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2.3 Status of Beryllium Study for Fusion in RF

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The main directions of research activities in the field of Beryllium application in fusion science and technology carried out in Russia during 2001-2003 have been reviewed. The main results of these investigations have been highlighted.

First wall and port-limiter. The investigation on the actively cooled components with beryllium cladding is under progress objecting on the clarification of their ultimate thermo cycling capabilities. The study on behavior of bulk Beryllium and the boundary region of the contact with the cooling structure under the intensive thermo cycling loading and neutron irradiation have been the object of consideration in particular.

The works on the optimization and modifications of industrial fabrication processes for commercial scaled production of Beryllium tiles were also under the way.

The influence of neutron irradiation. The new experimental data on the nuclear properties of several Russian Beryllium grades has been obtained. The samples have been subjected to the high neutron dozes. The influence of low temperature (70-200°C) neutron irradiation on the thermal conductivity has been examined in particular. The interrelations of the Helium inventory and temperature of neutron irradiation with Tritium release out of irradiated Beryllium samples have been analyzed.

The Beryllium associated safety questions. The experiments on the modeling of normal working conditions and conditions imitating the plasma disruption events in ITER performance scenario have been continued. The new experimental information on the coefficients of pulverization of Beryllium and the accumulation of deuterium in Beryllium under the action of proton beam has been collected.

The dependence of the reaction rate constant for the Beryllium oxidation by the water vapor for different conditions has been analyzed. The compact, porous and powder Beryllium samples have been tested at the wide range of temperature, pressure and duration of reaction with water vapor. The calculating codes have been proposed for determination of the rate of Hydrogen generation.

The theoretical investigation on the correlation of the plasma disruption with Tritium inventory in Beryllium components of the First Wall of ITER reactor and on the penetration of Tritium through First Wall material into coolant agent has been performed.

Tritium breading blanket. The investigation on optimization of fabrication processes of porous Beryllium with guaranteed predetermined open porosity in the range of 10–30 % altogether with properties characterization has been carried out.

1. INVESTIGATION OF ACTIVELY-COOLED COMPONENTS WITH BE ARMOUR [1,2]

The researches aimed at the estimation of the life-time and studying the behaviour of Be and Be/Cu joints subjected to a cyclic heat loads and neutron irradiation. In particular, estimation of the life-time for Be which uses CuCrZr heat-sink additionally protected with inserted Stainless Steel (SS) tube was continued. Basing on the promising results of the thermal fatigue testing of the First Wall (FW) mock-up which also uses CuCrZr heat-sink additionally protected with inserted SS tube, an attempt to propagate such heat-sink technology to another in-vessel component (port-limiter) was done. With that view actively-cooled mock-up with Be armour was manufactured and tested. This mock-up successfully withstood 2000 thermal cycles at 7 MW/m^2 (i.e. expected in ITER) (Fig. 1). This result allows considering mentioned heat-sink technology as a candidate for ITER port-limiter.

Optimization of the FW manufacturing method using vacuum casting was continued. All previous testing results were obtained in unirradiated condition. To estimate ITER FW life-time under the neutron flux, the "in-pile" test (the FW mock-up with Be armour will be subjected to cyclic heat flux directly in nuclear reactor) are being prepared now. The cross-section of the mock-up is planed to be identical to the ITER project (Fig. 2). The number of heat pulses are

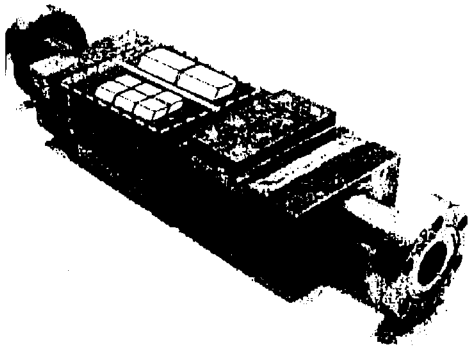
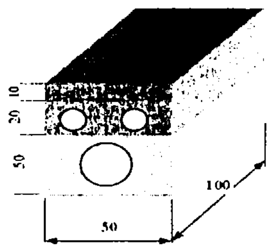


Fig. 1. Actively-cooled Be armoured mock-up with inserted SS tube successfully withstood 2000 thermal cycles at 7 MW/m^2)

planned to be $2x10^4$, heat flux – 0.5 MW/m^2, radiation damage in CuCrZr – up to 0.6 dpa. The heater for in-pile experiment is being tested now.



