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Section 6. Beryllium

Beryllium for fusion application – recent results

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Abstract

The main issues for the application of beryllium in fusion reactors are analyzed taking into account the latest results since the ICFRM-9 (Colorado, USA, October 1999) and presented at 5th IEA Be Workshop (10–12 October 2001, Moscow, Russia). Considerable progress has been made recently in understanding the problems connected with the selection of the beryllium grades for different applications, characterization of the beryllium at relevant operational conditions (irradiation effects, thermal fatigue, etc.), and development of required manufacturing technologies. The key remaining problems related to the application of beryllium as an armour in near-term fusion reactors (e.g. ITER) are discussed. The features of the application of beryllium and beryllides as a neutron multiplier in the breeder blanket for power reactors (e.g. DEMO) in pebble-bed form are described.

1. Introduction

Nowadays, three main areas of beryllium application in fusion reactors may be outlined, which are characterized by specific sets of requirements [1–3]. These areas are (1) first wall protective armour, (2) neutron multiplier in a solid breeding blanket, and (3) some in-vessel components of the plasma diagnostic system.

Beryllium has been used with success as plasma facing material in UNITOR and ISX-B. Extensive experience has been acquired in JET where use of Be significantly improved plasma performance [4]. For ITER application, beryllium is selected as reference armour material for the first wall and port limiter because of its low main plasma pollution, oxygen gettering capability, acceptable erosion lifetime and capability to protect the actively cooled first wall structure [5]. For the breeding blanket application, beryllium is required to increase the tritium breeding ratio (TBR) performance. In the helium cooled pebble bed (HCPB) blanket design, beryllium is used in the form of a single size pebble bed, which gives such advantages as elimination of the swelling effect and enhancing of tritium release. Beryllium safety aspects, such as its reactivity with steam and water under accident conditions for ITER and DEMO, are special issues for successful operation of fusion reactors. Neutron irradiation effects on the degradation of the properties and performance of the beryllium in different forms for near- and long-term applications are also very important.

Various aspects of beryllium application in fusion were discussed recently during 5th IEA Be Workshop,

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10–12 October 2001, Moscow, Russia [3]. The main results from this Workshop and status of the current R&D programs after ICFRM-9 [4] are presented in this paper.

2. First wall armour for ITER

Beryllium has been chosen as the armour material for the first wall and port limiter. The main reasons for the selection of Be are the low effect on plasma contamination, low radiative power losses, good oxygen gettering ability, absence of chemical sputtering (in comparison with carbon), low bulk tritium inventory, and the possibility of in situ (or in hot-cell) repair of damaged surfaces using plasma spray. Its main function is to protect the actively cooled wall structures from high heat fluxes and direct contact with plasma in order to satisfy the required component lifetime and plasma compatibility. Commercially available beryllium grades from the USA and from the RF have been evaluated as candidate materials. Among various grades, S-65C VHP (vacuum hot pressed) grade (manufacturer Brush Wellman Inc., USA) has been selected as a reference grade. The main reasons were lowest BeO and metallic impurity content among the other structural grades; high elevated temperature ductility and excellent low cycling thermal fatigue performance, and thermal shock resistance. The similar grade DShG-200 (manufactured by the RF) was selected as a backup. The base properties of these grades are well documented [6]. The key issue for application of Be as armour is its behavior after neutron irradiation and thermal transient events. Neutron irradiation typically leads to degradation of Be properties [7]. Embrittlement of Be at low temperature could lead to brittle destruction of the tiles and affect thermal erosion of Be during transient events. The study of thermal erosion and damage of neutron irradiated Be S-65C VHP confirms that this grade has better thermal shock resistance. The use of Be tiles without critical defects that may result in crack initiation could partly solve this problem.

