NATURE & SCIENCE International Online Scientific Journal. Impact Factor: 2.509. 2024 / Volume: 6 Issue: 8 / 38-43
ISSN: 2707-1146
e-ISSN: 2709-4189

https://doi.org/10.36719/2707-1146/47/38-43

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Metal-Based Matrix Materials

Abstract

One of the components of the fuel composition of dispersion nuclear fuel is a non-fissile material (matrix), which ensures its high radiation resistance. Despite the fact that dispersion nuclear fuel is used in reactors of various purposes (research, power, nuclear power plants, etc.), the operating conditions of which vary significantly, there are a number of requirements that must be taken into account when choosing the matrix material. Metals (aluminum and zirconium, as well as their alloys) have found wide application as matrix material in the production of dispersion-type fuel elements for research reactors and reactors of naval nuclear power plants (NPPs). Stainless steel matrix is used in dispersion nuclear fuel in power reactor plants.

Keywords: *metal, matrix, materials, components, composition*

Introduction

In table 1 provides data on the absorption cross-section of thermal neutrons of metals. The thermal neutron absorption cross section of alloys is defined as the sum of the products of the nuclear concentration and the absorption cross section for each component of the alloy, divided by the total number of nuclei in a gram of alloy (Degueldre, 1999, p. 274).

Table 1.
Absorption cross-sections of thermal neutrons of metals

Elements	Thermal neutron absorption cross section 10^{-24}sm^2
Beryllium	$10 \cdot 10^{-2}$
Carbon	$3,7 \cdot 10^{-3}$
Nitrogen	1,9
Oxygen	$2,0 \cdot 10^{-4}$
Magnesium	$7,0 \cdot 10^{-2}$
Aluminum	0,24
Silicon	0,16
Zirconium	0,18
Niobium	1,16

ISSN: 2707-1146
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Chromium	3,1
Titanium	5,8
Vanadium	5,0
Manganese	13,2
Iron	2,6
Nickel	4,6
Copper	3,8
Molybdenum	2,7
Tungsten	19,2
Tantalum	21,0

Aluminum and its alloys

Aluminum and its alloys are widely used as a matrix of dispersion fuel composition for research reactor fuel elements due to good nuclear and thermal properties, as well as excellent technological qualities. However, aluminum has low strength and unsatisfactory corrosion resistance. At the operating temperatures of most research reactors (100–150 °C), these disadvantages have little effect on the performance of fuel elements (Skorov, 1979, p. 344).

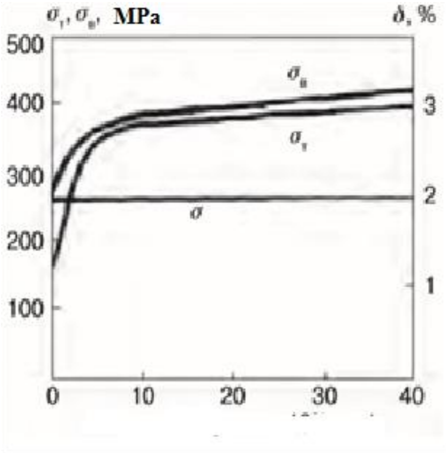
To improve the mechanical properties of aluminum, it is alloyed with magnesium, zinc, silicon, copper, silver, lithium, and gallium. The greatest effect in improving the mechanical properties of aluminum is achieved with simultaneous alloying with several elements. Aluminum alloys after heat treatment with precipitation hardening have good long-term mechanical properties (Tsykanov, 2000, p. 249).

Table 2.
Mechanical properties of some aluminum alloys before and after irradiation with a neutron flux

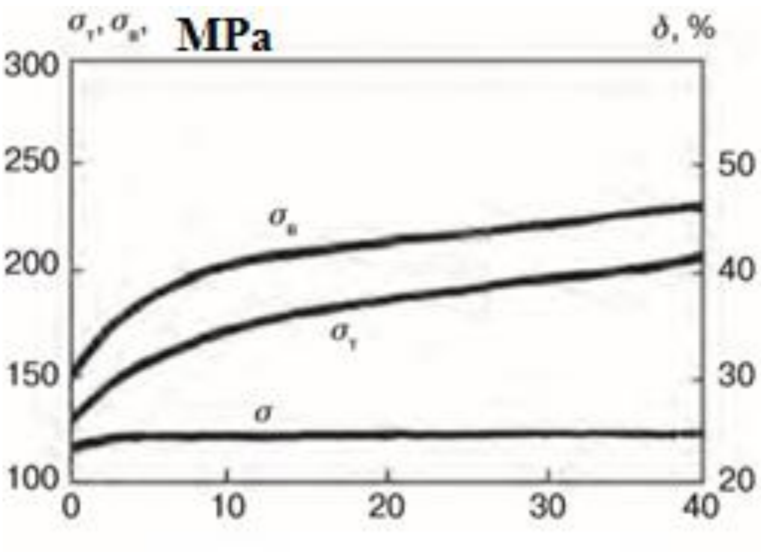
Alloy brand	□, *MPaB*	□0,2□, *MPa*	□ , %
	Before/after irradiation	Before/after irradiation	Before/after irradiation
AD1	95/180	47/120	38/21
SAB– 1	290/330	245/280	11/10
6061 (USA)	315/350	270/310	17,5/16

