TECHNIQUES AND RESULTS OF EXAMINATION OF FISSION PRODUCT RELEASE FROM VVER FUEL RODS WITH ARTIFICIAL DEFECTS AND A BURNUP OF ~60 MWd/kgU AT THE MIR LOOP FACILITY

A.V. Burukin, A.V. Goryachev, S.A. Ilyenko, A.L. Izhutov,
V.V. Konyashov, V.Yu. Shishin, V.N. Shulimov
(JSC "SSC RIAR", Dimitrovgrad)
L.M. Luzanova, V.N. Miglo (RRC "Kurchatov Institute", Moscow)

INTRODUCTION

When high-burnup fuel cycles are implemented at VVER NPP, it is necessary to justify radiation safety of NPP taking into account the effect of burnup level on radioactive fission product release from defective fuel rods into the primary coolant of the operating reactor. One of the methods of such justification is performance of tests at the loop facilities involving irradiation of the fuel rods with artificial through defects on their claddings.

Complex of equipment and several techniques were developed for one of the loop facilities (the PV-1 loop facility) of the MIR reactor (JSC "SSC RIAR") to perform the above tests under conditions simulating typical design-basis operation scenarios of a power reactor [1].

During the first test, which was conducted at the PV-1 loop facility, release of gaseous and main dose-forming volatile radioactive fission products from an experimental defective re-fabricated fuel rod was measured; this fuel rod was prepared using a full-size VVER fuel rod operated in the NPP power reactor. The re-fabricated fuel rod was irradiated under quasi-steady-state conditions which correspond to operating parameters of the VVER-1000 fuel rods. The experimental re-fabricated fuel rod had an average fuel burnup of ~ 60 MWd/kgU; the artificial defect was a through hole 1 mm in diameter applied on the cladding in the active part with the maximum heat release (the maximum linear heat generation rate (LHGR) during testing made up ~ 15 kW/m).

That was a methodical test which aimed at testing of new equipment and techniques in order to assess their appropriateness for the further test run.

1. EQUIPMENT AND TECHNIQUES

Preparatory stages before testing involved the following activities:

- analysis of technical characteristics of the MIR loop facilities in order to select the most appropriate one taking into account predicted coolant activity and implementation of different design-basis scenarios for fuel operation providing compliance with safety requirements;
- analysis of characteristics of the modern spectrometric equipment in order to assess whether measurements of predicted coolant activities are sufficient taking into account acceptable measurement intervals which allow tracing of activity change kinetics during the test;
- design, manufacture and installation of equipment for irradiation test at the MIR loop facility; the equipment should provide application of required examination techniques;
- development of equipment for application of calibrated artificial through defects on claddings of re-fabricated fuel rods.

During preparation and performance of the test the following activities were carried out:

- neutronic, thermophysical and hydraulic calculations to determine irradiation conditions simulating the design-basis scenario as well as safety requirements for the test;
- preparation (calibration and testing) of instruments for radiation, neutronic, thermophysical, hydraulic and water-chemistry measurements according to the test program;
- irradiation tests of the experimental re-fabricated defective fuel rod in accordance with the test program requirements for irradiation conditions and scope of measurements of test parameters (using a

special measurement and control system (MCS)) and radiation coolant parameters (using an on-line spectrometer installed in immediate proximity to the primary pipeline of the loop facility as well as a laboratory spectrometer for measurement of gamma spectra of water and gas coolant samples);

- analysis of measurement results to determine applied irradiation parameters and changes in the coolant activity during the whole test;
- PIE of the experimental re-fabricated fuel rod in the hot cell.

1.1 PV-1 loop facility of the MIR reactor, equipment used for radiation measurements and sampling

Layout PV-1 loop facility selected for irradiation testing is shown in Figure 1.

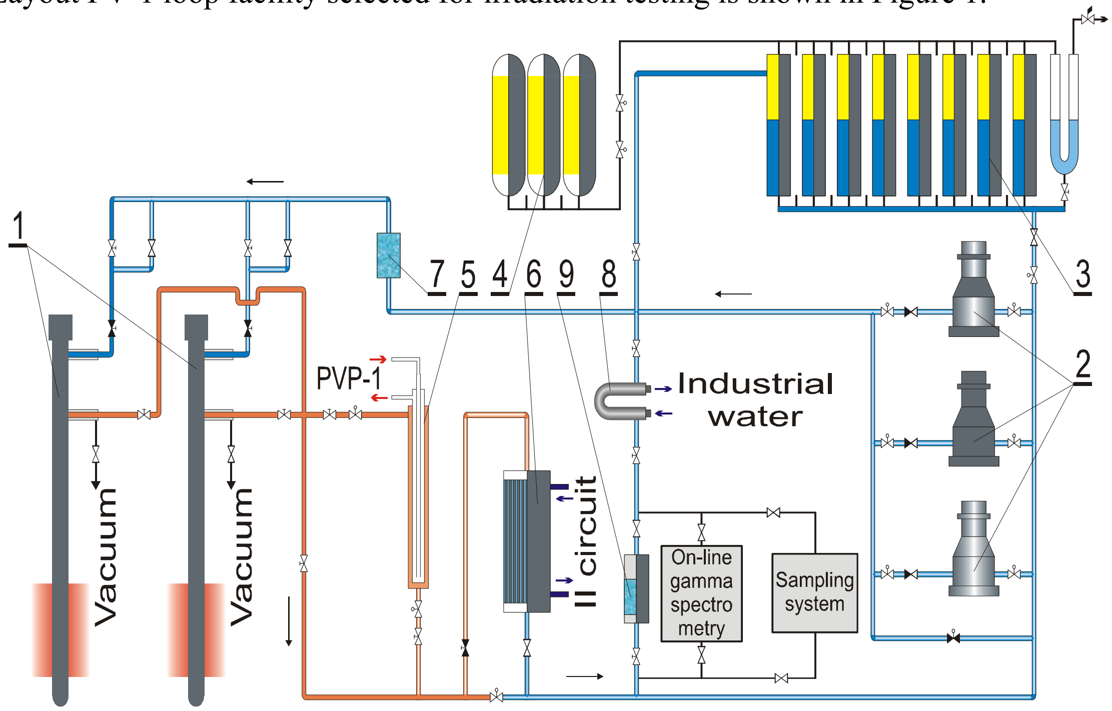


Figure 1. Layout of PV-1 loop facility of the MIR reactor:

1 – loop channels;
2 – circulating pumps;
3 – pressure compensators;
4 – gas tanks;
5 – heat exchanger of PVP-1 loop facility;
6 – main heat exchanger;
7 – mechanical filter;
8 – purification heat exchanger;
9 – ion-exchange filter

Primary circuit of this loop facility has a small volume; it makes radiation measurements easier and requires a minimum upgrade of the equipment in order to obtain necessary experimental data in addition to the available standard systems for primary coolant gamma activity control (gamma activity control system – GAC system) and integrity control with delayed neutrons (cladding integrity control system – CIC system).

