Power ramping and cycling testing of VVER fuel rods in the MIR reactor

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Abstract. Loop-type test reactor MIR was purposely designed to perform long-term lifetime tests of fuel assemblies, fuel rods and structural materials of various reactors. Since 1990s, the MIR reactor has been used to develop, implement and improve fuel rod power change test techniques, as well as to simulate abnormal operating conditions and power cycling. For these tests specific techniques, experimental rigs and in-pile gages have been developed and successfully applied.
During this period of time, about 120 VVER fuel rods have been tested under the power ramping and step-up modes (see *FIG*. below); about 20 VVER fuel rods have been tested under power cycling conditions [1…5].
The paper presents information about techniques applied at the MIR reactor as well as some tests and post-irradiation examinations results.

1. Introduction

A series of experiments has been conducted at the MIR reactor with simulating power ramp, step-up and cycling conditions to experimentally confirm the maximum inner pressure and stress limits for the water-cooled reactor fuel rod claddings both under abnormal operating conditions and power cycling. As a result of these experiments, linear heat rate (LHR) and cladding ultimate residual strain have been obtained at which fuel leakage occurs or does not occur, as well as gas release from the fuel composition and source data to calculate cladding stress.
The safety-critical criteria for the nuclear power plants (NPPs) operating VVER are identified by the following initial events [2]:

- uncontrolled withdrawal of control rods;
- unintended decrease of boron concentration in coolant;
- improper connection of the main circulation loop;
- human error when damping xenon variation.

Tested were refabricated or full-size VVER fuel rods and gadolinium fuel rods with different burnup from standard or modernized fuel assemblies (FAs) operated at NPPs.
In the first power ramping stage, the fuel rods were tested for at least 12 hours under thermal loads and parameters specific to the end of irradiation at a NPP. Then the power was increased at a rate of 5...200 W/(cm×min) depending on the simulated initial event, techniques applied in similar tests and requirements of the national licensing authorities. The linear heat rate was increased up to the level at which fuel leakage was expected to occur or fuel rods performance was confirmed. After that the fuel rods were tested for about 12... 17 hours at constant power. These experiments show that fuel rods leakage occurs according to the stress corrosion cracking mechanism (SCC) [2,6] that depends on the following parameters [7]:

- nuclear fuel burnup;

- LHR ratio after and before power increase (power increase amplitude);
- LHR increase rate;
- time interval under stress.

The amplitude, LHR increase rate, cycling mode, time of operation at reduced and increased power depend on the simulated process, power increase method and reactor design.

2. Fuel rod power change techniques

Different techniques are used at research reactors to change fuel rods power [2]:

- axial displacement of absorbing screens heightwise the core and tested fuel rods; for this purpose, a scram rod is introduced into the core [8];
- axial displacement of the fuel rods relatively the screen; in this case, the power increase field moves along the fuel rod [9];
- azimuthal displacement of the fuel rods or absorbing screens in the test rig (TR); in this case, there is no considerable impact on reactor operation [2, 10, 11];
- horizontal movement of the fuel rods along with the channel relatively the core with no considerable impact on reactor operation [12];
- change in the pressure (amount) of absorbing gas (He-3) in the tubes serving as absorbing screens [13].

Each technique has its pros and cons.
For instance, when using gas absorber the tubes can probably burst and gas (He-3) can escape from the core with positive reactivity insertion. Therefore, when there is gas absorber the maximum power change amplitude (ratio of the maximum power to power before the ramp) is limited by the maximum admissible gas pressure in the tubes. It is clear that contribution of excitation to reactivity depends as well on the reactor type (power, core dimensions and design, experimental channel position in the core and reflector, etc.). The similar procedure was applied at the MR reactor (Moscow, Russia) and HBWR (Halden, Norway).

