NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 9(1): 51–58
DOI 10.3897/nucet.9.102492

Research Article

Examination of fuel elements irradiated in the reactor of the World's First NPP after long-term storage*

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Academic editor: Georgy Tikhomirov ♦ **Received** 11 May 2022 ♦ **Accepted** 16 January 2023 ♦ **Published** 17 March 2023

Citation: Ivanov SN, Porollo SI, Shulepin SV, Baranaev YuD, Timofeev VF, Kharizomenov YuV (2023) Examination of fuel elements irradiated in the reactor of the World's First NPP after long-term storage. Nuclear Energy and Technology 9(1): 51–58. https://doi.org/10.3897/nucet.9.102492

Abstract

Examinations of fuel elements with two different fuel compositions, U-Mo+Mg and UO_2+Mg, irradiated in the AM reactor after their long-term storage do not reveal any visible defects on the surface of their outer claddings. However, in the fuel elements with U-Mo fuel, an increase in the diameter of the outer cladding is observed. This is most noticeable in the upper part of the fuel element. Storage of the fuel elements with UO_2 fuel for 15–22 years does not lead to a change in their diameter within the measurement accuracy. At the same time, metallographic studies have shown that on the external surface of the outer cladding and the internal surface of the inner cladding of the fuel elements with U-Mo+Mg and UO_2+Mg fuel compositions, after long-term storage, defects are observed in the form of intergranular and irregular frontal corrosion, pits and pittings up to 20 μm deep. No interaction is found at the points of contact between the fuel claddings and the fuel composition of the layers. There is no noticeable decrease in the thickness of the outer and inner claddings of the fuel elements after long-term storage, nor does the thickness of the claddings at the locations of defects go beyond its minimum initial value, taking into account the technological tolerance for variations in thickness. It is noteworthy, however, that cracks are found in both types of fuel elements both in the fuel grains and in the magnesium matrix. As a result of long-term storage of the fuel elements with U-Mo fuel for 45–55 years, the mechanical properties of their outer claddings gradually degrade, due to which the plasticity of the cladding is significantly reduced.

Keywords

long-term storage, fuel assembly (FA), fuel element, cladding, corrosion, fuel composition, metallographic studies, mechanical properties

Introduction

Countries developing nuclear power inevitably face the problem of storing spent nuclear fuel (SNF). The duration of SNF storage may vary depending on many factors, such as the possibility of its reprocessing and transport, the condition of the fuel, etc. At present, only spent fuel elements of VVER-440, BN-600, BN-800 reactors, as well as fuel elements of transportable nuclear power plants and research reactors are reprocessed in Russia. Fuel elements of VVER-1000 reactors are reprocessed in a limited amount, and most of them, like the fuel elements of RBMK reactors, are stored in specialized storage facilities. The same applies to the fuel elements of the EGP-6

* Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2022, n. 3, pp. 106–119.

reactor at the Bilibino NPP, the storage life of which is approaching 50 years. Issues related to the safety of nuclear power plants and the state of fuel rods after long-term storage have been the subject of numerous studies (Sidorenko 1994; Solonin et al. 2000; Volkova et al. 2003; INES 2010; Golosov et al. 2013a; Golosov et al. 2013b; Mukhamadeev et al. 2018; Parafilo et al. 2018).