The PV-1 systems for sampling and radiation measurements, which were specially developed and installed for this test, are shown in Figure 2 and Figure 3. These systems made it possible to integrate an on-line spectrometer as well as perform sampling into a tight vessel (without loss of dissolved gas fraction) and measure gamma spectra of the selected samples.

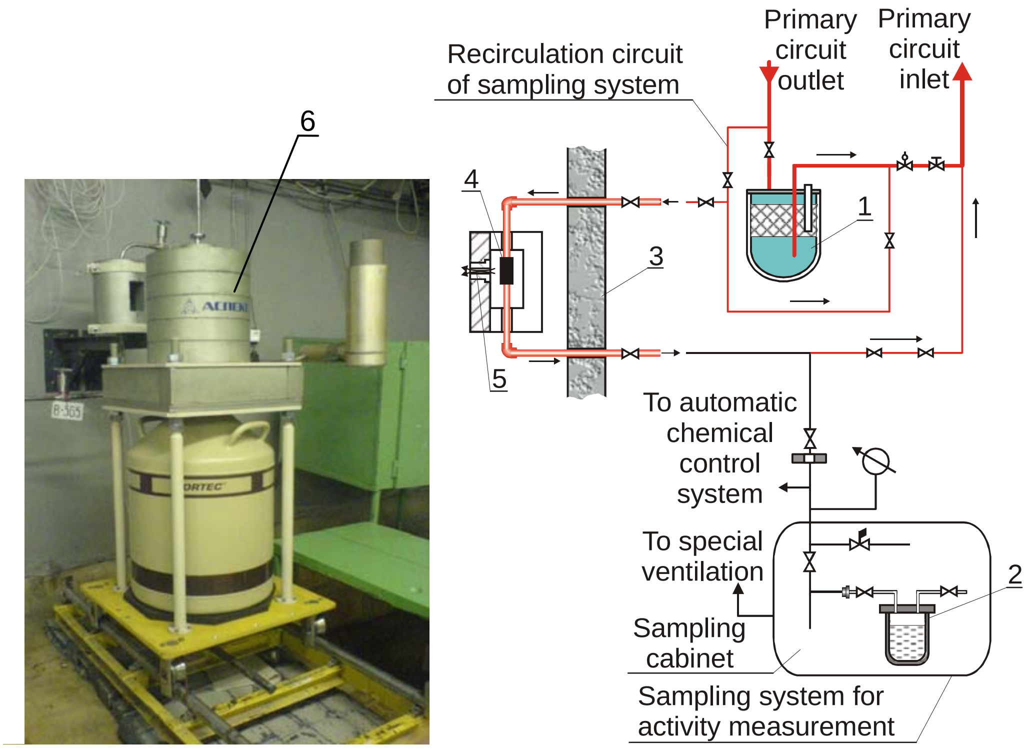


Figure 2. Equipping of the PV-1 loop facility primary circuit with auxiliary radiation measuring instruments:
1 – ion-exchange filter;
2 – portable sample container;
3 – chamber wall of PV-1 loop facility;
4 – flow-through measuring tank;
5 – collimator;
6 – HPGe detector

Gamma spectra measurements of the coolant samples and coolant immediately in the loop pipeline are performed using the ORTEC spectrometers equipped with the HPGe detectors. Primary processing of measuring data is performed using a control PC or an experimentator PC by a standard DSPec Plus hardware with MAESTRO or GAMMA-VISION software. This software provides for appropriate nuclear constant libraries and calibration of registration efficiency in the real geometry of "source-detector" which were determined using set of gamma ray spectrometer reference sources. Sufficiency of coolant gamma spectrum data obtained during the on-line measurements in the measuring tank is provided by the gamma ray collimation and radiation shielding around the detector and measuring tank. Sufficient gamma spectum data of the coolant samples are provided by shadow shielding that reduces background gamma radiation intensity of ~1 MeV near the detectors (due to adjacent pipelines, surrounding process equipment of the reactor and loop facility) by ~2 orders of magnitude. Radiation collimators with different slit width (for the on-line system) and variation in distance between the source and the detector (sample spectrum measurements) provide an optimal loading of the spectrometer path within a wide range of recorded radiation intensity.

The on-line gamma spectrometry measurement area for the coolant samples is shown in Figure 4.

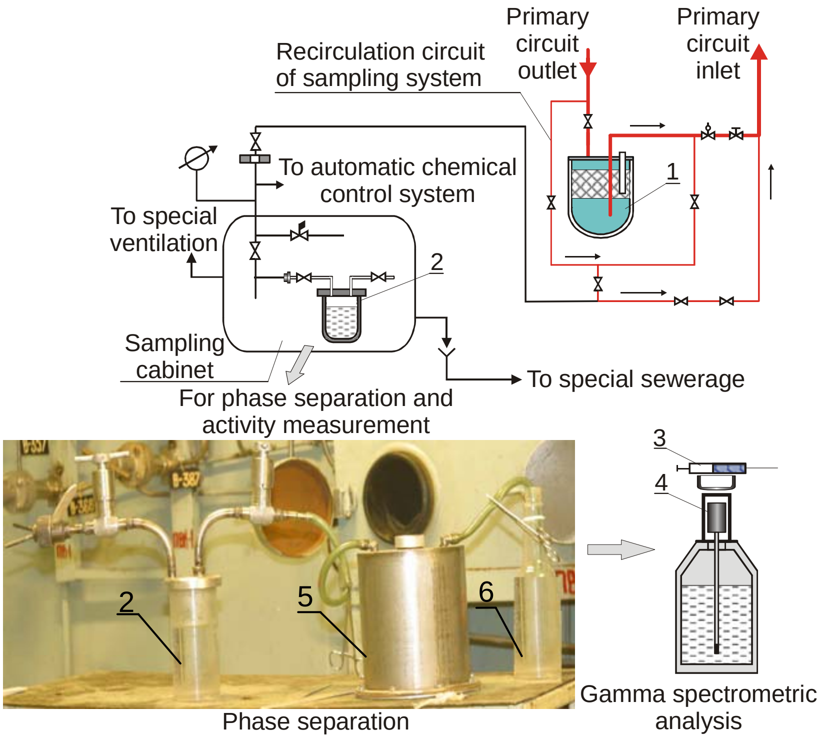


Figure 3. Flowsheet of coolant sampling system and sample handling operations:

1 – ion-exchange filter;
2 – portable sample container;
3 – sample syringe;
4 – HPGe detector;
5 – buffer gas vessel;
6 – water vessel

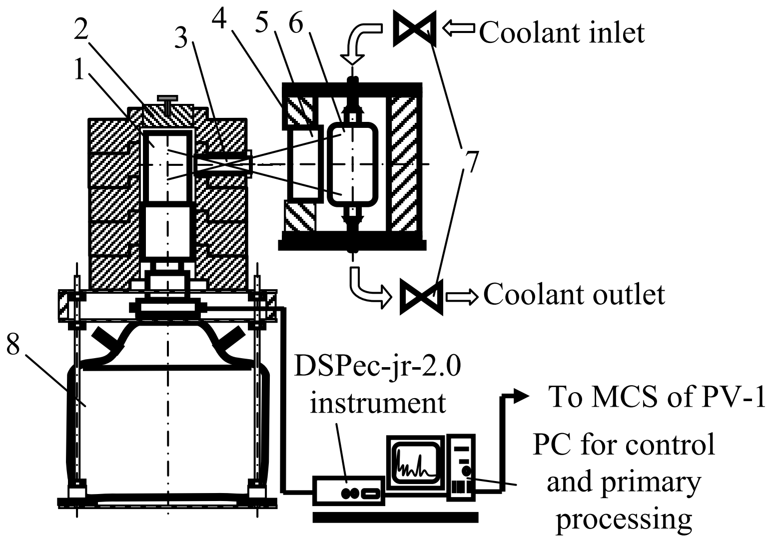


Figure 4. Gamma spectrometry measurement area for the coolant samples at PV-1 loop facility:

1 – gamma ray detector;
2 - radiation (background) shielding of detector;
3 - radiation collimator;
4 - shadow shielding of measuring tank;
5 – hole in the shielding of measuring tank;
6 - measuring tank;
7 - cutoff valves of measuring tank

1.2 Procedure of fission product sampling and activity measurement

Complementary procedure and system for measurement of specific activity of radionuclides can be applied simultaneously during irradiation testing:

- on-line measurement system which provides detailed information about short-lived high-activity fission products with a short half-life period (5-10 min);
- sampling technique which provides quantitative information about long-lived low-activity fission products.

Efficiency of these techniques depends on the testing task. The on-line measurement is more advanced for short-term tests involved rapid changes in the coolant activity; the sampling technique is of greater importance for long-run tests involved slow changes in the coolant activity. Therefore, an optimal measurement procedure can differ, especially due to a great but not always important information content obtained during the on-line measurements.

Because of a wide range of tasks, the first test included the following activities:

- determination of the minimum exposure time required for a statistically sufficient spectrum buildup at realizable values of specific coolant activity (the on-line system). The exposure time varied from 5 to 30 min; measurements were performed sequentially during the whole test;
- repeatability test for activity values of the main water and gas phase emitters in the coolant samples from measurement to measurement as well as increase of statistically sufficiency of measurements by the sampling technique with due regard to half-life periods of recorded nuclides. Measurements are performed immediately after coolant phase separation in ~ 1.5 hours and in 1, 3 and 7 days after sampling. The sampling was performed every day at the same time.

1.3 Irradiation rig and experimental fuel rods

The program of the first test provided a comprehensive evaluation of the developed equipment and test techniques for a re-fabricated fuel rod irradiated under steady-state design-basis conditions; the re-fabricated fuel rod was made from a fuel rod operated in the VVER-1000 reactor up to a burnup of ~ 60 MWd/kgU. For this purpose, an irradiation rig was developed and manufactured; it provided thermophysical parameters and a coolant velocity diagram near the experimental fuel rods which are similar to the standard ones in a full-size VVER-1000 fuel assembly.

The experimental fuel assembly, which contained seven fuel rods and was similar to the fragment of the VVER-1000 fuel assembly at the cross-section, formed a part of the irradiation rig. The experimental re-fabricated fuel rod was inserted into the central cell and surrounded by six fuel rods with unirradiated fuel.

The active fuel length was ~ 1000 mm; it corresponds to the MIR core height. Design of the experimental fuel rod was selected from considerations of its similarity to the full-size VVER-1000 fuel rod as to the active part and gas plenum volume. The artificial defect was applied on the cladding in form of a through hole 1 mm in diameter; it was applied in the active part with the maximum heat release at a height of ~ 240 mm from the fuel rod end plug.

2. RESULTS OF IRRADIATION TESTS

Duration of re-fabricated fuel rod irradiation for the first test depended on a standard cycle of the MIR reactor and made up ~22 days; it allowed achieving equilibrium activity levels of most of the recorded radioactive fission products in the PV-1 primary coolant. All requirements established for irradiation conditions and data content were met.

Figures 5 and Figure 6 show change of the main measured and calculated thermophysical parameters of the experimental re-fabricated fuel rod during the first test. LHGR and maximum fuel temperature values ($T_{f\ max}$) were in the ranges typical for standard operation of fuel rod at a constant LHGR in the VVER-1000 reactor.

Figures 7 and Figure 8 show change in activity of all main nuclides from the inert radioactive gas and halogen series in the PV-1 primary coolant (on-line measurements).