One of the key problems in the application of Be armour in ITER first wall was the development of the effective bonding of Be–Cu alloys. The main issue is that Be reacts with almost all metals at moderate and high temperatures and forms brittle intermetallic phases that are detrimental to joint reliability and fatigue lifetime. To solve this problem and to provide good quality Be/Cu joints, different approaches have been studied.

In recent years, the main efforts were aimed at the optimization of joining technology and manufacturing and testing of different mock-ups. The most recent results in this field are outlined below.

The EU Home Team has recently manufactured primary wall, baffle, and limiter mock-ups [8]. The primary wall mock-up has steel cooling tubes and was manufactured by brazing at 780 °C. It endured 1000 cycles at 1.5 MW/m^2 (ITER design value 0.5–0.8 MW/m^2). The baffle mock-up (Fig. 1) had a dispersion strengthened (DS) Cu cooling tube with a 0.2 mm thick steel liner. It was manufactured by hot isostatic pressing (HIPing) at 580 °C and 100 MPa for 2 h. It survived 1000 cycles at 5 MW/m^2 (ITER design value 2–3 MW/m^2).

With the same HIP technology, a limiter mock-up was manufactured. It had DS-Cu cooling tube without the steel liner. It was tested for 1000 cycles at 7 MW/m^2 plus 740 cycles at 10 MW/m^2 before failure (ITER design value 8 MW/m^2).

Vertical displacement event (VDE) simulations were carried out on miniaturized Be–CuCrZr mock-ups with Cu–Mn braze [9]. Three Be–CuCrZr mock-ups with 3-, 5- and 8-mm beryllium armour were loaded by VDE shots of 100 and 300 ms. The power density of each shot was 60 MJ/m^2. In spite of heavy melting of the beryllium, no indications for a detachment of the armour were observed (Fig. 2). All three mock-ups successfully survived the thermal fatigue test that followed (1000 cycles at 5 MW/m^2) without detachment.

In order to avoid a number of problems inherent to the HIP method (precise machining, canning, possible annealing of CuCrZr), the RF Home Team proposed an alternative technology for first wall manufacturing, which uses so-called 'fast brazing' of Be tiles in high vacuum at a temperature of 700 °C [10]. The required high heating rate (≃1.5 K/s) was provided by the e-beam TSEFEY facility (Fig. 3). The Be/Cu joints in

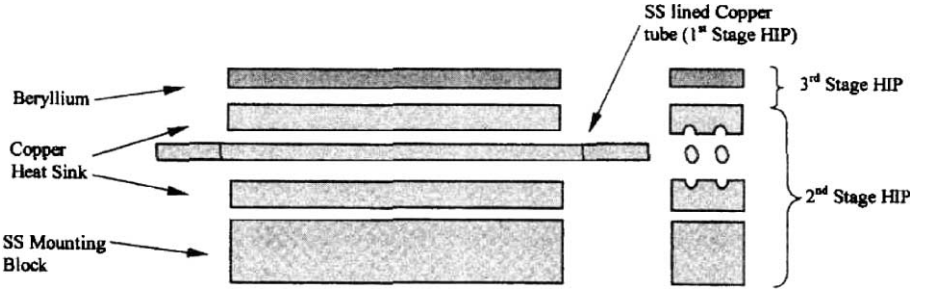

Fig. 1. Manufacturing sequence of the baffle mock-up [8].

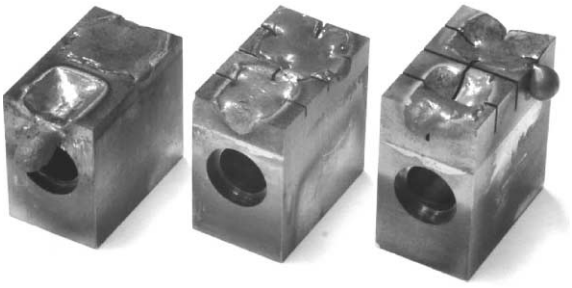

Fig. 2. Be–CuCrZr mock-ups after VDE testing [9]. Craters in front: $t = 300$ ms; craters in the back: $t = 100$ ms; left: 3 mm, middle: 5 mm, right: 8 mm Be armour.

