GA–A25928

OVERVIEW OF LIQUID METAL TBM CONCEPTS AND PROGRAMS

by

C.P.C. WONG, J.-F. SALAVY, Y. KIM, I. KIRILLOV,
E. RAJENDRA KUMAR, N.B. MORLEY, S. TANAKA, Y.C. WU,
and TBM TEAMS

AUGUST 2007

GENERAL ATOMICS

DISCLAIMER

GA–A25928

OVERVIEW OF LIQUID METAL TBM CONCEPTS AND PROGRAMS

by

C.P.C. WONG, J.-F. SALAVY,* Y. KIM,† I. KIRILLOV,‡
E. RAJENDRA KUMAR,# N.B. MORLEY,¶ S. TANAKA,§ Y.C. WU,Δ
and TBM TEAMS

This is a preprint of an invited paper to be presented at the 8th Int. Symp. on Fusion Nucl. Technology, in Heidelberg, Germany, September 30, 2007 through October 5, 2007 and to be published in the *Fusion Eng. and Design.*

*CEA, Saclay, France.
†KAERI, South Korea.
‡Efremov Institute, St. Petersburg, Russia.
#Institute for Plasma Research, Gujarat, India.
¶UCLA, Los Angeles, California.
§The University of Tokyo, Tokyo, Japan.
ΔChinese Academy of Sciences, Hefei, China.

Work supported in part by
the U.S. Department of Energy
under DE-FC02-04ER54698 and DE-FG03-86ER53266

GENERAL ATOMICS PROJECT 30200
AUGUST 2007

Abstract

In support of the ITER Test Blanket Module (TBM) program and coordinated by the Test Blanket Working Group, ITER party members have been focusing on the liquid metal blanket design concepts, most of which have been extensively explored. For the demonstration power reactor (DEMO) design, we will have to accommodate the neutron wall loading and first wall heat flux, breed and extract adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency. Most proposed liquid metal TBMs have the potential of achieving similar DEMO goals and requirements. Furthermore, all liquid metal TBMs are to satisfy ITER safety requirements and to be operated and tested within ITER operation scenarios. For the development of liquid metal TBM concepts, many R&D elements are common to a few designs such as the areas of Reduced Activation Ferritic/Martensitic Steel (RAFM, also abbreviated as FS in the following) or V-alloy fabrication, thermal fluid MHD, FS/PbLi, FS/Li and V-alloy/Li compatibility, irradiation effects on different materials, tritium extraction, and SiC flow channel insert (FCI) development, etc. With a well-coordinated ITER TBM program, different parties' R&D activities can supplement and complement each other via collaborations. This paper will present respective designs and R&D programs from seven ITER parties.

1. Introduction

Performing integrated tritium breeding blanket experiments has been a principal objective of ITER since its inception, which is actively coordinated by the ITER Test Blanket Working Group (TBWG) [1]. A principal mission of the ITER Test Blanket Module (TBM) program is to develop, deploy, and operate DEMO-relevant test blanket modules that can provide integrated experimental data and operational experience in a true fusion environment. Two basic tritium breeder choices have been investigated among ITER parties, the solid ceramic breeder (CB) [2] and the liquid breeder (LB) and both have been proposed to be tested in ITER. This paper will focus on the development of LB TBM concepts. We will tabulate the characteristics of respective DEMO blanket designs. LB TBM conceptual designs and corresponding R&D and development programs from seven ITER parties will be summarized. Japan has not proposed a specific liquid breeder TBM design, but plans to contribute to TBM testing by collaboration with other parties.

2. DEMO Blanket Design

TBM blanket concepts were initiated from the design of corresponding blanket concepts for DEMO. Respective DEMO design parameters are presented in Table 1.

