# The Main Results of Investigation of Modified Dispersion LEU U-Mo Fuel Tested in the MIR Reactor

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**Abstract.** The report offers the results of the tests and of post-irradiation investigations of low-enriched U-Mo fuel tested in the MIR reactor under the RERTR program. The results of post-irradiation investigations of mini fuel elements with modified dispersion fuel containing 5%, 13% of silicon in the matrix and protective layers on fuel particles in the form of ZrN, tested up to the mean values of U-235 burn-up ~ 60% and ~ 84%, have been drawn. Data proving the reduction of the process of interaction between U-Mo alloy and the aluminum matrix containing silicon additives is afforded. It is demonstrated that the interaction with the matrix does not happen until 84% of burn-up when protective ZrN cladding is available on U-Mo particles. The impact of irradiation conditions on the interaction layer growth is discovered.

## 1. INTRODUCTION

The facilities from INR Piteşti allow the testing, manipulation and examination of nuclear fuel and irradiated materials. The most important facilities are the TRIGA SSR research and material test Reactor and the Post-Irradiation Examination Laboratory (PIEL).

The purpose of this work is to determine by post-irradiation examination, the behaviour of CANDU fuel, irradiated in the 14 MW TRIGA reactor. The results of post-irradiation examination are:

- Visual inspection of the cladding;
- Profilometry (diameter, bending, ovalization) and length measuring;
- Determination of axial and radial distribution of the fission products activity by gamma scanning and tomography;

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- Microstructural characterization by metallographic and ceramographic analyses;
- Mechanical properties determination;
- Fracture surface analysis by scanning electron microscopy.

The irradiation of a fuel element can lead to defects in the cladding. This is due mainly to a combination between a strain quite high and a low ductility of the cladding material. In CANDU reactors, the fuel elements are subjected to power ramps severe enough when reloaded during the functioning of the reactor.

The CANDU reactors from Cernavodă Nuclear Power Plant (NPP) are using as nuclear fuel bundles of 37 elements each, assembled by some edge grids. This bundle has a length of 495 mm, a diameter of 103 mm and weight of 24 kg. The CANDU fuel element contains cylindrical pellets of $UO_2$ syntherized, placed into a Zircaloy-4 tube (also known as sheath or cladding), closed at both edges with endcaps. It has a length of 492 mm and a diameter of 13.08 mm.

In order to check and improve the quality of the Romanian CANDU fuel, power ramp tests on experimental fuel elements were performed in our TRIGA SSR reactor. The irradiated fuel elements were further subjected to examination in the PIEL laboratory.

During the irradiation, the fuel elements suffer dimensional and structural changes, and also modifications of the cladding surface aspect, as result of corrosion and mechanical processes. This can lead to defects and even the integrity of the fuel element can be affected.

The performance of the nuclear fuel is determined by the following elements:

- Status of cladding surface and the effects produced by corrosion;
- Cladding integrity;
- Dimensional modifications;
- Distribution of fission products in the fuel column;
- Pressure and volume of the fission gas;
- Structural modifications of the fuel and cladding;
- Cladding oxidation and hydration;
- Isotopic composition of the fuel;
- Mechanical properties of the cladding.

## 2. CANDU FUEL CHARACTERIZATION

### 2.1 The aspect of the cladding surface

After irradiation, the fuel rod was kept in the reactor pool for three months, for cooling. The fuel rod was then transferred to the INR hot cells where it was subjected to detailed examinations.

An image of the fuel element is given in Fig. 2.1. It was obtained using a periscope, coupled with an OLYMPUS digital camera. The aspect of the cladding surface indicates a normal behaviour of the fuel element.

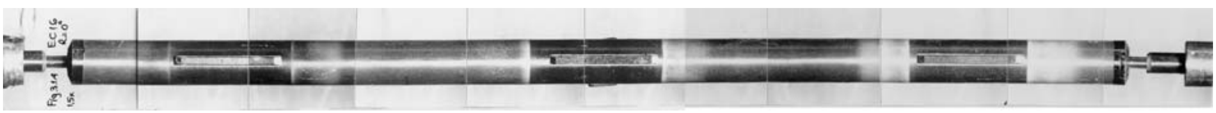


FIG. 2.1. Fuel element CANDU tested in the power ramp.

## 2.2 Profilometry

The diametrical profile, diametrical increasing, ovality and the arrow of fuel element were determined. In Fig. 2.2 is presented the average diameter profile of the fuel element. The average diameter is 13.149 mm. The average diametrical increasing is 0.087 mm (0.67 %), with respect to the diameter before irradiation.

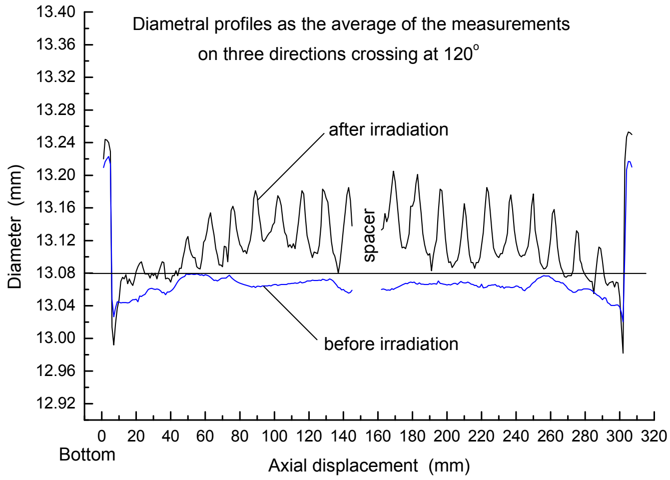



FIG. 2.2 Average diameter profile after irradiation.

Ovality profiles of the fuel element for two different positions on the vertical axis, Z = 97 mm and Z = 172 mm, are presented in Fig. 2.3. The graphic representation was made based on the measurements performed at these positions on three directions (0°, 120° and 240°). The profiles of bending are presented in Fig. 2.4.