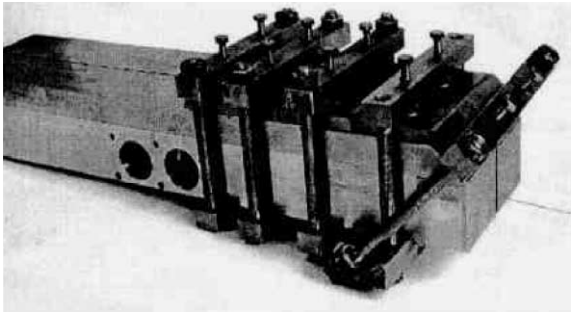

Fig. 3. FW mock-up with Be tiles assembled for brazing [10].

this mock-up have survived 5000 cycles at heat flux $\simeq$1 MW/m^2 and 500 cycles at 1.5 MW/m^2.

Joining by HIP between beryllium and DS-Cu with an interlayer has been investigated in Japan [11,12]. An interlayer of single or several thin layers of Cr, Ti, Cu and Al was deposited on the bonding surface of Be or solution-treated CuCrZr. The Be and CuCrZr were bonded with the interlayer by HIP (temperature: 520–610 °C, pressure: 150 MPa). The bonding that applied the interlayer of Cr/Cu showed the best performance (the specimens of the bending test were not broken even under the stress about 500 MPa). Heat removal and thermal cycle tests with heat flux of 5 MW/m^2 for 1000 cycles were carried out. It should be noted that a mock-up with an Al interlayer showed good heat removal performance including the ITER normal operation.

Based on the ITER R&D several joining technologies, which could be applied for the manufacturing of the ITER beryllium armoured components, have been developed. Some additional developments with the goal of increasing reliability and reduction of costs are still required.

3. Beryllium in breeding blanket

In a HCPB blanket for power reactors, the beryllium neutron multiplier and the breeder material are arranged as pebble beds. The maximum temperatures in the breeder and the beryllium pebble beds are in the range of 900 and 650 °C, respectively. Because of large temperature differences between pebble beds and the structural material, different thermal expansion coefficients, and irradiation effects, constrained strains occur. The large compressive stresses in the pebble beds might result in plastic deformations of pebbles. These deformations influence the thermal conductivity of the bed, which in turn determines the temperature distribution. Therefore, for the thermal–mechanical blanket design, the thermal conductivity as function of strain and stress must be reliably known. At Forschungszentrum Karlsruhe activities are on going to investigate these issues. Hot wire (HW) experiments were performed with an uniaxial compression test facility (UCT) in He atmosphere [13,14]. The pebble bed comprised an electrical heater with 1-mm outer diameter in order to calculate the effective thermal conductivity of the pebble bed at different pressures and temperatures.

Fig. 4 shows the stress–strain dependence for a beryllium pebble bed at 475 °C and the evaluated conductivities at different pressure levels for 1 mm Be pebbles manufactured by the Japanese Company 'NGK'. The increasing elastic and plastic pebble deformation is the reason for the increase of conductivity with increasing pressure. The conductivity essentially keeps the value reached at the highest pressure; only at small pressure do the values become smaller. This tendency indicates that the increase of conductivity during pressure increase is mainly caused by plastic deformations. In these experiments, a distinct increase of the thermal conductivity during creep was observed (from 11 to 14 W/m K).