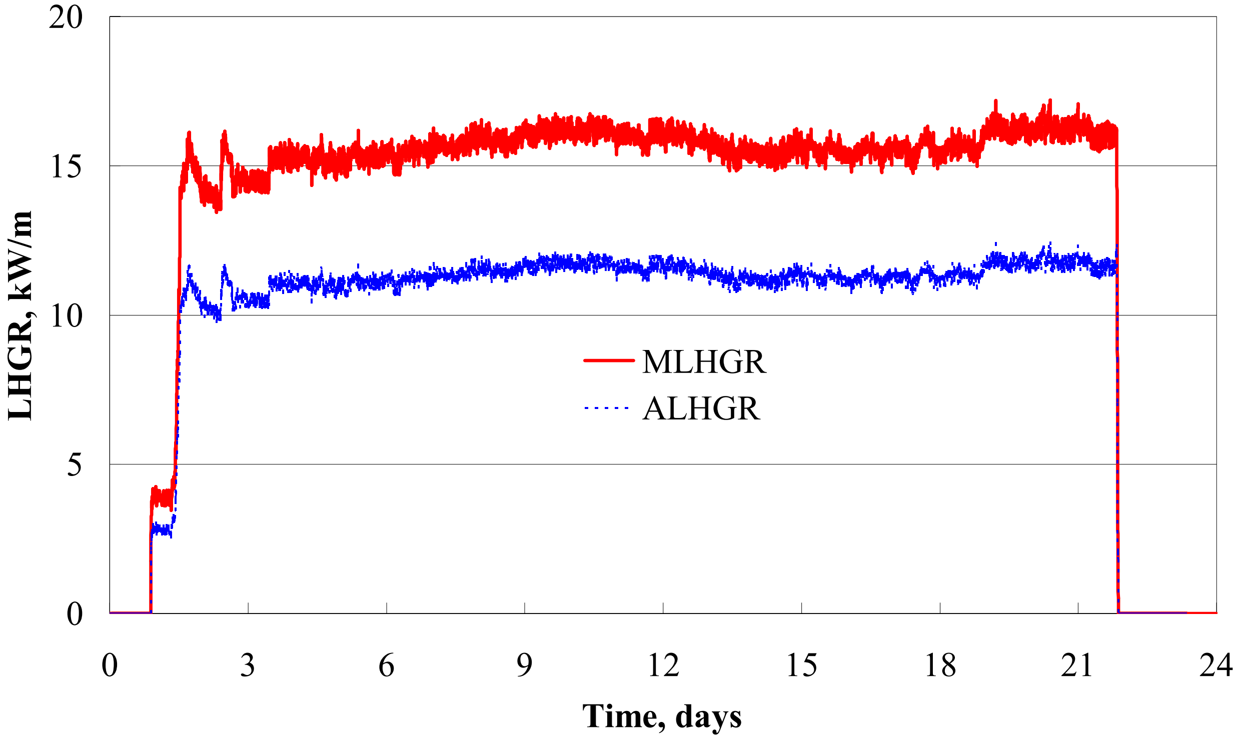


Figure 5. Change in maximum (MLHGR) and average (ALHGR) LHGR of the fuel rod with artificial defect during testing

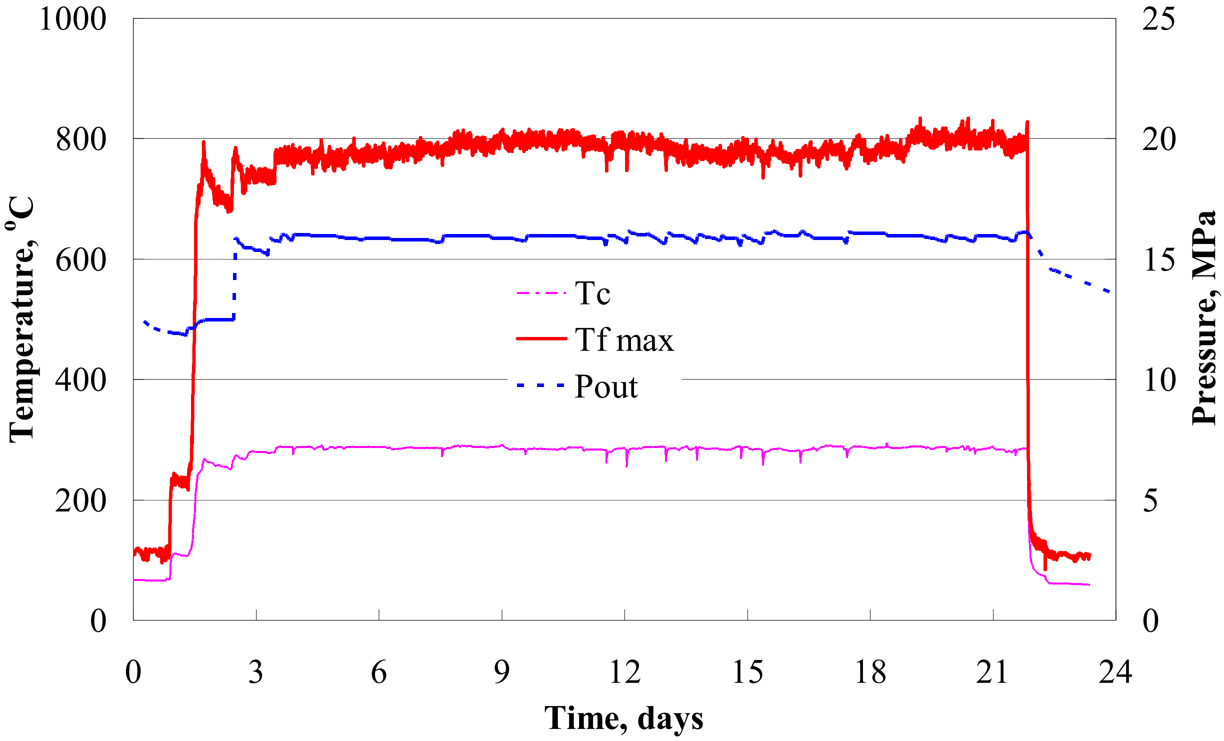


Figure 6. Change in maximum fuel temperature, temperature (T_c) and pressure (P_{out}) of PV-1 primary coolant during testing

