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# The status of beryllium technology for fusion ☆

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**Abstract**

Beryllium was used for a number of years in the Joint European Torus (JET), and it is planned to be used extensively on the lower heat-flux surfaces of the reduced technical objective/reduced cost international thermonuclear experimental reactor (RTO/RC ITER). It has been included in various forms in a number of tritium breeding blanket designs. There are technical advantages but also a number of safety issues associated with the use of beryllium. Research in a variety of technical areas in recent years has revealed interesting issues concerning the use of beryllium in fusion. Progress in this research has been presented at a series of International Workshops on Beryllium Technology for Fusion. The most recent workshop was held in Karlsruhe, Germany on 15–17 September 1999. In this paper, a summary of findings presented there and their implications for the use of beryllium in the development of fusion reactors are presented. 

## 1. Introduction

Beryllium has been used with success as a plasma-facing material (PFM) in several tokamak devices (e.g. UNITOR [1], ISX-B [2] and JET [3]) and is being considered for use as a neutron multiplier in breeding blankets and as constituents in inertial confinement fusion hohlraums. Beryllium as a component of the molten salt, Flibe is also finding interest in novel approaches to the plasma-structure interface, and it has been specified as a plasma-facing material for the RTO/RC ITER reactor.

Beryllium has advantages including a lower $Z$ number than carbon, and it is an excellent oxygen getter. Beryllium is considered as a neutron multiplier material in solid breeder blankets. Depending on the design selected, it may appear as pebbles in chambers near breeder zones [4,5], or it may be as orthogonal blocks. Disadvantages include its low melting temperature and high vapor pressure; its high physical sputtering yield; mechanical property degradation during neutron irradiation; its toxicity; its chemical reactivity with steam; and its relatively slow tritium release kinetics. [1] There are many developmental issues associated with its fabrication and joining of beryllium to itself and other materials, particularly for components subjected to high heat fluxes, such as those in the divertor.

Beryllium in the form of the molten salt mixture of LiF and $BeF_2$ commonly called Flibe has been considered as a renewable plasma-facing surface in advanced concepts for fusion reactors and also as a coolant. Flibe serves as a high temperature, low pressure, heat transfer

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[1] Whether the low permeability of tritium through Be is an advantage or a disadvantage depends on the context in which the determination is being made. It is a disadvantage in that tritium produced by neutronic transmutations tends to remain in the material, thus posing a significant potential risk. However, under off-normal conditions where temperatures do not exceed about 550°C, its ability to retain the tritium will be an advantage in reducing potential release.

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fluid that achieves high thermal efficiencies, acts as a good radiation shield, and has modest to good fluid flow properties [6]. Disadvantages include potential materials compatibility and corrosion issues with some structural materials and the potential for release of free fluorine [7,8].

A series of international workshops on beryllium technology for fusion has been organized to facilitate acquaintance of researchers and exchange of information on these topics. The first meeting in this series organized under the auspices of the International Energy Agency (IEA) was held in Karlsruhe, Germany in 1993. Since that time, meetings have been held under IEA auspices in Jackson Lake Lodge, Wyoming (USA) in 1995; Mito City, Japan in 1997; and again in 1999 at Karlsruhe.

## 2. Beryllium research

Beryllium research is being conducted at several centers of excellence. Obviously, those companies who mine beryllium and make it available to the public are involved. Because of its weapons and fission reactor applications, there are centers of beryllium expertise at several government laboratories around the world. Private institutes also perform research on beryllium. Among the technical issues associated with the use of beryllium in fusion machines are:

- Fabrication methods,
- Mechanical properties,
- Thermal properties,
- Plasma interactions,
- Neutron radiation effects,
- Chemical reactions,
- Health and safety.

## 3. Summary of work ongoing

### 3.1. Status of Be utilization/design in ITER

Changes in the general layout of the reactor have been introduced resulting in a size reduction of the machine by about 30%. For the reduced technical objective/reduced cost (RTO/RC) ITER, it is proposed to retain the same basic design shield blanket concept as in the 1998 ITER design. Beryllium was selected as the armor material for the main plasma-facing components. In the RTO/RC ITER, the plasma-facing armor and the heat sink materials remain the same as in the 1998 design, i.e. beryllium bonded to a copper heat sink with steel structure.