3. Fuel rod power change techniques applied at the MIR reactor

Loop-type test reactor MIR was purposely designed to perform long-term lifetime tests of fuel assemblies, fuel rods and structural materials of various reactors. MIR is a pool-type reactor with beryllium moderator and reflector [4,5,10]. Figure 1 shows reactor core (1 m high) arrangement. Since 1990s, the MIR reactor has been used to develop, implement and improve fuel rod power change test techniques. For such experiments, specific techniques have been developed and successfully applied as well as experimental rigs and in-pile gages.
During this period of time, about 120 VVER fuel rods have been tested under the power ramping and step-up modes (see *FIG. 2.*); about 20 VVER fuel rods have been tested under the power cycling conditions [1…5]. In 2011 the professionals from VNIINM and RIAR developed a new program [2] for a series of ramp tests with Russian-made fuel rods and gadolinium fuel rods of VVER and PWR including thinned cladding fuel rods and increased diameter fuel pellets with no hole. Since 2012 five experiments (NG1…NG5) have been performed in which three full-size Gd fuel rods and eleven full-size fuel rods (FSFR) have been tested. The next experiments are being prepared.

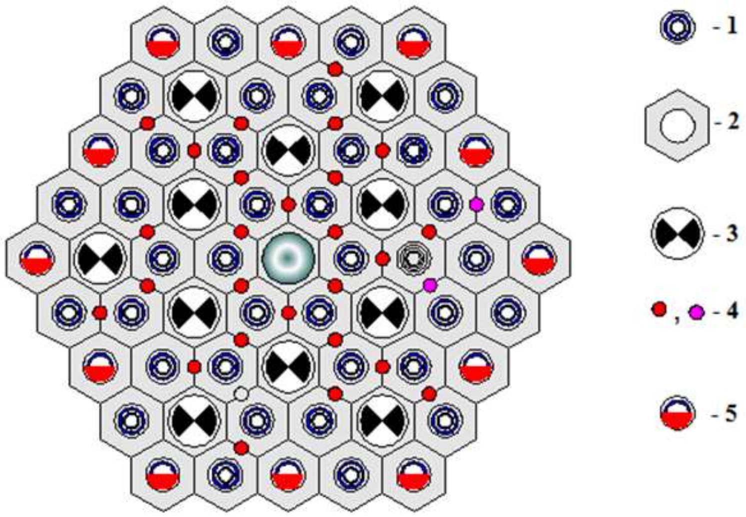

FIG. 1. MIR core:
1 – standard FAs; 2 – beryllium blocks; 3 – loop channels; 4 – control rods; 5 – combined FA–absorber.

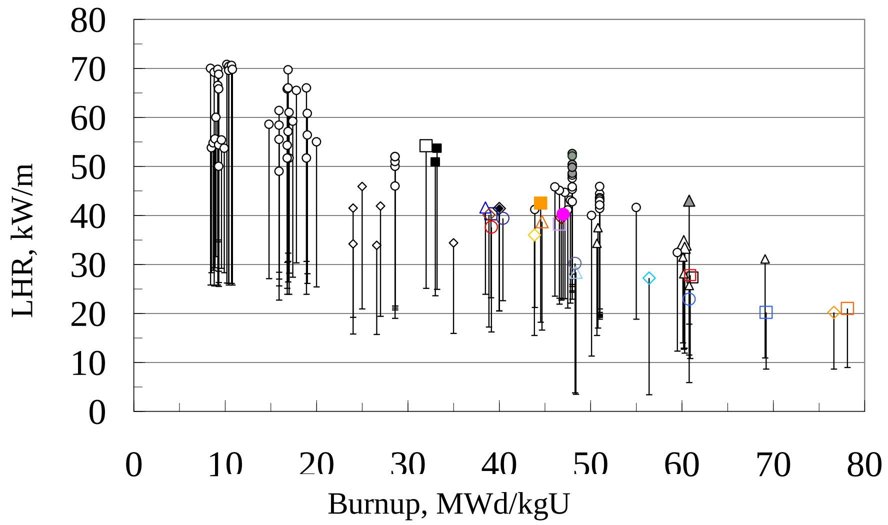

FIG. 2. LHR increment for fuel rods under the power ramping and step-up modes.

Black open symbols in the Figure indicate fuel rods tested in 1990…2000, and colored open ones show fuel rods that have been tested since 2012 (these are fuel rods with an increased amount of fuel due to an increased diameter of fuel pellets having no central hole, Gd fuel rods and fuel rods of more than 70 MWd/kgU). Solid black and colored symbols indicate fuel rods that have lost their integrity in these experiments, and solid grey symbols show fuel rods with radial nonthrough cracks in the cladding.
The MIR reactor physical and design parameters enable power ramping tests without using extra absorber in the test rig [10]. In such tests the reactor was brought to power providing for the required initial conditions in the loop channel. The control rods closest to the loop channel with the test rig were in the low position. After all parameters had become stable at this power level, there was a power ramp with the necessary amplitude. At the first stage the closest control rods were removed in parallel with compensation of inserted positive reactivity by immersing the control rods in other core areas. In this very moment the TR power was kept

constant, and the reactor power was decreasing. Then a power ramp was done by increasing the total reactor power within the set time interval. After that, the TR power was kept constant by removing periodically the control rods.