Heat flux
0.5 MW/m^2
Number of pulses
(3-10) 10^3
Dose
up to 0.6 dpa
Water temperature
70-100 °C

Fig. 2. Scheme of the FW mock-up with Be armour for the in-pile testing.

Post-mortem analysis of the actively-cooled mock-up with Be armour after previous in-pile campaign (1000 heat pulses at 7.5 MW/m^2, Dose 0.15 dpa) has been finished. Metallographic analysis of Be and Be/CuCrZr joint revealed no defect after testing in Be and Be/CuCrZr joining zone. (Fig. 3)

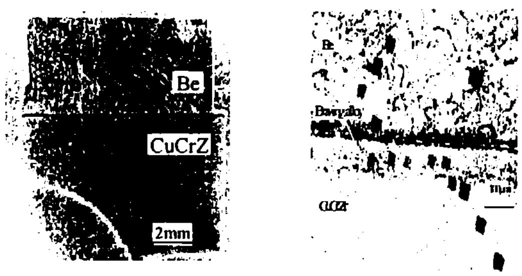


Fig. 3. Cross-section and brazing zone of Be/CuCrZr joint after in-pile testing

In some design using of "non-flat" Be armour is necessary. Attempts to propagate technology of fast brazing of Be armour from flat geometry to the monoblock or saddle-block geometry were done. In particular, Be armoured mock-up with saddle-block geometry successfully withstood 1000 heat pulses at 0.5 MW/m^2 (Fig. 4.).

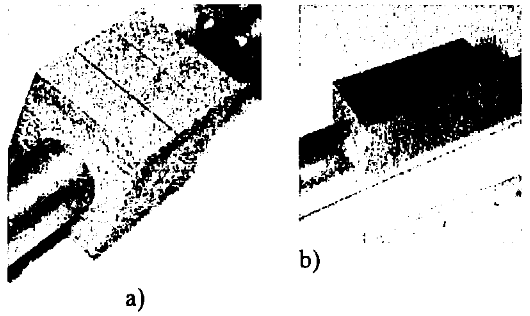


Fig. 4. Be monoblock (a) and saddle-block (b)

2. EFFECT OF NEUTRON IRRADIATION

In the frame of the agreement on a cooperation between VNIINM and NIIAR, concerning the investigation of beryllium damage under low temperature irradiation, new data were obtained on radiation properties of several Russian beryllium grades (TE-56, TE-30, TIP, DIP) irradiated up to a high neutron dose [3-5].

The investigations on the effect of neutron irradiation at 70°C with fluence of $(6\text{-}11.1)\cdot10^{22}$ cm^{-2}, E> 0.1 MeV on swelling and mechanical properties show that swelling for these beryllium grades does not exceed 1-2.5 % and progressively increases with a fluence rise. All the samples characterized by a total embrittlement and a significant reduction of ultimate stress. Nevertheless, cracking of specimens was not found even at maximum fluence $(10\text{-}11.1)\cdot10^{22}$ cm^{-2}, E>0.1 MeV [4].

Neutron irradiation at 70-200°C with fluence (0.6-11.1) 10^{22} cm^{-2}, E> 0.1 MeV results in lowering the thermal conductivity (TE-56) by a factor 3-7, depending on fluence value. An abrupt reduction of thermo conductivity occurs at a beginning of irradiation (up to 1 10^{22} cm^{-2}) then a rate of a reduction becomes slower. Short-time post-irradiation annealing (500°C for 1 hour) leads to partial recovery of thermo conductivity (from 53 to 150 Wt/mK) [.

