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Status of international collaborative efforts on selected ITER materials

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Abstract

The paper presents an overview of the performance of refractory metals, beryllium, and copper alloys, for the international thermonuclear experimental reactor (ITER) high heat flux structures. High temperature brazing, hot isostatic pressing (HIP), friction welding, explosive bonding, and other methods were explored to join copper alloys to 316 stainless steel for first wall and limiter applications. It is concluded that the main material problems for the ITER high heat flux components are: (a) degradation of properties after the manufacturing cycle (especially for Be/Cu and Cu/stainless steel (SS) joints); (b) helium embrittlement of Be, and Cu, and; (c) radiation-induced loss of fracture toughness for Be, W, and Cu alloys.

1. Introduction

High thermal conductivity materials are proposed as plasma facing and heat sink materials in high heat flux components of the international thermonuclear experimental reactor (ITER) such as the first wall, divertor, and limiter. Tungsten, beryllium, and C–C composites are favored as plasma facing materials, and high strength copper alloys are preferred as structural and heat sink materials [1]. At the beginning of the ITER activity, the database on the irradiation performance of these materials was very limited.

During the past six years the Russian Federation (RF) Home Team jointly with the United States (US) Home Team, and in cooperation with the European Community (EU) Home Team and the Japan (JA) Home Team have carried out numerous irradiation experiments using Russian nuclear reactors such as the SM-2 (mixed spectrum reactor) and the BOR-60 (fast spectrum reactor) in Dimitrovgrad.

High strength copper alloys GlidCop Al25 and Cu–Cr–Zr are the main candidate materials for the ITER high heat flux components [2,3], but the database on mechanical properties of these materials in the operation temperature range 100–400°C is limited. It is also evident that, in generating these data, the impact of manufacturing technology on the properties of copper alloys must be accounted for [4].

The scope of the investigations required to substantiate lifetime predictions for copper alloys and Cu/Stainless steel (SS) joints is summarized in Table 1.

2. Copper alloys and Cu/SS joints

Based on the results of a six-year investigation through close cooperation between the RF, US, EU and JA teams, a database was created on the properties of copper alloys and Cu/SS joints. These experiments provided, for the first time, data on the radiation resistance of the GlidCop Al25 IG alloy recommended by the Joint Central Team as the main material for the ITER heat sink system [5].

2.1. Summary of the irradiation performance of the first and second generations of copper base alloys and copper/steel joints

2.1.1. First generation of joints

Seventeen irradiation capsules with more than 3600 specimens of copper alloys and copper/steel joints were

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Table 1
Available database on copper alloys before the beginning of ITER activity (1993)

Material	Physical properties[a]					Tensile properties[a]				Fracture toughness[a]		LCF[a]		Creep[a]		Irradiation effects
	E	α	C	λ	ρ	σ_{ij}	σ_y	δ_{unif}	δ_{total}	Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated	0.1–9 dpa $T_{irr} = 100$–$350°C$
Glipcop Al25	+	+	+	+	+	+	+	+	+	Very limited	No	No	No	Very limited	No	Very limited
Glipcop Al15	+	+	+	+	+	+	+	+	+	Very limited	No	No	No	No	No	Very limited
MAGT0.2	+	+	+	+	+	+	+	+	+	Very limited	No	No	No	No	No	Very limited
Glipcop IG	+	+	+	+	+	No	No	No	No	No	No	No	No	No	No	No
Cu–Cr–Zr (SA + c.w. + aged)	+	+	+	+	+	+	+	+	+	No	No	Very limited	No	Very limited	No	Limited
Cu–Cr–Zr–Mg (SA + c.w. + aged)	+	+	+	+	+	+	+	+	+	No	No	Very limited	No	No	No	Very limited
Cu–Cr–Zr IG	+	+	+	+	+	No	No	No	No	No	No	No	No	No	No	No

[a] $T_{test} = 20\ldots350°C$.