Table 1. Design Parameters of DEMO Blanket Concepts

	EU HCLL [3]	**China DFLL [4,5]**		**US DCLL [6,7]**	**RF Li/V [8]**	**Korea HCML**	**India LLCB**
		DLL mode	SLL mode				
Structural material	EUROFER	CLAM	CLAM	F82H or EUROFER	V-Cr-Ti	ODS FFS	IN-LAFMS
Breeder	PbLi	PbLi	PbLi	PbLi	Li	Li	PbLi and Li_2TiO_3
Module dimension, WxHxD, m	2x1.8x1	2.2x2x1.2	2.2x2x1.2	1x2x~1	1.25-145x2.9-7.3x0.5	TBD	~2x2x1.2
Neutron multiplier	NA[a]	NA	NA	NA	Be or none	None or Be	NA
Neutron reflector/shielding	NA	NA	NA	NA	WC	Graphite	WC
Flow channel insert	NA	SiC_f/SiC	NA	SiC_f/SiC or metallic	NA	NA	NA
MHD insulation (coating or laminated wall)	NA	NA	Al_2O_3	NA	Different options[b]	NA	(Al_2O_3 TBD)
DEMO availability and fusion power:							
DEMO availability goal: %	>30	>30	>30	>30	>60	>30	> 30
Nuclear performance: Fusion power range, MW	2200-3300	2500	2500	~2500	2500-3500	2500-3500	2500
Key Coolant Power Cycle Loop:							
Primary coolant	He	He	He	He	Li	He	He
LB self-cooled	NA	PbLi	NA	PbLi	Li	Li	PbLi
Intermediate coolant with LB	NA	He	NA	He	Na	NA	He
FW heat flux, MW/m^2	0.5 peak	0.7 peak	0.7 peak	0.5 peak	0.7 peak	<1	0.5 peak
Neutron wall loading, MW/m^2	2.4 peak	3.54 peak	3.54 peak	3.0 peak	3.4 peak	>2	1.7 avg.
Primary Coolant Parameters:							
Inlet/outlet temperatures, °C	300/500 (He)	300/450 (He) 480/700[c] (PbLi)	300/450 (He)	350/450 (He) 460/700[c] (PbLi)	350/600 (Li)	250-350/ 550 (Li)	350/480 (He) 370/480 (PbLi)
Pressure, MPa	8 (He)	8 (He)	8 (He)	8 (He)	1.0 (Li)	8 (He)	8 (He)
T_{max}, FW, °C	~550	557	—	~550	650-800 as a function of FW thickness	TBD	520
T_{max}, breeder/FS or breeder/FCI, interface °C	544 FS/PbLi	538 FS/PbLi	—	<480 PbLi/FS <1000 PbLi/FCI	700	TBD	500
Power cycle	Rankine	Brayton	Rankine	Brayton	Water/steam Rankine	Brayton	Brayton
Gross thermal efficiency, %	~40	~47	~32	>40	44-45	>35	~40
Radial thickness, m	0.55–0.8	0.75	0.75	0.75	0.5-0.75	<1	1.0
Li-6 enrichment, %	90	90	90	90	50	<15	90%
3-D TBR	1.15	1.2	--	1.17[d]	1.05-1.09[c]	1.05-1.10	1.2

[a]NA: not applicable or not available.
[b]CaO, AlN, Er_2O_3, Y_2O_3, or multi-layer..
[c]Advanced alloy external to the blanket will be needed for tritium extraction, piping and HX components.
[d]No tritium breeding in divertor area (assumed 12% of total FW area).

3. TBM Design

After the definition of conceptual LB blanket designs for DEMO, respective design concepts were modified to fit the test port geometry and testing environment of ITER. A key constraint is the given test port area with a half-port height and width dimensions of 1.70 m and 0.524 m, respectively. The amount of LB that can be used is also limited due to safety concerns, which is limited by the generation of hydrogen to be <2.5 kg under the accidental condition of LB and water interaction. Figure 1 illustrate the schematics of TBM designs from EU, China, US, RF, Korea and India.

Characteristics of the ITER-TBM designs are tabulated in Table 2. All TBM designs will have to be designed to withstand transient events like edge localized modes (ELMs) and disruption from ITER. Since most of the TBM designs are using ferritic steel as the structural material, the localized ferromagnetic effect on the ITER confinement magnetic field has to be taken into consideration. These effects are best tested during the HH-phase of ITER, such that, if necessary, modifications can be made to the TBM design or ITER correction coil currents.