The effect of helium generation and irradiation temperature on tritium release was studied in [5]. Beryllium specimens (TE-56 and TshG-56) were irradiated at temperatures of 70-100°C and 650-700°C up to fluence $(0.32\text{-}2.0)\cdot10^{22}$ cm^{-2}, E>0.1 MeV. Helium content in the specimens was in the range from 521 to 3061 appm. It was shown that irradiation temperature and helium generation significantly affects the tritium release. After low temperature irradiation (70-100°C) a fraction of tritium of 44-74 % is released from beryllium at an annealing temperature below 800°C, while for samples exposed to high temperature irradiation (650-700°C) tritium release did not exceed 14 %. Majority of tritium (~68 %) is released within a temperature range from 800 to 920°C. The increase of helium generation from 521 appm to 3061 appm results in lowering the initial temperature and the upper temperature of tritium release by 100-130°C and 200-240°C, correspondingly. At temperatures below 900°C the diffusion mobility of tritium for the sample irradiated at high temperatures (650-700°C) is lower than for the samples irradiated at low temperatures (50-100°C).

Recently in the frame of the agreement on collaboration between RF (VNIINM) and EU (EFDA NRG) on the investigation and optimization of beryllium for ceramic breeding blanket of DEMO reactor, VNIINM has fabricated over 950 beryllium specimens with the required characteristics (grain size, chemical composition, density and porosity) for the characterization of beryllium at high dose (helium generation up to 6000 appm) and high temperature irradiation (400-800 °C) in HFR reactor (NRG, Petten, The Netherlands) [7].

For the adjustment of technical solutions for Russian conception of lithium self-cooling blanket, RF has started the investigation on corrosion behavior of beryllium in the Be- liquid Li- V4Ti4Cr system. Testing has been performed by isothermal annealing at 600°C, 700°C and 800°C with duration of 200-500 hours. First results show that the modern beryllium grades are much more corrosion-resistant in lithium than those grades, which were investigated earlier [6].

3. STUDIES ON FUSION REACTOR BERYLLIUM SAFETY.

The beryllium protective armour on the first wall of the International Thermonuclear Experimental Reactor (ITER) will operate at hard conditions. Under normal operation it will be exposed to pulse impact of deuterons and tritons with an energy of about 100 eV. The total intensity of incident ion fluxes will be up to 10^{20} m^{-2} s^{-1}. The ion irradiation dose at that will be up to 2.6 10^{27} m^{-2}. The heat load on the first wall under plasma disruptions is expected to be up to 20 MJ/m^2 [8]. Since any studies of the beryllium protective armour behaviour and working capacity in the fusion reactor conditions are impossible, big efforts are devoted to modelling of hydrogen isotope ion interaction with Be-elements under conditions simulating those expected in the ITER vacuum chamber.

A group of specialists of N.E. Bauman Moscow State Technical University together with scientists of Kurchatov Institute and M.V. Lomonosov Moscow State University performed such modelling studies in the facility MAGRAS with ion energy of 200-600 eV and ion flux density on the target up to 3 10^{21} $м^{-2}$ c^{-1}. An average dose of the target irradiation by ions was up to 3 10^{25} $м^{-2}$.

It was shown that under the experimental conditions (target temperature 420-700 K, proton and deuteron energy 200 eV) Be-sputtering yield (ratio of sputtered atoms to the number of incident ions) Y 0.02.

Use of three different materials (Be, W and C) as protective armour of the plasma facing components inevitably will result in formation of mixed layers in ITER. Therefore studies of carbon and tungsten effect on beryllium behaviour in the conditions of ITER vacuum chamber were performed. For this purpose targets made of Be and C or Be and W were used. The component composition of the co-deposited Be-C-D layer was determined (Fig. 5). It was shown also that presence of carbon and, to a lesser extent, of tungsten in the re-deposited Be-layers increases D-content by 2-6 at.% [9].

