Design Concept and Testing Strategy of a Dual Functional Lithium Lead Test Blanket Module for ITER and EAST

Y. Wu*, H. Chen, S. Liu, the FDS Team
Institute of Plasma Physics, Chinese Academy of Sciences,
P.O. Box 1126, Hefei, Anhui 230031, China
Phone/Fax: +86 551 559 3326 E-mail: ycwu@ipp.ac.cn

Abstract:
A Dual Functional Lithium Lead (DFLL) TBM (Test Blanket Module) concept for testing in ITER has been proposed by Chinese Party to demonstrate the technologies of the liquid lithium lead (LiPb) breeder blankets, including Quasi-Static Lithium Lead (SLL) breeder blanket and the Dual-cooled Lithium Lead (DLL) breeder blanket, which emphasizes the balance and the consistence between the risk and the attractiveness of blanket technology development.

Considering the confliction between the limited ITER resources for TBM testing and the requirement of various blanket concepts proposed by Parties, an effective testing strategy has been proposed to achieve the target of testing both SLL and DLL blanket concepts, technologies, and design tools. It covers three phases: materials R&D and small-scale out-of-pile mockup testing in loops, middle-scale TBMs pre-testing in EAST (the superconducting tokamak in China), and full-scale consecutive TBMs corresponding to different operation phases of ITER during the first 10 years.

Description of TBM system concept and the testing strategy combining TBMs testing, in sequence and in parallel, in EAST and ITER, are presented in this contribution.

Key words: Test Blanket Module (TBM); Lithium lead blanket; Dual function; Testing strategy

1. Introduction

Liquid lithium lead (LiPb) breeder blankets, including Quasi-Static Lithium Lead (SLL) breeder blanket with a moderate outlet temperature of up to 450 ^{o}C and the Dual-cooled Lithium Lead (DLL) breeder blanket with a high outlet temperature of up to 700 ^{o}C, are considered by several parties of ITER program as a design option of DEMO blankets for fusion power reactors due to their potential attractiveness of economy, safety and relatively mature technology base **[1-2]**. In the framework of the Chinese series lithium-lead breeder blankets development, a Dual Functional Lithium Lead (DFLL) Test Blanket Module (TBM) concept has been presented and proposed for testing in ITER by Chinese Party to demonstrate the technologies of both SLL and DLL blankets concepts, which emphasizes the balance and the consistence between the risk and attractiveness of blanket technology development .

On the another hand, it should be noted that there will be very limited resources (only three ports) in ITER for testing of more than ten TBM concepts which have been proposed by the participant parties, it is necessary to develop effective testing strategies to achieve the target of the TBM testing program considering limited testing space and testing time. A testing strategy has been proposed to achieve the target of testing both SLL and DLL blankets by including the earlier pre-testing of TBMs in EAST **[3]** (the Chinese Experimental Advanced

Superconducting Tokamak) prior to the installation of TBMs in ITER and the paralleled testing of TBMs in EAST while ITER is in operation.

2. Design Concept of the DFLL-TBM System

To balance the reduction of potential risk and the pursuit of attractiveness of the technology development, the DFLL-TBM developed based on the DEMO blanket features should be able to demonstrate the technology of SLL and DLL breeder blanket concepts, which would lead in future fusion reactors to tritium self-sufficiency, extraction of high-grade heat, and electricity generation.

2.1. Basic concept

The basic structure of DFLL-TBM, 484mm (t) x 1660mm (p) x 585mm (r) rectangular steel box placed in 1/2 ITER port with the reference dimensions 524mm (t) x 1700mm (p), is very similar to that of typical DEMO blanket modules. The structure box is reinforced by one radial-poloidal (r-p) and four toroidal-poloidal (t-p) stiffening plates (SPs), containing the LiPb flow channels as schematically shown in **Fig.1**. It is closed in the rear by the back plates (BPs) also acting as helium manifolds. The structure steel is made of the reduced activation Ferritic/Martensitic steel, e.g. CLAM (China Low Activation Martensitic steel) **[4-5].**

