Journal of Nuclear Materials 386–388 (2009) 400–404

Contents lists available at ScienceDirect

Journal of Nuclear Materials

journal homepage: www.elsevier.com/locate/jnucmat

Status and strategy of fusion materials development in China

Q.Y. Huang [a,*], Y.C. Wu [a], J.G. Li [a], F.R. Wan [b], J.L. Chen [a], G.N. Luo [a], X. Liu [c], J.M. Chen [c], Z.Y. Xu [c], X.G. Zhou [d], X. Ju [b], Y.Y. Shan [e], J.N. Yu [f], S.Y. Zhu [f], P.Y. Zhang [f], J.F. Yang [g], X.J. Chen [h], S.M. Dong [i]

[a] *Institute of Plasma Physics, Chinese Academy of Sciences; School of Nuclear Science and Technology, University of Science and Technology of China, Hefei, Anhui 230031, China*
[b] *University of Science and Technology Beijing, Beijing 100083, China*
[c] *Southwestern Institute of Physics, Chengdu, Sichuan 610041, China*
[d] *National University of Defense Technology, Changsha, Hunan 410073, China*
[e] *Institute of Metal Research, Chinese Academy of Sciences, Shenyang, Liaoning 110016, China*
[f] *China Institute of Atomic Energy, Beijing 102413, China*
[g] *Xi'an Jiaotong University, Xi'an, Shaanxi 710049, China*
[h] *China Academy of Engineering Physics, Mianyang, Sichuan 621900, China*
[i] *Shanghai Institute of Ceramics, Chinese Academy of Sciences, Shanghai 200050, China*

ABSTRACT

The liquid metal and solid ceramic pebble breeder blankets have become the most promising blankets for ITER-TBMs or DEMO reactors in China and the world due to their potential advantages. In recent years the corresponding research work on fusion reactor materials mainly focuses on structural materials, plasma facing materials and the functional materials for the blanket such as breeder, coating and flow channel insert etc. for the successful application of fusion energy in the near future. The R&D on those materials in the two kinds of blankets is being carried out widely in China, including fabrication and manufacturing techniques, physical/mechanical properties assessment before and after irradiation, joining techniques for structural materials, compatibility evaluation, and the development and verification of the criteria for fusion material designs. The progress on main R&D activities of fusion reactor materials in China is introduced and prospected in the paper.

1. Introduction

According to the requirement for energy in China due to the national economy development at the high speed and the nuclear energy development strategy, a lot of fission reactors will be constructed before 2050. Considering the limitation of the fission fuels and the characteristics of fusion energy such as the unlimited fusion fuel, inertial safety, environmental and economical advantages etc., the fusion energy is considered as one of the key ways to successfully solve the energy problem, while the development of fusion materials is one of the keys to successful achievement of fusion power system.

In fusion reactors, the plasma facing components (PFC), such as the first wall (FW) and the divertor, and the breeding blanket will be exposed to plasma particles and electromagnetic radiation and will be suffered from irradiation by an intense fluence of high-energy neutrons. The neutrons will induce the degradation of series properties of the materials and result in residual radioactivity of the exposed materials. In addition, the fusion reactor materials must show high performance, high thermal stress capacity, good compatibility with coolant(s) and the other materials, long lifetime, high reliability, adequate resources, easy fabrication, reasonable cost, good safety and environmental behavior etc. [1].

Much progress has been achieved in China for researches on the fusion reactor materials under strong support from the government and hard research work with oversea collaboration. The liquid metal and solid ceramic pebble breeder blankets have become the most promising blankets for ITER-TBMs or DEMO reactors in China [2–4] due to their potential advantages. Current R&D activities on materials for these two kinds of blankets in China mainly focus on the structural, plasma facing and functional materials, which are done with many participants from several institutes and universities in China and under wide overseas collaboration. This paper is to introduce main achievements obtained and the near-term perspectives in the different fields in China.

2. Candidate structural materials

Candidate structural materials for future fusion reactors mainly include reduced activation ferritic/martensitic (RAFM) steels, V-alloys and SiC_f/SiC composites, and the RAFMs are considered as the most promising structural materials for the near future for their qualified fabrication routes, welding technology and available general industrial experience.