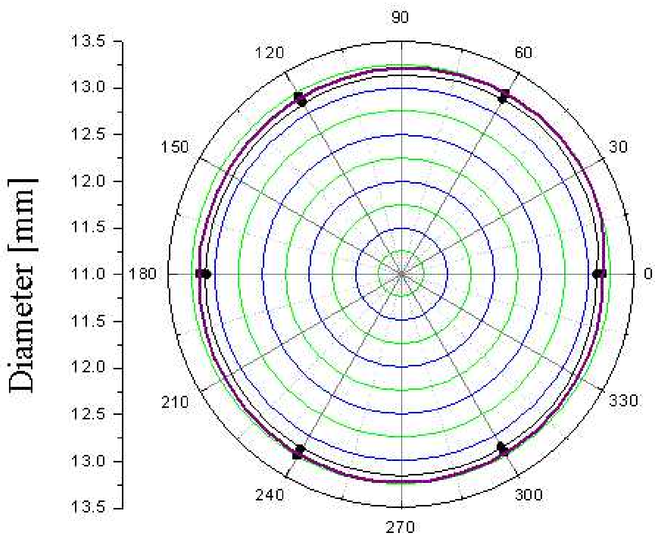



*FIG. 2.3. Ovality profiles for Z = 97 mm and Z = 172 mm.*

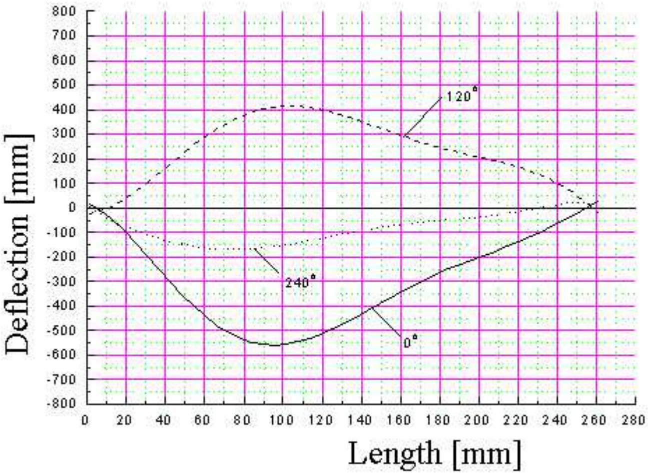



*FIG. 2.4.: Profiles of bending after irradiation.*

## 2.3 *Gamma scanning and tomography*

The gamma scanning equipment consists of a vertical fuel rod positioning machine equipped with SLO-SYN step-by-step motors, a collimator, in the hot cell shielding wall, a PGT intrinsic Ge detector and a multi channel analyzer.

For axial gamma scanning, the slit of the collimator was horizontal, having an aperture of 0.5 mm. The gamma acquisition along the fuel rod was performed at regular intervals of 0.5 mm; the acquisition time per step was 200 s. Fig. 2.5a shows the fuel rod axial gross gamma activity profile. A prominent depression of count rate at fuel pellet interfaces is observed, which means there is no interaction between the pellets. This gamma activity profile highlights practically a symmetric loading of the fuel rod.

A method of tomographic reconstruction based on a maximum entropy algorithm has been developed as described in Ref. [1–2]. The data acquisition was done while the fuel rod was moved transversally step-by-step at regular intervals of 0.25 mm after every 72º rotation in front of a vertical collimator slit (which is 50 mm high and has a 0.25 mm aperture). Fig. 2.5b shows, qualitatively, the tomographic image of the radial distribution of $^{137}$Cs gamma activity in the cross section of the fuel rod, in the flux peaking area. This tomography indicates that the $^{137}$Cs isotope migrated from the middle to the periphery of the fuel rod and was redistributed according to the temperature profile.

The $^{137}$Cs isotope was used as burn up monitor. For an accurate determination of the burn up, the gamma self-absorption coefficient was calculated using the distribution of $^{137}$Cs activity in the cross section of the fuel rod. The burn up of the fuel rod is 8.77 MW·d·kg$^{-1}$ U (for 192 MeV fission of U). The fuel rod burn up determined by mass spectrometry is 9 MW·d·kg$^{-1}$ U (for 192 MeV fission of U). These results are in good agreement.

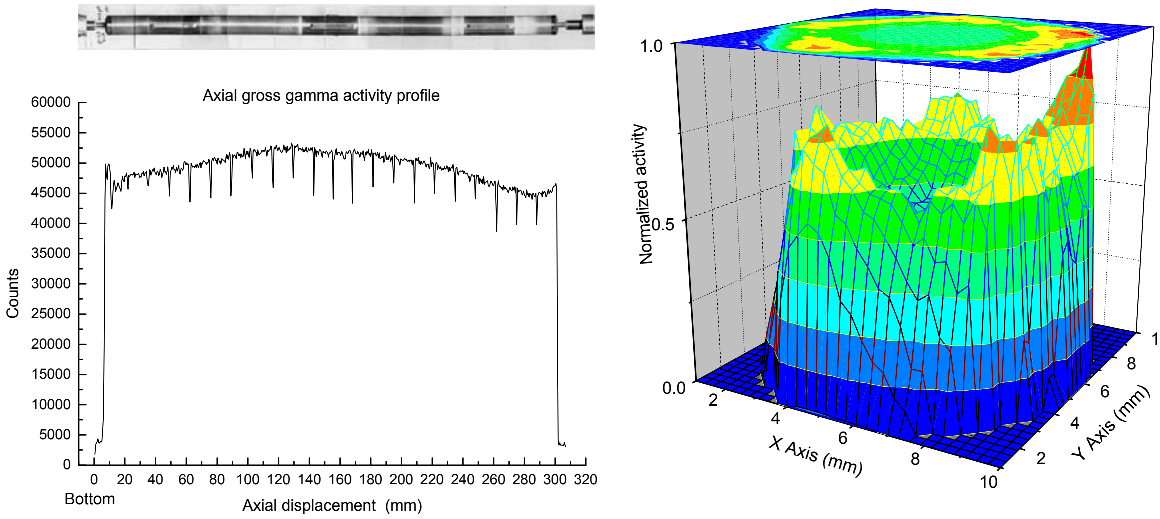
## 2.4 Metallographic and ceramographic examination

A LEICA TELATOM 4 optical microscope, with a magnification of up to 1000 times, was used for macrographic and microstructural analysis of the irradiated fuel rod. A computer-assisted analysis system is used for the quantitative determination of structural features, such as grain and pore size distribution.

The preparation of the samples includes precise cutting, vacuum resin impregnation, sample mounting with epoxy resin in an acrylic resin cup, mechanical grinding and polishing, chemical etching [3].

The analyses by optical microscopy provide information concerning:

- the aspect of pellet fissure (Fig. 2.6);
- the structural modifications of fuel and the sizes of the grains (Fig. 2.7);
- the thickness of the oxide layer and the cladding hydriding.




a. b.