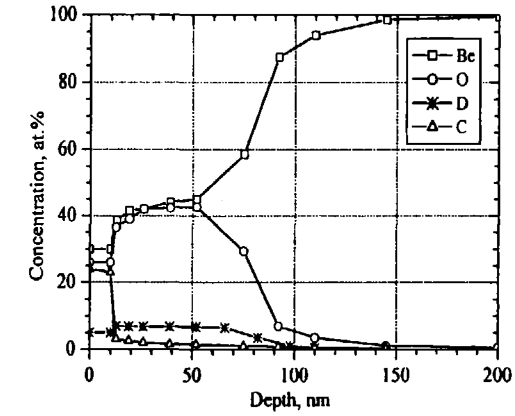


Fig. 5. Depth distributions of chemical elements in the co-deposited Be-C-D layer

Similar studies of Be-behaviour in the presence of carbon-fiber composite (CFC) were performed in electrodynamic plasma accelerator MKT in Troitsk Institute of Innovative and Thermonuclear Investigations (TRINITI) under simulation of plasma disruptions. Beryllium and CFC were irradiated by 10 pulses of deuterium plasma with an energy density of 900 kJ/m^2·pulse and a pulse duration of 60 μs. Deuteron energy was up to 1-2 keV. About 60 nm thick mixed layers were formed on the Be-target surface. The concentrations of Be, C and O in them were 58.2, 30.4 and 11.4 at.% respectively. Be-surface was cracked along the grain boundaries. Beryllium droplets, from 0.3 to 3.0 μm across, settled on grain surface. Integral deuterium concentration in the mixed layers was (6.4-6.5) 10^{19} m^{-2} [10].

The reaction constant (RC) for exothermic oxidation of compact, porous and powder beryllium by steam were measured in A.A. Bochvar Institute of Inorganic Materials. The kinetics of the oxidation was studied at pressures of 0.5, 1.3 and 3 atm. The experiments with compact and porous Be were performed in the temperature range from 500 to 1000 °C. The oxidation of Be-powder was studied at temperatures from 400 to 700 °C. RRC is an exponential function of inverse temperature.

The initial value of RC for oxidation of compact and porous beryllium by steam depends on the source material type. Oxide layer on the surface of such specimens favours RC-increase since it hampers the removal of heat generated in the reaction, though it creates diffusion resistance to the steam entry to reaction surface. It was found that under the interaction of compact and porous beryllium with steam, at certain conditions and at certain oxidation degree, growth of RC becomes burst-like when the basic part of the powder is oxidized during 1-2 minutes. This phenomenon has thermal nature and can be explained in the frames of the theory of thermal explosion as a spontaneous acceleration of oxidation at insufficient heat removal from the reaction zone. A relationship taking into account variation of sizes of Be-specimens, which are spherical or cylindrical pellets, under oxidation was obtained.

The initial value of RC for oxidation of Be-powder layer by steam is close to the values typical for porous beryllium. Oxide layer on the powder particles decreases RC, hampering steam entry to the particle surface in the layer depth. Oxidation RC-to-time, temperature and pressure ratio was revealed. Algorithm for computation of hydrogen formation rate in the investigated range of pressures and temperatures was suggested [11].

Be-dust produced under fusion reactor operation is dangerous as potential source of explosive hydrogen and due to its toxic and radiological hazard at accident situations. It is expected that such a dust will accumulate in the slots of plasma facing components in the reactor vacuum chamber. In Bochvar Institute was carried out an experimental study of chemical interaction of superheated steam with Be-dust (purity of 99%, size 15 μm), located in slots (40 mm long, 0.5-1.0 mm wide and 10mm deep) of stainless steel at temperatures from 400 to 700 °C. Chemical interaction between superheated steam and thin layers (200-350 μm) of Be-powder, uniformly scattered on a plate, was studied simultaneously. Relationships for maximum specific hydrogen generation rates under steam interaction with Be-dust at its different location (in slots and on a plate) to temperature as well as plots of temperature variation for experimental runs on different temperature levels were obtained on the basis of these experiments [12].