* Corresponding author.
E-mail address: qyhuang@ipp.ac.cn (Q.Y. Huang).

0022-3115/$ - see front matter © 2008 Elsevier B.V. All rights reserved.
doi:10.1016/j.jnucmat.2008.12.158

2.1. RAFM steel

A lot of work on China Low Activation Martensitic (CLAM) steel has been done in recent several years with wide collaboration in domestic and overseas [5]. According to the research and results, CLAM steel has good properties and most of them are similar to those of the other RAFMs, which have been widely researched in the world [5]. Some main properties of CLAM steel were included in the following sections.

2.1.1. Mechanical properties

The ultimate tensile strength (UTS) of the CLAM steel under room temperature (RT) and 600 °C for bar is 668 MPa and 514 MPa, respectively, the yield stress (YS) at RT and 600 °C is 334 MPa and 293 MPa, respectively. The tensile properties are comparable with those of Eurofer97 [6]. The Ductile–Brittle Transition Temperature (DBTT) for standard specimens is about −100 °C, which is lower than most reported DBTT values of the RAFMs [7]. This shows that CLAM has good fracture toughness under low temperature. Preliminary parallel creep test for CLAM and Eurofer97 was performed under 250 MPa stress at 550 °C, which showed that the creep rate and the rupture time of CLAM was less than that of Eurofer97 [5]. Another parallel creep tests for CLAM and JLF-1 showed that the rupture time of CLAM is longer than that of JLF-1 under the same test conditions [8].

2.1.2. Joining techniques

Hot Isostatic Pressing Diffusion Welding (HIP-DW) is a promising technology to fabricate the FW and cooling plates of ITER-TBM (Test Blanket Module) due to its many advantages. A lot of research on HIP-DW of the RAFMs has been done in the world [9,10]. Several HIP experiments on CLAM have been performed [11]. The impact absorbed energy of some joints reached the level of base metal. Further research on optimization of the HIP-DW technique is being carried out. A small scale FW module by HIP-DW is also planned. Researches on other joining methods such as EBW (Electron Beam Welding), LBW (Laser Beam Welding), DW and TIG (Tungsten Insert Gas) welding for CLAM are also being carried out in order to explore suitable techniques for joining the blanket components.

2.1.3. Irradiation tests

Preliminary neutron irradiation on CLAM was carried out at 250 °C to 0.02 dpa with a fast neutron fluence of $1.51*10^{19}$ n/cm^2 ($E > 1$ MeV). The results of mechanical properties before and after irradiation show that the UTS and DBTT increased for about 10 MPa and 5 °C, respectively, while the total elongation decreased about 4% [12]. The electron and ion irradiations on CLAM show the similar results with those of other RAFMs [5], CLAM has good irradiation swelling resistance.

2.1.4. Compatibility

The compatibility of RAFMs with flowing eutectic LiPb is a concern for LiPb blanket concepts and 5000 h corrosion test for CLAM has been completed in DRAGON-I loop. The temperature and the flow rate for the corrosion test are 480 °C and 0.08 m/s, respectively.

For the first 500 h exposure in LiPb, the specimen surfaces didn't suffer from a direct attack of the LiPb, due to the existence of passivated oxides on the surfaces which inhibited the contact of LiPb and the specimen surface [13,14]. After 2500 h, the passivated oxides layers in most area of the surface fell off, and the corrosive attack occurred in some parts of the steel matrix. And the steel elements such as Fe and Cr were dissolved into the LiPb. However, there were still areas without any attack [15,16]. Meanwhile, a slight weight change for CLAM was observed after 500 h and 1000 h, but a significant weight loss was observed after 2500 h with the value of 0.38 mg/cm^2. Longer time corrosion test for CLAM is underway.

2.2. V-alloy

V-alloys are candidate materials for structural components of fusion blanket because of their low long-term activation [17], high operating temperature [18] and acceptable post-irradiation properties [19]. The work on them in China mainly focuses on oxidation behavior [20], hydrogen embrittlement behavior [21], synergetic effects of hydrogen and oxygen [22], strength and thermal stability [23,24], and precipitation behavior [25]. The results are as follows.