To justify the possibility of safe long-term storage of SNF from EGP-6 reactors, it is necessary to have information on the actual state of spent fuel assemblies in storage. Unfortunately, due to the lack of hot chambers at the Bilibino NPP, it is impossible to conduct a detailed examination of the irradiated assemblies. Most of the necessary data can be obtained as a result of the study of the fuel elements of the AM reactor of the world's first NPP after their irradiation and subsequent storage. The fuel elements of the EGP-6 reactor are in fact analogous to the fuel elements of the AM reactor, both in the design and in materials used in their manufacture. The storage life of the fuel elements of the AM reactor after operation exceeds 50 years, due to which it is possible to make a fairly reasonable forecast on the behavior of the spent fuel elements of the EGP-6 reactor during long-term storage. In the event of an emergency depressurization of the canisters containing the irradiated assemblies of the EGP-6 reactor, water from the storage pool (SP) will come into contact with the fuel elements, causing their corrosion and subsequent water pollution. In Gaiazov et al. 2018; Ivanov et al. 2019; Gaiazov et al. 2021, the results of corrosion tests in the aqueous medium of the spent fuel elements of the AM and VVER-440 reactors after long-term storage are presented, which are necessary for predictive assessments of the release of radioactive products from leaky fuel elements of the EGP-6 reactor into the aqueous medium of the pool when the canisters are depressurized.

Previously, in the hot laboratory of IPPE, a study of the fuel elements of the AM reactor was carried out immediately after their irradiation and after storage for 38 years (Ibragimov et al. 1964; Ivanov et al. 2000; Ivanov et al. 2002; Ivanov et al. 2011). It was shown that storage of the fuel elements did not lead to any noticeable damage to them, and the properties of the fuel cladding material did not undergo significant changes compared to the properties of the cladding immediately after irradiation. To obtain more complete statistically reliable information, an additional study was carried out; it involved 100 fuel elements of the AM reactor (with the storage life of 22–55 years). The results of this study are presented in this article.

Design and conditions for irradiation of fuels assemblies

The fuel elements of the AM reactor at the First NPP have two coaxially arranged cylindrical claddings made of 1X18H9T (1X18H9T is a Russian grade of corrosion-resistant heat-resistant steel, where X = Cr, H = Ni and T = Ti) stainless steel, between which there is a fuel composition consisting of fuel grains (U-Mo or UO_2 alloy) dispersed in a magnesium matrix. The length of the active part of the fuel elements was 1700 mm, the total length was 1885 mm. The outer diameter and thickness of the outer cladding of the fuel elements were 14.0 and 0.2 mm, and the inner cladding was 9.0 and 0.4 mm, respectively (Ivanov et al. 2011). The fuel assembly (FA) of the AM reactor consisted of graphite bushings, inside which there was a central downcomer tube with a diameter of 15 mm and four annular fuel elements connected in the upper and lower parts to peripheral tubes made of 1X18H9T steel. The coolant (water) descended along the central tube to the lower distribution chamber, from which it rose upwards along the four peripheral tubes of the annular fuel elements, removing heat from the surface of the inner claddings of the fuel elements (Ushakov 1959). During irradiation, the fuel elements were cooled with water at a pressure of 10 MPa; the water temperature was 150–190 °C at the FA inlet and 210–280 °C at the FA outlet. The maximum temperature of the outer surface of the fuel element, which was in contact with the gaseous medium of the composition O_2 = 0.2%, H_2 = 0.2%, CO_2 = 0.5%, N_2 = the rest, did not exceed 370 °C (Ibragimov et al. 1964; Ivanov et al. 2011).

Storage of fuel elements after irradiation, selection of fuel elements for studies

Immediately after the SFAs were removed from the core of the AM reactor, they were placed in the near-reactor spent fuel pool to reduce the energy release. During storage, the SFAs did not come into contact with water; the sealed cells, in which the irradiated SFAs were placed, were constantly ventilated. After the afterpower was reduced, the SFAs were dismantled, the fuel elements in the amount of 50–100 pieces were packed into canisters, which were transported to an independent spent fuel storage installation (ISFSI) located on the territory of IPPE. The underground part of the storage facility is equipped with forced ventilation; the canisters with fuel elements are stored in air at a temperature close to the temperature of the central hall (~ 20 °C).