The first wall (FW) covers most of the reactor surface exposed to the plasma (several hundreds of square meters) and will be subjected to a relatively low heat flux (maximum 0.5 MW/m$^2$) during normal operation and to an intense heat shock during plasma off normal conditions such as a plasma vertical displacement event (VDE) or runaway electron impact (30–60 MJ/m$^2$ during 0.3–1 s). The FW of the limiter is instead subjected to high heat fluxes (up to $\sim$8 MW/m$^2$) during the start-up and shut-down phases of each pulse. In this case small and relatively thin (4–5 mm) tiles are used to assure high thermal performance and to limit the maximum tile operating temperature. The FW of the baffle module will have somewhat intermediate requirements (up to 1 MW/m$^2$) and armor dimensions closer to those of the primary FW.

Extensive research and development have been performed to develop, select and characterize the material. As a result of the work on beryllium forming and joining techniques and plasma spray coating, Be S–65C (or equivalent) has been selected as the reference material for all the components. Hot isostatic pressing (HIP) with specific interlayers (Ti or AlBeMet) seems to be the preferred joining technology for the primary and baffle FW. A special high performance fast brazing technology with the use of amorphous braze (CuInSnNi) seems to be the preferred method for the limiter.

### 3.2. Production

Different blanket concepts are presently under consideration within the European long-term technology program, and some of them foresee the use of beryllium as a neutron multiplier. Solid breeder blankets with a lithium ceramic breeder and steel or silicon carbide composites ($SiC_f$/SiC) as structural materials require beryllium to increase the tritium breeding ratio (TBR) performance. For the helium cooled pebble bed (HCPB) blanket for DEMO, beryllium is being considered as the neutron multiplier in the form of pebbles for several reasons. Under neutron irradiation, beryllium becomes brittle and swells. Within small pebbles, the temperature differences are small, thus the stresses caused by thermal gradients and different swelling rates (swelling is strongly temperature-dependent) are considerably smaller. As it is usually important to achieve a high beryllium pebble bed thermal conductivity density in the blanket, a binary bed of larger and smaller beryllium pebbles (the ratio between the diameter of the large and the small pebbles being at least 10) is presently used.

The larger pebbles (nominal diameter ($\Phi = 2$ mm) considered so far are produced either by means of the fluoride reduction process (FRP), or by means of the rotating electrode process (REP) [9,10].

The FRP beryllium pebbles are the result of an intermediate step in the process of extracting beryllium from ore. The beryllium is in the form of roughly spherical pebbles which have been used up to now in EU for breeder blanket research in the pebble bed area. Size

distribution of the pebbles ranges from approximately 1 mm to above 60 mm. For fusion research, the pebbles are separated into the required sizes ($\Phi = 2 \pm 0.2$ mm) using industrial screens (sieves). As a consequence, the FRP production yield is too low from this process for the supply of 2 mm beryllium pebbles to either ITER or DEMO blankets. The shape of these pebbles is generally not perfectly spherical, and they contain significant amounts of impurities (notably fluorine and magnesium), which could very soon justify discontinuing further work on this concept.

The rotating electrode process consists of arc melting the end of a long cast cylinder, which is rotating about its axis in a chamber filled with an inert gas (e.g. helium). Molten droplets of metal are thrown off the end of the rotating cylinder and solidify in flight. The size and quality of the REP beryllium pebbles depend essentially on the material used and on the production process parameters (e.g. electrode impurity content, electrode diameter, electrode angular velocity, cooling time, etc). Usually, this method produces almost perfectly spherical pebbles the mean size of which ranges from 0.2 to 2 mm or more, and with a very small surface roughness. Contrary to the FRP, the REP pebbles have been specifically developed and optimized for thermonuclear fusion application. However, because the REP pebbles are a relatively new product, extensive further R&D work is required, especially to determine its thermal, mechanical, swelling and helium/tritium release behaviors under fast neutron irradiation.

An alternative production method called the shot process (SP) was for several years under development at the Brush Wellman Inc., and it is indicated as being capable of development and potentially able to produce high-quality pebbles.