*FIG. 2.5. Axial gamma scanning (a) and tomography (b) on a CANDU fuel rod irradiated in the INR TRIGA reactor in a power ramping test.*

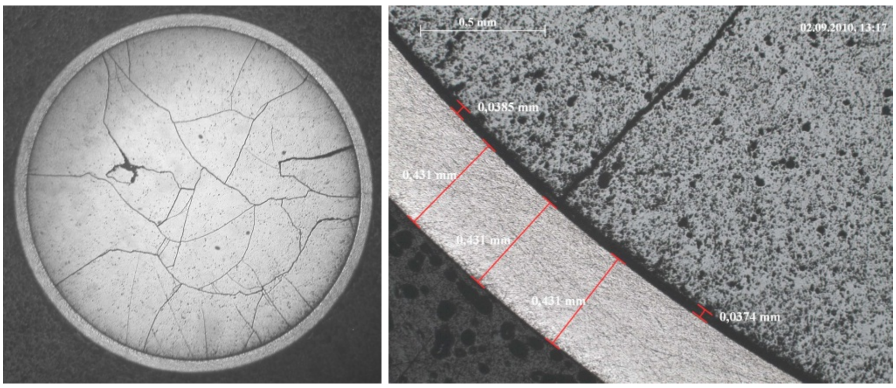


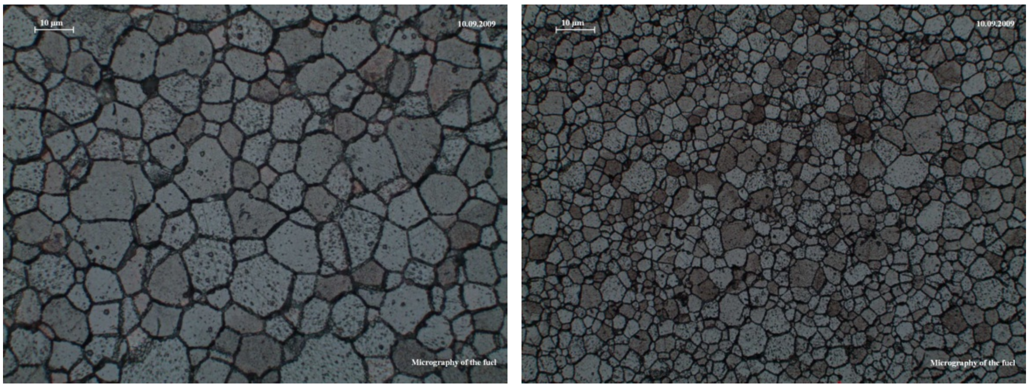
*FIG. 2.6. The cross section of the fuel pellet.*

The cross section of the fuel pellet (× 8) presents radial and circular fissures on the whole section. The cladding doesn't present nonconformities, the thickness of this being 0.431 mm. There are no visible effects on fuel sheath, due to mechanical or chemical interactions.

The hydride precipitates are orientated parallel to the cladding surface. A content of hydrogen of about 120 ppm was estimated by means of hydriding charts [4]. The fuel element presents on the outer side of the cladding a continuous and uniform zirconium oxide layer (Fig. 2.8). The thickness of the cladding oxide layer is 2.5 µm.

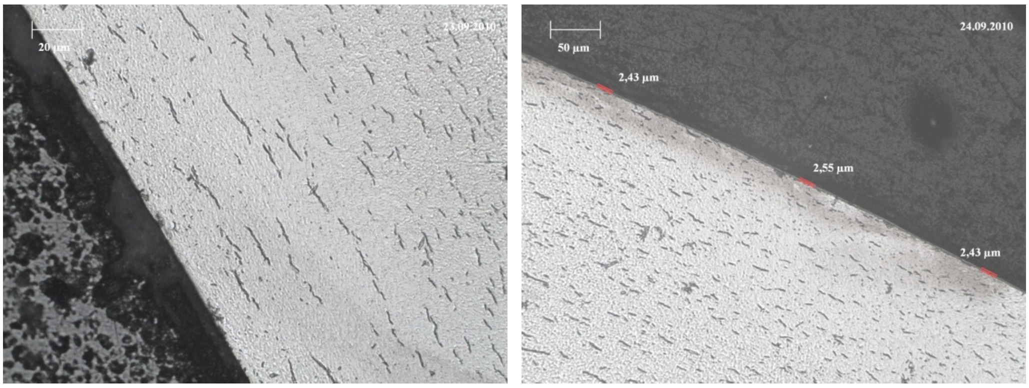
## 2.5 Determination of mechanical properties

After the preliminary tests, three ring samples (5 mm long each) were cut from the fuel rod, for further tensile tests (Fig. 2.9). The samples were prepared according to the shapes and dimensions given in Ref. [5–6].




a) Equiaxial grains b) Unaffected grains

*FIG. 2.7. The structural modifications in the fuel pellet.*




a) Cladding hydrading b) Outer oxide layer on cladding

*FIG. 2.8. Cladding aspect.*

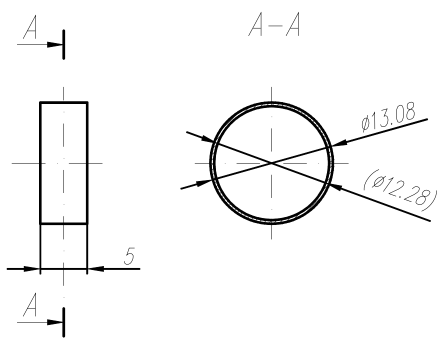



*FIG. 2.9. Ring test sample.*

The samples are tested in order to evaluate the changes of their mechanical properties as a consequence of irradiation. The tensile testing machine used is an INSTRON 5569 model. The machine uses the Merlin software for data acquisition and analysis.

The tests were done under the following conditions: constant testing temperature (300°C), 25N preload and constant tensile strain ($v$=0.05 min$^{-1}$).