After all the fuel was unloaded from the AM reactor, there were about seven thousand fuel elements with metal (OM-9 alloy) and UO_2 fuel in the storage installation. Of greatest interest for this study were the fuel elements with U-Mo fuel, which were operated in the reactor in 1954–1965, i.e., having a maximum storage life, as well as the fuel elements with UO_2 fuel, which have been operated in the reactor since 1987. At the same time, another selection factor was the degree of fuel burnup, which affects the degree of damage to the fuel cladding. Based on these considerations, two canisters with the fuel elements containing U-Mo fuel and two canisters with the fuel elements containing UO_2 fuel were selected for further work. According to the datasheet specifications, all the fuel elements in these canisters were sealed when sent for storage. The characteristics of the fuel elements located in these canisters are presented in Table 1.

Table 1. Characteristics of the AM reactor fuel elements

ISFSI slot No.	Fuel type	Number of fuel elements, pcs.	Life time in the reactor	Burnup range (average), MW×day/kg	Storage duration in SP and ISFSI, years
92	U-Mo	108	1954–1957	2.4–13.1 (7.3)	53–55
233	U-Mo	72	1954–1965	0.9–20.2 (12.3)	45–55
56	UO_2	80	1988–1992	7.9–13 (11.4)	18–20
50	UO_2	92	1988–1995	11.4–24.3 (18.6)	15–22

Visual inspection of fuel elements

For research purposes, four canisters were delivered from the ISFSI to the hot laboratory of IPPE: two canisters with fuel elements based on U-Mo fuel located in Slots 92 and 233 of the storage installation and two canisters with fuel elements based on UO_2 fuel located in Slots 50 and 56 (see Table 1). Twenty-five fuel elements were randomly removed from each canister for visual inspection. A total of 100 fuel elements, 50 of each type, were selected for the inspection.

The fuel elements were inspected in the hot chamber through binoculars with ten-power magnification. The main purpose of the inspection was to assess the state of the surface of the outer claddings of the fuel elements and the presence of any defects on them due to long-term storage. Each inspected fuel element was assigned a conditional number; photographs were made of various parts of the surface, including the detected features of their condition (tint colors, type of films and deposits, risks, scratches, etc.).

Figures 1 and 2 show the most characteristic areas of the outer surface of the examined fuel elements. The inspection showed neither visible damage to the outer claddings of the fuel elements nor defects in the form of cracks, pits, pittings and other corrosion damage. Nor were there local swellings found on the outer claddings, indicating the presence of through destruction in a given place. The surface of all the fuel elements with U-Mo fuel was covered with a thin film of light gray color, on which one could see scrapes, scratches, traces of abrasion and scoring as well as spots of tarnish colors (Fig. 1a, b). There were spots with films and deposits of gray, brown and white color on the surface of many fuel elements (Fig. 1c, d) and of rust color on some fuel elements. Most of the surface of 45 of the 50 examined fuel elements with UO_2 fuel is covered with a thin layer of gray or dark gray color (Fig. 2a, b), the surface of other fuel elements is light silvery (Fig. 2d). On the surface of all the fuel elements, one could find scrapes, scratches, traces of abrasion, scoring and tarnishing as well as spots with films and deposits of white, gray, light brown and black color (Fig. 2b, c, d).

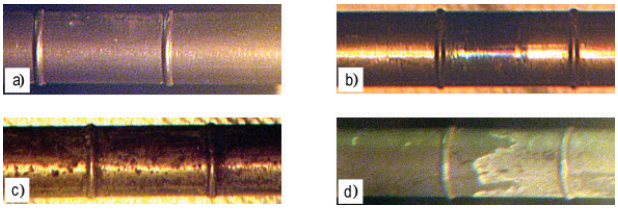

Figure 1. Typical view of the outer surface of the spent fuel elements with U-Mo+Mg fuel from the AM reactor after long-term dry storage for 45–55 years: **a)** gray surface; **b)** area with tarnish colors; **c)** multiple spots of brown plaque; **d)** area with a white film.