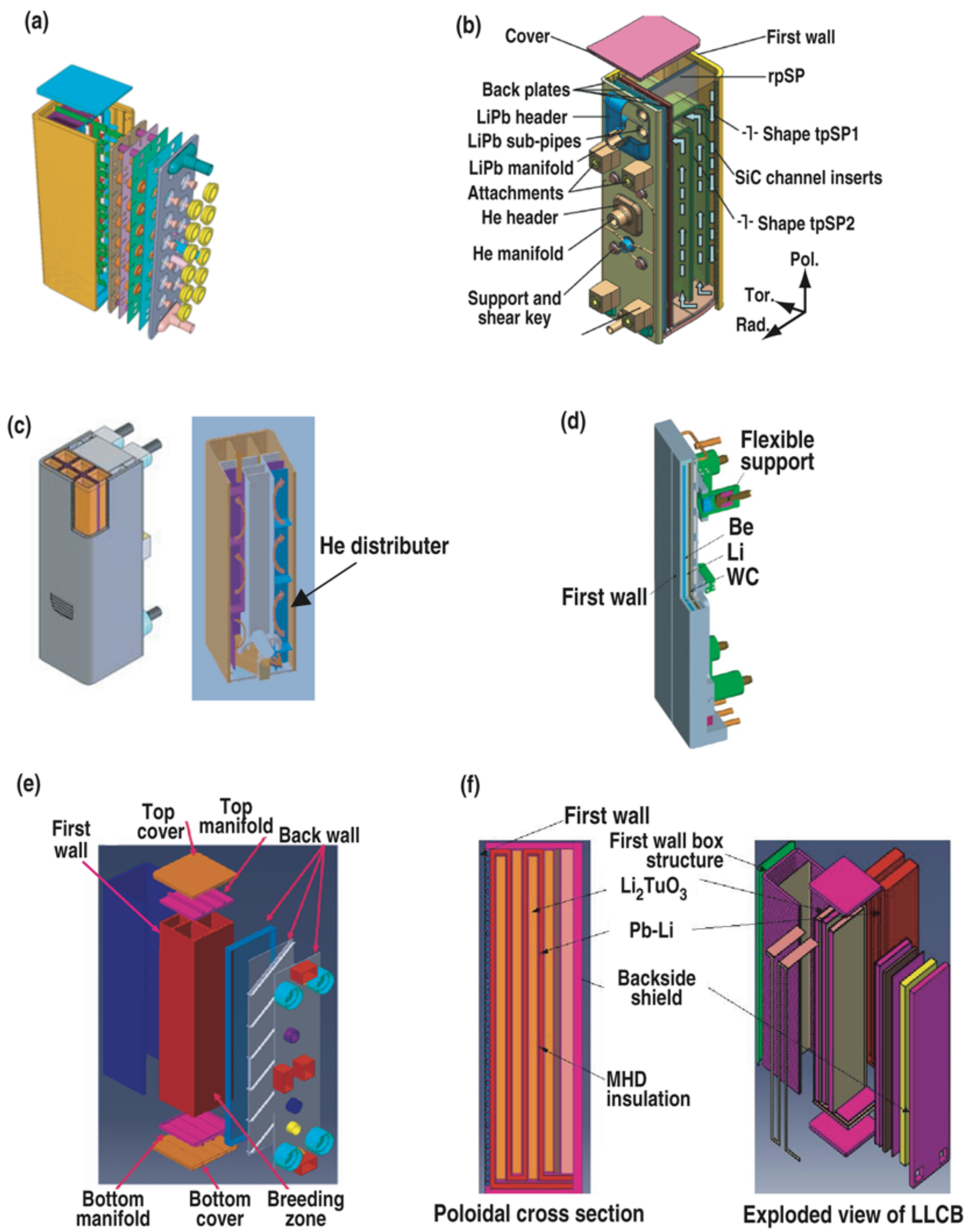

Fig. 1. (a) EU: HCLL-TBM, (b) China: 3-D view of DLL-TBM, (c) US: DCLL TBM sub-assemblies, (d) RF: Li-cooled TBM, (e) Korea: HCML TBM concept, and (f) India LLCB TBM concept.