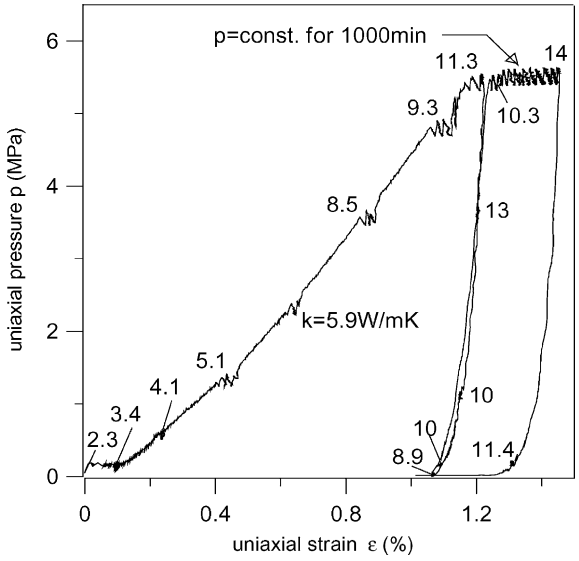

Fig. 4. Thermal conductivity of Be pebble bed for UCT at $T = 475$ °C [14].

Under similar conditions, the temperature dependence of the electrical resistance of a 1-mm single-size pebble bed was also investigated [15]. It was ascertained that after mechanical cycling the bed showed an irreversible behavior, and the electrical resistance never reached its initial value for zero pressure.

3.1. Beryllide

During the last years, the Japanese team studied the possibility of using beryllides (mainly $Be_{12}Ti$) as the neutron multiplier in a solid breeder blanket instead of beryllium. $Be_{12}Ti$ has advantages over beryllium from the perspectives of higher melting point, lower chemical reactivity with water steam, lower swelling and lower tritium inventory [16]. Preliminary neutronics estimation concerning DEMO blanket design showed that the TBR in the case of $Be_{12}Ti$ pebbles was 1.1, and this TBR is 10% smaller than TBR in blankets with Be pebbles [17].

Compatibility tests were performed [18], and their results showed the advantage of $Be_{12}Ti$ as the material for high temperature use:

- The thickness of reaction layer between $Be_{12}Ti$ and SS316LN was much smaller that that of Be and SS316LN. The thickness at 800 °C was one tenth of that for beryllium.
- The structure of the reaction layer for $Be_{12}Ti$ and SS316LN showed a characteristic difference from that for Be. $Be_{11}Fe$ was not observed in the layer for $Be_{12}Ti$ and SS316LN.
- The calculated effective thermal conductivity in the $Be_{12}Ti$ pebble bed was estimated close to that of beryllium (Fig. 5) [19].

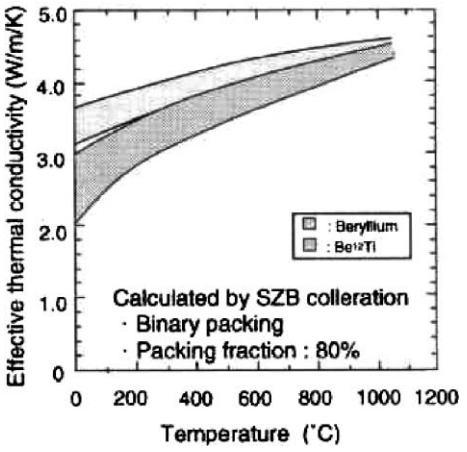

Fig. 5. Thermal conductivity of $Be_{12}Ti$ pebble bed calculated by SZB model [19].

Between now and 2005, the Japan team will seek the answer to the following question: 'How do beryllides work compared with beryllium metal?' [16].

4. Neutron irradiation effects

A number of data on radiation damage of beryllium are available in the literature, and the general behavior of beryllium at different conditions is quite well understood (see for example the latest reviews [7,20]). Nevertheless, some additional data were generated recently that provide some new important information.