Following this algorithm, the power ramp was used within 5…30 minutes, and the ratio of power after ramp to power before ramp (amplitude) made up 1.5…3 relative units (see *FIG. 2*).

Test irradiation rigs (IR) with moving hafnium absorbing screens (plates) were specifically designed for fuel power cycling tests at the MIR reactor [11]. Their layouts are presented in the figure. Several such tests have been conducted when the power change amplitude made up ~1.5 relative units (see *FIG. 4*) and the power change time was 10…20 minutes.

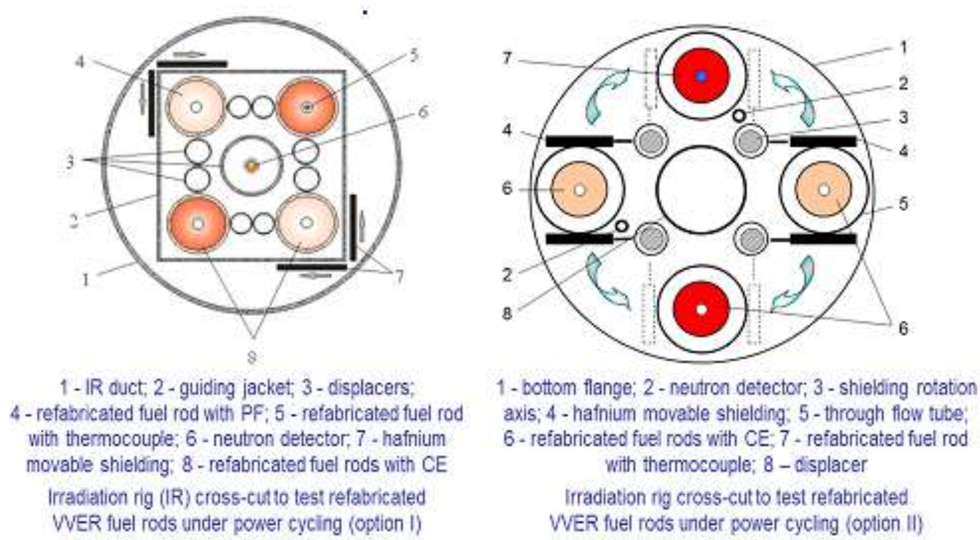

FIG. 3. Irradiation rigs to test fuel rods under power cycling.

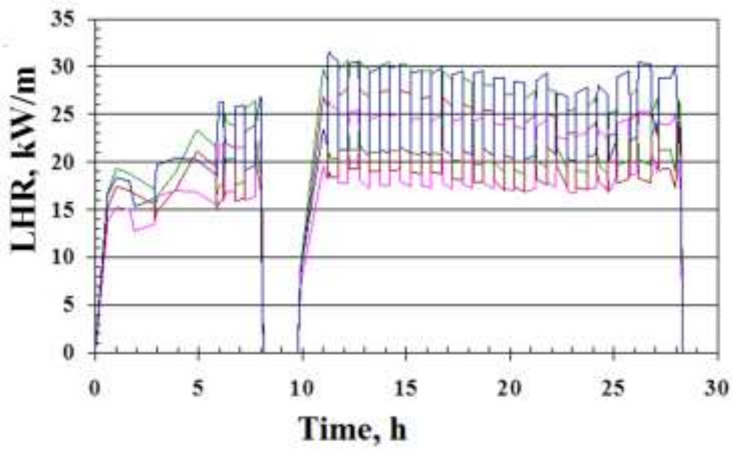

FIG. 4. LHR of different fuel rods during power cycling tests.