The smaller pebbles can be produced either by inert gas atomization (IGA) or by the rotating electrode process (REP) [9–11]. Spherical beryllium powder has been made by inert gas atomization and centrifugal atomization at both Brush Wellman Inc. and NGK Insulators Ltd. in limited quantities.

### 3.3. Impurity control

Thermal, mechanical, and irradiation behaviors of different Be grades depend on the impurity levels as well as on the thermal-mechanical treatment, which affects grain size, anisotropy, dislocation density, distribution of BeO and other impurities, etc.

The results of an analysis of the tensile stress–strain curves for S-200F HIP beryllium [12] suggest that the yield stresses of unirradiated beryllium are determined principally by the impurities. However, the tensile strengths do not show a consistent dependence on the BeO content of beryllium in the range of 0.5–1.2 wt%. The non-uniform tensile ductility generally decreases with increasing BeO and/or impurity content at test temperature in the range 310–605°C.

The most important single factor controlling the creep strength of beryllium is also the impurity content. The principal impurities in beryllium are beryllium oxide, carbon, iron, aluminum, silicon and magnesium. The beryllium-oxide impurities have an indirect influence on creep strength through their effects on the recrystallization temperature and grain boundary mobility, while iron and carbon appear to have only a minor influence on creep strength. On the contrary, above temperatures of about 600°C, the beryllium creep strength is considerably reduced by the presence of aluminum, magnesium and silicon, and this reduction becomes more marked as the temperature increases [13].

The irradiation behavior of beryllium seems to be positively influenced by the presence of BeO impurities. Irradiation tests performed in Russia show that the helium-induced swelling of different beryllium grades tends to decrease with increasing BeO content.

It would be desirable to develop an isotropic Be with a sub-micron grain size and relatively low BeO and elemental impurity contents in an attempt to increase the cleavage fracture stress and the ductility and toughness at low temperature and, at the same time, to reduce as much as possible the helium-induced swelling under fast neutron irradiation.

### 3.4. Joining and high heat flux performance

One of the main requirements to use Be as a candidate for plasma-facing components (PFC) is providing a reliable joint between Be and the Cu-alloy heat sink structure. A unique fast brazing process of joining beryllium to Cu-alloy that allows joint survival at heat fluxes higher than 10 MW/m$^2$ during thousands of heating/cooling cycles without seriously damaging the armor material, has been developed in Russia [14]. Comparative tests of Be/CuCrZr and Be/GlidCop joints were carried out. At room temperature, both Be/CuCrZr and Be/GlidCop joints showed a shear strength of about 150 MPa. Thermal cyclic tests of Be/CuCrZr and Be/GlidCop joints produced by fast brazing were carried out in the electron-beam test facility TSEFEY in St. Petersburg at simulated disruption loads ($\sim$5 MJ/m$^2$) and by subsequent thermal cycling ($\sim$5 MW/m$^2$, 1000 cycles) [15]. These experiments revealed no macroscopic damage to the grades tested, although significant differences in crack formation and propagation were observed. In general both Be/CuCrZr and Be/GlidCop joints successfully withstood high heat fluxes. Therefore, this Russian fast brazing technique could represent a promising joining technique capable of providing Be/CuCrZr and Be/GlidCop joints for ITER application.

Thermo-mechanical modeling has shown the desirability of using a brush-like structure for plasma-facing

components armor because of the reduction in stress at the armor/heat sink interface. Small-scale divertor mock-ups have been fabricated with beryllium and tungsten brush armors at Plasma Process Inc. [16]. The brush armor materials fabricated to date have been made with preformed tungsten and beryllium rods, which are backed with specific matrix materials. The results of high heat flux testing experiments by Sandia National Laboratories (SNL) in the electron beam testing system (EBTS) show that PFC mockups using the tungsten brush armor have survived cyclic loading (10 s beam-on, 10 s beam-off) for 500 cycles a piece at 5, 10, 15, 22 and 30 MW/m$^2$ with no damage to the heat sink and little or no damage to the armor.