Several Russian beryllium grades (TE-400, TE-56, TE-30, TIP, DIP), produced by different technologies, were irradiated in the SM and BOR-60 nuclear reactors at temperatures of 70–450 °C in the neutron fluence range of $(0.5–16) \times 10^{22}$ cm^{-2} ($E > 0.1$ MeV) [21]. After irradiation at 70 and 200 °C up to $(1.4–3.9) \times 10^{22}$ cm^{-2}, swelling ranged from 0.2% to 1.5% depending on the corresponding neutron fluence. Swelling of all the beryllium grades irradiated at 70 and 200 °C was in the range 0.2–1.5% and gradually increased with the neutron dose. No strict dependence of swelling on irradiation temperature was observed. There was strong embrittlement of all grades and a reduction of fracture stress, which was observed also earlier. Neutron irradiation up to $(1–6) \times 10^{22}$ cm^{-2} resulted in decrease of thermal conductivity (TE-56) by a factor 3–6 depending on the corresponding neutron fluence, while that decreased by 15% after irradiation at 400 °C up to 1.6×10^{22} cm^{-2}. Post-irradiation short annealing (500 °C, 1 h) led to the partial recovery of the thermal conductivity (from 53 to 150 W/m K) (Fig. 6). This reduction of the thermal conductivity could be explained only by generation of a large number of the defects, because the swelling of these materials remained at the level of several percents. This effect seems not important for ITER due to the much lower expected neutron damage dose, but for application in future reactors it should be taken into account.

The effects of irradiation dose on tritium and helium release were reported in [22]. Beryllium (TE-56) was irradiated with a neutron fluence of $(0.5–5) \times 10^{22}$ cm^{-2} ($E > 0.1$ MeV) at 70–100 °C. The total amount of helium accumulated in irradiated beryllium varied from 520 to 6000 appm. The increase of irradiation dose by a factor 8–10 resulted in a threshold shift of a sharp acceleration of helium release to a lower temperature, from 1075 to 700–775 °C, while the helium accumulation level significantly affected tritium release behavior. For the samples irradiated to $(4–5) \times 10^{22}$ cm^{-2}, maximum tritium release rate was concurrent with a sharp acceleration of helium release rate. The apparent diffusion coefficients of helium and tritium were calculated for the sample irradiated with a fluence $F_s = 4 \times 10^{22}$ cm^{-2}:

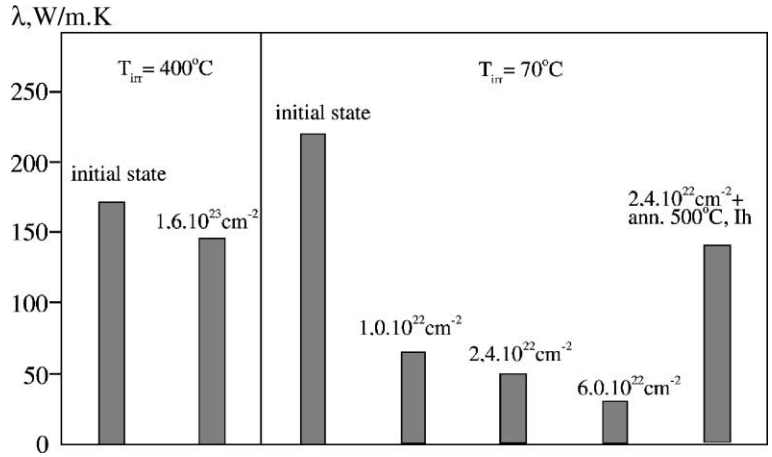

Fig. 6. Thermal conductivity of irradiated beryllium, measured at room temperature [21].

For tritium (m^2/s):

$$\ln D = -(1.1 \pm 0.8) - (123.9 \pm 3.6)/\mathrm{RT}. \quad (1)$$

For helium (m^2/s):

$$\ln D = -(23.1 \pm 0.7) - (61.3 \pm 2.9)/\mathrm{RT} \quad (300 < T < 750\ ^\circ\mathrm{C}). \quad (2)$$

The tritium release from $Be_{12}Ti$ irradiated at 330, 400 and 500 °C up to 4×10^{21} cm^{-2} was studied in [23]. The apparent diffusion coefficient of $Be_{12}Ti$ appears to be about two orders of magnitude higher than that of beryllium at 600–1100 °C. A small amount of released tritium from irradiated samples at 500 °C was observed because almost all tritium had already been released during the neutron irradiation. Swelling of $Be_{12}Ti$ under irradiation varied from 0.5% to 1%.