Under reactor irradiation conditions, the properties of structural materials change, and their plasticity is greatly reduced. Reactor irradiation has little effect on the mechanical properties of aluminum and its alloys. It is noteworthy that aluminum and its alloys practically do not lose plasticity when irradiated, but the strength of aluminum and alloys increases. The effect of irradiation on the mechanical properties of aluminum and its alloys is characterized by the data presented in figure 1.

ISSN: 2707-1146
e-ISSN: 2709-4189



a)



b)

Figure 1. Effect of irradiation on the mechanical properties of technical aluminum (a) and Al + 7 % Si alloy (b)

From the analysis of the presented dependencies (Figure 1) it follows that the greatest change in properties occurs up to a neutron flux of (3–5)•1021 cm–2. The available data show that technical aluminum containing iron, silicon, copper and other impurities can be successfully used as a matrix of the fuel composition of dispersion nuclear fuel up to temperatures of 100–130 °C. Technical aluminum with the addition of nickel is used as a matrix of the fuel composition of dispersion nuclear fuel in nuclear power reactors with pressurized water at temperatures up to 215–230 °C (Samoilov, 1982, p. 224).

Zirconium and its alloys

Zirconium and its alloys are widely used in nuclear power engineering due to the combination of their properties: nuclear, chemical and technological. However, these properties manifest themselves if zirconium is purified from hafnium, with which it is usually found in nature. Before use in nuclear technology, zirconium is usually subjected to iodine refining. Zirconium has found wide application as a matrix material in reactors of naval nuclear power plants (Beskorovayny, 1995: 324).

Currently, the existence of three phases of zirconium has been established: α-phase (T > 862 °C), β (T = 862–1855 °C) and unstable ω-phase (P > 6 GPa). During phase transformation, volumetric changes occur, which must be taken into account when determining the operating conditions of zirconium products (EMT, 1964, p. 127).

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Figure 2 shows the effect of alloying elements on the strength of zirconium. The most important alloying elements used in zirconium alloys are niobium and tin.

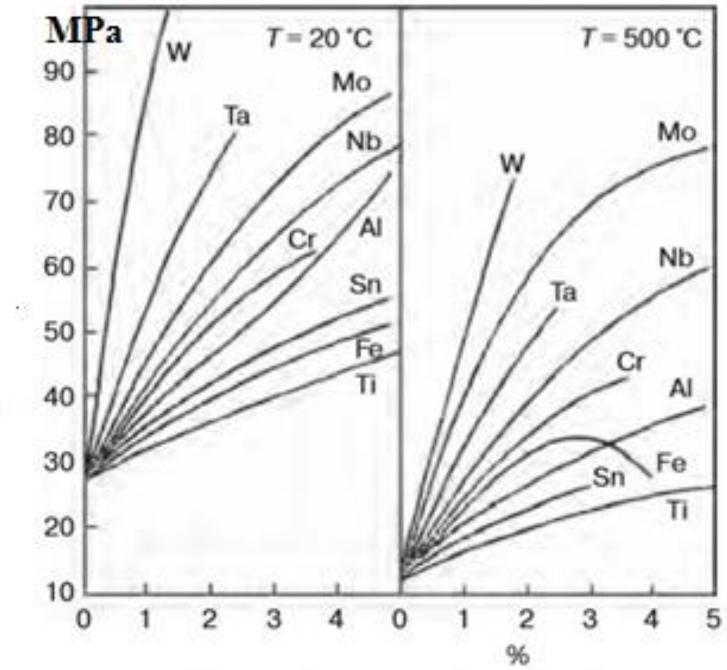

Figure 2. Effect of alloying element content on the strength of zirconium at different temperatures

As follows from the data in Figure 2, niobium, having a relatively small cross-section for capturing thermal neutrons, significantly increases the strength of zirconium both at room temperature and at 500 °C. In our country, zirconium alloys with a niobium content of 1 % (E110) and 2.5 % (E125) are mainly used. In the USA, complex alloys are used (Eidenson, 1969, p. 352).

