. 11. Change in the fuel temperature during the experiment.

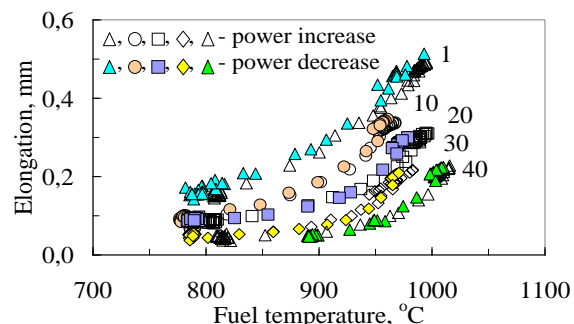


FIG. 12. Fuel rod elongation vs. fuel temperature under the power cycling (in figures are marked the number of cycles).

### 3.4. Design-Basis LOCA

The experiments simulated conditions typical for the second and third stage of the VVER-1000 maximal design-basis accident with the main coolant circuit break. Techniques and IRs were developed and are applied to test both VVER FA fragments and single fuel rod (FIG. 13). The first IR is a fragment of a VVER 19-rod FA where some fuel rods are refabricated, including those with high burnup. The other IR is used to test single instrumented refabricated fuel rod [4, 7, 9].

The purpose of experiment is to get the following data: deformation of the fuel rod bundle to further use this information for the thermo-mechanical state calculations; fragmentation of high-burnup fuel, its axial movement and release into the coolant in case of a cladding rupture. The tests are performed according to the temperature scenario and based on the calculations and algorithm to implement transient thermal processes that are peculiar in generating a medium phase boundary with a high steam content at the FA top.

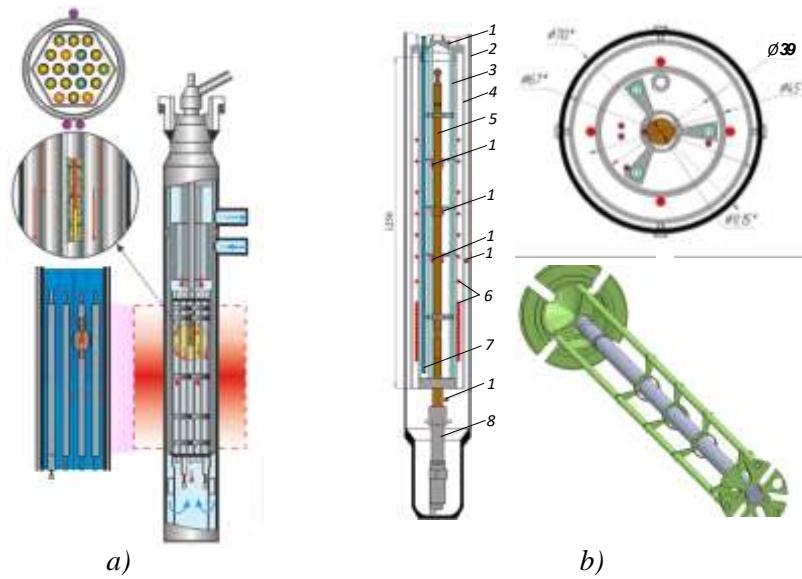


FIG. 13. IRs to test an FA fragment (a) and a single fuel rod (b) under simulated LOCA conditions 1 - thermoelectric converter; 2 - TR jacket; 3 - basket; 4 - insulator; 5 - fuel rod; 6 - heater; 7 - water supply tube; 8 - pressure transducer.

The rewetting of fuel rods is done by a quick power drop with further increase in the flow rate of coolant, of which temperature at the fuel rod bundle inlet makes up about 100°C. During testing, the following parameters are measured online: fuel and cladding temperature, gas pressure under the cladding, coolant temperature at the inlet and outlet of the experimental FA and loop channel, heat rate in the loop channel (by means of a neutron detector). FIG. 14 shows the recorded change in temperature during the LOCA experiment. FIG. 15 and 16 present the appearance of an FA fragment and a single fuel rod after the tests [4, 7, 9].

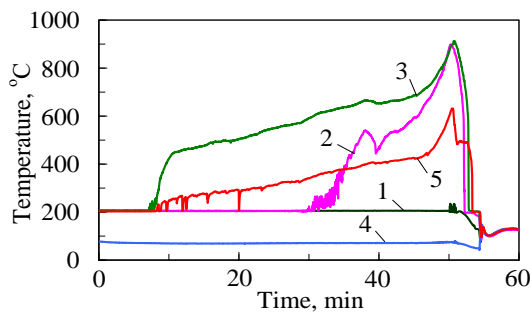


FIG. 14. Change in the cladding temperature at a distance of 562 mm (1), 757 mm (2) and 887 mm (3) from the support grid. Change in the coolant temperature at the FA inlet (4) and outlet (5).