An in situ ion implantation experiment was performed for beryllium using the multi beam high voltage electron microscope [24] to evaluate He irradiation up to 20 000 appm and electron implantation using an ion accelerator as preliminary tests before actual irradiation testing. Microstructure evolution was evaluated in two kinds of beryllium that had a different impurity content as a function of irradiation temperature, dose and spontaneous He irradiation effects. Microhardness tests were performed to investigate mechanical property changes in beryllium before and after He irradiation up to 17 000 appm He. A peak was observed in the Vickers hardness for about 1000–3000 appm He, and it was practically stable up to 10 000 appm.

5. Beryllium safety

Beryllium is a toxic material. Inhalation of Be-aerosols is the most dangerous exposure. Non-activated Be-dust releases would come primarily from assembly and treatment of the reactor first wall. To ensure personnel protection, all the rooms, potentially contaminated with beryllium, are divided into three zones depending on a concentration of Be-aerosols in the air (C_{air}) and on the level of surface contamination with beryllium (C_{surf}) [25]:

	C_{air} (g/m^3)	C_{surf} (g/m^2)	
Uncontrolled zone	<0.01	<0.1	(3)
Controlled zone	0.01–0.2	0.1–10	(4)
Respiratory protection zone	>0.2	>10	(5)

Personnel have time-unlimited access to the uncontrolled zone. Only designated Be-workers are allowed access into controlled zone. Access time and protective equipment are determined by the activities with beryllium bearing equipment and the potential for airborne beryllium. A respiratory protection zone must be physically enclosed and outfitted with appropriate ventilation. Workers will require respiratory protection commensurate with the work and the hazards measured.

Nuclear transmutations result in accumulation of tritium in beryllium [20]:

$$^9\mathrm{Be} + \mathrm{n} \rightarrow {}^7\mathrm{Li} + {}^3\mathrm{H} \quad (6)$$

$$^9\mathrm{Be} + \mathrm{n} \rightarrow {}^6\mathrm{He} + {}^4\mathrm{He} \quad (7)$$

$$^6\mathrm{He} \rightarrow {}^6\mathrm{Li} + \beta\ (T_{1/2} = 0.8\ \mathrm{s}) \quad (8)$$

$$^6\mathrm{Li} + \mathrm{n} \rightarrow {}^4\mathrm{He} + {}^3\mathrm{H} \quad (9)$$

The first of these reactions has a threshold of about 12 MeV. The second one has the threshold of 600 keV and a strong resonance at 3 MeV. Therefore, tritium

production rate strongly depends on the neutron spectrum. The first reaction dominates in a fusion reactor. Under the neutron fluence of 0.5 MW a/m^2, 115 g of tritium will be bred in ITER first wall beryllium [25]. However, the main part of the tritium accumulates in Be-elements of a fusion reactor due to implantation of tritium particles having energy from several eV to some hundreds eV from plasma. Saturation effects limit the tritium mobile atom concentration at the near-surface Be-layers and thus inhibit diffusion of tritium into the bulk beryllium. The apparent cause is surface texturing and the development of interconnected porosity by migration of gaseous bubbles formed in the implantation layer. At ITER operating temperatures, implanted tritium never diffuses deep into beryllium. During a thermal excursion event, its mobility increases and some of the implanted tritium diffuses into the bulk of the material and becomes trapped. From the mobilization point of view, the neutron-bred tritium behaves as trapped tritium as well. At temperatures <500 °C trapped tritium is not mobilized from beryllium in any significant amount [26]. According to calculations [25,27], the total tritium inventory in the beryllium first wall armour due to implantation, diffusion, trapping, and breeding after 12 000 pulses will be of 100–250 g. This value essentially depends on trap density, which is expected to be from 0.1 to 1 at.%, and on trap energy.