Irradiation has a significant effect on the properties of zirconium and its alloys. When zirconium alloys are irradiated, they are strengthened with a simultaneous decrease in ductility.

Table 3 shows the data characterizing the effect of irradiation on the mechanical properties of pure zirconium. The data in Table 3 show that the change in the properties of zirconium as a result of irradiation is significant (Polmear, 2008, p. 464).

Table 3.
Effect of irradiation on the mechanical properties (at 20 °C) of zirconium annealed at 650 °C for 30 minutes

Irradiation temperature, ∘ *C*	Fast Neutron Stream 1023 neutron/m^2	□, *MPaB*		□, *MPa* 0,2		□ , %	
		Before irradiation	After	Before irradiation	After	Before irradiation	After
80	0,3	300	330	160	220	35	26
80	4,0	265	320	160	260	34	20
80	10,0	265	360	160	340	34	15
300	0,4	260	280	150	220	34	25

At a fast neutron flux of 1023 neutrons/m2, the ultimate strength increases by a maximum of 36 %, the yield strength by 112 %, and the elongation decreases by 68 %. According to, the zirconium alloy E635 is practically not subject to radiation growth at an irradiation temperature of 80 and 300 °C (Calliot, 1963, p. 1).

In nuclear technology, zirconium and its alloys are widely used as fuel element cladding materials. Information on the use of these materials as inert matrices for nuclear fuel dispersion compositions is extremely limited (Samsonov, 1978, p. 472).

Nickel

Nickel was used as a matrix for the dispersion fuel composition in the SM-2 reactor. It does not undergo allotropic transformations. The thermal conductivity of nickel is higher than that of stainless steel (CSNCT, 2001, p. 23).

At a temperature of 100 °C, its thermal conductivity is 83 W/(m•°C). A distinctive feature of nickel is the combination of high strength and elastic modulus with good plasticity (σ = 30–40 %) (SPR, 2001, p. 15).

Along with thermal conductivity, long-term strength characteristics have a significant impact on the performance of the dispersion fuel composition. Figure 3 shows the long-term strength data, and Figure 4 shows the dependence of the creep rate of nickel on stress at different temperatures (Alekseev, 2013, p. 240).

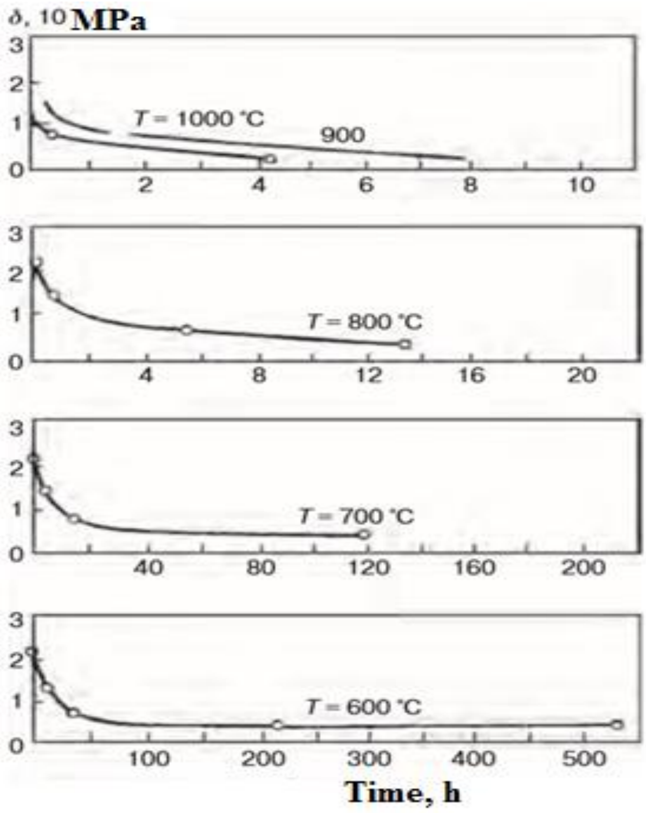

Figure 3. Long-term strength of nickel at different temperatures

Conclusion

Study of the effect of neutron irradiation in a reactor on the mechanical properties of nickel showed a sharp drop in plasticity in irradiated nickel at a temperature of 600 °C and above, while strength increases. Figure 5 shows the effect of temperature on the mechanical properties of technical nickel, unirradiated and irradiated to a neutron flux of 1.7•1020 cm–2, at a temperature of 150–200 °C. According to the data in Figure 5, up to a temperature of 400 °C, typical for research reactors, the plastic properties of nickel remain quite high.

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Received: 02.05.2024
Revised: 16.07.2024
Accepted: 28.07.2024
Published: 20.08.2024