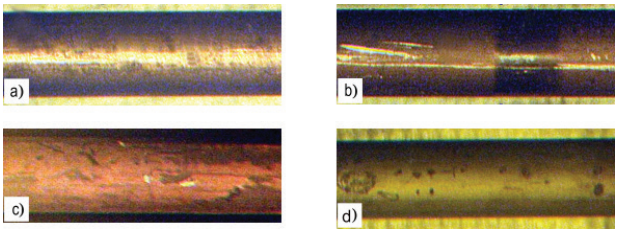

Figure 2. Typical view of the outer surface of the spent fuel elements with UO_2+Mg fuel from the AM reactor after long-term dry storage for 15–22 years: **a)** gray surface with dark gray plaque spots; **b)** dark gray surface with scrapes and scratches; **c)** surface with brown and white deposits; **d)** light silvery surface with gray plaque spots.

Methods of material science research

After the visual inspection of the fuel elements was completed, fuel element 33 with U-Mo fuel and fuel elements 8 and 38 with UO_2 fuel were selected for further material science studies; their surface conditions were different. Fuel element 8 had a light silvery surface covered with spots with light gray patches (see Fig. 2d), while fuel element 38 had a dark gray surface (Fig. 2b). To determine the degree of degradation of the properties of the cladding material and fuel during long-term storage, the results of post-reactor studies of fuel elements immediately after irradiation, as well as after storage for 38 years, were used. A brief description of the fuel elements examined immediately after irradiation and after irradiation and storage for 15–55 years is given in Table 2.

Material science studies included measuring the diameters of the fuel elements, metallographic examination of the fuel and fuel element cladding as well as measuring the short-term mechanical properties of the cladding. Measurements of the outer diameters of fuel elements 8, 33 and 38 were carried out using a micrometer (with an accuracy of ± 0.01 mm) every 100–110 mm along the length of the fuel elements in the sections located between the corrugations. Four measurements were carried out at each section with the fuel element rotated relative to the longitudinal axis by 45°.

Specimens for metallographic studies of the fuel were prepared in the hot chamber according to the standard technique, and microstructural studies were performed on a MIM-15 remote optical microscope. The microstructure

Table 2. Characteristics of the examined fuel elements

Fuel element No. (SFA No.)	Irradiation period in the reactor	Fuel	Burnup range (average), MW×day/kg	Storage duration, years
Fuel element 33	1954–1965	U-Mo+Mg	12.6	45–55
Fuel element 8	1988–1992	UO_2+Mg	12.8	18–20
Fuel element 38	1988–1995	UO_2+Mg	11.4–24.3	15–22
Four fuel elements (AMK-248)	1955–1958	U-Mo+Mg	17.6	38
Four fuel elements (D-85)	1979–1987	U-Mo+Mg	24.9	–
Four fuel elements (Zh-72)	1986–1987	UO_2+Mg	5.4	–
One fuel element (Zh-52)	1984–1987	UO_2+Mg	14.9	–

of the fuel claddings was studied after chemical etching of the fuel composition using an MMP-4 optical microscope.

Mechanical tests of ring samples (2–3 mm wide) of the fuel claddings after fuel etching were carried out on remote tensile testing machines MD-2T and 1689R-1 with an active gripper movement speed of 1 mm/min at a temperature of 25 °C. Samples were cut from the lower and middle parts of the fuel elements (4–5 samples from each section). As a result of the tests, the following mechanical characteristics of the fuel cladding material were obtained: ultimate tensile strength σ_B, yield strength σ_{02}, and total relative elongation δ_r.

Measuring the diameters of the fuel elements

The results of measuring the diameter of fuel element 33 with U-Mo fuel are shown in Fig. 3. Also shown for comparison are the results of measuring the diameters of two D-85 fuel elements immediately after irradiation. It can be seen that the diameter of fuel element 33 is larger than the diameters of the other two fuel elements. This difference is especially noticeable in the upper parts of the fuel elements.