Mobilization of implanted tritium occurs in two ways. One is diffusion, which is slow. The other is a non-diffusive burst release. The potential for tritium 'burst' mobilization from beryllium depends on temperature and microstructure, which itself depends on temperature, time at temperature, neutron damage/swelling and porosity. Normally, burst releases are not seen below 600 °C and occur following an incubation time of several hours. Such conditions are not reached in ITER off-normal situations [25].

However, at high temperature increase rates (20–100 K/s) the temperature of tritium burst release from beryllium irradiated with high neutron fluences ($E >$ 0.5 MeV) up to 10^{22} n/cm^2 decreases down to 450 °C [28]. This may apply to small fraction of the FW ($\simeq$1 m^2 or 0.2%) in case a VDE coincides with an accident [29].

It is expected that because of relatively low Be-temperatures, tritium located in the Be-bulk will not be released into environment at accidents [25]. It is estimated that a few tens of kg of Be-dust could be produced after the reactor is activated, during treatment of elements containing beryllium and contaminated with beryllium dust, in particular under replacement of such elements [30]. All such operations will be performed with the gas circulating through particulate filters with an efficiency of better than 99.9%. It is estimated [30] that a release of radioactive Be-dust under normal ITER operation would be <0.1 g/a.

Under accident situations with steam or water ingress into the vacuum vessel, the chemical reaction of beryllium with water will lead to likely oxidation and subsequent hydrogen production. Here, also, the most danger is from the presence of fine (submicron) Be-dust. For these reasons, the amount of the Be-dust on hot surfaces must not exceed 6 kg. In this case, complete reaction of dust would produce 2.5 kg H_2, which is below the deflagration limit for the ITER-FEAT vacuum vessel (4 kg). The total mobilisable amount of dust (particles smaller than 100 m in diameter) within the 1st confinement barrier (vacuum vessel) due to environmental release limits must not exceed 100 kg [25].

Some beryllium safety studies were performed during the last two years [29,30].

(1) Chemical interaction of different Be-forms with steam, air and carbon was studied at the A.A. Bochvar Institute of Inorganic Materials. Studies of the chemical interaction of dense and porous beryllium with superheated steam were performed at temperatures $T = 813$–1058 K and steam pressures from 50 to 400 kPa. A plot of cumulative hydrogen yield and temperature as a function of time was obtained. The main efforts were focused on the chemical interaction of commercial beryllium powder (with particles size of <56 μm) and superheated steam. These experiments covered the temperature range $T = 673$–873 K at pressures from 50 to 300 kPa. A mathematical model describing cumulative hydrogen yield (y) resulting from such an interaction was developed [31,32]:

$$(y)^2 = (0.5 \pm 0.3)\exp[-(1.07 \pm 0.06) \times 10^4/T]\tau \quad (\mathrm{mol}^2/\mathrm{m}^4), \tag{10}$$

where τ is in minutes.

Studies of steam chemical reactivity with Be-powder (with coarseness of <15 μm) placed in slots and on a flat surface at temperatures of 773, 873 and 973 K and a pressure of 200 kPa have shown that at $T < 773$ K, the hydrogen production rate in slots is higher than on the flat surface. At $T > 773$ K, the hydrogen production rate in slots is slower than on the flat surface. The law describing the reactivity changes from a parabolic (at $T < 773$ K) to a linear one (at $T > 973$ K) [33].