Firstly, the oxidation rate was reduced by the addition of Al in the V–4Ti alloy, but it was slightly affected by the Cr element with ~4wt% in it. Then more protection for the matrix metal was given by the segregation of Al to the surface layer and the formation of the oxide Al_2O_3 [20].

Secondly, the resistance to the H embrittlement could be increased significantly by the addition of Ti/Al, due to the strong binding force of the alloying elements to the H in solid solution [21].

Thirdly, O didn't cause the ductility loss of the alloy. However, shifting of the critical H concentration to a lower level was resulted in, which made the alloy more sensitive to the H embrittlement. The synergetic effects of H and O on the ductility mainly result from the individual effect of O on the grain boundary strength and H on the grain strength. Both the grain boundary weakening by O and the grain hardening by H make the alloy with a high trend to intergranular fracture [22].

Fourthly, effective strengthening for the V-alloy was caused by W, comparing with Ti or Cr, and the H embrittlement properties was not affected by the addition of W [23]. V–4Cr–4Ti alloy could be greatly strengthened by ageing process and cold rolling, with insignificant loss of ductility or even higher static tensile fracture toughness. Both strengthenings were thermally stable below 500 °C. At higher temperature, the alloy in the state of complete recrystallization showed better thermal stability than the alloy in solid solution state [24].

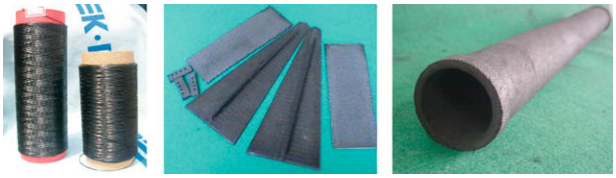
Finally, precipitation hardening was found in solid solution treated alloys after annealing below 900 °C and ageing at 600 °C. And in the V–6W–4Ti alloy, the precipitates were Ti and C-rich ones without enrichment of W. The growth of the precipitates was controlled by the diffusion of interstitial solution of C, N and O [25].

2.3. SiC_f/SiC composite

Continuous SiC fiber reinforced SiC matrix composites (SiC_f/SiC) are considered as one of the promising structure materials in fusion reactor research for their high strength, low density and high tolerable temperature. Lots of work has been carried out for years in China, and great progress has been achieved [26–28].

Recently, research of the fabrication technology of SiC_f/SiC composite focuses on the method of "CVI (Chemical Vapor Infiltration) + PIP (Precursor Infiltration Pyrolysis) + CVD (Chemical Vapor Deposition)" process [28]. Firstly, the SiC fibers were covered with a layer of SiC coating by CVI/CVD method. Secondly, the prefabrication as being designed was weaved by SiC fiber. Then, the PIP process was used to form the SiC_f/SiC matrix. Finally, the compact coating of SiC on matrix was produced with CVD process.

Based on development of the fabrication technology, some key productions have been achieved successfully, such as SiC fiber, plates of SiC_f/SiC composites and a SiC_f/SiC pipe with the length of 0.80 m, and so on, as shown in Fig. 1. The pipe was designed to be used in the DRAGON-II loop in order to achieve high operation temperature (>700 °C). Meanwhile, the SiC_f/SiC capsule will



s

(a) SiC fiber (b) 2D SiC_f/SiC plates (c) 2.5D SiC_f/SiC pipe

Fig. 1. Achievement of SiC_f/SiC composites in China.

be fabricated to study the compatibility of the structural materials with the static LiPb under ~1000 °C. Further work on some key issues, such as optimization of the SiC_f/SiC composites, joining technology between the SiC_f/SiC composites themselves or with other metals etc. is underway.

3. Plasma facing materials

Plasma facing material (PFM) is an essential part of PFCs. And researches on PFMs have been carried out for many years in China [29]. The low-Z PFMs, e.g. graphite, are being used in tokamak experiments and many R&D activities on low-Z materials have been carried out. The high-Z PFMs, e.g. tungsten, are increasingly considered as the primary candidate armor materials facing the plasma in tokamaks for medium to high heat flux components. Tungsten-based alloys are planned to be used as PFM in the long-term operation in EAST (Experimental Advanced Superconducting Tokamak) [30]. A new liquid diverter and liquid first wall concept has been put forward and study on its possibility for commercial pure fusion reactors is underway [29].

