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ScienceDirect
Journal of Nuclear Materials 367–370 (2007) 1410–1415
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Design status and development strategy of China liquid lithium–lead blankets and related material technology

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Abstract

A series of fusion reactors (named FDS series) have been designed and assessed in China, with four types of liquid lithium lead blankets including the RAFM steel-structured He-cooled quasi-static LiPb tritium breeder (SLL) blanket, the RAFM steel-structured He–LiPb dual-cooled (DLL) blanket, the RAFM steel-structured refractory material thermally-insulated high temperature LiPb (HTL) hydrogen production blanket and the RAFM steel or optionally the austenitic stainless steel-structured He–LiPb dual-cooled high level waste transmutation (DWT) blanket. To demonstrate and validate the feasibility of the candidate blankets for fusion energy application, the three-phases-strategy of TBM (test blanket module) development, i.e. material R&D and out-of-pile experimental mockup, EAST-TBM and ITER-TBM have been proposed. A brief overview of the four types of LiPb blanket designs and their goals are given. Material technology requirement and development strategy are also presented in this paper.

1. Introduction

A series of fusion reactors (named FDS series) have been designed and assessed by the FDS (fusion design study) Team in China [1]. Up to now, four reactor concepts have been developed, which are the fusion-driven subcritical system (named FDS-I), the spherical tokamak-based compact reactor (named FDS-ST), the fusion electrical generation reactor (named FDS-II), and the fusion-based hydrogen production reactor (named FDS-III). In the same time, four types of liquid lithium–lead (LiPb) blanket concepts have been developed, including the He–LiPb dual-cooled high level waste transmutation (DWT) blanket, the He-cooled quasi-static lithium lead tritium breeder (SLL) blanket, the He/LiPb dual-cooled lithium lead (DLL) blanket and the high temperature liquid LiPb (HTL) blanket considering SiC_f/SiC composite or other refractory materials as FCIs (flow channel inserts) in the LiPb flow channels. The four types of blankets are possessed of many common features, e.g. the RAFM (reduced activation ferritic/martensitic) steel as the structural material, the LiPb eutectic as tritium breeder, He/LiPb as coolants etc., as well as some different requirements for material technology development due to different goals requiring corresponding outlet temperatures of coolants.

In this contribution, in addition to a brief introduction to the four types of LiPb blankets of the

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0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2007.04.031

FDS series fusion reactors, the requirements for blanket materials and the development strategy (test blanket modules, in-pile/out-of-pile testing, etc.) in China are presented based on the features of the four types of blankets.

2. LiPb blanket concepts for the FDS series fusion reactors

2.1. He–LiPb dual-cooled high level waste transmutation (DWT) blanket

The DWT blanket concept is designated for the fusion-driven subcritical reactor FDS-I which is to transmute the long-lived nuclear wastes from fission power plants and to produce fissile nuclear fuel for feeding fission power plants as an intermediate step and early application towards final application of fusion energy on the basis of easily-achieved plasma physics and engineering technology [2–4]. The plasma core parameters are being optimized assuming the constraint of lower than ITER levels (the neutron wall loading $\sim$0.5 MW/m^2, fusion power $\sim$500 MW). The DWT blanket design (see Fig. 1) focuses on the technology feasibility and concept attractiveness to meet the requirement for fuel sustainability, safety margin and operation economy. A series of design scenarios, with emphasis on circulating particle or pebble bed fuel forms considering geometry complexity of tokamak and frequency of fuel discharge and reload (including design of an emergency fuel discharge subsystem to improve the safety potential of the system), are being evaluated and optimized considering various blanket module structure and fuel forms. A design and its analysis on the DWT blanket with carbide heavy nuclide particle fuel in circulating liquid LiPb coolant (named DWT-CPL) has been studied for years. Other concepts, such as the DWT blanket with oxide heavy nuclide pebble bed fuel cooled in circulating helium gas (named DWT-OPG) and with nitride heavy nuclide particle fuel in circulating helium-gas (named DWTNPG) are also being investigated [4]. The RAFM steel or optionally the austenitic stainless steel (e.g. 316SS)-structured DWT blanket concept is adopted and the coating is considered to reduce LiPb corrosion and to reduce MHD pressure drop. The pressurized helium gas and liquid lithium lead are selected as coolants working in the temperature range of 250–450 °C.