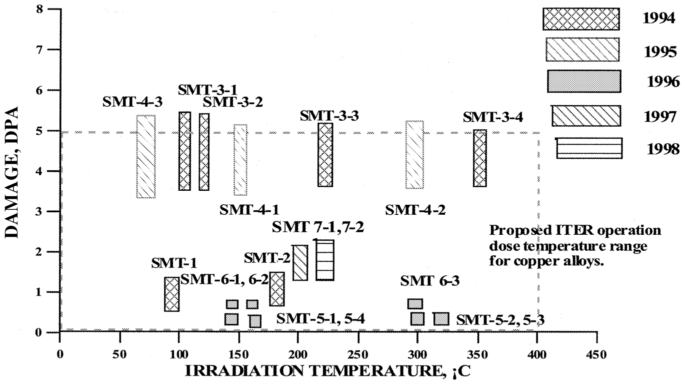

Fig. 1. Summary of irradiation experiments on copper alloys, Cu/SS joints, and PFC materials in the SM-2 reactor during the period 1994–1998.

irradiated. As seen from Fig. 1, the irradiations cover almost the full operation range of copper alloys and copper/steel joints in ITER – 0.2–2 dpa and irradiation temperature range 80–300°C.

Our results demonstrated that GlidCop Al25 IG had satisfactory strength and maintained satisfactory uniform strain (>10%) for doses up to 0.2 dpa in the temperature range 150–300°C [5]. This behavior distinguishes this alloy favorably from the initial version of GlidCop Al25, which suffers from low temperature radiation-induced embrittlement [3].

A comparison between three different technologies for copper/steel joints made by the hot isostatic pressing (HIP) method in the US, RF, and EU showed that the HIP technology and the properties of the base copper alloy had a profound effect on the material properties after HIP, both in the initial and irradiated states [5–7]. It was concluded that GlidCop Al25 IG holds good promise for radiation resistance both in the manufactured condition and after HIPping.

The study of high strength (up to 1000 MPa) Cu–Be, Cu–Ni–Be and Cu–Cr–Ni–Si alloys showed these alloys to be highly susceptible to helium embrittlement and intergranular fracture at increased temperatures [8]. Taking into account the high rate of helium generation expected in ITER, the application of these materials is rather doubtful.

The investigations undertaken on the first generation of base alloys and joints permitted comparison of the behavior of oxide-dispersion-strengthened (ODS) and precipitation-hardened (PH) alloys for ITER applications [7].

In comparing the strength and ductility properties of the two best representatives of ODS and PH copper alloys, it can be shown that (a) at a test temperature (T_{test}) and irradiation temperature (T_{irr}) equal to 150°C, the Cu–Cr–Zr HIPped alloy has nearly a factor of two lower strength, but nearly ten times higher ductility than GlidCop Al25, and (b) at $T_{test} = T_{irr} = 300$°C, both the base alloy and HIPped Cu–Cr–Zr have nearly a ten-fold advantage in uniform elongation and practically the same strength as GlidCop Al25.

It is concluded that the Cu–Cr–Zr alloy has an apparent advantage for operation at 300°C, as it is the only material which does not tend to exhibit brittle failure after HIP and irradiation.

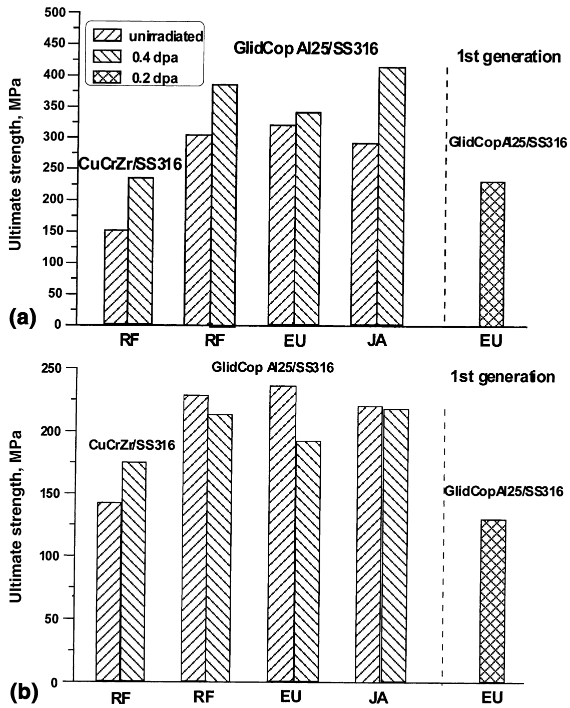

Fig. 2. Effect of neutron irradiation on the ultimate strength of GlidCop Al25/SS316 HIP joints and Cu–Cr–Zr/SS316 HIP joints, unirradiated and irradiated to 0.2 and 0.4 dpa at $T_{irr} = 150$°C, $T_{test} = 150$°C (a); unirradiated and irradiated to 0.2 and 0.4 dpa at $T_{irr} = 300$°C, $T_{test} = 300$°C (b).