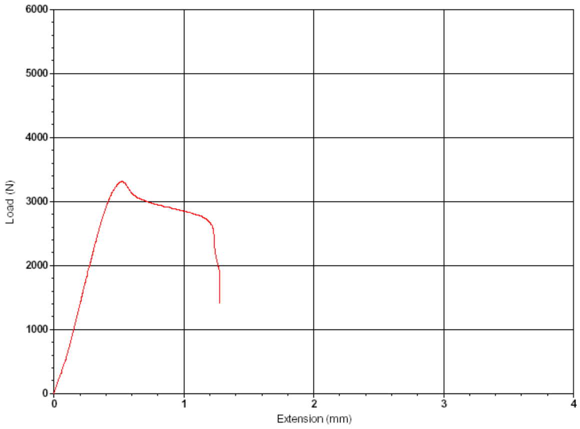



*FIG. 2.10. Load-extension diagram.*

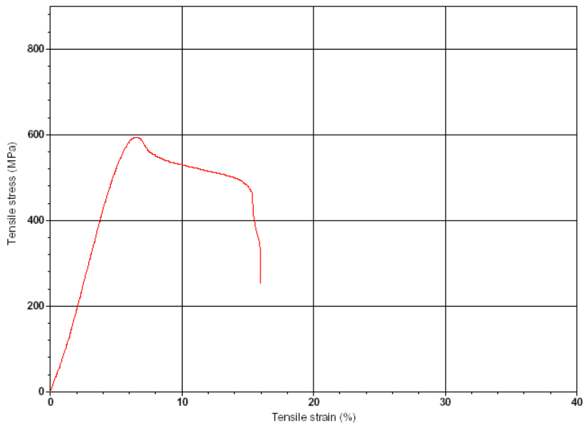



*FIG. 2.11. Strain-stress diagram.*

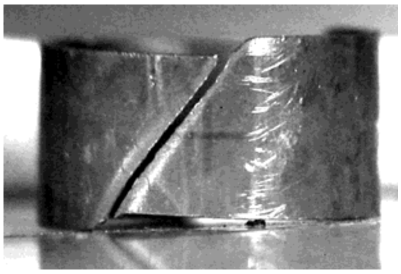


*FIG. 2.12. Ring sample after test.*

The tests have been performed in order to record or evaluate the following mechanical characteristics:

- the strain–stress diagrams and load extension (Figs 2.10–2.11);
- the yield strengths (offset method at 0.2%);
- the elastic limit;
- the ultimate tensile strength of the samples.

The tests were done according to the procedures and standards given in Ref. [7–8]. The aspect of the ring sample after the test is presented in Fig. 2.12.

### 2.6 Fracture surface analysis by scanning electron microscopy (SEM)

For sample analysis an electron microscop model TESCAN MIRA II LMU CS with Schottky field emission and variable pressure was used. The magnification range is × (4–1000000). An outstanding depth field, much higher than in the case of optical microscopy characterizes the scanning electron microscopy (SEM). This makes SEM very appropriate for analyzing fracture surfaces of Zircaloy 4 cladding resulted from tensile test.

Because of the ring shape of the sample, for rupture surface visualization, the sample was split in two parts, which were mounted in microscope chamber as in Fig. 2.13.

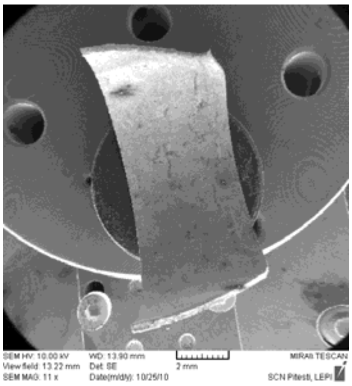
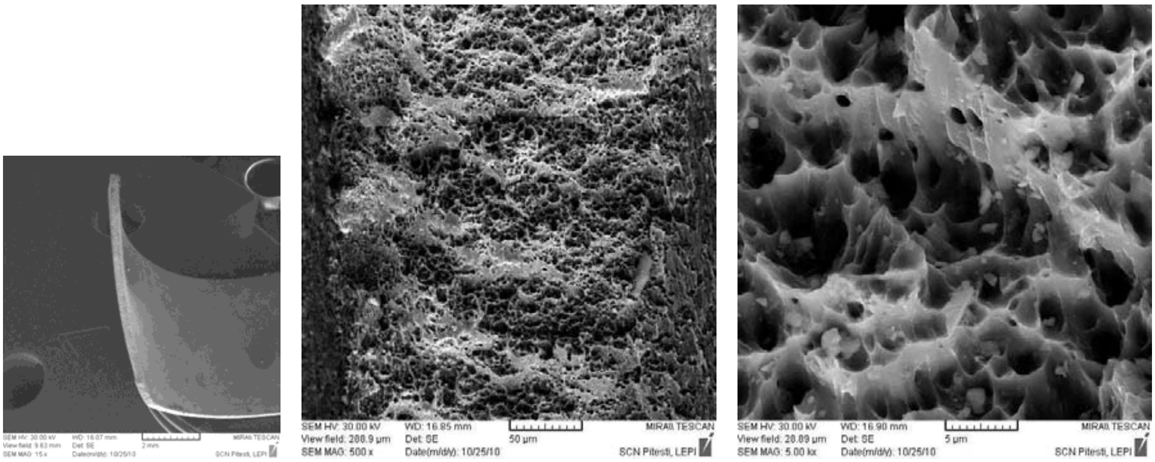


*FIG. 2.13. Sample fixture on the electronic microscope table.*

Both sides of the tensile fracture were analysed on each half of the ring. The dimples from the central zone are rather deep, whereas the ones on the outer side are tilted and smaller.

The central zone of the fracture presents equiaxial dimples (Fig. 2.14).



a) general view × 15 b) × 500 c) × 2000

*FIG. 2.14. The aspect of the central zone of the fracture.*

## 3. CONCLUSION

After irradiation, the fuel rod was kept in the reactor pool, for cooling and then it was transferred to the INR-PIEL hot cells where it was subjected to detailed examinations.

- First of all, visual inspection of the cladding was done. The aspect of the cladding surface indicates a normal behaviour of the fuel element.
- The diametrical profile, diametrical increasing, ovality and the arrow of fuel element were determined.
- The tomography indicates that the $^{137}$Cs isotope migrated from middle to periphery of the fuel rod and was redistributed according to the temperature profile.
- By metallographic and ceramographic examination we determinated that the hydride precipitates are orientated parallel to the cladding surface. A content of hydrogen of about 120 ppm was estimated. The cladding doesn't present nonconformities. The fuel element presents on the outer side of the cladding a continuous and uniform zirconium oxide layer 2.5 μm thick.

- After the preliminary tests, three ring samples were cut from the fuel rod, and were subject of tensile test on an INSTRON 5569 model machine in order to evaluate the changes of their mechanical properties as a consequence of irradiation.
- Scanning electron microscopy was performed on a microscop model TESCAN MIRA II LMU CS with Schottky FE emitter and variable pressure. The analysis shows that the central zone has deeper dimples, whereas on the outer zone, the dimples are tilted and smaller.

A full set of non-destructive and destructive examinations concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the cladding was performed. The obtained results are typical for CANDU 6-type fuel.