For fuel elements 8 and 38 with UO_2 fuel, no regularity in the changes in the diameters along the length of the fuel elements is observed. No regularity is found for the changes in the diameters along the length of the fuel elements in the fuel assemblies Zh-72 and Zh-52, which were examined immediately after irradiation. Most of the diameters of the fuel elements with UO_2 fuel do not go beyond the manufacturing tolerance.

Table 3 shows the results of measuring the diameters of the fuel elements irradiated in the AM reactor with U-Mo- and UO_2-fuel before and after storage. It can be seen that the diameters of the fuel elements with U-Mo fuel slightly increase during storage, but such a conclusion cannot be drawn for the fuel elements with UO_2 fuel due to insufficient statistics.

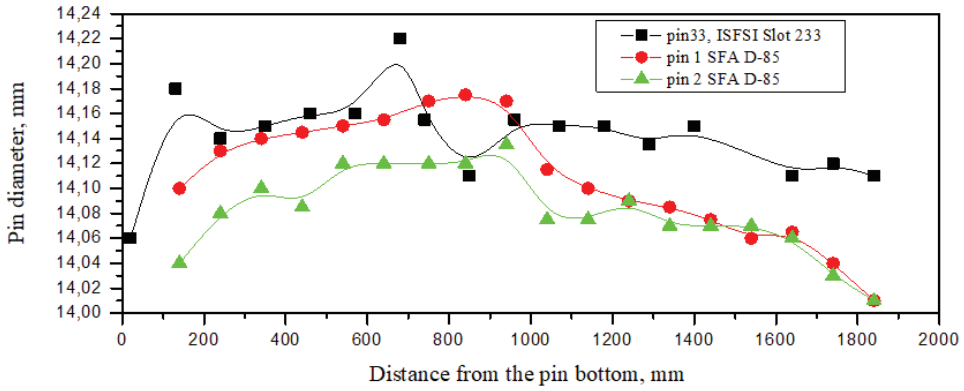

Figure 3. Changes in the diameters along the length of fuel element 33 after long-term storage and D-85 fuel elements immediately after irradiation with U-Mo fuel.

Table 3. The results of measuring the diameters of the fuel elements with U-Mo and UO_2 fuel

Fuel element No. (SFA No.)	Fuel	Average fuel element diameter, mm	Fuel element diameter range, mm	Maximum relative diameter change*, %
Fuel element 33	U-Mo+Mg	14.15	14.15–14.22	1.57
Fuel elements 1, 2, 3 (AMK-248)		14.10	13.98–14.17	1–1.25
Fuel element 1 (D-85)		14.11	14.10–14.17	1.25
Fuel element 2 (D-85)		14.08	14.04–14.14	0.93
Fuel element 8	UO_2+Mg	14.02	13.98–14.08	0.57
Fuel element 38		13.96	13.94–13.98	–0.14
Fuel element 1 (Zh-72)		14.00	13.97–14.03	0.21
Fuel element 1 (Zh-52)		13.98	13.97–13.99	–0.07

* The maximum change in the diameter of the fuel elements was determined relative to its nominal value, equal to 14.00 mm.

Metallographic study of the fuel elements with U-MO fuel

During metallographic studies of the D-85 fuel element carried out immediately after irradiation, no defects were found on its outer and inner claddings. The study of the AMK-248 fuel assemblies after 38 years of storage revealed that the outer surface of the outer cladding and the inner surface of the inner cladding of the fuel elements had only small defects up to 10 μm deep in the form of small pits and intergranular corrosion with chipping of individual grains. The depth of interaction between the fuel and the cladding did not exceed 1 μm.

Samples for metallographic studies of fuel element 33 were cut in the hot chamber with a diamond disk on a remote milling machine. After chemical etching of the fuel from the sample cut from the middle part of the fuel element, it was found that the resulting annular sample of the outer cladding had fractures that had occurred during the cutting process (Fig. 4a). The destruction of the cladding on both sides had a similar character and had occurred as a result of a brittle transcrystalline cleavage going at an angle of 20° from the inner surface of the cladding (Fig. 4b). In addition, a transcrystalline crack is observed near the cleavage (indicated by an arrow in Fig. 4b).