3.1. Low-Z plasma facing materials

Graphite is being widely used in current tokamak devices. A series of B-, Ti- and Si-doped graphites have been recently developed as low-Z PFMs to reduce its strong chemical sputtering (CS) and radiation enhanced sublimation (RES) at elevated temperature and widely used in tokamak machines [31]. The functionally graded material (FGM), e.g. SiC/C FGM instead of the SiC coating only, was the most successful one [32], and the thermal stress in PFC during thermal shock obviously decreased when FGM was adapted. The SiC coated multi-element doped graphite has successfully been applied as PFM in HT-7 tokamak [29] and it has been selected as PFM of EAST.

Many other Low-Z PFMs, e.g. C/C and SiC_f/SiC composite, Be etc., have also been studied [29].

3.2. High-Z plasma facing materials

The High-Z PFMs, e.g., W, Mo etc., have higher sputtering threshold energy and hence lower erosion rate and longer lifetime in comparison with that of graphite. W is more favorable in future breeders for its relatively low activation property. Some researches on W-alloy [33], W-coating [34,35] and W/Cu FGM [32] are being carried out.

The Oxide Dispersion Strengthening (ODS) technique could be used to enhance mechanical properties of tungsten, and the strengthening mechanism was studied by addition of 1% La_2O_3 to pure W. The results shows the flexural strength of the alloy reached 475 MPa, which is 35% higher than that of the pure tungsten with the same density [33].

3.3. Joining techniques

Joining technique is also a key issue for PFM and heat sink material in PFC, e.g., W and Cu, due to their great difference in thermal expansion, while FGM supplies a good solution for it. For example, W/Cu FGM developed recently offers transition layer between W and Cu [36]. Vacuum Plasma Spray (VPS) tungsten coatings on CuCrZr with W/Cu interlayer was fabricated and tested under high heat flux by electron beam with active cooling. Results showed that after 10 MW/m^2 thermal shock experiment, the W/Cu interlayer was not damaged except a few pores, exfoliation and cracks appeared.

4. Functional materials

4.1. Breeding materials

The liquid metal LiPb and solid ceramic pebble Li_4SiO_4 and Li_2TiO_3 have become the most promising tritium breeders for the blankets of ITER-TBMs or DEMO reactors in China and the world.

4.1.1. Liquid breeding materials

Compared with the solid breeder blanket, the liquid metal LiPb blanket has the following characteristics [37]: (1) Pb has higher neutron's economy, which can enhance the tritium breeding performance with no neutron multiplier such as Be added; (2) LiPb can fit tokamak's geometry well; (3) it is easy to extract tritium from liquid LiPb online.

The preliminary specifications for impurity levels of LiPb are listed in Table 1 [38]. The chemistry should be well controlled during the operation in the reactors, especially for the dynamic control of Li content and follow-up of tritium concentration etc.

The production of LiPb eutectic has been studied for years on the technology to control the impurities. 300 Kg LiPb has been produced (as shown in Fig. 2). And ~2 tons of LiPb eutectic are planned to be produced next year for DRAGON-IV loop, which is a forced convection loop to achieve the multi-functions such as researches on MHD effect, corrosion test under high temperature, stress corrosion, mini-TBM tests etc. and will be finished the construction at the end of 2009.

4.1.2. Solid breeding materials

The solid breeder has the following attractive characteristics [39]: (1) no MHD effect, (2) less danger for the reaction between

Table 1
Specifications for impurity levels of LiPb (wppm).

Element	Ag	Al	As	Au	Cr	Cu	Cd	Fe	Ga
wppm	<1	1	40	<1	<3	<1	<1	13	<1
Element	Hg	In	Li	Ni	P	Sb	Su	Ti	Bi
wppm	<1	<1	0.68%	7	<10	<10	<5	<10	0.02%

Fig. 2. Heats of LiPb produced in China.