2.1.2. Second generation of joints

Modifications of the HIP procedure resulted in significant improvements in the radiation resistance of GlidCop Al25 IG/316SS type joints, and enhancement of the strength properties of the joints to a level of the base alloy (ultimate tensile strength ~400 MPa) at 150°C [7,8].

The results of the investigation of the second generation of GlidCop Al25/316LN (EU, JA, RF, US) [HIP] demonstrated (Fig. 2) that these joints have a high maximum fracture strength (about 400–450 MPa) at $T_{test} = 20$–300°C, when irradiated to 0.4 dpa, and are not susceptible to brittle failure at $T_{test} = 150$°C and 300°C, as distinct from the joints of the first generation. The HIPped Cu–Cr–Zr (IG)/316SS joints demonstrated the best ductility ($\delta_{tot} \sim 10\%$) after irradiation, compared with all other joints, and they demonstrate a satisfactory level of strength ($\sigma_u \sim 200$ MPa) even after the HIP procedure [7–9].

3. Summary of progress in the development of Be and W armor and joining techniques with Cu-based heat sink

Various Russian and foreign grades of Be and W have been investigated as plasma facing materials. During the period 1994–1998, numerous tests in conditions specific to the fusion environment were performed (Table 2). It was shown that most grades of Be have an excellent thermal strength and that the mechanical integrity of tiles is preserved under nominal thermal cycling, under the impact of plasma disruption, and under combinations of these conditions. Surface cracks produced by thermal loads are not dangerous because they are stable under multiple and longtime loading [10,11]. There are no significant differences in behavior among the best Russian and American grades of Be during thermal cycling and after neutron irradiation; thus, there is some flexibility in the selection of manufacturers [10–14].

In the recrystallized conditions, tungsten also showed good thermal stress resistance under high thermal loading conditions. Tungsten tiles preserve their heat conducting properties and their mechanical integrity under the combined action of thermal cycling and thermal shocks in plasma disruption simulations. Doped tungsten grades that have some advantage in their initial condition are, in general, comparable with lower cost powder metallurgy grades of pure tungsten. The lower cost grades also have the advantage of better activation properties and introduce fewer impurities into the plasma. However, the effects of neutron irradiation on the thermal strength of W alloys must still be investigated because they become brittle after small doses at low temperatures.

A wide range of technologies for joining Be and W to a Cu-based heat sink has been considered: brazing by

Table 2
The features of experiments with armor materials in fusion relevant conditions

Type of testing candidate materials	Comparative thermal fatigue testing (jointly with SNLA, US) [10]	Thermal shock testing at disruption simulation (jointly with FZJ Germany) [11,12]	Thermocycling after disruption [13]	Neutron irradiation influence on phys. –mech. properties [14]	Integrated in-pile testing [15]
Be-grades					
RF: DShG-200, TShG-56, TR-30, PVD-Be, others	Surface thermal loading (e-beam) ~ 250 MW/m^2	e-Beam loading ~ 5 MJ/m^2	Surface thermal loading (e-beam) ~ 5 MW/m^2	Fast mixed neutron spectrum $T_{irr} = 150$–800°C, D = 2–5 dpa, properties: UTS, YS, TE, UE, swelling, microstructure	*Completed experiment*: Surface heat load = 3 MW/m^2 $N = 100$ cycles D ~ 0.2 dpa
$\tau \sim 50$ ms	$\tau \sim 5$ ms	$\tau = 10/10$ s pulse/pause			
US: S65, others	$N = 2500$ cycles		$N = 1000$ cycles		*Under preparation*: Surface load 5–8 MW/m^2 $N = 1000$ cycles D ~ 0.2 dpa
W-grades					
RF: W–1 Mo (cast), single crystal, pure sint. W, W–Re (cast), others	Surface thermal loading (e-beam) ~ 250 MW/m^2	Plasma gun loading ~ 10 MJ/m^2	Surface loading ~ 15 MW/m^2		*Under preparation*: Surface load ~ 10 MW/m^2 $N = 1000$ cycles $D \sim 0.2$ dpa
	$\tau = 50$ ms	$\tau = 0.1$–0.4 ms	$\tau = 15/15$ s	–	
EU: $W–La_2O_3$, $W–CeO_2$, others	$N = 1.4 \times 10^4$ cycles		$N = 1000$ cycles		
US: $W–La_2O_3$, others					