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# The main results of investigation of modified dispersion LEU U-Mo fuel tested in the MIR reactor

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**Abstract.** The report offers the results of the tests and of post-irradiation investigations of low-enriched U-Mo fuel tested in the MIR reactor under the RERTR program. The results of post-irradiation investigations of mini fuel elements with modified dispersion fuel containing 5%, 13% of silicon in the matrix and protective layers on fuel particles in the form of ZrN, tested up to the mean values of U-235 burn-up ~ 60% and ~ 84%, have been drawn. Data proving the reduction of the process of interaction between U-Mo alloy and the aluminum matrix containing silicon additives is afforded. It is demonstrated that the interaction with the matrix does not happen until 84% of burn-up when protective ZrN cladding is available on U-Mo particles. The impact of irradiation conditions on the interaction layer growth is discovered.

## 1. INTRODUCTION

Development of high-density low-enriched uranium fuel for research reactors is a worldwide tendency within the framework of mass-destruction weapons non-proliferation and anti-terrorism policy.

Reactor conversion to a new fuel of reduced enrichment should be carried out provided specific requirements related to active core and fuel assemblies are observed:

- reactor core design should be retained;
- annual consumption of FAs with low-enriched fuel should not exceed the number of FAs before conversion;
- main operating parameters for research reactor should be retained.

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Development of new low-enriched fuel for low-flux research reactors in Russia was initiated in 1978. Cooperation between Russian and foreign experts within the program “Reduced Enrichment for Research and Test Reactors” (RERTR) was started in the 1990s.

It should be noted that the Russian program is the only one among national programs within the framework of which a new type of fuel composition is developed together with a new design of fuel elements. It is supposed to use currently developed unified dispersion fuel rods instead of standard FAs with tubular fuel elements.

Based on the examination results, the RERTR program member-countries came to the conclusion that U-Mo alloy is the best material for high-density fuel of research reactors.

One of the main problems for low-enriched dispersion U-Mo fuel is interaction of the U-Mo alloy particles with Al matrix. As a result of this interaction, fraction of the matrix material reduces with the fuel burn-up, physical and mechanical properties of fuel meat significantly change, including considerable reduction of heat conduction, increase of fuel meat temperature and formation of gas bubbles. The paper [1] described gas bubbles formation and unpredicted fuel meat swelling caused by intensive interaction of U-Mo particles with Al matrix. Within the RERTR program, further comprehensive examinations were carried out in order to find a way to reduce this interaction. The authors of paper [2] showed possible reduction of interaction by adding silicon to Al matrix.

Within the Russian RERTR program, to solve this problem much attention was paid to the development of protective coatings on the U-Mo particles surface. Preliminary examinations showed that zirconium nitride coating on particles was determined as the most promising technology. It was decided to conduct comparative reactor tests of mini-rods with modified dispersion U-Mo fuel coated with ZrN and those with silicon additions in Al matrix.

This paper presents the results of fuel meat examinations using SEM and EPMA of non-irradiated mini-rods and those tested up to uranium-235 burn-up of ~ 60% and ~ 84%. SEM and EPMA were used to obtain data on the fuel meat structure and composition.

## 2. BASIC CHARACTERISTICS AND TEST CONDITIONS OF MINI-RODS

Mini-rods with various U-Mo modifications were tested and examined in 2003–2006. Mini-rods with silicon additions in the matrix and protective coatings on fuel particles have been tested in 2008–2010. Basic characteristics and test conditions of mini-rods are given in Table 2.1. General view and cross-section of mini-rods are presented in Fig. 2.1.

TABLE 2.1. BASIC CHARACTERISTICS AND TEST CONDITIONS OF MINI-RODS

| Parameter | Mini-rods with dispersed fuel meat, irradiation rig 1 | Mini-rods with dispersed fuel meat, irradiation rig 2 |
|---|---|---|
| Cross-section shape | Square with ribs at angles | Square with ribs at angles |
| Fuel | U-9.4 %Mo | U-9.4 %Mo |
| Type and size of U-Mo particles | Granules, ~ 100–140 μm | Granules, ~ 100–140 μm |
| Matrix material | Al; Al+2%Si; Al+5%Si; Al+13%Si | Al; Al+2%Si; Al+5%Si; Al+13%Si |
| Uranium density of fuel meat, [$g\cdot cm^{-3}$] | ~ 6.0 | ~ 6.0 |
| Clad material | SAV-6, alloy 99 | SAV-6, alloy 99 |
| Heat flux density, [$MW\cdot m^{-2}$] - aver. / max. | 0.6 / 1.2 | 1.0 / 1.8 |

TABLE 2.1. BASIC CHARACTERISTICS AND TEST CONDITIONS OF MINI-RODS (cont.)

| | | |
|---|---|---|
| Temperature of outer surface, [ºC] -aver. / max. | 90 /140 | 120 /180 |
| Burn-up 235U, [%] - aver. / max. | ~84 / 93.0 | ~60 / 67.5 |
| Fission rate in particles, [$10^{14}$ $cm^{-3}s^{-1}$] - aver. / max. | ~2.3 / ~4.6 | ~3.6 / ~6.8 |
| Fission density in particles, [$10^{21}$ $cm^{-3}$] - aver. / max. | ~5.6 / ~6.2 | ~4.0 / ~4.5 |
| Duration of tests, [days] | 285 | 130 |

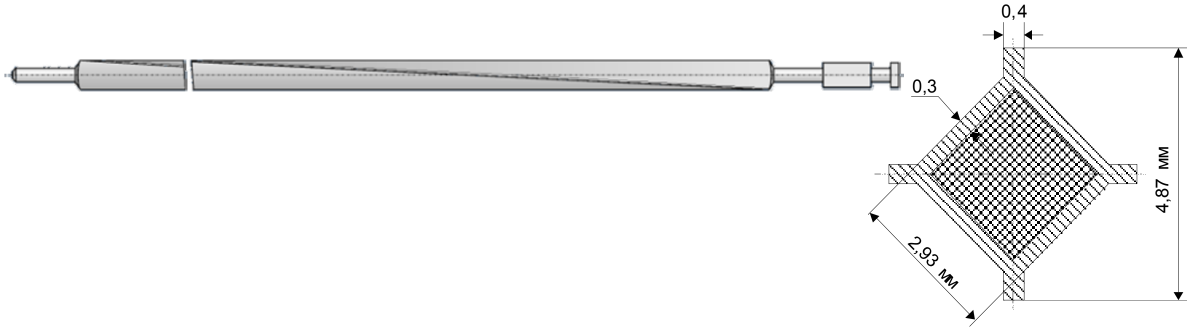



*FIG. 2.1. General view and cross-section of mini-rods.*

## 3. PREPARATION OF SPECIMENS

For SEM and EPMA examinations, full cross-sections of mini-rods were prepared. Preliminary examination of specimens using optical microscopy showed that the width of the interaction layer (IL) between U-Mo particles and matrix is significantly scattered. Therefore, it was impossible to evaluate an extent of the fuel-matrix interaction based on the IL width for separate particles. It was necessary to get SEM images of the entire cross-section of mini-rods and determine volume fraction of the IL.