Li and other elements, (3) good thermal and chemical stability, (4) good tritium release characteristics. Li_4SiO_4 and Li_2TiO_3 etc. are considered as the promising candidate solid breeders in China [40,41]. Li_4SiO_4 is considered as the primary candidate breeder, and Li_2TiO_3 and other acceptable ceramics as alternative materials. The R&D on solid breeders mainly focuses on γ-$LiAlO_2$, Li_2ZrO_3, Li_4SiO_4 and Li_2TiO_3 in China [40]. Systematic work has been done on γ-$LiAlO_2$ system including the thermal, mechanical, electrical properties, especially on in-pile and out-of-pile tritium release. And a wet process fabrication technology of the Li_4SiO_4 pebbles has been developed. The sphericity of the pebbles is higher than 90%. The phase purity of Li_4SiO_4 is larger than 95% according to the XRD data. And the SEM photograph (shown in Fig. 3) shows that the pebbles have abundant cellular structure. Further research work of their physical, chemical and tritium related properties are being performed.

4.2. Coating materials

Coatings have been proposed as the solution to critical materials constrains for most of the blanket concepts under development for fusion power applications. The work on coatings for fusion research in China included SiC coating on 316L by Radio Frequency Sputtering (RFS), TiC and TiN + TiC + TiN/SiO_2 coatings on 316L by Physical Vapor Deposition (PVD) etc. In addition, preliminary research on Al_2O_3 coating on CLAM by CVD with comparison experiment on Eurofer97 was done and the results show it had higher micro-hardness and higher density than that of Eurofer97 steel. The study on compatibility with LiPb, H/D/T permeation, in situ self-healing on defects and irradiation effect will be performed [42–44].

For China ITER-DEMO design, the proposed criteria is a TPRF (Tritium Permeation Reduction Factor) >1000 for gas-phase permeation tests and a TPRF > 100 for tests in liquid LiPb compared to the reference permeation rates through bare CLAM steel at the typical wall temperature of $\sim$550 °C, the product of the electrical resistivity of the coating and its thickness should exceed a value of $\sim 0.01\ \Omega\,m^2$, and the coating has the potential ability for in situ self-healing on defects. While for China ITER-TBM, the TPRF in LiPb should be larger than 10, the other parameters are the same as the DEMO design [45].

Al_2O_3 is considered as the primary candidate coating material for LiPb blankets, while researches on the other coating materials, such as SiC etc., are underway. Considering the complicated geometry of LiPb blanket, development of coatings has been focused on Al_2O_3 formed on aluminized CLAM steels by the methods of HDA (Hot-Dipped Aluminization) and CVD etc. In addiation, research on a multilayer coating i.e. Al_2O_3 + SiC coating on CLAM for LiPb blankets is also undergoing [46–48].

On the other hand, Al_2O_3 coating and TiC/TiN coating or their composite are also selected as candidate tritium permeation barrier for the ceramic pebble blanket, and the latter with good properties were fabricated by hollow cathode ion plating assisted with reaction gas of C_2H_2. Further work will be performed in the near future.

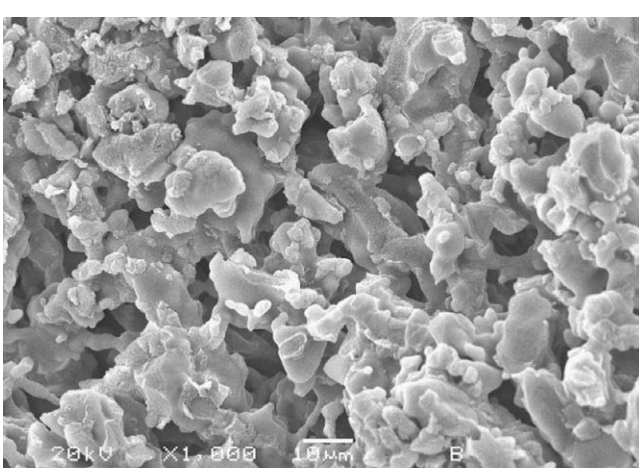


Fig. 3. Microstructure of the pebbles.