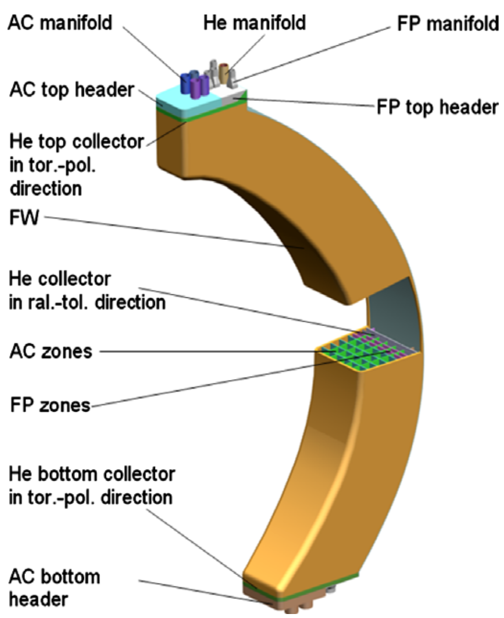

Fig. 1. The reference model of DWT blanket.

2.2. He-cooled quasi-static lithium lead (SLL) and He/LiPb dual-cooled lithium lead (DLL) blankets

Both SLL and DLL LiPb blanket are adopted for the fusion electrical generation reactor FDS-II, which is designated to exploit and evaluate potential attractiveness of pure fusion energy application, i.e. obtaining a high-grade heat for generation of electricity on the basis of conservatively advanced plasma parameters (fusion power 2–3 GW, neutron wall loading 2–3 MW/m^2) which can be limitedly extrapolated from the successful operation of ITER [5]. It is understandable that the FDS-II requirement for plasma technology could be met by the development of ITER and/or FDS-I.

The DLL blanket, which adopts the RAFM steel as structural material, is a dual-cooled LiPb breeder system with helium gas to cool the first wall and main structure and LiPb eutectic to be self-cooled. The FCIs, e.g. SiC_f/SiC composite, are designed as the thermal and electrical insulators inside the LiPb flow channels to reduce the magnetohydrodynamic (MHD) pressure drop and to allow the coolant LiPb outlet temperature up to 700 °C for high thermal efficiency. The exploded 3D view of DLL blanket is shown in Fig. 2.

The SLL blanket is another option of the FDS-II blanket if the critical issues of the DLL blanket could not be solved and validated by testing in ITER. To avoid or mitigate those critical problems resulting from MHD effects and FCI technology,

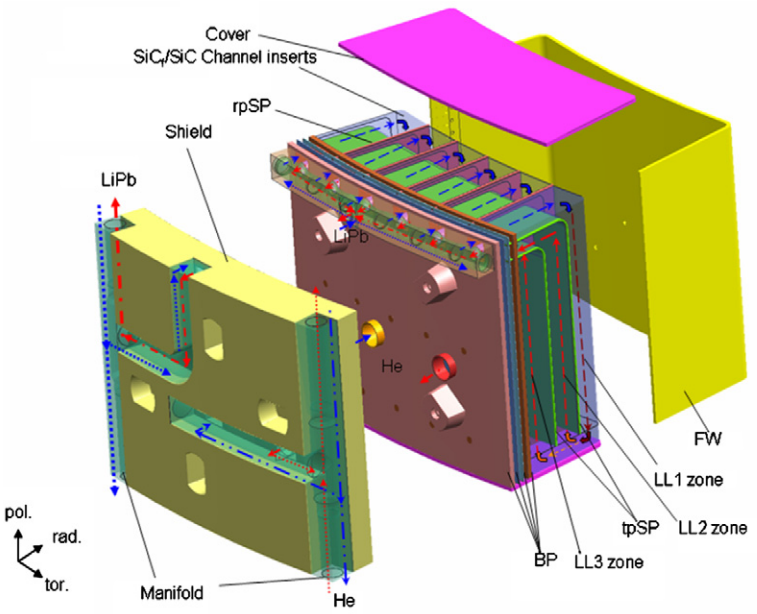

Fig. 2. The exploded 3D view of DLL blanket.

the SLL blanket is designed to use quasi-static LiPb flow instead of fast moving LiPb with the similar structure configuration and the structure material as of the DLL blanket. The heat in the SLL blanket is removed by pressurized helium gas with the outlet temperature of ~450 °C.