Specimens were prepared in the specially-equipped hot cells. The cross-section was made by fixing a specimen in the holder and its polishing to get a surface of the required quality. The specimen was fixed in the holder by Wood alloy.

Upon completion of preparations, specimens were transported using an inter-cell transporter to a hot cell with microscope Philips XL 30 ESEM-TMP. Specimens were installed in the microscope working chamber by means of manipulators.

The SEM area at RIAR (Dimitrovgrad, Russia) allows the examination of radioactive specimens with high level background ionizing radiation. This area is arranged so that the microscope column is placed on a separate base inside the hot cell (Fig. 3.1) and controlled remotely from the operator's room. This area and its SEM and EPMA capabilities were described in details at the HOTLAB conference in 2004 [3].

*FIG. 3.1. View of the microscope from the operator's room through the hot cell window.*

# 4. RESULTS OF EXAMINATIONS

## 4.1. State of fuel meat of non-irradiated mini-rods

The examination of unirradiated fuel compositions showed no interaction between fuel particles and matrix during manufacturing of mini-rods of all types.

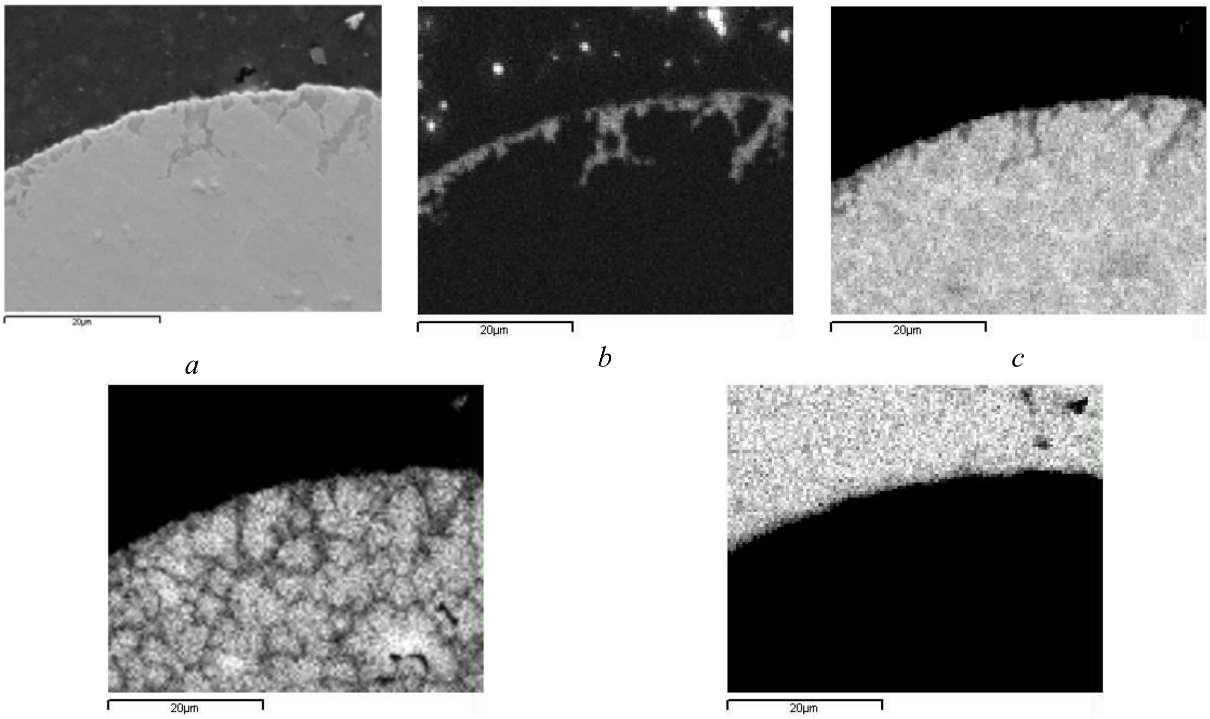
For mini-rods with 5% Si matrix, silicon interacts with U-9%Mo alloy. A silicon-saturated layer is formed on the surface of fuel particles. It doesn't form a continuous coating of particles, its size differs for various particles and is not uniform along the perimeter of one and the same particle.

Distribution maps of silicon, uranium, molybdenum and aluminum were plotted using EPMA (Fig. 4.1). The grain boundaries of the U-Mo alloy have reduced amount of Mo and increased amount of U, as it was mentioned in [4]. The distribution map analysis makes it possible to identify that Si diffuses inside U-Mo particles along the grain boundaries.

No traces of Al penetration into U-Mo alloy were found.

For 13% Si in the matrix, no qualitative change in the interaction between silicon and fuel particles is observed as compared to its 5% content in the matrix. Silicon penetrates deeper into U-9%Mo alloy diffusing along the grain boundaries. The silicon-saturated layer becomes thicker on the surface of fuel particles. It covers larger surface of fuel particles without forming a continuous film.

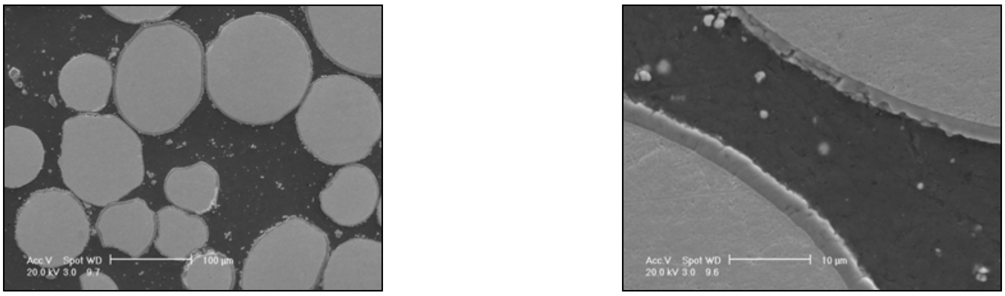
For fuel elements with ZrN coating on fuel particles, the coating layer thickness makes up 2–3 μm (Fig. 4.2). For some U-Mo particles, ZrN coating layer is observed partially or not observed at all.