4.3. FCI material

Flow Channel Insert (FCI) component would act as an electrical and thermal insulating layer in liquid blanket designs to reduce MHD effect and to elevate the outlet upper temperature of liquid metal, which can enhance the efficiency of the reactor [49]. It will divide the flow channel into two regions. The large inner region contains the flowing LiPb with high temperature up to 480–700 °C and the thin outer region contains a cooler stagnant layer of LiPb that reduces the potential erosion/corrosion of the steel walls in LiPb blanket designs [50].

SiC_f/SiC composites are now considered as the primary candidate materials for FCI in LiPb blankets due to its resistance to irradiation, high temperature creep and high temperature compatibility with LiPb up to 800–1000 °C. The required properties for SiC_f/SiC-FCI are low transverse thermal and electrical conductivities, e.g. below 2–20 W/m K and 5–500 $\Omega\,m^{-1}$, respectively [46]. The typical thickness of FCI is $\sim$5 mm. Considering the geometry and required performance of FCI, 2D-SiC_f/SiC composites can be chosen and the fabrication method can be CVI or PIP. Its compatibility with LiPb, influence on MHD and the irradiation effects should be studied.

5. ITER related R&D activities

Based on China ITER-TBM designs, a mini-TBM made of 316L with dimensions of $330 \times 170 \times 195\ mm^3$ was fabricated successfully to test the techniques for fabrication of the future TBM. Further experiment on it is planned to study the corrosion behavior of the overall mockup. On the other hand, EAST can serve as a pre-testing platform for ITER-TBM because the basic electromagnetic parameters and the average heat flux density at the FWs is comparable to those of ITER. So a one third scale mockup made by CLAM will be manufactured and tested in EAST firstly.

The other work is R&D on ITER shield blanket modules applied to the neutral beam port recent years [51,52]. Small Be/Cu mockup of $50 \times 50 \times 32\ mm^3$ with one Be title to one CuCrZr plate and $80 \times 280 \times 84\ mm^3$ with three Be tiles for ITER shield blankets were fabricated using the HIPing technology [53].

6. Near-term strategy

ITER-TBM will be put into ITER Tokamak on the first day of its operation to perform the integral test to verify the material performance and the design of the TBM. So development of fusion material in China will focus on the requirement of the TBM fabrication.

For the structural material, development of RAFM steel, e.g. CLAM, will be the emphasis. Rounded database of CLAM steel is needed for the design of TBM and obtainment of the license to be used in ITER. To fabricate qualified TBM, welding techniques and processing technology of CLAM are needed to be studied further. Also, properties of welding joints before and after neutron irradiation are needed to be tested as well.

For the plasma facing materials, the development of fabrication techniques of W-coating and joining techniques of W to heat sink material is very important and need further research on it.

For the breeding materials, their fabrication for higher impurity control and better properties is needed for both LiPb and solid breeding materials. The compatibility of breeding material with structural material and protective coating should be studied. Also, research on extraction of tritium from Li_4SiO_4 and Li_2TiO_3 should be performed. As for the tritium permeation coating, R&D activities should focus on the fabrication technology which can be used for large and complex configurations. And the development of large-scale SiC_f/SiC composite with low transverse thermal conductivity and low electrical conductivity is needed for the requirement of FCI.

7. Summary

Commercial application of pure fusion power is the ultimate goal of the fusion research and still need great efforts from each country in the world. Material is one of the key issues for the realization of fusion energy. The R&D on the structural, plasma facing and functional materials in the two kinds of blankets is being carried out widely in China with strong support from the government, including fabrication and manufacturing techniques, physical/mechanical properties assessment before and after irradiation, joining techniques for structural materials, compatibility evaluation, and the development and verification of the criteria for fusion material designs. And much progress has been achieved with oversea collaboration. Currently, further work will focus on the requirement of TBM. And there are still many difficulties and wide international collaboration is needed.

Acknowledgements

This work was supported by the National Natural Science Foundation of China with the Grant Nos. 10775135, 10375067, 50871108 and 10675123, the Knowledge Innovation Project of Chinese Academy of Sciences, the State Major Basic Research Development Program with Grant No. 2007CB209900, the JSPS-CAS Core University Program.