2.3. The high temperature liquid LiPb (HTL) blanket

The HTL blanket is designed for the fusion-based hydrogen production reactor FDS-III, which aims to obtain the high temperature heat in the blanket of fusion reactor for efficient production of hydrogen based on the promising extrapolation of material technology. The innovative HTL blanket design is considered to obtain high temperature heat with the special multi-layer FCIs inside the LiPb flow channels based on the relatively mature and most promising RAFM steel (allowed temperature up to 550 °C) as structural material [6]. In this multi-layer flow channel concept, Low temperature LiPb flows into the channel, then meanders through the multi-layer flow channel inserts, the temperature of the coolant LiPb is improved step by step, at last it is exported from the blanket in the high outlet temperature. The refractory materials (e.g. SiC_f/SiC composite) with low thermal conductivity are adopted as flow channel electrical and thermal insulators to allow the outlet temperature of coolant LiPb up to about 1000 °C. The 2D and 3D schematic views of the LiPb channel concepts for the HTL blanket are shown in Fig. 3.

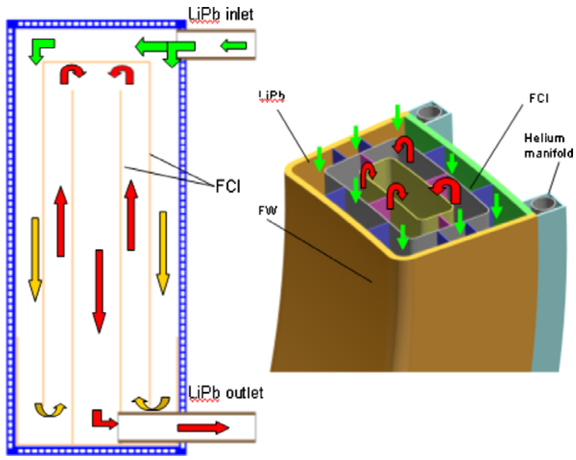

Fig. 3. The schematic views of LiPb channels for HTL blanket.

3. Requirements for materials

Structural steel: The RAFM steel is the primary candidate of the structural material for the China LiPb blankets because of its better swelling resistance, better thermo-physical and thermo-mechanical properties compared with Type 316 stainless steel in the range of temperature [7]. The improved austenitic steel may be considered optionally as the candidate structural material for the DWT blanket because of lower fusion neutron fluence (neutron wall loading $\sim$0.5 MW/m^2) and lower operation temperature (max. coolant outlet temperature $T_{max} \sim 450$ °C) in the DWT sub-critical blanket. Development of structural steels for China LiPb blankets will focus on irradiation resistance to high energy neutrons and corrosion resistance to high temperature LiPb.

Coating: For the LiPb blankets, coatings are needed to achieve the following functions: to reduce tritium permeation, to reduce helium leakage into the plasma chamber, to mitigate MHD effects in self-cooled liquid metal flow systems, and to reduce corrosion of structural steels by LiPb [8].The determination of tritium permeation reduction factor (TPRF) of the TPB(tritium permeation barrier) coating is a complex task which needs to take into account the site release limit of tritium and many factors in specific reactor designs. A typical number for TPRF may be 100–1000 for the China LiPb blanket designs. The MHD coating requires a high product of electrical resistivity and thickness above 0.01 Ω m^2. To assure complete coverage, a coating thickness of 5–10 µm is considered desirable, and therefore, an electrical resistivity (ρ_e) of 10^3–10^4 Ω m would be adequate. In addition, the principal requirement for the coating is compatibility between the flowing LiPb and the substrate wall steel in the environment of high temperature and high irradiation. An Al_2O_3 coating is the primary candidate for the LiPb blankets.

FCI: FCI is designated to reduce the MHD effect of the flowing LiPb as an electrical insulator and to elevate the exit temperature of LiPb as a thermal insulator [8]. SiC_f/SiC composites are the primary candidate materials for FCIs for the LiPb blankets. Thermal and electrical conductivities (σ_{th}, σ_e) below 2–20 W/mK and 5–500 $(\Omega\ \mathrm{m})^{-1}$ are required respectively. Open issues for FCIs are the fabrication of SiC_f/SiC composite flow channel inserts and their compatibility with flowing LiPb.

Fission materials: The chemical compositions and forms of fission nuclear fuels for the DWT blankets need to be optimised further considering the effectiveness of waste transmutation/fuel breeding and the expected material technology [3]. TRISO(Tri-ISOtropic)-like coated actinide carbide particle suspended in the LiPb slurry is considered as one of the options in the design of actinide fuel form referring to the maturity of HTGR (high temperature gas cooled reactor) fuel fabrication. Silicon carbide (SiC) as the peripheral cladding is considered as the most compatible material with LiPb currently. Besides following the development of the HTGR and LMFBR materials, the compatibility between fission fuels and flowing LiPb needs attention.

The requirements and working conditions of the FDS LiPb blankets for material technology development are briefly summarized in Table 1.