Several mock-ups were fabricated using brazing techniques and tested at the TSEFEY and EBTS electron-beam facilities [17]. The mock-ups were divided into two groups: low-heat-flux elements with armor thickness of 10 mm and tile (castellation) dimensions in the range of $40 \times 40$–$10 \times 10$ mm$^2$; and high-heat-flux (HHF) elements with armour thickness of 5 mm and tile dimensions in the range of $20 \times 20$–$5 \times 5$ mm$^2$. The influence of tile planar dimensions on the mock-up reliability was investigated by cyclic HHF tests by finding the number of thermal cycles required for armor debonding for several heat flux values. The mockup with the tile dimensions of $5 \times 5 \times 5$ mm$^3$ demonstrated the best results during the HHF tests at the EBTS facility. During a thousand cycles with a heat flux of 13.5 MW/m$^2$ no damage in the Be/CuCrZr joint occurred as confirmed by a metallographic investigation of the tested and non-tested cross-sections.

### 3.5. Mechanical properties

Complicated mechanical tests have been recently performed by Brush Wellman Inc., expanding the elevated temperature beryllium database [18]. The elevated temperature (ambient to 648°C) thermo-mechanical properties of two beryllium grades made by hot isostatic pressing (HIP) were compared: S-65H (made from impact ground powder) and GA (made from gas atomized powder). Valid beryllium $K_{IC}$ fracture toughness results were obtained for the first time at temperatures above the room temperature. The elevated-temperature properties (ultimate tensile strength, yield strength, percent elongation, reduction in area) of S-65H generally exceed those of HIP'd GA Be up to 650°C, but are consistent with the vacuum hot-pressed version of S-65 (i.e., S-65C). On the other hand, the elevated-temperature tensile properties of HIPed GA are comparable to a vacuum hot pressed grade with a similar chemical composition (i.e., S-200F). GA has consistently higher fracture toughness values than S-65H. Over the temperature range of ambient to 205°C, the toughness increases approximately 50% in magnitude from approximately 10–11 to 15–16 MPa $\sqrt{m}$.

At temperatures of 260°C and higher, non-linearity in fracture load traces as well as insufficient specimen thickness and crack length invalidate the $K_{IC}$ data. Physical measurements of crack extension reveal that slow, stable crack advance or tearing does not generally occur at temperatures less than 480°C. Minimal amounts of slow, stable cracking were observed as the temperature increased. At the highest 650°C condition, significant crack tearing and load relaxation (creep) were observed.

At 90% yield strength load, GA and S-65H have roughly the same behavior at room temperature through 205°C. From 315°C to 540°C there is some indication that S-65H has better fatigue performance. One cannot distinguish between the two grades at 650°C because both grades have run out of behavior.

An assessment has been made of the ambient- and elevated-temperature tensile and fracture toughness properties of unirradiated and aged (~2000 h at temperatures in the range 185–605°C) hot isostatic (HIP) and vacuum hot (VHP) pressed S-65 and S-200F beryllium grades [12]. The effects of material (powder consolidation method, beryllium oxide and elemental impurity contents and grain size) and temperature variables were analyzed to further the development of composition-structure-property relationships for unirradiated and irradiated beryllium. The results of the analysis demonstrate that the tensile yield and ultimate strengths of the reference and aged beryllium grades at a given test temperature increase with the inverse square root of the grain diameter in accordance with the Hall–Petch relationship. The tensile yield strengths are also determined by the impurity elements and precipitates.

### 3.6. Chemical reactivity

Recent work has investigated the rate of reaction of beryllium pebbles with steam [19]. Pebbles of 2 and 0.1–0.2 mm diameter and uncompacted powder were examined for their specific surface area using the BET (Brunauer, Emmett, Teller) technique. The larger pebbles, obtained from the fluoride reduction process showed only 0.12 m$^2$/g while the smaller pebbles showed 0.24 m$^2$/g. For the powder, specific surface area varied from 0.66 to 1.21 m$^2$/g. Hydrogen generation rates were measured by both mass-spectrometry and weight gain and were found to compare well with those on fully dense compacted powder metallurgy product measured previously [20] when compared on a specific surface area basis. In some cases, with the starting temperature depending on porosity, the reaction was found to be self-sustaining or 'ignited'.