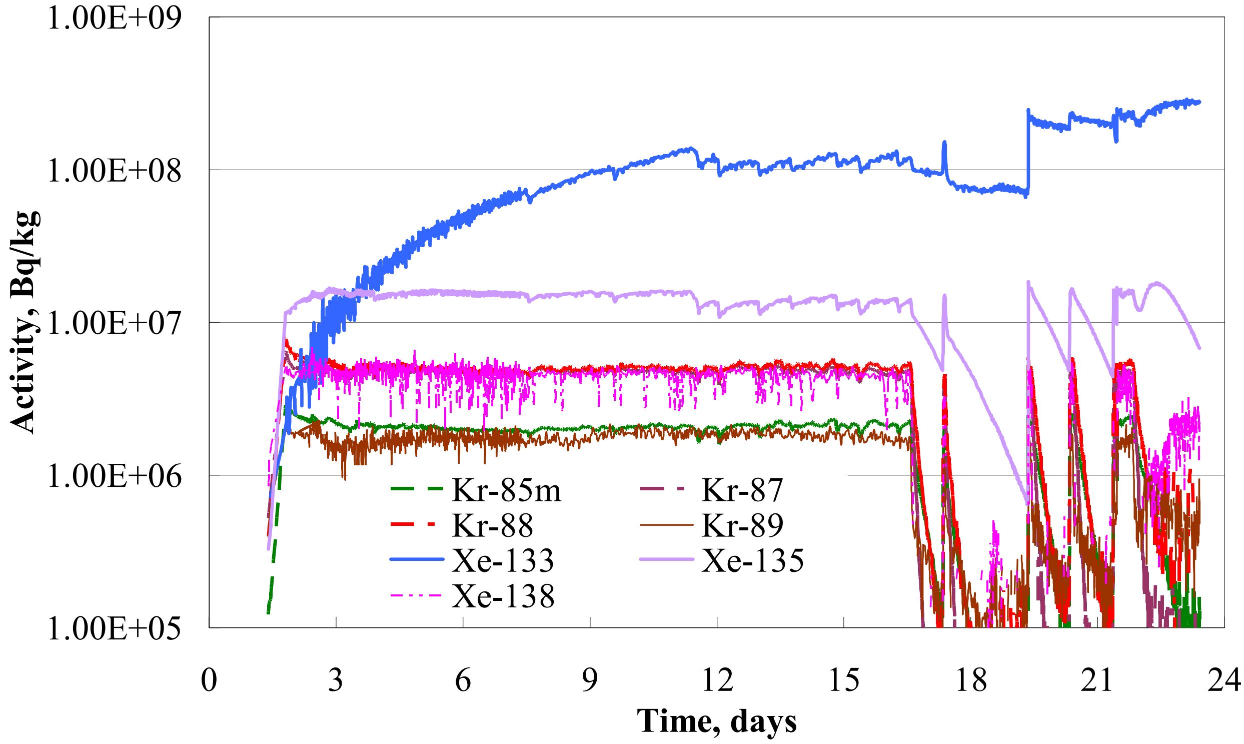


Figure 7. Change in specific activity of inert radioactive gases in the coolant during testing

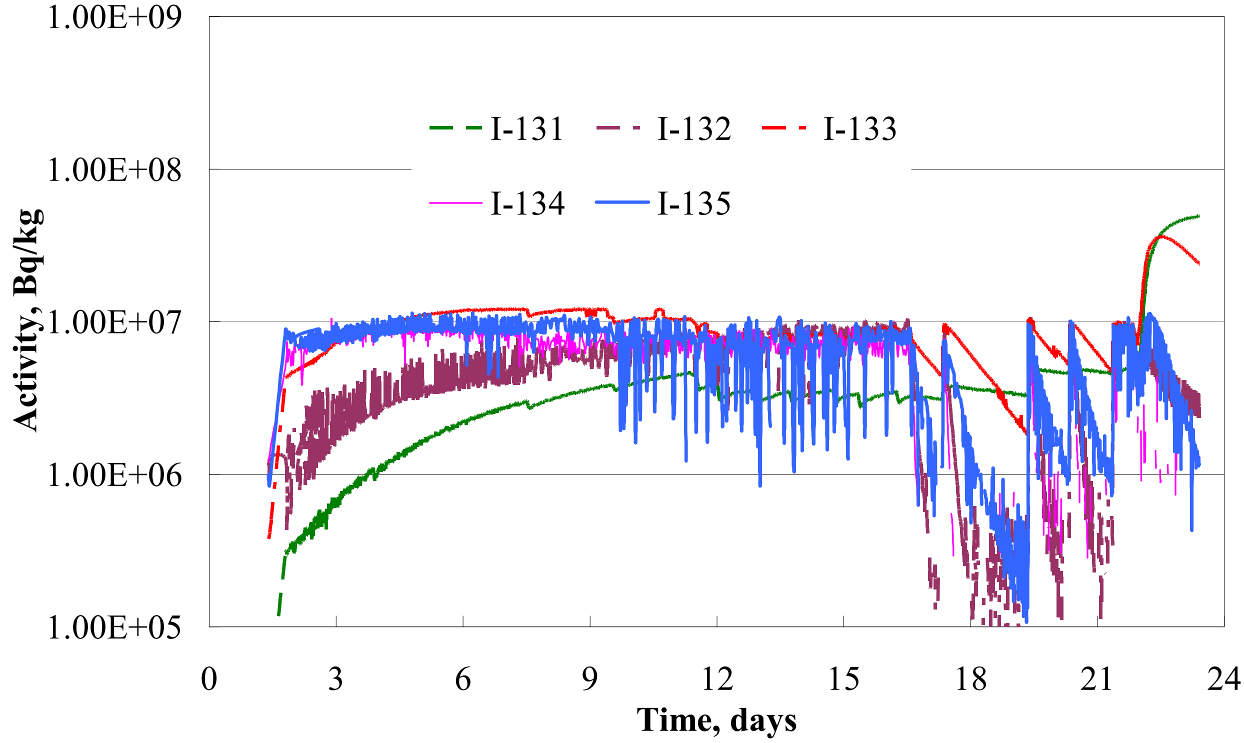


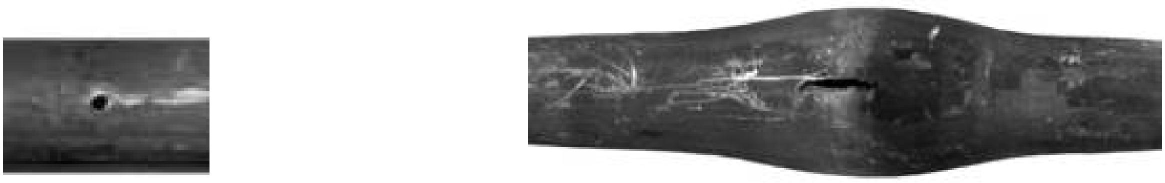
Figure 8. Change in specific activity of iodine radionuclides in the coolant during testing

3. MAIN RESULTS OF POST-IRRADIATION EXAMINATION

The primary inspection of the experimental fuel rod showed a cladding swelling in the gas plenum area. There was a longitudinal through crack 7mm in length and ~ 1mm in width in the middle of strained area; no change in appearance of the artificial cladding defect was observed.

Then the **non-destructive examination** of the experimental re-fabricated fuel rod was performed: detailed inspection and photographing of the fuel rod, gamma scanning, profilometry measurements and X-ray radiography. Test performed by the above techniques did not reveal any noticeable changes of fuel rod state except for the strained area with a crack in the gas plenum area. Cladding surface of the active part, the artificial defect and the areas near this defect didn't undergo any noticeable changes. Figure 9 shows photographs of the artificial defect and a defect generated during testing.