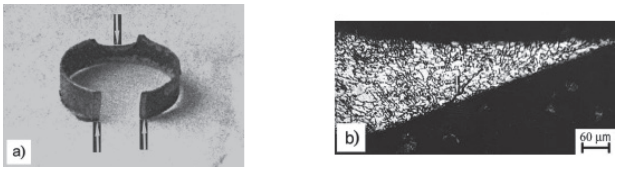

Figure 4. View of the sample of the outer cladding of fuel element 33 after fuel etching (the arrows show the fracture sites) (**a**) and the microstructure of the outer cladding material in the fracture zone (**b**).

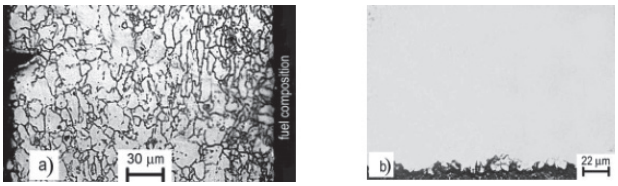

Figure 5. A wedge-shaped pit on the outer side of the outer cladding (**a**) and intergranular corrosion with grain chipping on the inner side of the inner cladding (**b**) of irradiated fuel element 33 with U-Mo fuel (in a section of 400 mm from the bottom of the fuel element) after storage for 45–55 years.

On the outer side of the outer cladding (on the side of the cladding contact with the gaseous medium of the reactor during irradiation) of fuel element 33, intergranular corrosion up to 9 μm deep, pittings 4–5 μm deep, and rounded or wedge-shaped pits 10–20 μm deep are observed (Fig. 5a). On the inner side of the cladding (on the side of contact with the fuel), there are only pits 1–3 μm deep to be observed (Fig. 5a).

On the outer side of the inner cladding (on the side of the fuel), local depressions 80–150 μm in length along the cladding perimeter and up to 18 μm deep were found, the formation of which was most likely associated with the fuel element manufacturing technology rather than caused by irradiation or long-term storage. On the inner side of the inner cladding (from the side of the water coolant during irradiation), almost along the entire perimeter, corrosion damage was found in the form of intergranular (Fig. 5b) and uneven frontal corrosion up to 15 and 20 μm deep, respectively.

In fuel element 33, the fuel composition consisting of U-Mo alloy particles in a magnesium matrix retained its integrity. The fuel grains were tightly linked to the matrix, and around the fuel grains in the matrix, dark gray belts 25–35 μm wide were observed (Fig. 6). Tight adhesion is maintained between the magnesium matrix and the claddings, but single microcracks were found in the magnesium matrix near the claddings (Fig. 6b).

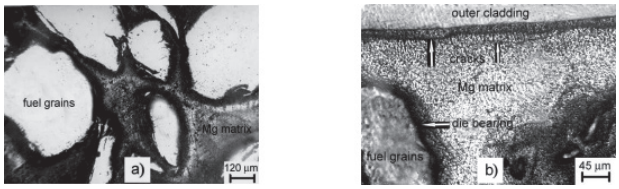

Figure 6. (**a**) Microstructure of the U-Mo fuel and (**b**) contact zone of the fuel composition with the cladding of fuel element 33.

Metallographic study of the fuel elements with UO_2 fuel

In the lower section of the Zh-72 and Zh-52 fuel elements, examined immediately after irradiation, only small pittings up to 3 μm deep were found on the outer side of the outer cladding, while there were no defects on the inner side of the cladding.