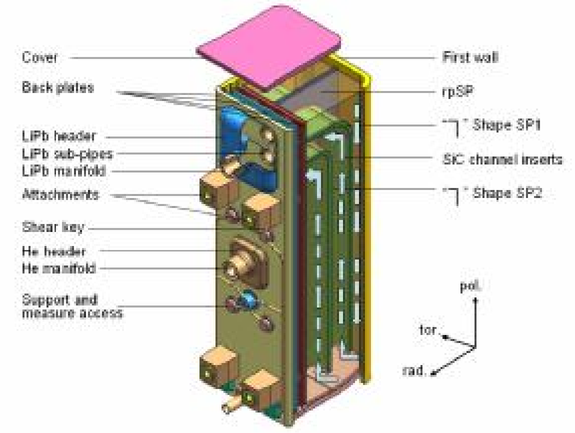

Fig.1 3D structure View of DFLL-TBM

DFLL-TBM will be used to test SLL blanket technology on the early phase of ITER, then the DLL blanket technology on the later phase. On the early phase of ITER, DFLL-TBM is designated to use quasi-static LiPb flow. All the heat of the first wall and structure and LiPb is cooled by helium gas. Maximum outlet temperature of LiPb is about 450 ^{o}C. On the later phase of ITER, DFLL-TBM is designed to use fast flowing LiPb for LiPb self-cooled. The FCIs (Flow Channel Inserts) e.g. SiCf/SiC composite, are designed as thermal and electric insulators inside the LiPb channel to reduce the MHD pressure drop and to allow the higher outlet temperature 700 ^{o}C of LiPb. The later TBM scheme is the major target of the concept, but can be replaced by the early scheme to continue the TBM testing if the critical issues (e.g. MHD effects and FCI technology) could not be solved.
The main reference design parameters of DFLL-TBM are summarized in **Table 1**.

Table 1 Main reference design parameters of the DFLL-TBM

	Parameters
Heat Flux Neutron Wall Load	Ave.0.3MW/m^2, Max. 0.5 MW/m^2 0.78 MW/m^2
Structural material	China Low Activation Martensitic steel (RAFM Steel)
TBM dimensions	Pol. 1660 mm × Tor. 484 mm × Rad. 585 mm Gap between TBM and Frame = 20 mm
Total deposited power	0.66 MW
He coolant	$T_{in/out}$ = 340/402 °C; P_{in} = 8 MPa; Q_{tot} =1.49 kg/s
First Wall	U-shape; Toroidal He cooling; 4 paths; Thickness: 30 mm (5/15/10) Cooling channels: (15 x 20)mm^2, pitch 25 mm
	$T_{in/out\ He}$ = 340 / 395 °C; V_{He} = 50 m/s
r-p Stiffening plate	Thickness: 10 mm (3/4/3); Cooling channels: (4 x 9)mm^2, pitch 12 mm
	V_{He} =60m/s; $T_{in/out\ He}$ = 395/402 °C
t-p Stiffening plate	Thickness : 10 mm (3/4/3); Cooling channel: (4 x 8)mm^2, pitch 11 mm
	V_{He} =64m/s; $T_{in/out\ He}$ = 395/412 °C
Cover	Thickness: 32 mm; 8 parallel cooling channels; (8 x 16) mm^2, pitch13mm
	V_{He} =60m/s; $T_{in/out\ He}$ = 395/400 °C
He collector	3-stage collector; radial direction size: 20/20/10/20/10/20/20 mm
Breeder/multiplier LiPb	2 rows poloidal flowing channel
	V_{LiPb}=14/5.5/5.5 mm/s V_{LiPb}=4.33 kg/s; $T_{in/out}$ = 480/700 °C