Table 2. Design Parameters of ITER LL TBM Blanket Concepts. All dimensions are designed to a half module width = 0.484 m and height = 1.66 m (with a 20 mm gap to the frame)

	EU HCLL [9]	China DFLL [4,5]		US DCLL [6,7]	RF Li/V [10]	Korea HCML [11]	India LLCB
		DLL mode	SLL mode				
Structural material	EUROFER	CLAM	CLAM	F82H or EUROFER	V-Cr-Ti	RAFM	IN-LAFMS/ EUROFER
Breeder	PbLi	PbLi	PbLi	PbLi	Li	Li	PbLi and Li_2TiO_3
LB volume, m^3	0.2	0.229	0.258	~0.28	0.022	<0.033	~0.2
Power deposition, MW	0.7	0.66	0.64	0.87	0.65	0.675	0.85
Neutron multiplier	NA[(a)]	NA	NA	NA	Be or none	NA (or Be)	NA
Neutron reflector/shielding	NA	NA	NA	NA	WC	Graphite	WC/SS316
Flow channel insert	NA	SiC_f /SiC	NA	SiC_f /SiC or metallic	NA	NA	NA
MHD insulation	NA	NA	Al_2O_3 or other choice	NA	Different options[(b)]	NA	Al_2O_3 or other choice
Tritium barrier	Natural oxide/ Al_2O_3	Al_2O_3 or other choice	Al_2O_3 or other choice	Natural oxide/ Al_2O_3	NA	NA	Al_2O_3 or other choice
Primary coolant	He	He + PbLi	He	He + PbLi	Li	He	He and PbLi
Intermediate loop	NA	He	NA	He	Organic	NA	He
Primary Coolant Parameters:							
He inlet/outlet, °C	300/500	340/402	340/420	350/420	Varies[(a)]	300/406	350/480
He coolant pressure, MPa	8	8	8	8	NA	8	8
He velocity, m/s, FW/side plate/central plate	68/11/8	50/60	54/64	58	NA	50	45/60
T_{max}, FW at 0.5 MW/m^2, °C	546	534	547	554	~675	614	550
He pressure drop in module, MPa	0.21	0.277	0.295	0.24	NA	1.15	0.3
Pumping power, kW	83	82	91	49	10 kW(e)	TBD	48
LB Parameters:							
Breeder inlet/outlet, °C	300/480	480/700	~450	360/470	250-450/350-550	300/411	350/460
Breeder pressure, MPa	<1	1	–	2	0.5	TBD	<1.2
Breeder T_{max}, °C	568	700	480	500	550	458	480
Breeder/structural material interfacxe T_{max}, °C	548	434	478	<480	590	<500	~450
Breeder mass flow rate, kg/s	0.22-2.2	5.2	0.1~1	21	1.07	0.0114	42
Breeder flow max. velocity, m/s	<0.001 in BU	0.014 in breeder zone	0.003	0.1	0.002-0.004	<0.5	1.0
TBR	0.45	0.44	0.46	0.741	0.56	0.36	TBD
Li-6 enrichment, %	90	90	90	90	90	<15	90

(a)NA: not applicable or not available.
(b)CaO, AlN, Er_2O_3, Y_2O_3, or multi-layer.
(c)Advanced alloy external to the blanket will be needed for tritium extraction, piping and HX components.
(d)No tritium breeding in divertor area (assumed 12% of total FW area).

4. TBM R&D Programs

R&D items were identified for all proposed LB TBM concepts. A few R&D items are common to several TBM concepts. In the following we will summarize the status of development of the twelve items, which are at different stages of development. Japan at this time has not proposed any specific LB TBM concept, but has R&D activities in support of several relevant items.

Ferritic Steel Fabrication. Reduced activation ferritic-martensitic (RAFM) steel is a structural material under development for fusion reactor applications; this development includes the establishment of fabrication procedures that will maintain RAFM steel favorable properties under projected high fusion neutron fluence conditions. Several fabrication processes for the production of TBM sub-components and assembly using EUROFER have been investigated in the EU R&D program. For the fabrication of sub-components (first wall, stiffening plates, cooling plates and caps) different options can be considered: a two-step HIP process, a one step process with high pressure HIP with metal inserts, the weld and HIP process, the improved HIP forming process for rectangular tubes and the diffusion weld (DW) process by adjustable pressure. Mechanical properties and the leak tightness of the produced joints in the different structures will be assessed. The most successful process with respect to reliability and performance will be chosen and tested with sub-component mockups. Assembly and fabrication methods have been generally assessed by EU industry, and they are being developed and analyzed in parallel with EU laboratories. Correspondingly, China is developing the Chinese RAFM (CLAM) steel, and India the LAFMS. R&D activities including the technology for fabrication have been initiated in the US, China, and India. Japanese F82H development is also in the phase of component R&D phase. Testing of oxide dispersion strengthening (ODS) ferritic steel structures has also been initiated at Kyoto University.