4. TBMs and development strategy

To check and validate the feasibility of the China LiPb blankets, the dual-functional lithium lead-test blanket module (DFLL-TBM) system, which is

Table 1
The working conditions and performance requirements for blanket materials

Function	Material candidates, working conditions and performance requirements			
	DWT	SLL	DLL	HTL
Structure	RAFM ($T_{max} \sim 550$ °C, anti-irradiation, anti-corrosion)			
Tritium breeder	Lithium lead eutectic			
Coating	Al_2O_3 or others ($T_{max} \sim 500$ °C, anti-irradiation, anti-corrosion, low σ_e)			
Coolant (max. temp.)	He and LiPb (450 °C for LiPb)	He (450 °C)	He and LiPb (700 °C for LiPb)	He and LiPb (900–1000 °C for LiPb)
FCI	No/yes	No	SiC_f/SiC or others ($T_{max} \sim 700$ °C low σ_{th} and low σ_e)	SiC_f/SiC or others ($T_{max} \sim 700$ °C low σ_{th} and low σ_e)
Fission fuel	HLW/U/Pu	No	No	No

designated to demonstrate the integrated technologies of both He single coolant (SLL) and He–LiPb dual-coolant (DLL) blankets, is proposed for test in ITER [9,10]. The DFLL design allows the strategy of an earlier test of SLL model, evolving to later test of DLL model after the issues related to FCIs and MHD effects can be solved. That means two modes of TBM should be tested in ITER with as similar as possible basic structure and auxiliary system except for including FCIs and faster flowing LiPb in the DLL model. The integrated test and validation of the remaining critical issues related to the DWT blanket and the HTL blanket can be conducted after successful operation of ITER.

The TBM development program covers three-phases, i.e. (1) Materials R&D and out-of-pile blanket mockup (e.g. 1/3 size) test; (2) middle scale (e.g. 1/2 size) TBM (called EAST-TBM) test in the EAST superconducting tokamak; (3) full size TBM (called ITER-TBM) test in ITER.

4.1. Materials R&D and out-of-pile blanket mockup test

The materials R&D for the China LiPb blankets focus on the development of CLAM (China low activation martensitic/ferritic) steel [11] as the structural material, Al-based coatings as the TPB [12], SiC_f/SiC FCIs for MHD control. R&D for the CLAM focus on the activation impurities control in large scale melting, the radiation hardening and embrittlement behaviour at 300–400 °C, the mechanical properties such as creep, fatigue and fracture toughness at 450–600 °C, the corrosion performance in flowing LiPb, the optimization of several welding techniques such as HIP (hot isostatic pressing) diffusion bonding, EB (electron beam) welding and laser welding, etc. and the properties study before and after irradiation. As for coatings, main attention will be paid to development of Al_2O_3, including its fabrication on CLAM steel by means of HDA (hot dip aluminizing) and CVD (chemical vapor deposition), etc., tests of compatibility with LiPb, H/D/T permeation, in situ self-healing capability to repair defects and the performances under neutron irradiation. R&D for SiC_f/SiC FCIs is addressed on the fabrication and joining technology, the chemical compatibility with LiPb at high temperature and the influence on MHD effect. Moreover, FCIs should be characterized by the tests of thermal conductivity, electrical conductivity and mechanical properties under irradiation.

An out-of-pile test of small scale mockup (e.g. 1/3 size) under the out-of-tokamak conditions is mainly focused on the TBM fabrication route and techniques, the thermo-mechanical/thermo-fluid dynamic performances, the MHD effects of flowing LiPb, the compatibility between flowing LiPb and structural steel, and the reliability and safety with regard to the EAST/ITER tokamak operation standards. A few experimental loops need to be constructed. Thermal convection LiPb loops are designated to test the compatibility between RAFM steels, coatings, FCIs and very slowly flowing LiPb at high temperatures (500–700 °C). The forced convection LiPb loop is designated mainly to test the MHD effect and thermo-fluid dynamics performance of fast flowing LiPb at strong magnetic field of a few Tesla. The impurities purification techniques will be developed as well. A He gas loop is to be built to test He gas thermo-fluid performance under the conditions of high temperatures and high pressures and to validate the heat removal capacity from the blanket mockups.