2.2. Cooling Auxiliary Systems

LiPb auxiliary system is an important external loop equipment of TBM for liquid LiPb blanket. According to the testing strategy of DFLL-TBM which can demonstrate the technologies of the SLL and DLL blankets, the design of the LiPb anxiliary system is performed to support the dual functional blanket designs. On the early operation phase, It enables the TBM feeding and the tritium extraction from the liquid metal outside the TBM and On the later operation phase it serves as the tritium and heat extraction from the liquid metal outside the TBM. The flow diagram of LiPb auxiliary system is shown in **Fig.2.**

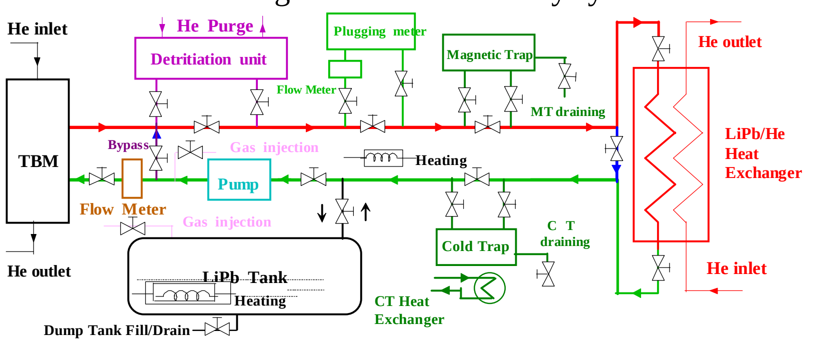

Fig.2 LiPb auxiliary system layout

Helium auxiliary system is also designed to meet the requirements for DFLL-TBM testing purpose. The flow diagram of the helium cooling systems for the DFLL-TBM is shown in **Fig. 3**. This system consists of two helium loops, one of which, the primary heat transport loop (Structure-He loop) is to cool the FW and structure stiffening plates using helium, while the secondary helium loop is to exchange heat with the liquid LiPb breeder and serves as secondary loop of the liquid breeder loop (LiPb-He loop). Both loops are connected to the helium to water loop which is considered to be a part of the ITER tokomak cooling water system (TCWS).

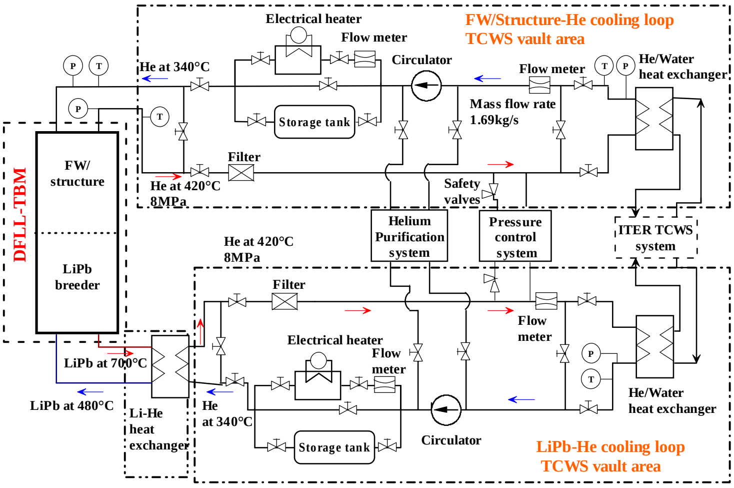

Fig. 3 Helium cooling system layout

3. Performance Analysis

The feasibility and reliability of the DFLL-TBM program has been validated with a series of performance analyses, such as neutronics analysis, thermal-hydraulics (including MHD effects of liquid metal flow) analysis, thermo-mechanics analysis and safety analysis under operational and accidental conditions, on the basis of 3D real models. Preliminary results, listed in the **Table 2**, show TBM design can satisfy the material requirement and ITER safety design requirement. The detailed performance analysis can be found in **Ref [6-7].**