Results of gamma scanning didn't indicate redistribution of the radioactive fission products and fuel wash-out through the artificial defect (see Figure 10). Results of profilometry measurements performed in four planes with an azimuth step of 45^{o} showed that the fuel rod diameter didn't change over the full active fuel length. The X-ray radiography results showed that there were no gaps between the fuel pellets; the pellets were not damaged.



a) b)

Figure 9. Areas of the fuel rod cladding with the artificial defect after testing: (a) artificial defect, (b) through defect in the gas plenum area

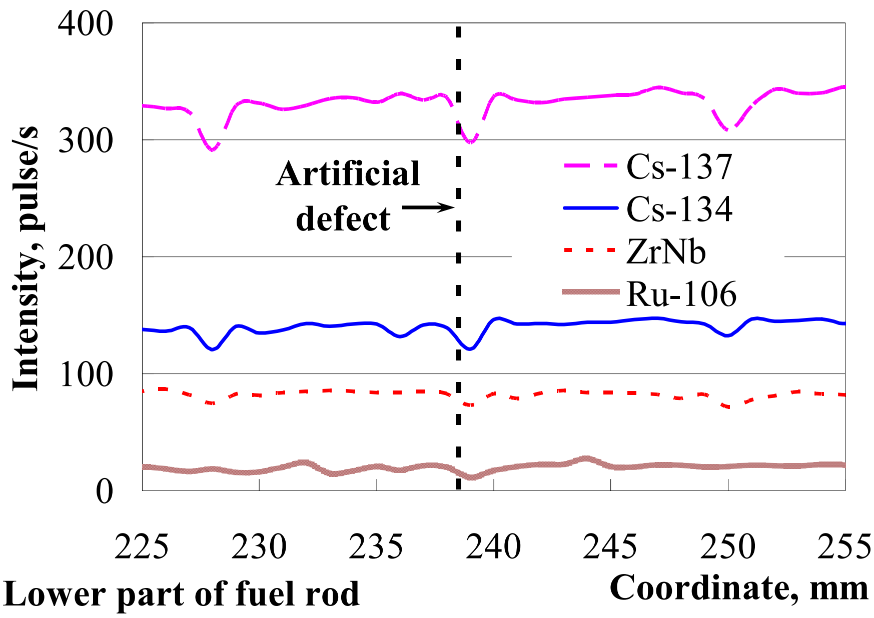


Figure 10. Results of gamma scanning of the experimental fuel rods in the area of the artificial defect after testing

Destructive examination involved metallographic examination and electron probe microanalysis at different cross-sections over the length of the re-fabricated fuel rod. Among the results, which were obtained by the destructive examination techniques, the parameters of the experimental re-fabricated fuel rod should be noted which have effect on fission product release into the primary coolant of the loop facility during testing. These are macro- and microstructure of fuel, width of fuel-cladding gap, diameter of central hole in the fuel column.

Examination of the **fuel structure** showed that:

- the fuel is fragmented into 4-6 parts by radial cracks mainly;
- diameter of central hole in the fuel column is 2.3–2.4 mm over the full length of the experimental fuel rod;
- the rim-layer is ~ 60 μm in width;
- there is no fuel-cladding gap over the full length of the experimental fuel rod;
- the fuel-cladding interaction layer is ~ 10 μm in width.

The above parameters correspond to a typical state of the VVER fuel with a burnup of ~ 60 MWd/kgU.

Fuel microstructure for two cross-sections of the experimental fuel rod – at the defect location and in the distance from it – is shown in Figure 11. Average grain size for both cross-sections makes up ~10 μm; it is characteristic for an average grain size in the fuel before testing.

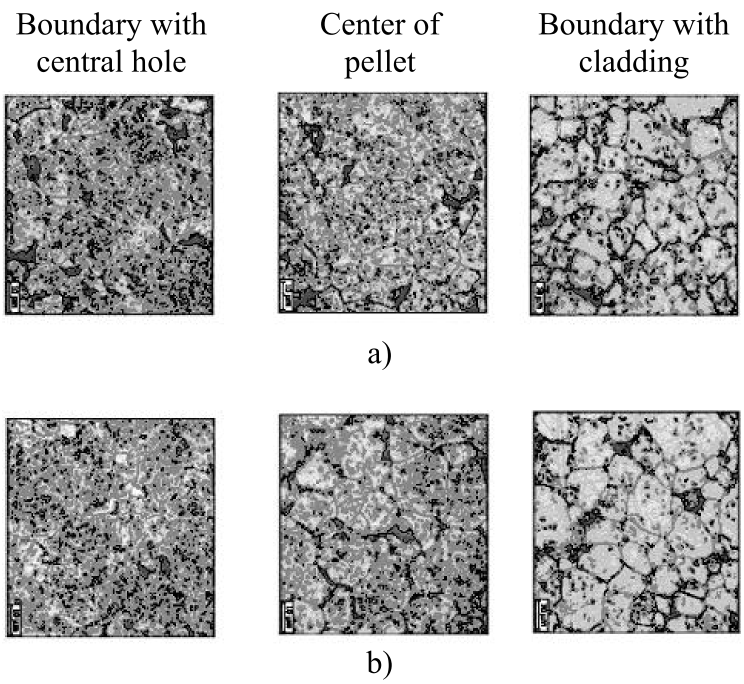


Figure 11. Fuel microstructure of experimental fuel rod with artificial defect after testing at different cross-sections: (a) cross-section ~500 mm above end plug, (b) cross-section near the artificial defect

4. RESULTS AND DISCUSSIONS

Results of measurement of radioactive fission product activity in the primary coolant of the PV-1 loop facility obtained during the first test were analyzed using PIE results. This analysis based on the final objective of the MIR test run, i.e. study of effect of fuel burnup on radioactive fission product release from fuel rod under the cladding as well as from fuel rod into the coolant in case of its failure.

Summarized results of the above study allow us to make up the following conclusions:

4.1 PIE showed that the fuel-cladding gap of the experimental fuel rod was bridged. A close thermal contact between fuel boundary area and cladding caused the rim-layer temperature to be within limits specified for normal operation of an original full-size VVER fuel rod. This observed effect and absence of fuel surface oxidation, which is probably caused by a close contact between fuel and cladding, suggest that only gaseous and volatile fission products, which were accumulated in the central hole of the fuel column, released into the primary coolant of the PV-1 loop facility through the cladding defects (artificial defect and defect formed in the gas plenum area during testing).