A rig was designed [2] to test both shortened (refabricated, experimental) and full-size VVER-1000 fuel rods (~3.9 m long) under rapid power change conditions. The fuel rod test part is 1…1.5 m high taking into account the MIR core height (1m) and extrapolated neutron escape areas above and under the core. Figure 5 shows the TR schematic representation. The TR enables changing the linear heat rate about twice during several and more seconds at constant reactor power. The power is changed due to screen rotation about the rig axis or fuel rods basket. The screen is rotated with the use of the driven shaft located in the space above the reactor. A similar rig provides for tests under power cycling conditions.

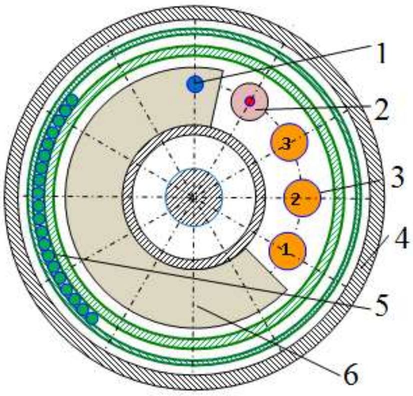

FIG. 5. TR to irradiate fuel rods under power ramp conditions (top view): 1 –neutron detector (self-powered detector – SPD); 2 – thermometric fuel rod; 3 – full-size fuel rods; 4 – loop channel; 5 – absorbing screen; 6 – central tube with a lateral displacer.

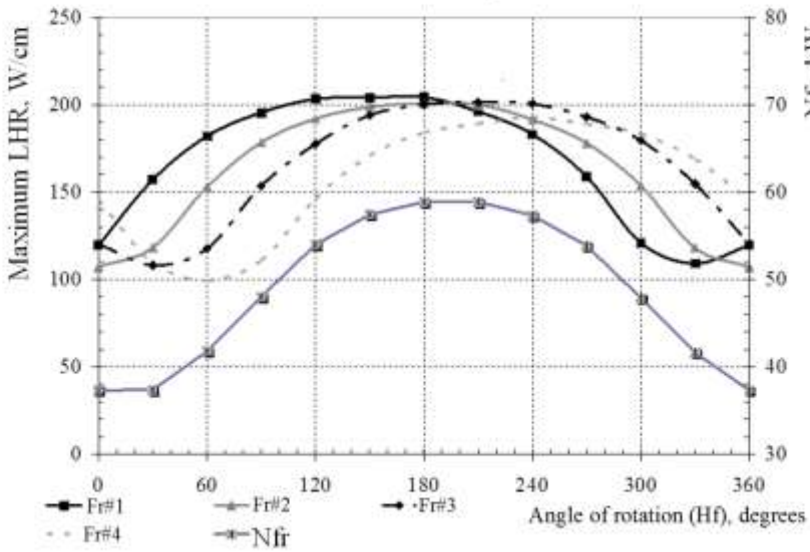

FIG. 6. Max LHR and total power of fuel rods vs. absorbing screen angle of rotation (counterclockwise) at specific reactor power.

A similar rig was tried out (see *FIG. 7*). The screens were rotated at constant power of FAs surrounding the experimental rig. The neutron detector and fuel temperature gage readings were recorded. The neutron detector readings were compared with the calculation data for the neutron flux density. Acceptable reproducibility was observed, the deviation not exceeding 3%. After check the reactor was shut down.