Table 2. Main performance parameters of DFLL-TBM

Parameters			Remarks
Neutronics analysis	Tritium breeding rate	0.0134	- Fusion power of 500MW
	Nuclear heating (MW)	0.47	- Enrichment 90% ^{6}Li in LiPb
Thermal-hydraulics analysis	He pressure drops (MPa)	0.134	Average heat flux of 0.3MW/m2
	He pumping power (KW)	42	
	LiPb pressure drops (MPa)	0.12	
	LiPb pumping power	72	
Thermo-mechanics analysis	Max. temperature (°C)	535	- Max. heat flux of 0.5MW/m2; - Assumed design life time for St is 10,000 hours - PL+Pb<KSm; PL+Pt/Kt<St - X+Y<1
	Max. Von Mises stress (MPa)	234	
	PL+Pb (MPa)	112	
	PL+Pt/Kt (MPa)	103	
	X+Y	0.25+0.70	
	Max. stress (MPa)	438	Assuming the entire TBM under 8 MPa
Activation analysis	Total afterheat at shutdown (MW)	0.018	8500 full power ITER pulses (high duty D-T operational phases)
	Total activity at shutdown (Bq)	6.8×10^{16}	
Tritium inventory/ permeation analysis	TES from LiPb (mg/yr)	833.6	-Tritium production rate: 0.346mg/pulse -TPRF=10 (in TBM) -10% of LiPb into TES -0.1% of He into CPS (efficiency :95%)
	TES from He coolant (mg/yr)	157.6	
	Residing in TBM system (mg/yr)	46.8	
	Permeation as HTO (mg/yr)	0.032	
Accident analysis	System pressurization (KPa)	22.4	Assuming all helium ingress into the VV
	Decay heat removal	ok	
	Hydrogen generation (kg)	<2.5	Assuming all LiPb ingress into the VV

4. Testing Strategy

Prior to the ITER testing, a size-reduced DFLL-TBM will be tested in EAST by considering that the designed parameters of EAST are comparable to those of ITER. The Main parameters comparison of ITER and EAST are listed in the **Table 3.** EAST can server as a valuable pre-testing platform for DFLL-TBM.

Table 3. Main parameters comparison of ITER and EAST

Device	EAST		ITER	
Phase	DD	HH	DD	DT
R (m)	1.95			6.2
A (m)	0.46			2
Bt (T)	3.5--4.0			5.3
Neutron rate (n/s)	10^{15}~10^{17}		10^{17}~10^{18}	1.77×10^{20}
Avg.HF(MW/m^2)	0.1~0.2	0.11		0.27
Port Size	0.97m x 0.53m		2.2 m x 1.7m	
Pulse (sec)	~1000	100-200		400

So, the testing strategy of DFLL-TBM covers three phases:

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

The materials R&D is focused on the development of CLAM steel as the structural material, Al-based coatings as the tritium permeation barrier material, SiC_f/SiC FCIs as the MHD effect control material, tritium control technology, and diagnosis and measurement technology etc. Out-of-pile test of small-scale mockup (e.g. 1/3 size-reduced) is mainly to concern the TBM fabrication route and techniques, the thermo-mechanical/ thermal-hydraulic performances, the MHD effects of flowing liquid LiPb, the compatibility between flowing LiPb and structural steel, the reliability and safety with regard to the EAST/ITER tokamak operation standards etc. Several experimental loops, such as the thermal convection LiPb loop, the forced convection LiPb loop, and the high temperature and high pressure He gas loop need to be constructed to carry out performance test for out-of-pile blanket mockup.

4.2. Test in EAST

The middle-scale (e.g. 1/2 size-reduced) TBM testing in the EAST is to validate the design tools and codes for electro-magnetics, thermo-mechanics and partially neutronics, and to assess the TBM influence on plasma performances, as well as to demonstrate the design feasibility of DFLL-TBM auxiliary system prior to ITER. **Fig.4** shows the exploded view of EAST-TBM. The date and experiences derived from EAST-TBM testing can be applied to optimize and improve the design of DFLL-TBM system.