*a* *b* *c*

*d* *e*

*FIG. 4.1. Image of the fuel meat region with 5% Si matrix (a) on which distribution maps of Si (b), U (c), Mo (d) and Al (e) were obtained.*



*a* *b*

*FIG. 4.2. Fragment of the fuel meat of the mini-rod with ZrN coating of U-9%Mo alloy particles (a). SE image of ZrN coating (b).*

### 4.2. State of fuel meat after irradiation

The U-Mo alloy interacts with Al under irradiation. The degree of interaction between the U-9%Mo particles and matrix from aluminum powder PA-4, 5% Si and 13% Si and ZrN coating was evaluated based on the volume fraction of the interaction layer. The results are presented in Table 4.1.

The table shows that with addition of silicon in the matrix, the IL significantly reduces. Increase of silicon concentration from 5% up to 13% considerably reduces the IL volume fraction.

For mini-rods with ZrN coating on fuel particles, the IL volume fraction remains at 6% for the above-mentioned irradiation conditions. Microstructure of irradiated fuel compositions in examined mini-rods is presented in Fig. 4.3.

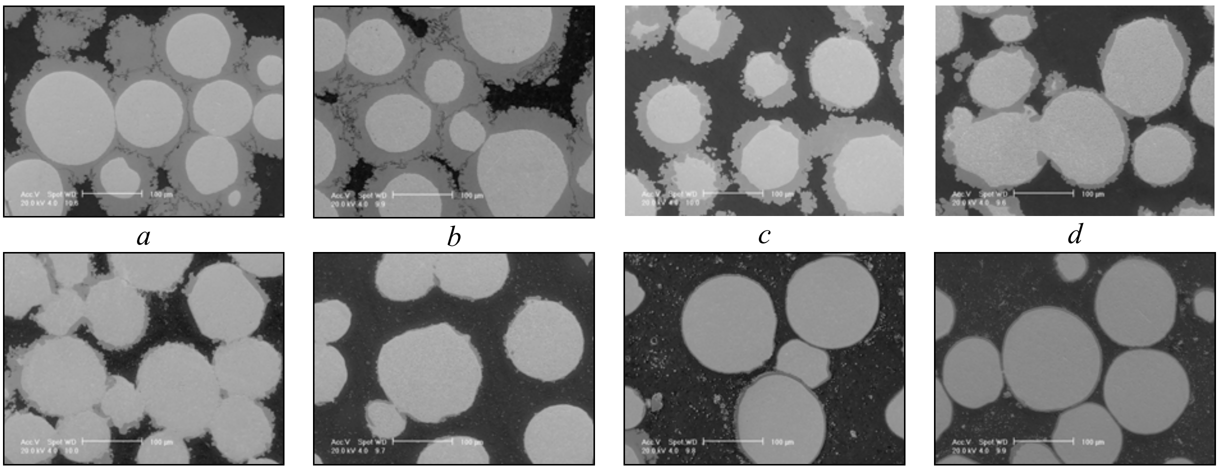
Analysis of the fuel meat state shows that the IL thickness is similar for all U-Mo particles of the U-9%Mo/Al (without Si) composition and makes up 25–30 µm. With addition of 5% Si to the matrix, the IL thickness becomes very non-uniform along the perimeter of fuel particles. Its values range from 2 to ~30 µm for high fission rate and a burn-up of ~ 60%, and from 2 to ~20 µm for low fission rate and a burn-up of ~ 84%.

For 13% Si added matrix the major area of the U-Mo particles surface has IL ~ (1–2)µm thick. At a high fission rate, this layer achieves its maximum thickness of 15 µm, while at low fission rate it is equal to 10 µm.

Protective ZrN coating completely prevents the interaction of U-Mo alloy with the Al matrix. The interaction layer is formed in the fuel meat of such mini-rods where there is no protective coating on the fuel particle surface, both for a burn-up of ~ 60% and ~ 84%.

TABLE 4.1. RESULTS OF MEASURING THE IL VOLUME FRACTION IN MINI-RODS AFTER IRRADIATION UP TO THE AVERAGE BURNUPS OF ~ 60% AND ~ 84%

| Fuel meat material | Rig No.2 | | Rig No.1 | |
|---|---|---|---|---|
| | Burn-up, [%] | IL volume fraction, [%] | Burn-up, [%] | IL volume fraction, [%] |
| Si<0.4% | ~ 60 | 40±4 | ~ 84 | 40±4 |
| A1+5%Si | ~ 60 | 25±3 | ~ 84 | 10±2 |
| A1+13%Si | ~ 60 | 10±2 | ~ 84 | 6±1 |
| ZrN coating Si<0.4% | ~ 60 | 6±1 | ~ 84 | 6±1 |



*a b c d e f g h*

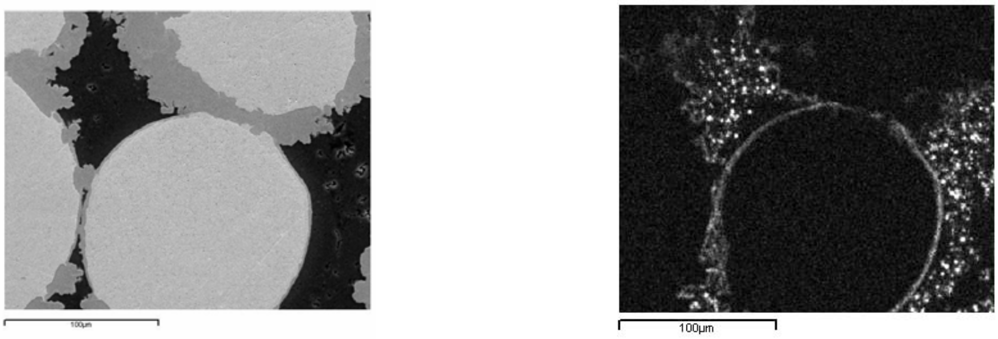
*FIG. 4.3. SEM-images of fuel meat microstructure: U-9%Mo particles in Al matrix. Burn-up ~ 60% (a) and ~ 84% (b); U-9%Mo particles with addition of 5% Si in the matrix. Burn-up ~60% (c) and ~84% (d); U-9%Mo particles with addition of 13% Si in the matrix. Burn-up ~60% (e) and ~84% (f); U-9%Mo particles with ZrN coating. Burn-up ~ 60% (g) and ~ 84% (h).*

EPMA was used to determine silicon effect on the IL thickness. Silicon distribution maps were plotted. They demonstrate that with high amount of silicon in the IL, its thickness is equal to 2–3 µm (Fig. 4.4) irrespective of irradiation conditions for this experiment. Quantitative EPMA shows that the silicon content in such IL ranges within 7–15 at.%.