V-alloy Fabrication. For the Li/V design, RF have produced V-alloy up to 1 ton in the form of sheets with dimensions of 80×1000×1200 mm and 5×1000×12000 mm and tubes 5-100 mm in diameter. Thermophysical and mechanical properties of V-(4-5)%Cr-(4-5)%Ti alloys have been studied at temperatures up to 800°C, both in non-irradiated and irradiated conditions up to a fluence of 49 dpa. V-alloy corrosion in lithium studies show that at temperatures up to 700°C, with a proper control of suitable additives in lithium, the corrosion rate is relatively small [12]. Recent efforts in production of these alloys with improved control of impurity content showed good results for increasing impact properties of the welded joints. In Japan, the development of V-alloy is coordinated and carried out by NIFS and universities. The reference V-4Cr-4Ti alloy heat (NIFS-HEAT) was produced, which is used for characterization and fabrication of TBM components. Various manufacturing technologies were developed using the alloy, including welding, tube manufacturing and W-coating.

Flow Channel Insert (FCI). For the FCI that applies to dual coolant TBM designs, research and development of the SiC materials for FCI requires low electrical conductivity, low thermal conductivity and good compatibility with PbLi at temperature above 1000°C. Assessments on its production, fabrication and cost have been carried out. The electrical and thermal conductivity properties need to be developed and tested, particularly after irradiation and in contact with PbLi, where surface contact resistance may contribute significantly to the equivalent resistivity of the material. Experiments in determining the compatibility between FCIs and PbLi are being performed. US R&D activity for SiC_f/SiC composite FCIs is currently underway in several areas. Capsule compatibility tests between SiC and PbLi at elevated temperatures are being conducted at Oak Ridge National Laboratory (ORNL). Results indicate a high compatibility temperature (>1000°C) for prolonged exposures. Measurements of transverse electrical conductivity of SiC_f/SiC composites as a function of temperature are being conducted at Pacific Northwest National Laboratory (PNNL). Results show a lower conductivity by about a factor of 100 compared to the in-plane conductivity for 2D composites [13]. Irradiation experiments of several FCI relevant SiC samples are in preparation at ORNL with the goal of getting the response of fundamental constituents for use in the modeling of composite. In addition, the planning on producing the best FCI composite architecture and the exploratory fabrication of SiC-foam are also underway. From Japan, simple shape components such as pipes, plates and rectangular pipes are available with SiC_f/SiC composite as well as more complicated structure such as cooling panel from Kyoto University, and compatibility and thermal conductivity experiments are performed in the temperature range between 500° to 900°C. Flow tests for FCI in PbLi, including compatibility experiments, are also underway. Irradiation tests, including tests of FS, FCI and mockup tests, are being planned in China and Japan, as well as under the US/Japan Jupiter-II program [14] and beyond. Collaboration between EU and JA under the "Broader Approach" program also includes SiC_f/SiC development related tasks.

MHD Coating [15]. For the V/Li TBM concept proposed by RF, the successful development of the MHD coating is essential for acceptable MHD pressure drops and pumping power. RF is investigating the possibility of multi-layered MHD barriers with V-alloy facing the Li breeder. In Japan, research in this area is being carried out by the domestic program and the Japan-US collaboration program JUPITER-II. High density and high purity coating materials like AlN, Y_2O_3, Er_2O_3 are being considered. Er_2O_3 was studied by bulk corrosion tests in Li. PVD coatings of crystalline Er_2O_3 have been shown to be stable to 1000 h in Li and with low mass loss <5 mg/cm^2 at 800°C. Correspondingly, double coating with metallic over layers and in-situ Er_2O_3 coatings are also being developed.