4.2 Defect in the gas plenum area as a result of cladding strain and rupture was formed in the earliest test stage due to thermal expansion of the coolant if the fuel rod reached a specified LHGR. This conclusion was made up on the basis of:

- readings of GAC and CIC systems (see Figure 12);
- PIE results of the experimental fuel rod;
- simulated processes inside the defective fuel rod containing coolant before its irradiation (during preparatory work the irradiation rig was installed into the loop channel and the loop facility was tested for leakage conditions at pressure increased up to 12-13 MPa before bringing reactor to power. The fuel rod interior was filled with the primary circuit water through the artificial defect).

The above conclusion is validated by change in activity of iodine radionuclides throughout the test. This change was recorded by both on-line system and sampling system. In particular, no marked increase in content of any radionuclide under control in the PV-1 primary coolant was observed during irradiation of the experimental fuel rod although such defect in the gas plenum area should cause this increase. A pronounced increase (more than 10 times) of ^{131}I activity and, at the same time, decrease of short-lived ^{134}I activity (see Figure 13) recorded after reactor shut-down is a typical spike effect rather than an additional seal failure of the fuel rod which could cause increase of activity of the iodine short-lived radionuclides but it was not observed.

4.3 Significant size of the defect, which was formed in the gas plenum area, reduced appreciably (or practically eliminated) inhibition of volatile and gaseous fission product release from the free volume of fuel rod into the coolant.

If we compare size and location of the artificial defect and the through defect formed on the cladding during testing, we can suppose that iodine released from the central hole of the fuel column completely and without interruption through the macrodefect in the gas plenum area.

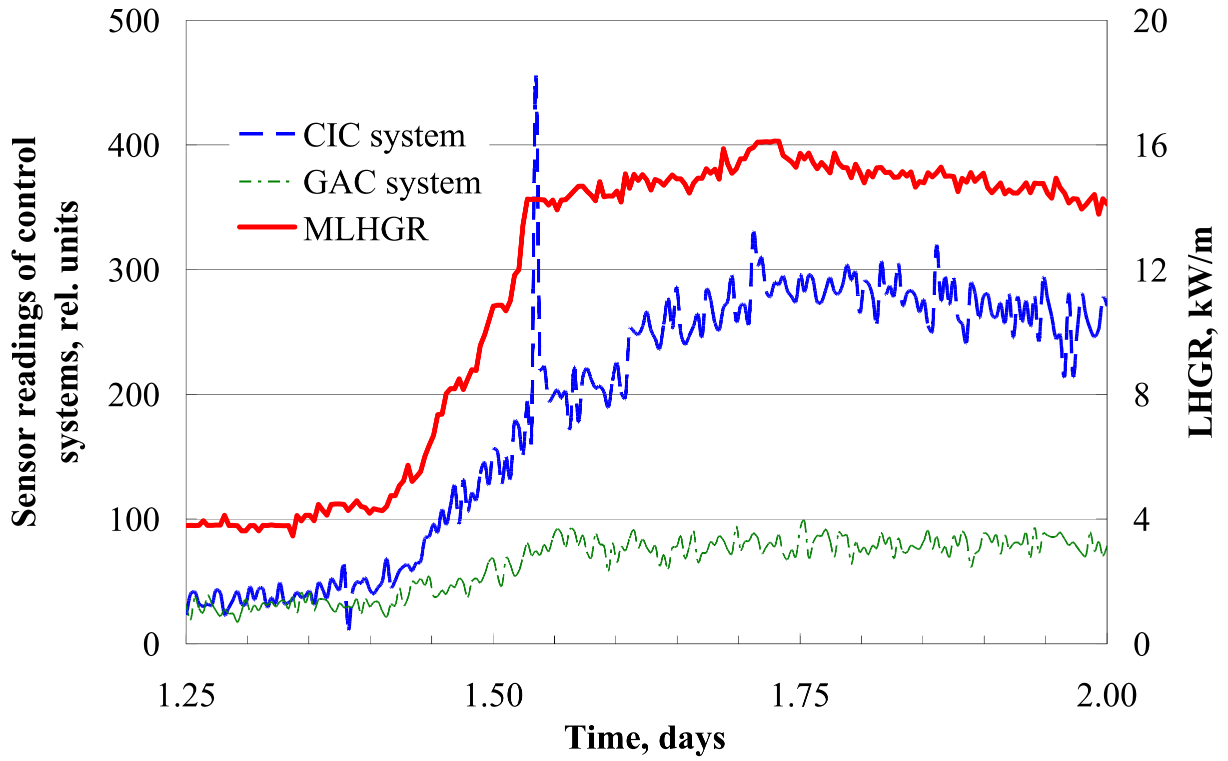
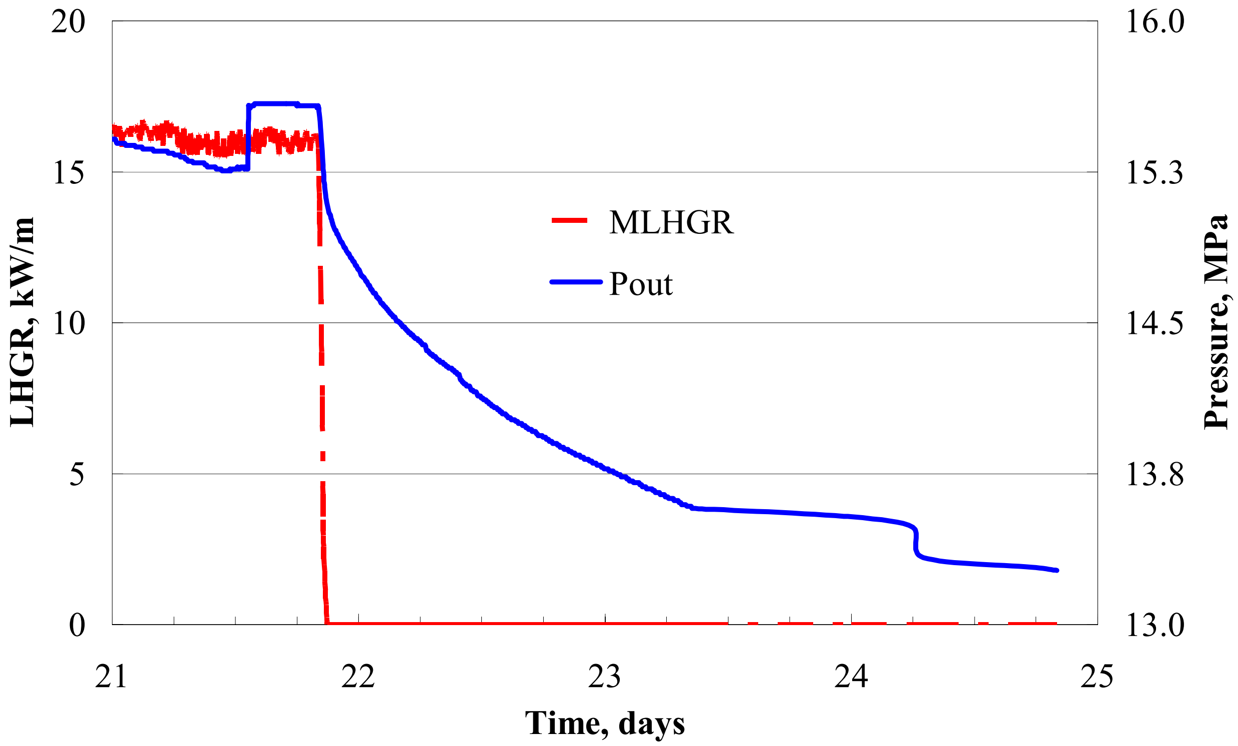
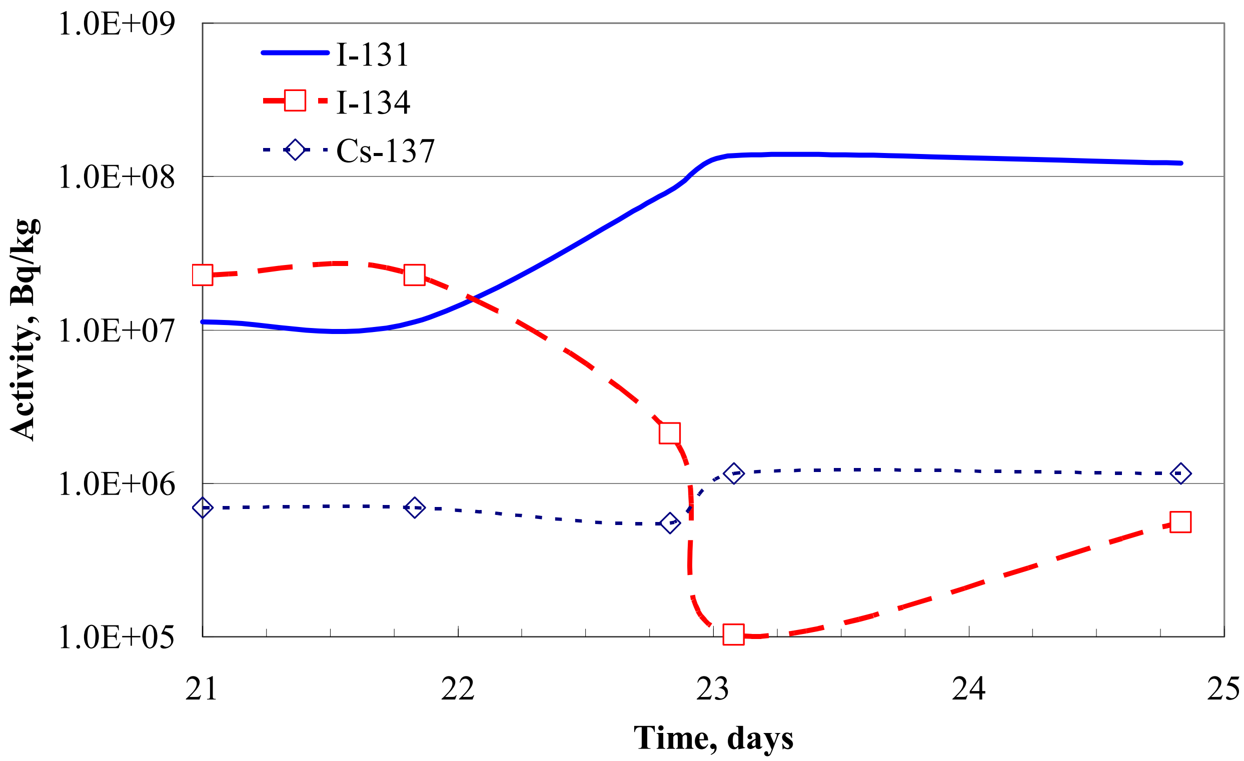