References

[1] N. Baluc, K. Abe, J.L. Boutard, V.M. Chernov, E. Diegele, S. Jitsukawa et al., Nucl. Fusion 47 (2007) 696.
[2] Y. Wu, FDS Team, Fusion Eng. Des. 81 (2006) 2713.
[3] Y. Wu, S. Zheng, X. Zhu, W. Wang, H. Wang, S. Liu et al., Fusion Eng. Des. 81 (2006) 1305.
[4] Y. Wu, FDS Team, Fusion Eng. Des. (2008), doi:10.1016/j.fusengdes.2008.06.048.
[5] Q. Hunag, C. Li, Y. Li, S. Liu, Y. Wu, J. Li et al., Chin. J. Nucl. Sci. Eng. 27 (2007) 41.
[6] M. Rieth, M. Schirra, A. Falkenstein, P. Graf, S. Heger, H. Kempe et al., FZKA 6911 (2003).
[7] Y. Li, Q. Huang, Y. Wu, T. Nagasaka, T. Muroga, J. Nucl. Mater. 367–370 (2007) 117.
[8] Y.F. Li, T. Nagasaka, T. Muroga, Q.Y. Huang, Y.C. Wu, J. Nucl. Mater. 386–388 (2009) 495.
[9] K. Furuya, E. Wakai, M. Ando, T. Sawai, K. Nakamura, H. Takeuchi et al., J. Nucl. Mater. 307–311 (2002) 289.
[10] P. Norajitra, G. Reimann, R. Ruprecht, L. Schäfer, J. Nucl. Mater. 307–311 (2002) 1558.
[11] C. Li et al., FDS Team, Fusion Eng. Des. 82 (2007) 2627.
[12] L. Peng, Q. Huang, C. Li, S. Liu, J. Nucl. Mater. 386–388 (2009) 312.
[13] Q. Huang, M. Zhang, Z. Zhu, S. Gao, Y. Wu, Y. Li et al., Fusion Eng. Des. 82 (2007) 2655.
[14] Q. Huang, S. Gao, Z. Zhu, M. Zhang, Y. Song, C. Li et al., Fusion Eng. Des. (2009), doi:10.1016/j.fusengdes.2008.12.038.
[15] M. Zhang, Q. Huang, Y. Wu, Z. Zhu, S. Gao et al., Mater. Sci. Forum 561–565 (2007) 1741.
[16] Z. Zhu, M. Zhang, S. Gao, Y. Song, C. Li, L. Peng et al., Fusion Eng. Des. 84 (2009) 5.
[17] R.H. Jones, H.L. Heinisch, K.A. McCarthy, J. Nucl. Mater. 271 (1999) 518.
[18] R.J. Kurtz, M.L. Hamilton, J. Nucl. Mater. 283–287 (2000) 628.
[19] D.L. Smith, H.M. Chung, B.A. Loomis, H.C. Tsai, J. Nucl. Mater. 233–237 (1996) 356.
[20] J. Chen, Y. Xu, Y. Deng, Z. Xu, Plasma Sci. Technol. 5 (2003) 1661.
[21] J. Chen, S. Qiu, T. Muroga, Y. Xu, T. Nagasaka, Y. Chen et al., J. Nucl. Mater. 334 (2004) 143.
[22] J. Chen, Y. Xu, Y. Deng, L. Yang, S. Qiu, Plasma Sci. Technol. 5 (2003) 2051.
[23] J. Chen, T. Muroga, S. Qiu, T. Nagasaka, W. Huang, M. Tu et al., J. Nucl. Mater. 329–333 (2004) 401.
[24] J. Chen, T. Muroga, T. Nagasaka, S. Qiu, C. Li, Y. Chen et al., Fusion Eng. Des. 81 (2006) 2899.
[25] J. Chen, T. Muroga, T. Nagasaka, Y. Xu, C. Li, S. Qiu et al., J. Nucl. Mater. 334 (2004) 159.
[26] S. Dong, Y. Ding, D. Jiang, K. Akira, J. Inorg. Mater. 20 (2005) 883.
[27] Z. Li, L. Zhang, L. Chen, X. Wang, Y. Xu, Aeronaut. Manuf. Technol. 7 (2005) 68.
[28] X. Zhou, C. Zhang, H. Zhang, R. Liu, H. Yu, J. Mater. Sci. Eng. 24 (2006) 815.
[29] J. Qian, Y. Wu, J. Li, J. Nucl. Mater. 307–311 (2002) 1637.