Thermal Fluid MHD [16]. With the proposals of LB blanket concepts it is essential to develop an understanding of 3-D MHD behavior in a complicated flow circuit including inlets, outlets, expansions, manifolds and with the inclusion of flow channel inserts and corresponding separation gaps. The main consequences of the MHD effect are pressure losses, which contribute to the required pumping power for the external circulator. A main concern for liquid metal blankets where the LB also serves as coolant is the strong MHD influence on the distribution of flow between parallel channels fed from a common manifold. For separately cooled LB blankets, the strongest MHD-effects are expected in the supply and return manifolds where the flow velocity of the PbLi is highest, especially for the inboard region where the magnetic field is also the highest. The appearance of induced electrical currents in the structure and LB and changes of the velocity distribution of the LB could evolve into coupled and complicated fluid motions, which could affect the heat transfer of the coolant and tritium permeation to the He coolant. For example, the first EU study on T permeation modeling, taking into account MHD velocity profiles, indicated that the effect is equivalent to a permeation reduction factor of about 30 [17]. Extensive MHD and heat transfer tests are being conducted in different countries to obtain a good understanding of flows in straight ducts with conducting and electro-insulating walls [18]. Less data are available for flows in ducts of complex geometry and/or 3-D magnetic fields, or ducts with imperfect electrical insulation. Mockup tests in strong magnetic fields with developed electro-insulating barriers will be needed.

The liquid PbLi in the HCLL, SLL and HCML concepts is circulated slowly for tritium extraction with much reduced MHD pressure drop. Theoretical calculations to describe the MHD effects are being performed and verified under simplified experimental conditions, e.g. the use of NaK instead of PbLi in Mekka facility [19, 20]. From China the circulation loop experiments are being planned for an out-of-pile experiment and in EAST [21], which has a test port that can accommodate a test module about 1/2 of the size of the ITER half-port TBM. To predict the coupled MHD effects on the US DCLL design, development and application of sophisticated numerical simulation tools for predicting 2-D [22] and 3-D [23] MHD flow profiles, flow distribution, buoyancy effects and drag – and their subsequent coupled effects on the thermal field – are underway. This is supported by experiments on fundamental MHD phenomena for FCI and manifold flows at the MTOR facility at UCLA using a gallium alloy as a PbLi stimulant. Partially integrated mockup experiments using PbLi are planned in subsequent years. Since the liquid lithium flow in Korea's HCML is very slow, it is expected to have very little MHD effects. However, a 3-D MHD analysis is underway with CFX and its electromagnetic module to characterize the MHD phenomena of the Li breeder of the HCML concept. In the RF self-cooled Li/V design, 3-D MHD flow effects are also important and TBM mockups tests in Li circulation loops are planned. In Japan, MHD measurements with circular tubes and square ducts of SiC composite are performed in a PbLi loop.

FS/PbLi Compatibility. For the LB design with the use of FS as the structural material, the key limitation on thermal performance is the compatibility temperature between PbLi and FS. The compatibility of EUROFER with PbLi was addressed in recent experiments at temperatures up to 550°C with PbLi flowing at up to 0.22 m/s for 5000 h. Results show that the erosion rate at higher temperature is about five times the rate obtained at 480°C. The higher erosion rate leads to the concern of possible loop blockages by precipitation formed in the cooler sections of the PbLi loop. Modeling has been performed and under similar boundary conditions, the results support rather well the observed erosion at 480°C and 550°C temperatures . The possibility of a magnetic field influence on corrosion has also been studied by exposing identical EUROFER samples in PbLi to magnetic fields of 1.7 T [24]. Similar experimental studies are being done and planned in China, Japan and India. Some experimental loops in China, such as the thermal convection PbLi loop (Dragon I) at 450°-500°C, the high-temperature thermal convection PbLi loop (Dragon II) with up to 700°C have been in operation, and the forced convection PbLi loop (Dragon III) are being constructed. A new V/Li flowing compatibility is underway in the US in collaboration with Japan as part of JUPITER-II program designed to address specifically the corrosion of V alloy by Li in a very controlled, all vanadium thermoconvection system [25].

Irradiation Tests. RAFM steel and V-alloy are the two most credible structural materials for DEMO. It becomes necessary to be able to project its properties under high fusion neutron fluence irradiation of up to 15-20 MW.yr/. TBM testing in ITER with a maximum fluence of 0.3 MW.yr/m^2 will provide integrated irradiation data in a tokamak environment.