Additional experiments were performed on vacuum hot pressed grade S-200F beryllium as: (1) unirradiated

dense material with a density of 99.9% of theoretical, (2) irradiated dense material, similar to the first but having been irradiated to $1.6 \times 1021$ n/cm$^2$ in the BR2 reactor, and (3) irradiated porous material irradiated to a fluence of $4.0 \times 10^{22}$ n/cm$^2$ and having a density of 97.2% of theoretical [21]. Thermogravimetric measurements and differential thermal analysis were performed in air and steam. The air experiments showed some scatter but generally kinetics were parabolic at 600°C and accelerating or linear behavior above 800°C. The steam reactions showed the same general results. The transition to linear kinetics occurred at about 800°C for the irradiated and unirradiated dense material and 700°C for the lower density irradiated material.

### 3.7. Irradiation effects

The mechanical properties of different beryllium grades for plasma-facing components have been studied to assess the effects of irradiation on the tensile yield, ultimate and fracture stresses, uniform and total elongations and reduction of areas of the S-65 HIP, S-65 VHP, S-200F HIP and S-200F VHP beryllium grades produced by Brush Wellman [12,22].

Irradiation produces marked embrittlement at test irradiation temperatures of 185, 235 and 310°C but there is some recovery of ductility at higher temperatures of 485, 540 and 600°C. The enhanced tendency to cleavage fracture at the lower temperature is associated with a larger irradiation-induced increase in the yield than in the fracture stress. The cleavage fracture shows a dependency on the inverse square root of the grain size, again in agreement with the Hall–Petch relationship.

The irradiated beryllium usually shows intergranular and/or ductile dimple fracture at the higher temperatures (485, 540, and 600°C) and such an intergranular cracking probably results from the stress-induced growth and coalescence of adjacent grain boundary helium bubbles.

In general, there is little difference between the various beryllium grades (i.e. S-65 HIP, S-65 VHP, S-200F HIP, and S-200F VHP) with respect to their resistance to radiation damage at the lower irradiation/tensile test temperatures (lower than 310°C) as they are all brittle. However, the S-65 VHP and, in particular, the S-200F HIP grades are more ductile at 435–600°C.

Analysis of beryllium samples irradiated in the BR2 reactor at 216, 473 and 607°C and at fluences between 2 and 3 dpa shows that the formation of helium during irradiation seems to have an influence on the microstructure only for the samples irradiated at 473°C and 607°C [23]. For these samples, helium bubbles are clearly observed at grain boundaries.

At 200°C, there is only a small change with respect to the unirradiated samples. Irradiation seems to increase the dislocation loop density. Differences in the fabrication methods seem to have only a small influence, if any, on the beryllium microstructure after irradiation.

The behavior of various Russian beryllium grades different by manufacture technology, grain size and BeO contents (i.e. TE-56, TE-30, DIP-30, TIP-30) irradiated in the SM reactor at 60–75°C up to $0.62–2.37 \times 10^{22}$ cm$^{-2}$ ($E$>0.1 MeV) has been investigated [24]. All beryllium grades showed a corrosion rate in water from 0.1 to 5.4 g/cm$^2$ (irradiation time period of 2700 h) or, in other words, 1.3–10.8 μm/y. The swelling ranges from 0.07% to 0.91% depending on the corresponding neutron fluence. There was strong embrittlement of beryllium in all grades studied and a reduction of fracture stress.

The effects of neutron irradiation such as swelling and helium and tritium retention/release on different Russian beryllium grades (i.e. TIP-30, TIRR, DIRR) were also investigated by VNIINM [25]. Beryllium was irradiated with a neutron fluence of $2.6–3.5 \times 10^{21}$ cm$^{-2}$ ($E$>0.1 MeV) corresponding to neutron damage of 1.3–1.8 dpa at temperatures of 550, 620 and 790°C. The total amount of helium accumulated in the irradiated beryllium varied from 240 to 620 appm. A significant part (25–65%) of the generated tritium was released during the irradiation at 790°C, while almost all helium was retained in the material. In general, the swelling decreased while the tritium retention increased by increasing the BeO content in the range 2.1–3.9 wt%.

### 3.8. Plasma interactions

Beryllium exposed to fusion plasmas is acted on in various ways by the impinging particles. One process is sputtering where atoms of beryllium are removed by physical collisions of the ions and charge-exchange neutral atoms. Because of the absence of affinity of beryllium for hydrogen, there is effectively no chemical sputtering. That process is a problem for carbon-faced structures.