A design-theoretical study of the same problem was performed in D.V. Efremov Institute [13]. A computing code modelling chemical interaction of steam with Be-powder located on open surface and inside a thin slot in a temperature range from 700 to 900 °C in the case of an accident was developed. The results of computations are in agreement with the experiments performed at temperatures from 850 to 900 °C and pressure of 100 kPa.

The tritium accumulation in the ITER first wall and its permeation through the first wall into the coolant were computed in the Kurchatov institute using codes TMAP-4 and TMAP-5. About 24 g of tritium will accumulate in the ITER first wall during 10 000 pulses. About 0.5 g of tritium will permeate into the first wall coolant during 6000 pulses (2 years of the reactor operation). Numerical modelling has shown that plasma disruptions insignificantly decrease tritium accumulation (by ~0.7 % during 10 000 normal pulses) and its permeation into the coolant (by 2.2 % during 5 500 normal pulses). Studies of the effect of Be-surface changing, resulted from the plasma disruptions, on the tritium accumulation in the beryllium first wall and on the tritium permeation through it has shown that such an effect is also insignificant [8].

4. POROUS BERYLLIUM

Porous (20–30 %) beryllium is proposed as possible material for neutron multiplier in ITER experimental modules of DEMO breeding blanket [14-16] developing in Russia at present. There must be a material with fully open porosity for helium and tritium to be eliminated efficiently. One of the possible ways for producing of such a material is the method of obtaining of a porous beryllium with the use of nanocrystalline beryllium developed in VNIINM [14]. The essence of method lies in thermal treatment of pre-compacted mixture of beryllium metal powder and beryllium hydride BeH_2. As the result of thermolysis of beryllium hydride BeH_2, at temperature of 250-350°C, to produce nanocrystalline beryllium (nano-Be) in the form of microcellular structure and to evolve hydrogen which promote to formation of fully open porosity [15-17]. This nano-Be plays a part of bonding between more coarse particles of beryllium metal powder. It is shown that proposed method permits to get articles with fully open controllable porosity and with a minimal difference in density [18, 19]. In changing the procedure parameters (temperature and compacting pressure) can be regulate an overall porosity of material.

The new data on the microstructure and compression strength recently were obtained in Bochvar Institute (Fig.6.). One can see that material have quite satisfactory strength – minimum value is 74.3 MPa and maximal value is 379.4 MPa under porosity of 34 and 10 % respectively. Material with such strength can machine handle – turning, drilling, etc., what extends the possibilities of technology.

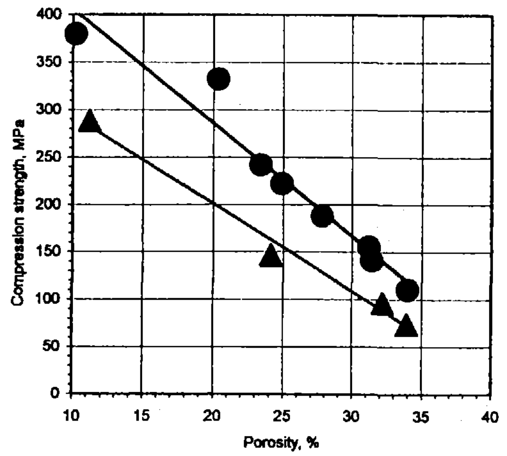


Fig. 6 Dependence of compression strength on porosity.

It was shown also that the structure of material consists of single of beryllium powder particles and an interparticle space filled full with a nano-Be phase. This phase has a microcellular structure with an average cells size of 1.26 μm. Walls thickness of a microcells are ~ 100-300 nm. The material has fully open porosity.

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