[30] G. Luo, X. Zhang, D. Yao, X. Gong, J. Chen, Z. Yang, et al., Phys. Scr. T128 1–5, doi:10.1088/0031-8949/2007/T128/001.
[31] J. Yu, Z. Yao, G. Yu, F. Chu, X. Tang, Y. Zeng et al., J. Nucl. Mater. 283–287 (2000) 1077.
[32] C. Ge, J. Li, Z. Zhou, W. Cao, W. Shen, M. Wang et al., J. Nucl. Mater. 283–287 (2000) 1116.
[33] Y. Chen, Y. Wu, F. Yu, J. Chen, Rare Mater. Eng. 36 (2007) 822.
[34] X. Jiang, F. Gitzhofer, M.I. Boulos, Plasma Sci. Technol. 8 (2006) 164.
[35] F. Chong, J. Chen, J. Li, Surf. Technol. 33–37 (2005) 37.
[36] F. Chong, J. Chen, J. Li, J. Nucl. Mater. 363–365 (2007) 1201.
[37] Y. Wu, W. Wang, S. Liu, Q. Huang, S. Zheng, H. Wang et al., Chin. J. Nucl. Sci. Eng. 25 (2005) 347.
[38] O. Gastaldi, L. Sedano, LiPb specifications and the chemistry control, in: Coordinating Meeting on R&D for Tritium and Safety Issues in Lead–Lithium Breeders (PbLi-T 2007), Idaho Falls, ID, USA, June 11–12, 2007.
[39] T. Yuan, K. Feng, X. Wang, Z. Chen, Nucl. Fusion Plasma Phys. 26 (2006) 195.
[40] X. Chen, H. Wang, Y. Luo, K. Feng, Z. Wen, Nucl. Fusion Plasma Phys. 26 (2006) 210.
[41] K. Feng, New progress and status of China HC-SB TBM program, presented at TBWG-18 meeting, Aix en Provence, France, March 20–22, 2007.
[42] P. Wang, J. Liu, Y. Wang, B. Shi, Surf. Coat. Technol. 128&129 (2000) 99.
[43] Z. Yao, J. Hao, C. Zhou, C. Shan, Atom. Energy Sci. Technol. 34 (2000) 65.
[44] X. Li, G. Yu, J. Yu, K. Wang, Q. Huang, J. Nucl. Mater. 329–333 (2004) 1407.
[45] Y. Wu, FDS Team, J. Nucl. Mater. 367–370 (2007) 1410.
[46] Z.L. Yan, Q.Y. Huang, Y. Song, Z.H. Guo, Y.C. Wu, FDS Team, Chin. J. Nucl. Sci. Eng. 28 (2008) 289.
[47] Z.H. Guo, Q.Y. Huang, Y. Song, Z.L. Yan, Y.C. Wu, FDS Team, Chin. J. Nucl. Sci. Eng. 28 (2008) 295.
[48] Q. Huang, Y. Song, Z.H. Guo, Z.L. Yan, C.J. Li, Y.L. Wang et al., Chin. J. Nucl. Sci. Eng. 28 (2008) 256.
[49] G. E. Youngblood, R.J. Kurtz, R.H. Jones, SiC/SiC design for a flow channel insert, Pacific Northwest National Laboratory (PNNL), Department of Energy by Battelle Memorial Institute Under Contract DE-AC06-76RLO-1830.
[50] Y. Wu, FDS Team, Nucl. Fusion 47 (2007) 1533.
[51] F. Zhang, W. Kang, J. Wu, Y. Fu, J. Chen, E. Filippo, ITER shield blanket design activities at SWIP, in: Annual Report of SWIP, 2006.
[52] H. Yan, Cooling enhancing about front lid of the shield blanket in ITER, in: Proceedings of the 9th China–Japan Symposium on Materials for Advanced Energy System and Fission and Fusion Engineering (CJS-9), Guilin, China, October 23–26, 2007.
[53] Fusion reactor and materials division, progress in fusion technologies and materials researches, in: Annual Report of SWIP, 2006.