#### 3.8.1. Sputtering and erosion

Recent experiments related to beryllium erosion and sputtering have examined erosion of mixed material layers [26,27]. Such layers are formed as beryllium is sputtered and then redeposited, usually in combination with carbon or oxygen. Experiments at the Max Planck Institut für Plasmaphysik in Garching investigated mixed layers of beryllium, carbon and oxygen formed under $10^{-8}$ Pa vacuum on beryllium surfaces exposed to $C^+$ and $CO^+$ ions with energies between 3 and 12 keV [26]. Surface reactions were studied with Rutherford backscattering spectroscopy. When only carbon was applied, a carbon film was formed that eventually prevented erosion of beryllium. This process was well-modeled by the TRIDYN code. When $CO^+$ atoms were implanted, however, the beryllium was able to diffuse through the film and erosion of the beryllium was

continuous. The results are verified by X-ray and UV photoelectron spectroscopy that reveal the surface chemical state.

At the Kurchatov Institute, mixed layers of tungsten and beryllium were prepared by simultaneously sputtering beryllium and tungsten onto a beryllium substrate using 20-keV $Ar^+$ ions [27]. The films were about 500 nm thick and were nominally 35% beryllium, 35% tungsten, and 30% oxygen.

Where the film was removed by plasma ion sputtering, deuterium atom concentrations of up to $5 \times 10^{21}$ D/cm$^3$ were measured near the surface while the deuterium concentration in the melted film zones was only one fifth that value or less. Analysis of the sputtered particles collected on basalt fiber filters showed spherical droplets of tungsten up to 20 μm in diameter and flakes of tungsten–beryllium mixture, suggesting some segregation occurs during sputtering.

In a separate magnetron experiment at the University of Missouri, the objective was to deposit $Be_2C$ films for inertial confinement fusion target applications [28]. The sputtering of beryllium into a methane/argon plasma, excited at an audio frequency, was found to generate crystals of $Be_2C$ in a beryllium-rich matrix. There was no evidence of beryllium or BeO crystals, therefore, the excess beryllium must have been amorphous. When such films were heated to 750°C using thermogravimetric analysis, there was evidence of beryllium sublimation. When they were exposed to air and steam at elevated temperatures, there was greater resistance to oxidation in dry air. At ambient temperature, no oxidation was evident in neither environment over a period of 24 h nor in the air environment over 30 days.

*3.8.2. Hydrogen interactions*

Thermal desorption spectroscopy (TDS) experiments have been performed to compare deuterium retention in surface BeO films grown thermally with films generated by ion sputtering [29]. Thermally grown films were found to have similar release characteristics for deuterium implantation in the range 300–900 K, but for the ion-sputtered films, release characteristics depended much more strongly on the ion implantation temperature. The implication is that at higher temperatures, implanted deuterium is able to migrate to bubbles at grain boundaries for the films produced by sputtering while at lower implantation temperatures such a migration is not possible.

In somewhat similar experiments, BeO layers were formed by magnetron sputtering of beryllium in an equal mixture of hydrogen and argon [30]. BeO films were nominally 25 μm thick. TDS experiments after deuterium implantation showed two deuterium peaks, one at 760–800 K and the other at 920–970 K. Typical deuterium concentrations in the implanted layer were $3800 \pm 200$ appm. In experiments where the BeO layer was enriched with carbon, retention increased to $9600 \pm 200$ appm. These experiments clearly showed that under these conditions, deuterium retention was strongly controlled by the surface films.

In similar experiments with low ion energy (200 eV) and high flux density ($13 \times 10^{21}$ D/m$^2$ s) in the MAGRAS magnetron facility, studies using Rutherford backscattering with $H^+$ ions showed that except for a nominally 100-nm surface layer, the 500-nm redeposited film consisted entirely of BeO. The H:Be and D:Be concentration ratios varied from 0.15 at 350 K to 0.06 at 570 K plasma exposure temperatures, in agreement with previous measurements made in the US [31]. Again, the importance of surface films in tritium inventories is confirmed.