RAFM Steel. EU irradiation campaigns of EUROFER have been continuing with a fluence level of up to 80 dpa (~8 MW.yr/m^2) at temperature in the range of 300°-325°C [26]. No saturation of the hardening effect of the yield strength or of the shift of the DBTT has been observed for samples irradiated to approximately 40 dpa. In 2003 the modeling of the irradiation processes to obtain a better understanding of the evolution of the microstructure in bcc RAFM steels was started. The following modeling tools will be used: ab-initio and molecular dynamics calculations, Monte Carlo (MC) simulations and dislocation dynamics. Irradiation experiments of FS samples are also underway in the US and in collaboration with EU and Japan. The “Broader Approach” collaboration between EU and Japan identifies RAFM steel R&D as an important subject. An extensive database is available for F82H for the application to CB blanket, which will also be applicable to LB TBMs.

V-alloy [27]. From RF, mechanical properties of irradiated V alloys are available at temperatures up to 600°C and up to 30-49 dpa. Tests are planned in BN-600 reactor at irradiation temperatures of 400°-750°C and 50-120 dpa. From Japan irradiation tests of vanadium alloys, including characterization of weld and joints and irradiation creep properties, are carried out using JMTR, JOYO and ion accelerators under domestic programs and the US HFIR irradiation under the JUPITER-II collaboration program.

Tritium Extraction [28]. Tritium Extraction System (TES) – From EU various processes and equipment are considered with focus on the recovery of the tritium dissolved in the PbLi:

- *Gas-liquid contactors*. Spray columns, plate columns, bubble columns and packed columns were tested in the past. Best results were obtained with an efficiency of only 30% in a packed column, which was not optimized.

- *Use of vanadium getters*. Vanadium could be used as a getter for hydrogen in the PbLi stream because hydrogen has a far stronger affinity to V, and PbLi and V show good compatibility.

- *Permeators*. Membranes with high permeation are discussed to allow the permeation of tritium from the liquid PbLi into a separated volume either purged with an inert gas or evacuated. This vacuum permeator is also the main approach being investigated in the U.S. at the Idaho National Laboratory.

- *Tritium Extraction System (TRIEX) from PbLi*. Part of the above mentioned concepts for tritium recovery from the PbLi shall be tested experimentally in the system. TRIEX is a facility presently being built in ENEA-Brasimone, Italy.

From Japan, tritium transfer in the system of PbLi and permeation through SiC will be measured in the dual coolant loop in 2007-2008. Tritium recovery from DCLL concept will be demonstrated. Tritium solubility and equilibrium pressure in PbLi are regarded as the most important data to be obtained under the TITAN US-Japan collaboration. For the V/Li concept, the RF is developing methods of tritium extraction, including equilibrium molecular distillation and cold trap method. Different tritium extraction techniques and experiments, either for the PbLi or the Li system, are being planned and performed in China, Korea, and India.

Tritium Permeation in PbLi. Tritium permeation in PbLi is far from understood. EU foresees the launch of an ambitious program, with strong opportunities for international collaboration, to further detail the problem of tritium permeation (including material database, constitutive law and various effects, development and experimental validation of modeling tools, etc.). Tritium diffusivity in PbLi and the mass-transfer coefficient from PbLi to purge gas were obtained under fast neutron irradiation in the fast neutron source reactor “YAYOI” of the University of Tokyo.

Mockup Tests. To support the development and the acceptance tests for respective TBMs, mockup tests will be needed. From the EU perspective, in addition to sub-component testing and qualification, functional tests of TBM mockups integrating all components and reproducing completed with He cycles and hydrogen (tritium simulating) extraction are mandatory for the development of HCLL. The main issues addressed are: i) validation of the TBM design performances (heat removal, H extraction), ii) validation of the fabrication route, and iii) reliability and safety with regard to ITER standards (structural integrity).

Testing facilities envisaged for these tests are:

- *HeFUS3 at ENEA Brasimone*. A closed He loop capable of testing HCLL and HCPB mockups.
- *European Breeding Blanket Test Facility (EBBTF) at ENEA-Brasimone*. A closed PbLi loop capable of testing 1:1 HCLL TBM mockups with respect to the supply of liquid PbLi.
- *Tritium Extraction System for PbLi (TRIEX) at ENEA-Brasimone*.
- *Helium Loop Karlsruhe (HELOKA) at Research Centre Karlsruhe (FZK)*. A closed He loop for testing TBMs at temperatures up to 550ºC, pressure up to 8 MPa and He mass flows up to 1.8 kg/s is in the detailed planning phase.
- *Upgrade of HeFUS3 at ENEA-Brasimone*. Upgrading of the existing HeFUS3 loop from 0.35 kg/s to a He mass flow of 1.3 kg/s to allow full testing of at least 1:3 TBM mockups.
- Smaller test facilities are also available for central plate testing of HCLL, needing facilities such as DIADEMO (CEA-Cadarache) and HEBLO (FZK).