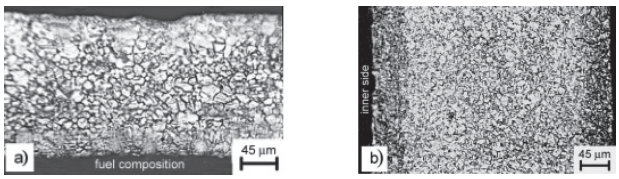

Figure 7. Microstructure of the material of the outer (**a**) and inner (**b**) cladding of fuel element 38 with UO_2 + Mg fuel after irradiation and storage for 15–22 years.

On the outer side of the outer cladding of fuel elements 8 and 38, after storage for 15–22 years, in the lower, middle and upper parts, separate areas with corrosion defects in the form of uneven frontal corrosion, pits and pitting were found, the depth of which was 6–15 μm (Fig. 7a). On the inner side of the outer cladding (on the side of the cladding contact with the fuel), depressions 4–18 μm deep were observed.

As noted above, fuel elements 8 and 38 differed in surface color: fuel element 38 had a dark gray surface, fuel

element 8 had a light surface. Metallography showed that in the central part of the outer cladding of fuel element 38 with a dark gray surface in a section 100–120 μm wide, along the grain boundaries, numerous precipitates were found (Fig. 7a), while in the cladding material of fuel element 8 with a light surface there were no precipitates along the grain boundaries.

It should be noted that the surface of the three fuel elements of the spent fuel assembly Zh-72 (without long-term storage) was dark gray, while the surface of the fourth fuel element was light silvery. Numerous precipitates were also observed along the grain boundaries in the middle part of the outer cladding of the fuel elements of this SFA with the dark surface along the grain boundaries, while there were no such precipitates in the cladding material of the fuel element with the light surface. Most likely, the dark surface of the irradiated fuel elements after long-term storage and the presence of precipitates along the grain boundaries in the material of the outer cladding of these fuel elements is due not to their operation in the reactor or long-term storage but to the technology of manufacturing these fuel elements.

On the inner side of the inner cladding of fuel elements 8 and 38, in some areas of the perimeter, uneven frontal corrosion with a depth of 7–20 μm, pits and pittings up to 15 μm deep are observed (Fig. 7b). On the outer side of the inner cladding (from the side of contact with the fuel composition) of the fuel elements, in some areas of their perimeter, there were depressions with a length along the perimeter of 50–100 μm and a depth of up to 10 μm.

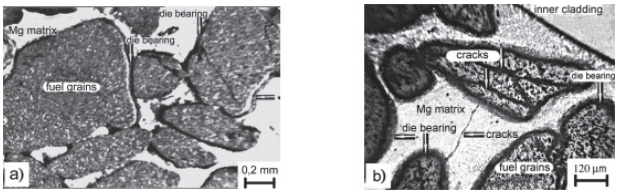

Figure 8. Microstructure of the material of the fuel composition of the irradiated fuel elements with UO_2 fuel: (**a**) before and (**b**) after long-term storage.

In the fuel composition of fuel elements 8 and 38 in the magnesium matrix around the fuel grains, dark gray belts 20–45 μm wide and microcracks in the fuel particles and the magnesium matrix were found (Fig. 8b). In the fuel composition of the irradiated fuel elements examined immediately after irradiation, there were no microcracks and the thickness of the belts around the fuel particles was 10–30 μm (Fig. 8a).

The belts around the fuel particles in the matrix of the dispersive fuel elements were formed during irradiation in the reactor as a result of damage to the matrix material by fission products (Weber and Girsh 1956; Samoylov et al. 1969). The thickness of these belts could also increase during long-term storage as a result of further damage to the matrix by α-particles emitted from the fuel grains.

Mechanical properties of the fuel element claddings

Table 4 shows the results of short-term mechanical tests of the cladding material of the fuel elements with U-Mo fuel immediately after their irradiation and storage for 38 years.