*3.8.3. Surface cleaning*

Some very interesting work was reported that made use of negative transferred-arc cleaning for removing carbon–hydrogen films from beryllium surfaces [32]. In this process a cathodic condition is created at the surface of the workpiece that results in the transfer of electrons from the workpiece to a plasma torch during an electric arc discharge. Such an electron transfer can remove oxide and carbonaceous films from the surface of beryllium. This process offers good prospects for the detritiation that must take place periodically in a machine like ITER to maintain tritium inventory goals.

*3.9. Pebble bed behavior*

The heat transfer parameters (i.e. thermal conductivity and heat transfer coefficient) of single-size as well as binary beryllium pebble beds have been obtained by experimental investigations [33,34]. Besides the relationship of the heat transfer parameters as a function of the differential thermal expansion $\Delta\ell/\ell$ (interference) and temperature, the effect of the pressure of the pebbles on the containing walls have also been measured.

Air cooling experiments indicate that for binary beryllium beds with an average temperature higher than 200°C, the thermal conductivity for $\Delta\ell/\ell = 0$ is practically independent of the temperature up to 600°C [34]. The effect of the constraint is quite large and the thermal conductivity increases linearly with $\Delta\ell/\ell$ in the interference range 0–0.1%. However, cycling tests show a relatively high hysteresis of the thermal conductivity as a function of higher values of $\Delta\ell/\ell$.

Uniaxial compression tests (UCT) with monosized and binary beryllium pebble beds were performed between ambient temperature and 480°C and pressures up to 8 MPa [35]. Empirical correlations for the modulus of deformation for the different bed types and the first measurements on thermal creep have been obtained.

Additionally, results for triaxial compression tests (TCT) for single-size and binary beds show that the

internal friction of these beds is significantly larger than that for spherical particles with relatively smooth surfaces (e.g. ceramic breeder pebbles) indicating that the ability of beryllium pebbles to flow is more suppressed. The experiments showed that the initial state of consolidation of the pebble bed is very important for the stress–strain dependence.

The electrical resistivity behavior of a beryllium pebble bed has been studied as a function of the temperature and pressure [36]. At room temperature the resistivity of a single size 2 mm pebble bed decreases drastically from $2 \times 10^{-2}$ to about $10^{-4}$ Ω m by applying an external pressure. After this first drop, the resistivity decreases slightly with increasing applied pressure. The same trend appears for a single-size 0.1–0.2 mm pebble bed, but the resistivity values are about one order of magnitude higher than in the case of the 2 mm pebbles. The resistivity behavior of the pebble bed with applied pressure is, at high temperature, qualitatively the same as that observed at room temperature. However, for the same applied load the pebble bed electrical resistivity increases almost linearly with the temperature due to the increased oxide content of the pebbles.

Detailed three-dimensional thermal-mechanical analyses in both steady and transient conditions on a medium scale mock-up for investigating the thermal-mechanical behavior of both beryllium and lithium orthosilicate pebble beds have been carried out [37]. The results obtained seem to show that the mock-up, in its actual layout, is able to withstand the thermal and mechanical loads it undergoes during normal operating conditions.

### 3.10. Molten salt

The molten salt $LiBe_2F_4$, commonly referred to as Flibe has become interesting for fusion. In addition to being a coolant, Flibe will breed the tritium required for fusion reactions. Tritium breeding may produce an excess of fluorine that must be managed to prevent HF or TF being released. Issues have arisen regarding the management of tritium in Flibe systems: keeping the tritium from escaping in heat exchanger tubes or other places where it is not wanted, and getting it out easily at locations where its removal is necessary.

Two experiments are presently in progress and more are planned to help answer questions of tritium production and management in Flibe. The FLIQURE (Fusion LIQUid RElease) experiment in Idaho will make use of an embedded $^{252}Cf$ source to generate neutrons that will produce tritium in a small (200 ml) crucible of molten Flibe. A similar experiment has already been placed in operation at the YAYOI reactor in Japan [8]. There, neutrons from the reactor core produce tritium in the Flibe. It was discovered that the form of tritium release depended heavily on the concentration of $H_2$ in the He sweep gas passed over the Flibe. When this concentration was 10% or more, tritium release was first-order in tritium concentration and was evolved as HT. At lower $H_2$ concentrations (1000 appm or less), the tritium evolved as TF and was apparently controlled by diffusion of tritium into the surface.