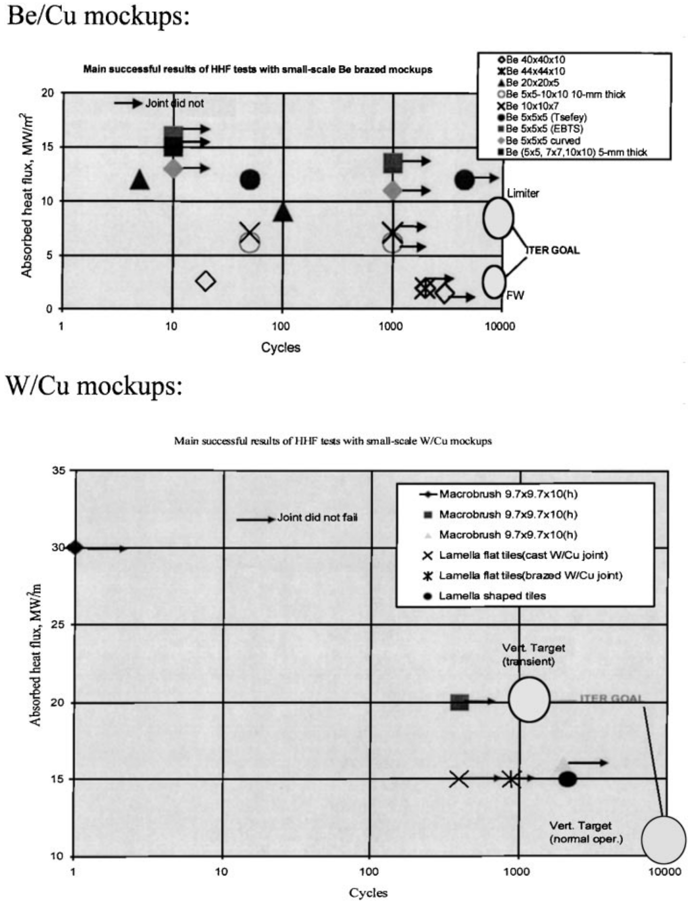

Fig. 3. Main results of high heat flux tests with small-scale mockups.

different brazing alloys, diffusion bonding, explosion welding, joint rolling, Cu-casting, HIP, and others [10,13,15,16]. The reliability of joints was estimated on the basis of thermal cycling tests of multilayered mockups with active cooling. Mechanical testing and structural analysis were used as additional methods. As a result of these investigations, the following joining technologies were recommended [10–16]:

- Be–Cu alloy joints: fast brazing by Cu–In–Sn–Ni alloy (brazing temperature T_{br} ~750°C),
- W–Cu joints: active metal casting of Cu on W,
- Cu–Cu alloy joints: fast brazing by Cu–In–Sn–Ni alloy (T_{br} ~750°C), and
- W–Cu alloy joints: fast brazing by CuMn alloy (T_{br} ~900°C).

Fast brazing inhibits the growth of brittle intermetallic compounds (Be_2Cu and others) and also limits the overaging of the PH–Cu alloy (Cu–Cr–Zr). The results of high heat flux testing of small Be–Cu and W–Cu mockups (Fig. 3) [11,17] have confirmed the reliability of the technologies selected.

4. Conclusion

Although rapid progress has been achieved in these areas, it is clear that further work is required to improve the unirradiated properties of individual materials and to further substantiate the irradiation performance of bonded structures for ITER applications.

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