According to reference data, the tensile strength of the non-irradiated ring specimens made of 1X18H9T steel tested at 25 °C is 590 MPa, and the percentage elongation is 32%. As the irradiation dose becomes higher, the strength properties of the cladding increase (the tensile strength increases to 1040 MPa), while the plasticity decreases to 22%. It should be noted that in (Ibragimov et al. 1964) during mechanical tests of the fuel cladding material with U-Mo fuel of three SFAs (fuel burnup 6.7–33.5 MW day/kg) immediately after irradiation, it was found that the plasticity of the outer cladding is less than that of the inner cladding and is 3–21% and 11.5–28%, respectively. A lower plasticity value of the outer cladding compared to the inner cladding, equal to 8.4–9.2 and 12%, respectively, was also observed in the study of fuel elements of a similar type irradiated in the Beloyarsk NPP reactor to a burnup of 8–12 MW day/kg (Tsykanov and Davydov 1977).

As a result of long-term storage, the strength properties of the cladding material remain at a high level (σ_B = 990–1080 MPa), while the plasticity of the outer cladding is significantly reduced (δ_r = 5.5%), while the plasticity of the inner cladding remains as before, at a high level (δ_r = 21–25%). Short-term mechanical tests of the cladding of fuel element 33 after the maximum storage period were not carried out, but the brittle fracture of the outer cladding of the fuel el-

Table 4. Short-term mechanical properties of the cladding material of the fuel elements with U-Mo fuel at a test temperature of 25 °C

SFA No.	D-85 after irradiation	AMK-248 after irradiation and 38 years of storage	
Cladding	Outer	Outer	Inner
Sample location	850 mm from the fuel element bottom	850 mm from the fuel element bottom	
Fluence of fast neutrons, 10^{24} m^{-2}	2.05	1.45	
σ_B, MPa	1040 ± 20	990 ± 20	855 ± 35
σ_{02}, MPa	1000 ± 10	810 ± 30	655 ± 25
δ_r, MPa	22 ± 1	9 ± 4	25 ± 5
Sample location	100 mm from the fuel element bottom	200 mm from the fuel element bottom	
Fluence of fast neutrons, 10^{24} m^{-2}	1.3	1.1	
σ_B, MPa	1000 ± 10	1080 ± 10	845 ± 10
σ_{02}, MPa	905 ± 10	895 ± 40	700 ± 30
δ_r, MPa	21 ± 3	5.5 ± 0.5	21 ± 2

ement during the preparation of samples allows us to state that the plasticity of the cladding is close to zero.

Conclusion

The studies of 50 fuel elements with U-Mo fuel irradiated in the AM reactor after storage for 45–55 years and 50 fuel rods with UO_2 fuel after storage for 15–22 years allow us to draw the following conclusions:

1. No visible defects were found on the surface of the outer cladding of all the examined fuel elements with the two different fuel compositions.
2. However, in the fuel elements with U-Mo fuel, an increase in the diameter of the outer cladding is observed. This is most noticeable in the upper part of the fuel element. Storage of the fuel elements with UO_2 fuel for 15–22 years does not lead to a change in their diameter within the measurement accuracy.
3. In some areas of the external surface of the outer cladding and the internal surface of the inner cladding of the fuel elements with U-Mo+Mg and UO_2+Mg fuel compositions, after long-term storage, defects are found in the form of intergranular and irregular frontal corrosion, pits and pittings up to 20 μm deep. No interaction is found at the points of contact between the fuel claddings and the fuel composition of the layers. There is no noticeable decrease in the thickness of the outer and inner claddings of the fuel elements after long-term storage, nor does the thickness of the claddings at the locations of defects go beyond its minimum initial value, taking into account the technological tolerance for variations in thickness.
4. As a result of long-term storage for both types of fuel elements, the formation of a few microcracks was found both in the fuel grains and in the magnesium matrix.
5. Storage of fuel elements with U-Mo fuel for 45–55 years leads to a gradual degradation of the mechanical characteristics of their outer claddings, due to which their plastic characteristics are significantly reduced.

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