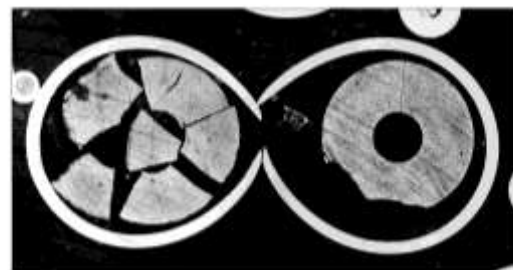


FIG. 15. Cladding deformation of a refabricated (a) and unirradiated (b) fuel rods after LOCA testing of an FA fragment.

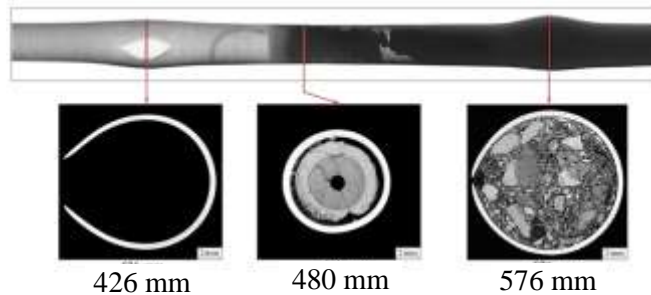


FIG. 16. Deformation of a single refabricated fuel rod cladding after LOCA test (distance from the fuel rod bottom).



### 3.5. Design-Basis RIA

The tests are performed to get experimental data on the behavior of fuel rods under the design-basis RIA, when a pulse appear in VVER-1000 with an amplitude achieving 4 seconds and half-width of (1.5-3) seconds [7, 9]. Unlike tests performed previously in pulse reactors, the one developed for the MIR reactor allows implementing the desired fuel rod LHR, primary coolant temperatures and required parameters for the pulse thus simulating in full the fuel rods operational conditions.

The test conditions are provided under the constant reactor power by removing the hafnium absorber. The IR contains a fragment of a VVER-1000 instrumented three-rod FA (FIG. 17), The insertion of positive reactivity is compensated by an additional absorber that substitutes the hafnium one in the core and has the same absorbing capacity. The experiment is finalized with the scram discharge actuated by the time switch signal. The exposure at the maximal LHR lasts for (0.5-3) seconds (FIG.18). The exposure period is selected fuel based on the necessity to provide the maximal enthalpy and radius-average temperature of fuel [9].

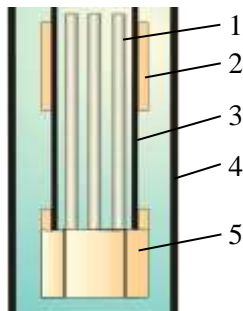


FIG. 17. IR for RIA experiments:

1 – fuel rod; 2 – displacer; 3 – absorbing screen; 4 – channel body; 5 – additional absorbing screen

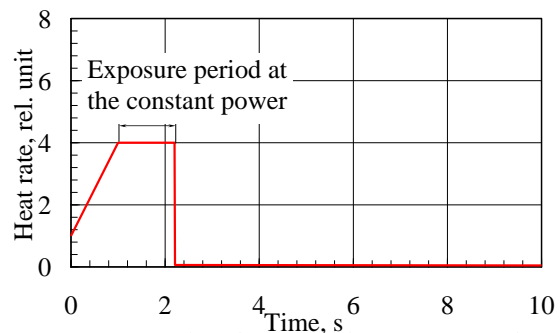


FIG. 18. Pulse shape for the RIA test in the MIR reactor.

### 3.6. Tests of Leaky Fuel Rods with Artificial Defects

Taking into account the peculiar state of VVER fuel rods with burnup more than  $\sim (45-50)$  MWd/kgU, the current experiments are being done to get new data for the development and verification of calculation codes, justification of criteria of the safe VVER operation in case of leaky fuel rods, prediction of changes in their state and radiation environment. To get such data, an experimental program was developed and experiments are being done in the MIR reactor loop with VVER refabricated fuel rods having different artificial defects on their claddings [7].

The key experimental tasks are:

- to plot parametric dependencies between the release of fission products (FP) of different physical-chemical groups and single nuclides and fuel burnup, power level and type, size and location of cladding defect;
- to determine the kinetics and peculiarities of the cladding defect evolution, including the secondary defect generation.

An IR was developed, of which design provides for the similarity of distribution of coolant velocity over the FA cross-section for all types of leading-through cells – central, horizontal and angle (FIG.19) [7].

The techniques to examine the FP into the coolant provide power change testing according to a specific scenario. To get information about the FP kinetics, an on-line gamma-spectrometer (GEM-detector + DSPec-analyzer) is used installed close to the loop primary pipeline. The sampling and further measurements are done according to the scenario with the account of changes in the emission intensity from different FP during the sample exposure.