Chemical interaction between commercial Be-powder (<56 μm) and air was studied in a temperature range from 773 to 1273 K at atmospheric pressure. A kinetic plot of cumulative Be-powder oxidation at 1073 K was drawn. In the temperature range $T = 773$–1273 K, the oxidation rate (V) can be described by an Arrhenius-type equation:

$$V = (1.4 \times 10^5 \pm 3.8 \times 10^5)\exp[(-1.1 \pm 0.3) \times 10^4/T] \quad (\mathrm{mg/m\ min}).$$

Studies of beryllium–carbon chemical interaction in conditions typical of the ITER divertor revealed the

dependence of beryllium carbide layer thickness (h) growth rate versus time (in seconds) at 1223 K [31,32]:

$$h^2 = 1.12\{(24 \pm 6) \times 10^{-6} \exp[-(102\,400 \pm 2500)/\mathrm{RT}]\}\tau \quad (\mathrm{m}^2). \tag{11}$$

(2) Another experimental study of steam chemical reactivity with beryllium powder on hot surface inside grooves was carried out in Efremov Institute (Russia) [34]. It was shown that at temperatures <700 °C Be-dust–steam chemical reactivity practically does not depend on dust allocation geometry. At 800 °C the early stage reactivity decreases significantly depending on the dust allocation geometry. At 900 °C, the hydrogen yield in slots and on open flat differs by a factor of about 30.

(3) Studies of beryllium pebbles' chemical reactivity with air and steam in the temperature range 300–1000 °C were performed in the Belgian Nuclear Research Institute in Mol [35]. The tests have shown that at temperatures <700 °C, the kinetics is approximately parabolic and is associated with the growth of a protective oxide layer. Above 700 °C, the kinetics is accelerating/linear and oxidation is non-protective.

(4) Beryllium–steam interaction experiments were carried out in Kazakhstan Institute of Atomic Energy (Kurchatov) [36]. The emissivity of oxidized beryllium was measured in the temperature range from 580 to 1120 K for different surface oxidation degrees. The emissivity factor (E)–temperature (T, K) plot in this temperature range can be described by the equation:

$$E = 35.6T^{-0.64}. \tag{12}$$

The correct data for Be emissivity after interaction with steam are needed for the assessment of the temperature response of the ITER first wall at accident conditions.

The interaction of the steam with a real first wall mock-up has been also reported in [36]. The steam temperature was between 367 and 425 K, the temperature of beryllium armour was 680, 880 and 1210 K. No self-sustained Be–steam chemical reaction at temperatures used in the experiments was observed.

(5) Analytical study of steam diffusion and chemical interaction with beryllium powder on the hot surface inside the grooves at temperatures above 600 °C has been carried out in the Efremov Institute [37]. It was shown that the rate of hydrogen production in grooves is considerably less than that on flat surfaces. Dust plugs may be formed at groove necks due to dust swelling under oxidation or sintering at high pressures (above 800 °C at 1 bar).

6. Conclusions

Due to the unique set of properties, beryllium is an important material for fusion. For ITER application it is mainly proposed as first wall armour material. During ITER-coordinated R&D activity, significant progress has been made in the assessment of the beryllium production, selection and characterization of the beryllium grades, developing of the promising Be/Cu joining technologies, investigation of the features of the plasma-beryllium interaction and the safety issues. The performances of the beryllium itself and beryllium armoured components in ITER are adequate to the ITER goals. Still more information is needed to support the lifetime prediction for ITER (e.g. behavior of the neutron irradiated Be at relevant fluences at disruption and VDE transient events and thermal fatigue). For the first wall components, the final selection of the joining technology have to take into account the reliability of the Be/Cu joints and cost of manufacturing and will be done in the first stage of the ITER construction.

There was also significant progress in the development of the HCPB breeding blanket and assessment of the beryllium and beryllium intermetallic compounds pebbles performance. The key remaining issue is the behavior of the beryllium pebble bed at high neutron fluence (more than $\simeq$10 000 appm He).

The safety aspects of the beryllium application are also very important. ITER made the significant progress in the understanding of these issues. For the breeding blanket application, the similar studies have to be performed in the nearest future.

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