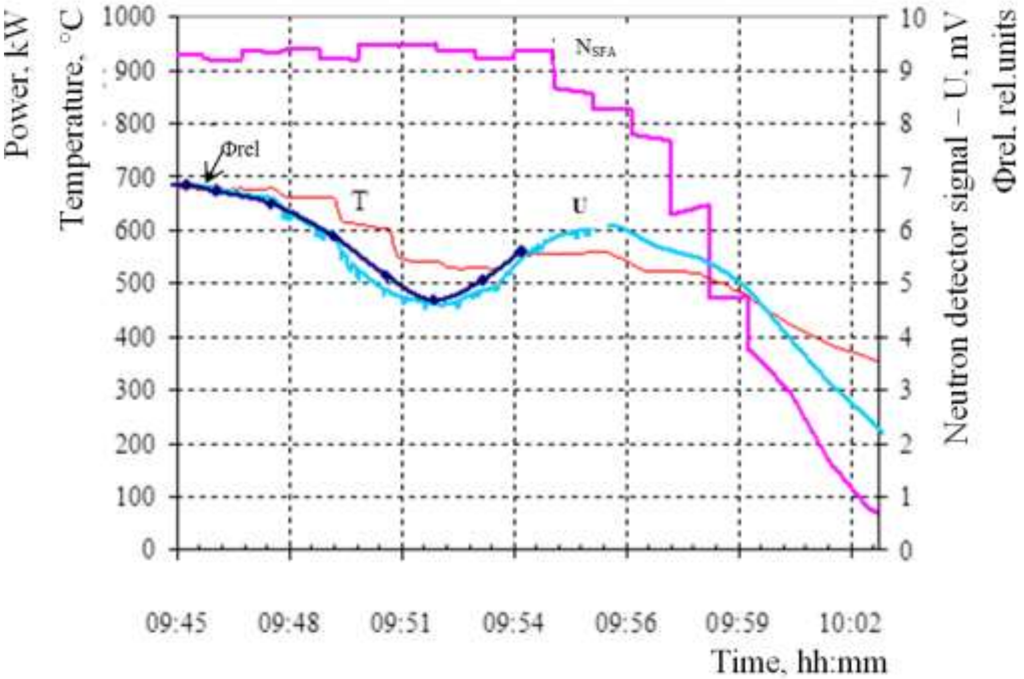

FIG. 7. TR trial operation.

Power ramping and cycling tests will be performed with the use of a similar rig.
Having four rows of absorber rods five-time power increase tests can be carried out.
When using such rigs, the introduced reactivity changes did not exceed the admissible limits, since absorber was not removed from the core – it was moved only in the local area.
For fast ramping combined techniques can be used, such as power change by displacement of the control rods closest to the loop channel or reactor power change and further power increase with the use of specific TRs. Such technique enables changing rapidly the amplitude within the specified range.
To maintain high stress in the fuel-to-cladding system after power ramp with the use of specifically designed TR, the reactor power can be gradually increased with unchanged position of the control rods closest to the loop channel. Thus, the fuel rod is kept at constant stress for a long time interval having corrosive reagents (I, H) under the cladding.

4. Measured parameters of the fuel rods and irradiation rig during tests

During in-pile tests cladding thermomechanical loading and stress relaxation can be monitored using, for example, fuel rod or fuel column elongation transducers; fission gas release under the cladding can be checked using pressure gages. The temperature in the fuel meat center is measured by chromel-alumel or tungsten-rhenium thermocouples depending on the measured temperature range. The key parameters of the gages used in the fuel rods and irradiation rigs at the MIR reactor are shown in Table 1 [2, 14]. Figure 8 presents schematic design of some in-pile gages developed at JSC “SSC RIAR”.

TABLE I: Types and parameters of the gages used in the fuel rods and TR.

Gage purpose	Design	Measurement range	Error	Dimensions, mm	
				diameter	length
Coolant and cladding temperature	chromel-alumel cable thermocouple	up to 1100 ºC	0.75%	0.5…3.0	-
Fuel temperature	chromel-alumel cable thermal probe	up to 1100 ºC	0.75%	1.0-1.5	-
Fuel temperature	tungsten-rhenium thermal probe (5/20, protection tube Mo+BeO)	up to 2300 ºC	~ 1.5%	1.2-2.0	-
Gas pressure in the plenum	bellows + DT[1]	0-20 MPa	See note[2]	16	80
Fuel cladding elongation	DT	0-10 mm	See note[2]	16	80
Fuel column elongation	DT	0-10 mm	See note[2]	16	80
Neutron flux density (relative units)	SPD (rhodium, hafnium)	10^{11}-10^{15} n/cm^2·s	~ 1%	2-4	50-100
1) DT – differential transformer; 2) determined by specific scaling.					