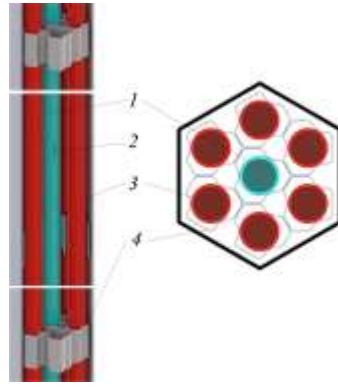


FIG. 19. FA layout and cross-section:

1 - experimental fuel rod; 2- the artificially applied defect; 3- non-irradiated fuel rod; 4 - spacer grid

At that, the gaseous and liquid phases of each sample are measured. Table 3 present the experimental data generated when studying the FP into the loop primary coolant [7].

TABLE III: Experimental data on the FP into the coolant during tests of leaky fuel rods.

Equipment	Measurement results	Target data
Standard loop system for cladding integrity control	Intensity of loop coolant neutron emission	Change in coming of delayed neutrons carrier into the coolant
Sampling system. On-line gamma-spectrometer	Activity of nuclides in the loop coolant and coolant samples with liquid and gaseous phase separation	Rate of FP through the cladding defect into the coolant
		Rate of fuel erosion with contact coolant in the area of defect
Standard loop detectors	Coolant parameters: flow rate, pressure, temperature.FA power	Coolant parameters: flow rate, pressure, temperature. Local power, cladding temperature, heat removal mode

### 3.7. Test of Research Reactors Fuels

Fuel rods and FAs from research reactors are tested in the MIR reactor both in the standard channels (inner diameter 75mm) and in special-purpose channels (diameter up to ~ 150 mm) or in capsules. The coolant pressure and temperature at the experimental channel inlet are determined by the results of measurements of the above parameters at the header inlet. The temperature and coolant flow rate are measured just in the channel outlet pipeline and coolant samples are taken to check the cladding integrity control.

The MIR reactor is also used for lifetime tests of full-size LEU IRT-3M lead test assemblies of different design. The key task of the lifetime tests is to proof the performance of the IRT-3M LTA up to the average  $^{235}\text{U}$  burnup of no less than 60%. At present, IRT-type FAs are used at many research reactors of Russian design: IR-8, IRT-MEPHI, IRT-T (Russia), LVR-15 (Czech) and VVR-SM (Uzbekistan) [11].

The IRT-3M LTA consists of tubular fuel rods of square section, and top and bottom end components. A peculiar feature of the IRT-3M LTA is square section of tubular fuel rods. So, to simulate hydro-dynamic test conditions the most close to the operational ones, an experimental channel was designed consisting of several parts with differently shaped through sections. At the active part level, there are displacers to shape a square section of the hydraulic path (FIG.20). Heat from the LTA is removed by the primary coolant going straightforwardly top-bottom [11].



FIG. 20. IRT-3M LTA (a) and experimental channel with LTA (b).

By now, tests have been completed for two 8-tube LEU IRT-3M LTAs with the uranium density in the fuel meat of about  $\sim 5 \text{ g/cm}^3$ . By the close of tests, the average  $^{235}\text{U}$  burnup made up 62% by volume, the maximal one made up 82% at a point. No cladding leakage was revealed during the tests. The experiments proved the LEU IRT-3M LTAs performance under operation at a thermal capacity of  $\sim 1300 \text{ kW}$  and maximal thermal flux of  $1.96 \text{ MW/m}^2$  that proves the use of this fuel in the pool-type reactors IR-8, IRT-MEPH, IRT-T [11].

#### 4. Conclusions

The techniques and experimental programs available at the MIR reactor allow getting experimental data on the performance and behavior of fuel rods under various operational conditions as well as on changes in the characteristics and properties of structural materials and FA components under irradiation. The data are used to improve the existing fuels and materials and to develop new ones, to check the compliance with the fuel licensing requirements, to improve calculation codes used to evaluate the fuel rods state and to predict radiation consequences in case of cladding leakage.

To further enlarge the MIR's experimental capabilities and develop promising areas of research, the following activities will be done:

- improvement of the techniques to control parameters and perform in-reactor measurements of fuel rods characteristics;
- reactor tests in justification of the improved and new types of VVER and PWR fuels under different designed conditions;
- use of a gas-cooled loop to examine core components and FA dummies of high-temperature gas-cooled reactors;
- reactor tests to improve and justify fuels for floating, low-power and ice-breaker reactors of new generation;
- design of a universal loop to simulate operational conditions of advanced power water-cooled reactors and upgrade loops under operation;
- upgrading of the MIR reactor and its equipment and extension of its lifetime, including replacement of Be blocks.

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