No silicon was revealed in the wide interaction layer. It concentrates at the boundary of wide IL with the matrix.

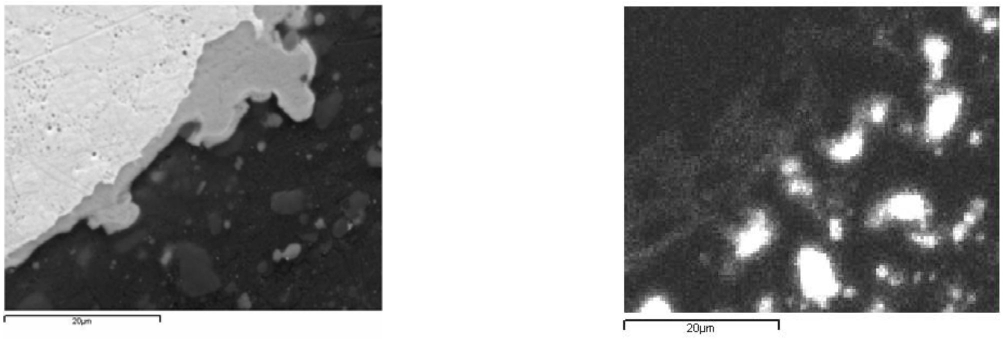
Summary of the EPMA results shows that the IL grows before contact with silicon particles. At that moment it stops to grow and further interaction is observed at adjacent silicon-free regions. With a great quantity of silicon particles, the IL growth stops in all directions. It is clearly seen in Fig. 4.5, where silicon particles prevent the IL growth in each direction.

No cases of silicon particles presence inside the IL were observed. Therefore, silicon particles effectively prevent the IL propagation.




*a* *b*

*FIG. 4.4. Non-uniform interaction layer in 5% silicon matrix. Burn-up of U-235 is ~ 60%. Silicon distribution around fuel particles. No silicon is found in the wide IL.*




*a* *b*

*FIG. 4.5. IL image. 13% Si matrix, U-235 burn-up is ~ 60%. The IL propagation is stopped by silicon particles in all directions (b).*

## 5. DISCUSSION

Results of examinations of mini-rods from two irradiation rigs where the testing conditions differed by thermal power (fission rate) are evident of significant effect of irradiation conditions on the IL formation intensity. For the first irradiation rig, an average burn-up of U-235 of ~ 84% (average fission density in fuel particle is ~$5.6 \times 10^{21}$ $cm^{-3}$) was achieved at full power operation of 285 days, average fission rate of $2.3 \times 10^{14}$ $cm^{-3}s^{-1}$. For the second irradiation rig, an average burn-up of U-235 of ~ 60% (average fission density in fuel particles is ~$4.0 \times 10^{21}$ $cm^{-3}$) was achieved at full power operation of 130 days and an average fission rate of $3.6 \times 10^{14}$ $cm^{-3}s^{-1}$. The following results were obtained
(Table 4.1):

- for silicon-free matrix, nearly the same volume fractions of the fuel-matrix interaction layer were obtained;
- for mini-rods with the 5% silicon in the matrix, the IL volume fraction is 2.5 times higher than that for mini-rods with the 13% silicon in the matrix during irradiation in the second irradiation rig up to a burn-up of ~60% at fission rate of $3.6 \times 10^{14}$ $cm^{-3}s^{-1}$;

- for mini-rods with the 5% silicon in the matrix, the IL volume fraction is 1.7 times higher than that for mini-rods with the 13% silicon in the matrix during irradiation in the first irradiation rig up to a burn-up of ~84% at fission rate of $2.3 \times 10^{14}$ $cm^{-3}s^{-1}$.

Therefore, with silicon addition to the matrix, interaction suppression effect of silicon concentration in the matrix is greater at relatively high fission rate ($3.6 \times 10^{14}$ $cm^{-3}s^{-1}$).

The above-mentioned effects can be significantly influenced by fuel meat temperature.

In the production of mini-rods with silicon additions in the matrix, dissolution of silicon near the surface of U-Mo particles is not completed. As a result, fuel particles or some regions of the fuel particle surface are not saturated with silicon. In this case, when on the surface of U-Mo particles a silicon-saturated layer is formed, it slows down the diffusion of uranium and molybdenum atoms from one hand and Al atoms from the other hand. The IL becomes 1–3 µm wide under irradiation. It was determined that to prevent intensive growth of the IL under irradiation in reactor MIR, 7 at.% silicon concentration is sufficient. If in the initial state silicon concentration on the fuel particle surface is not sufficient, formation of a wide interaction layer occurs during irradiation.

There is no silicon inside the wide IL. It is displaced to the IL boundary with matrix. With movement of the boundary inside the matrix the silicon concentration on it increases resulting in slow-down of the IL growth. If large silicon particles appear on the way of the IL boundary with matrix, the IL stops to grow in this area, while the interaction in regions free from silicon continues. It explains non-uniformity of the IL width in the fuel meat in the matrix of which there are silicon additions. Therefore, silicon particles form an effective barrier against U-Mo alloy interaction with Al. Increase of fuel meat temperature contributes to more intensive diffusion. It results in the increase of the IL volume fraction with increase of fission rate.

The most effective measure to stop the U-Mo alloy interaction with Al is application of ZrN coating on the surface of U-Mo particles. If the coating layer is 2–3 µm wide, interaction with the Al matrix under the above-mentioned irradiation conditions up to a burn-up of ~ 84% is not observed.

## 6. CONCLUSION

Post-irradiation examinations of mini-rods with modified dispersion U-Mo fuel tested in the MIR reactor up to average burn-ups of 60% and 84% of U-235 showed the following results:

- protective coating on U-Mo particles with ZrN layer 2–3 µm thick almost completely prevents their interaction with the Al matrix without using any additions up to high burn-ups at fission rates of $3.6 \times 10^{14}$ $cm^{-3}s^{-1}$;
- silicon added into the matrix decreases significantly the interaction between U-Mo particles and the matrix, both silicon content and fission rate being effective;
- effect of interaction suppression with increase of silicon concentration in the matrix is higher at relatively high fission rate.

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