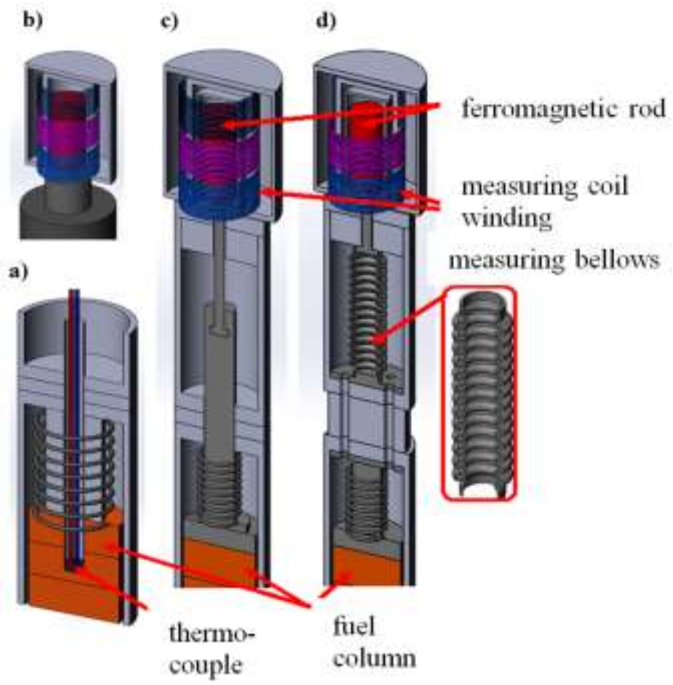

FIG. 8. In-pile gages to measure fuel rod temperature (a), fuel cladding elongation (b), fuel column elongation (c), gas pressure under the cladding (d).

To determine axial strain of the test part (1.5 m) for the full-size fuel rod (total length ~ 3.9 m) in the experiment, elongation differential transducers have been developed and tested [1]. To compensate the fuel cladding elongation when changing coolant temperature in the test rig zirconium tubes are installed. Figure 9 shows schematic representation of the rig with such elongation transducers and data related to measurement of LHR and fuel elongation in one of the experiments.

During tests at the MIR reactor fuel cladding integrity is checked using standard systems of monitoring the rate of gamma radiation dose from loop facility pipelines (see *FIG. 10*). The peak in the diagram shows that there is loss of fuel cladding integrity.

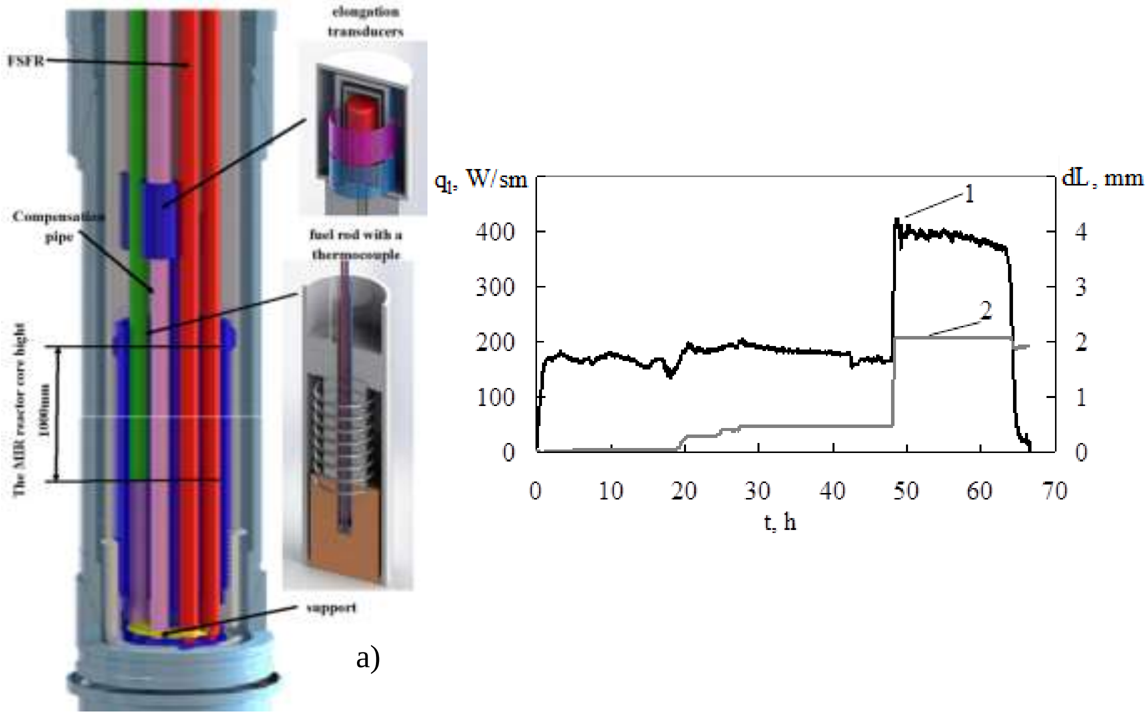

FIG. 9. Rig with elongation transducers intended for the full-size fuel rods test part (a) and results of measuring LHR (1) and fuel rod elongation (2) in one of the experiments.