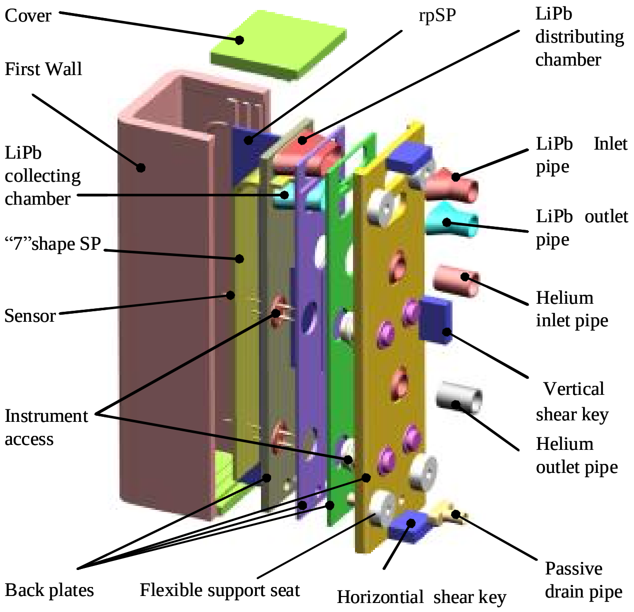

Fig4. The exploded view of EAST-TBM

4.3. Test in ITER

The full-scale consecutive TBMs are to be tested on the different operation phases of ITER during the first 10 years. Four 'act alike' DFLL-TBMs may be designed in turn as EM-TBM for test of EM effects on H-H phase, as NT-TBM for performance of neutronics on D-D phase, as TT-TBM for thermo-mechanics and tritium behavior on low duty D-T phase, and as IN-TBM for integrated performance test on high duty D-T phase, respectively. This program will allow to consecutively validate SLL/DLL blanket concepts, technologies and design tools with reliable and safe operation, and finally to demonstrate relevant technologies for the fusion DEMO reactors. This test program is developed assuming successful testing in earlier phases.

.

5. Summary

The DFLL-TBM systems including basic concept and auxiliary systems are described in this contribution. Performance analyses of neutronics, thermal-hydraulics, thermo-mechanics, etc. are presented to validate the feasibility and reliability of this concept and show TBM design can satisfy the material requirement and ITER safety design requirement. The testing strategy are proposed by combining the requirements of both SLL and DLL DEMO blanket concepts and combining the conditions of EAST and ITER operations.

Acknowledgement

This work is supported by the National Natural Science Foundation with grant No. 10675123.

References

[1] Y. Wu, the FDS teams, Conceptual Design Activities of FDS Series Fusion Power Plants in China, Fusion Eng. Des. (2006), in press.

[2] Y. Wu, W. Wang, S. Liu, et al., Conceptual Design Study on the Fusion Power Reactor FDS-II, Chinese J. of Nuclear Science and Engineering 1 (25) (2005):76-85.

[3] Y. Wan, P. Weng, J. Li, et al. Progress of the EAST Project in China. The 20th IAEA FEC, Vilamura, Portugal, Nov. 2004. 1~6.

[4] Q. Huang, J. Yu , F. Wan, et al. Overview on the Development of China Low Activation Martensitic Steel for Fusion Reactors. Chinese J. of Nuclear Science and Engineering, 2004, 1 (24):56~64.

[5] Q. Huang, C. Li, Y. Li, et al., Progress in Development of China Low Activation Martensitic Steel for Fusion Application, presented at the 12th Inter. Conf. On Fusion Reactor Materials, Dec.4-9, 2005, Santa Barbara, USA.

[6] Y. Wu, et al. Design and Analysis of the Chinese Dual-Functional Lithium Lead (DFLL) Test Blanket Module for ITER, presented at the 24th Symposium on Fusion Technology, Warsaw, Poland, Sep. 11-15 2006.

[7] Y. Wu, W. Wang, S. Liu, et al. Design Description Document for the Chinese Dual-Functional Lithium Lead-Test Blanket Module for ITER (2005).