4.2. TBM test in EAST

The He–LiPb dual-coolant loop and relevant testing facilities are to be constructed to test the middle-scale (1/2 size) TBM in the EAST super-conducting tokamak in order to validate the design tools and codes for electro-magnetics (EM), thermo-mechanics, and partially neutronics, to check the availability of diagnostic instruments and to assess the TBM influence on plasma performances and versus as well as to demonstrate the feasibility and availability of DFLL-TBM auxiliary system design before the DFLL-TBM system is installed in ITER. The goals of this test phase can be achieved on the basis of the expected EAST parameters of the port size ($0.97\ m \times 0.53\ m$), the first wall surface heat flux ($0.1–0.2\ MW/m^2$), the toroidal field (3.5–4 T), the D–D neutron rates ($10^{15}–10^{17}$ n/s) and the plasma discharge pulse length (up to 1000 s), which are comparable to the ITER parameters in the corresponding phases.

4.3. TBM test in ITER

The full-scale consecutive TBMs are to be tested during the different operation phases of ITER. Therefore, the 'act alike' DFLL-TBMs are designed in turn as EM-TBM for test of EM effects, as NT-TBM for performance of neutronics, as TT-TBM for thermo-mechanics and tritium behavior, and as IN-TBM for integrated performance test, respectively, on different phases of the ITER operation. This program will allow consecutive validation of the SLL/DLL blanket concepts, technologies and design tools with reliable and safe operation, and finally to demonstrate relevant technologies for the fusion DEMO reactors on the basis of the expected ITER parameters of 1/2 port size ($0.524\ \mathrm{m} \times 1.7\ \mathrm{m}$), the first wall surface heat flux (0.1–0.3 MW/m^2), the toroidal field ($\sim$5 T), the D–T neutron rates ($\sim 10^{20}$ n/s) and the plasma discharge pulse length ($\sim$400 s). This test program is developed assuming successful testing in earlier phases.

5. Summary

Four types of liquid LiPb blanket concepts for the FDS series fusion reactor designs are being developed and assessed in China, which are named the DWT (dual-coolant waste transmutation), the SLL (single coolant lithium lead breeder), the DLL (dual-coolant lithium lead breeder) and the HTL (high temperature liquid breeder) blankets. Commonly the RAFM steel is adopted as the structural material and LiPb eutectic as tritium breeder and pressurized He gas as the structure coolant. However, the four concepts are designated to obtain different grades of heat at different temperatures (450–1000 °C) with different designs.

The requirements for blanket materials including structural materials, breeder materials, coolant materials and functional materials (coatings and FCIs) have been summarized on the basis of operational conditions of the four types of blankets.

The three-phases-strategy of development of TBMs and material R&D have been proposed, i.e. the DFLL-TBM system is to be developed through the three phases: tests in out-of-pile loops, EAST and ITER. The ongoing R&D activities on China LiPb blankets can be found in the Ref. [13].

References

[1] Y. Wu, The FDS Team, Fus. Eng. Des. 81 (2006) 2713.
[2] Y. Wu, Plasma Sci. Technol. 3 (6) (2001) 1085.
[3] Y. Wu, J.P. Qian, J.N. Yu, J. Nucl. Mater. 307–311 (2002) 1629.
[4] Y. Wu, S. Zheng, X. Zhu, W. Wang, H. Wang, S. Liu, et al., Fus. Eng. Des. 81 (2006) 1305.
[5] Y. Wu, W. Wang, S. Liu, J. Li, H. Wang, H. Chen, et al., Chinese J. Nucl. Sci. Eng. 25 (1) (2005) 76.
[6] H. Chen, Y. Wu, in: The 21st International Atomic Energy Agency Fusion Energy Conference, 16–20 October, 2006, Chendu, China.
[7] N. Baluc et al., J. Nucl. Mater., in press, doi:10.1016/j.jnucmat.2007.03.036.
[8] C.P.C. Wong et al., J. Nucl. Mater., in press, doi:10.1016/j.jnucmat.2007.03.241.
[9] Y. Wu, W. Wang S. Liu, H. Chen, S. Zheng, Y. Bai, et al., Design description document for the Chinese dual-functional lithium lead-test blanket module for ITER, 2005.
[10] Y. Wu, the FDS Team, in: The 21st International Atomic Energy Agency Fusion Energy Conference, 16–20 October, 2006, Chendu, China, Nucl. Fus., to be published.
[11] Q. Huang et al., J. Nucl. Mater., in press, doi:10.1016/j.jnucmat.2007.03.153.
[12] X. Li, G. Yu, J. Yu, K. Wang, Q. Huang, J. Nucl. Mater. 329–333 (2004) 1407.
[13] Q. Huang Y. Wu, C. Li, Y. Li, Y. Feng, M. Zhang, L. Peng, Q. Li, in: Proceedings of the 8th China–Japan Symposium on Materials for Advanced Energy Systems and Fission & Fusion Engineering, 4–8 October, 2004, Sendai, Japan.