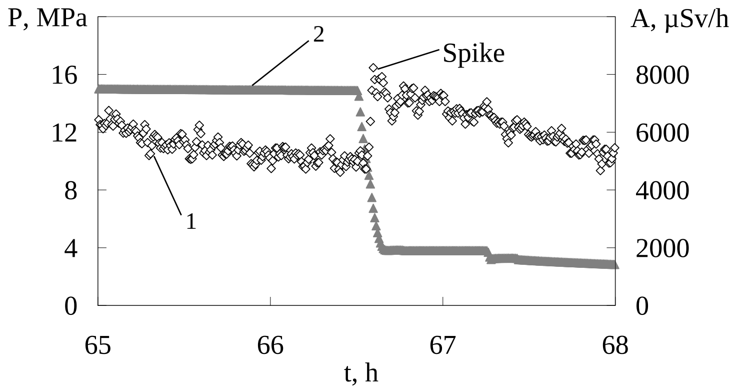

FIG. 10. Change in the radiation dose rate (1) from pipelines when reducing pressure (2) after reactor shutdown in one of the experiments.

5. Some results of post-irradiation examinations in the hot cells

After testing and decay heat decrease the fuel rods are sent for examinations to the RIAR's hot cells. In these examinations the pre-test calculation and test measurement results are confirmed.

Table 2 contains changed fuel rod length measured by transducers in the experiments and obtained through direct measurement after experiments (residual elongation).

TABLE 2: Fuel rod elongation.

Full-size fuel rod #	**Measured elongation at the maximum thermal neutron flux density/ after reactor shutdown, mm**	**Residual elongation direct measurement data, mm**
4	2.07 / 1.91(see *FIG. 10*)	1.91 ± 0.14
6	1.23 / 1.06	1.06 ± 0.14
8	3.21 / 2.55	2.47 ± 0.11
309	3.18 / 1.26	1.17 ± 0.11

When examining the fuel rod condition after power ramping, several fuel rods were detected having through cracks in the cladding resulted from stress corrosion cracking (see *FIG. 11*). The remaining fuel rods kept their integrity.

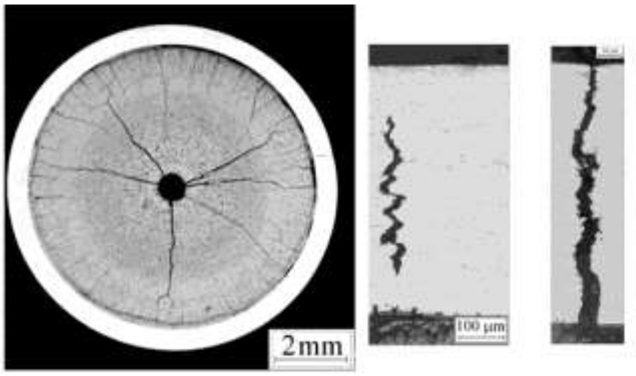

FIG. 11. Structure of a leaky fuel rod.

During the examinations gamma scanning (see *FIG. 12*) was carried out, fuel rod cladding length and diameter were measured (see *FIG. 13*), and gas composition in the plenum was examined. Change in the fuel rod diameter in different areas heightwise being specific to the relevant LHR increment in ramp and burnup, strain was observed in the areas 500…700 mm long. Figure 14 shows claddings residual strain for the VVER fuel rods containing fuel pellets with no central hole at the relevant LHR for different burnup fuel rods.

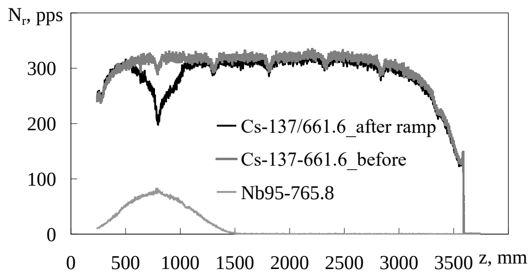

FIG. 12. Gamma intensity heightwise distribution of 137Cs before tests, 137Cs and 95Nb after tests for a leaky full-size fuel rod.