### 3.11. Health and safety

The main hazards of beryllium come from inhalation and skin contact. Therefore, toxicologically relevant exposure to beryllium is almost exclusively restricted to the workplace [38]. Although numerous regulations, directives, and recommendations have been laid down, none of these give a concentration limit value for the environment, and no specific European Union law exists right now. In most European Union countries, the American guidelines are used as a general reference. Most European countries adopt a beryllium limit concentration in the workroom of 2 μg/m$^3$ but the definition of the exposure limit may differ from place to place. Open environment limits are almost absent in the official regulations/directives. However, the 0.01 μg/m$^3$ established by the US Environmental Protection Agency in the 1940s prevails as a limit value in some EU countries [38].

According to the data from Russian and Japanese scientists [39], a sub-micron dust of metallic beryllium, which is not blocked by standard filter system could be found in the workrooms of beryllium production plants including high-temperature beryllium processing. Because of their larger specific surface and their capacity for reaching almost all parts of lungs, the sub-micron beryllium particles probably cause the largest health injuries. If beryllium has to be used on a large scale in the next generation of fusion reactors, a systematic experimental study on the effects of sub-micron beryllium particles on the human health will be necessary.

## 4. Conclusions

At the conclusion of the workshop, an experts' forum summarized the general status of the technology and pointed directions for future research. Among the points made were the following.

There is a need for better characterization of the differences between the various grades of beryllium available, manufacturing processes, impurities, etc. Whereas the present understanding of beryllium properties is now understood at the μm level, it may prove necessary to understand processes in beryllium at the nm level. An example of a parameter that may be important is the ratio of Fe to Al in the beryllium. Though the techniques for producing beryllium are well known,

many of the specifications are out of date and should be changed to modern ones.

There is some value in studying unirradiated material, but it was acknowledged that irradiated material behaves rather differently. For example, high BeO content will possibly reduce swelling due to helium gas bubble formation, but the low-temperature mechanical properties may be degraded by oxide addition. Attempts to modify materials for better performance under irradiation need to be performed circumspectly, with consideration for a wide range of other factors as well. Joints to other materials will be a particular problem area in the radiation environment.

There is a need for a low-activation material that can be cleared or recycled readily. It is not yet possible to recycle beryllium that has been activated by neutron irradiation. There is hope that after a 30- to 100-year holding period, it will be possible to recycle the material from fusion reactors, thus eliminating beryllium from the waste stream. Particular issues associated with disposal are levels of gamma activity and included tritium. The database on tritium retention in less than fully dense material is sadly lacking. Disposal of used beryllium will be an issue of increasing importance to fusion. Preventing corrosion of beryllium once in storage, with attendant release of toxic and radioactive substances, is a technology in need of development.

One of the new forms of beryllium that will probably be needed for fusion blanket applications is beryllium pebbles. Present designs and research show that different sizes will be needed for optimal packing density. Finding ways to produce such pebbles with sufficient dimensional and quality controls to give a good performance at an affordable cost is one of the challenges presently facing the industry. Determining microscopic and macroscopic performance parameters for heat transfer, geometric ratcheting due to thermal expansion, and swelling are also topics of importance now.

There is a need for high-fluence irradiation testing of beryllium. The main agent for property degradation is helium production. Irradiations up to 3000–6000 appm He per year at 700°C with a He:T ratio of about 100 are needed, but facilities for reaching such goals simultaneously are now not available. Fundamental studies of the effects of neutrons on beryllium are needed so that beryllium components can be properly sized.

New forms of beryllium may be required. Recent proposals have included beryllium salts and intermetallic compounds for novel plasma-facing component designs. Intermetallic compounds would exhibit low sputter rates and show good resistance to oxidation in the event of an accidental exposure to steam. They tend to be extremely brittle, however, and much needs to be learned before they could be used with confidence.

To sum up, beryllium is an important material for fusion, particularly as evidenced in the ITER design. Much has been learned in the last decade regarding the interactions of various aspects of the fusion environment with beryllium, but there is much yet to be learned, some of which we have identified, and much of which we probably are not yet aware of. The best solution is to plan and conduct experiments to provide answers to questions we now have and to hope Mother Nature will be kind as we seek new answers and applications.

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