Figure 12. Change in MLHGR of fuel rod and activity of PV-1 primary coolant in accordance with sensor readings in the course of bringing reactor to power



a)



b)

Figure 13. (a) Change in MLHGR of fuel rod and P_{out} before/after test completion in the course of MIR reactor shut-down and decrease of pressure in the PV-1 primary circuit, (b) change in ^{131}I, ^{134}I and ^{137}Cs activity in the coolant under the above conditions

CONCLUSIONS

1. Complex of equipment and several techniques for examination of radioactive fission product release from defective fuel rods were developed, prepared and tested at the PV-1 loop facility of the MIR reactor.

2. During the first test, which was conducted at the PV-1 loop facility and aimed at testing of developed equipment and techniques, measurement of radioactive fission product release from an experimental re-fabricated fuel rod with a burnup of ~60 MWd/kgU and an artificial defect was performed under design-basis steady-state operating conditions of the VVER-1000 reactor.

3. PIE of all main parameters of the experimental defective fuel rod did not reveal any state peculiarities which could be caused by the artificial defect, i.e. fuel and cladding characteristics in the defect area did not differ from the initial ones (before testing) as well as their characteristics in areas distant from the defect; they are typical for fuel rods with a similar irradiation history in the VVER NPP.

The gap in the experimental fuel rod was bridged due to close contact between fuel and cladding at increased fuel burnup; it can appreciable reduce release of radioactive fission products into the PV-1 primary coolant. This suggestion and quantitative characteristics of effect of gap bridging in a high-burnup fuel rod on radioactive fission product release should be investigated during the next tests performed at the PV-1 loop facility.

4. Values of radioactive fission product release measured during the first test at the PV-1 loop facility in the MIR reactor will be used for development of an empirical engineering model in order to take into account high burnup effects and their impact on fission product release from fuel and defective fuel rods.

ACKNOWLEDGMENTS

The authors would like to express profound gratitude to the specialists of JSC TVEL, especially to Mr. A.I. Sharikov, for his great assistance and active participation in management of the tests in the MIR reactor.

REFERENCES

1. S.A. Ilyenko, A.L. Izhutov, V.N. Shulimov et al, “Technique and some results of examination of fission product release from the VVER fuel rod with an artificial defect and burnup of 60 MWd/kgU in the MIR water loop”, Scientific and technical conference of JSC TVEL “New generation nuclear fuel for NPP. Development results, operation experience and development directions” (STC-2008), November 19-21, 2008, Moscow, Russia.