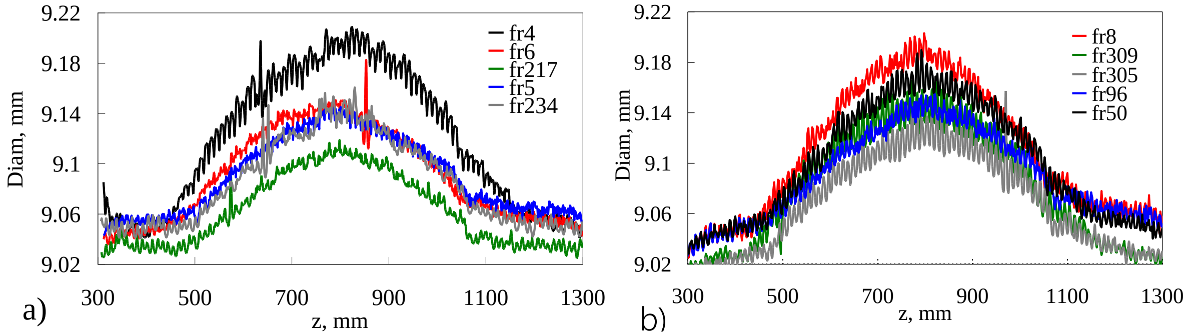

FIG. 13. Some diameter measurement results for fuel rods under power ramping (the results of NG4 (a) and NG5 (b) experiments).

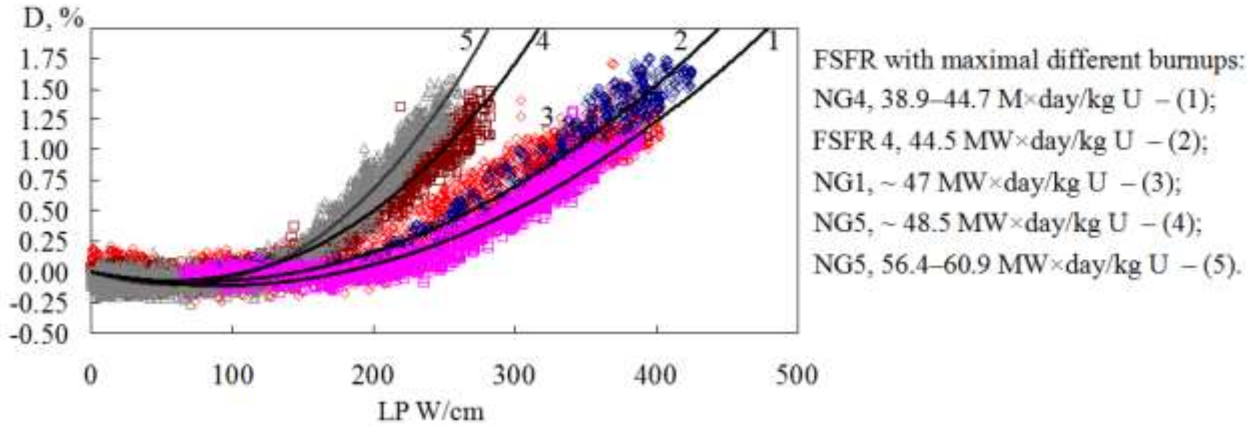

FIG. 14. Ratio of cladding residual strain to LHR for different burnup fuel rods [1].

6. Conclusions

About 120 VVER fuel rods have been tested under the power ramping and step-up modes; about 20 VVER fuel rods have been tested under the power cycling conditions.
Techniques and procedures have been developed for power ramping, step-up and cycling tests.
In-pile gages have been designed to measure fuel rods parameters. The in-pile measurement results have been confirmed by hot cell examinations.
Further work is being carried out to improve the experimental base of the MIR reactor and hot cells to perform power change tests of fuel rods.
The tests enabled obtaining the data related to VVER fuel rod behavior under power change resulted from abnormal operation. Good reliability of the tested fuel rods was demonstrated.
The gained experience provides for further tests under power change modes with other fuel rod types and different coolants (water, gas); ampoule tests are also possible.

7. References

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