Similarly, mockup tests with FW heat flux, MHD flow and heat transfer with PbLi and transient pressure environments are currently being planned in the US and China. An upgrade of the helium flow loop facilities at the PMTF test facility at SNL (US) and a new helium loop in China are being considered. The MTOR MHD facility at UCLA is being investigated with the addition of a PbLi flow loop. Similar scaled liquid metal loop, helium loop and mockup tests are being planned in Korea, India and Japan. From RF, tests of TBM mockups of different scale up to a full-scale mockup are planned, and corresponding test facilities are under consideration. An existing small-scale lithium loop is used for electro-insulating barrier characterization. EAST, the superconducting tokamak device in China is proposed to serve as an integrated pre-testing platform for TBM.

Tritium Permeation Barrier (TPB). The production of TPB in the EU R&D program follows two lines. The first one, linked with the compatibility/corrosion studies, consists of depositing a TPB (mainly Al_2O_3 coatings) on the PbLi side. Various methods have been investigated and have shown different results in terms of permeation reduction factor (PRF) when measured in gas or in PbLi phases. The second one consists of producing a TPB with natural oxides on the He side. This method has the advantages of insuring that the deposit would form in all the He tube surfaces, and that the deposition could be maintained during operation via the self-healing effect. Results of the permeation tests carried out at ENEA Brasimone showed that with an online oxidation generated self-healing coating, the PRF can be in the range 10–30. This process proved its feasibility and effectiveness, but further optimization is needed to address the impact of irradiation. Correspondingly, Al_2O_3 or other materials are proposed by China and India as a tritium permeation barrier and MHD coating.

Be Joining to FS. To match the first wall material of ITER, all TBMs are to have a Be coating of 2 mm on the test module. With the exception of RF, all ITER parties are looking into the technology for joining Be to FS, starting with basic studies in the materials development area. This is an area that international collaboration could achieve the most benefit by avoiding duplication of efforts, but active coordination and data exchange on the development will be needed.

5. Conclusions

With the overview of the LB TBM designs and development from the seven ITER parties, the following conclusions can be made:

- Most LB TBM designs for ITER have the possibility of being extended to high performance DEMO designs.
- The two dual coolant and two single PbLi coolant designs are very similar to each other in basic approaches and R&D needs. But the two single coolant designs have different flow configurations. EU-HCLL has the PbLi flowing in the radial direction, while the CH-SLL has the PbLi flowing in the poloidal direction. The two dual coolant designs have their own features with their manifold structures and He flowing schemes.
- The DFLL configuration has the goal of initiating the testing of SLL blanket technology and to test DLL blanket technology at later stages.
- The self-cooled Li/V design requires a robust MHD insulation coating and the dual coolant designs require robust FCI for thermal and MHD insulation.
- Relatively, HCML requires a very low Li-6 enrichment (<15%) to achieve a self-sustainable TBR when compared to the other blanket concepts considered in this paper.
- LLCB is a new blanket concept aiming to optimize the use of both PbLi and ceramic breeder with helium and PbLi as coolants. Critical issues will be identified and R&D activities will be performed to assess its performance for DEMO.
- All R&D items from different parties are in progress but at very different stages of development. EU is the most advanced party in this group.
- Japan plans to contribute to LB TBM by international collaboration although no specific LB TBM has been proposed.
- Many of the R&D items are common to a few designs, like the FS fabrication, thermal fluid MHD, FS/PbLi, V/Li compatibility, Be coating on FS, tritium permeation barrier, just to name a few. With a well-coordinated ITER TBM program, different parties can supplement and complement each other via active and strong collaborations. This is encouraged by all parties and details will have to be negotiated.

References

[1] L. Giancarli, et al., Breeding Blanket Modules Testing in ITER: An International Program on the way to DEMO, Fusion Eng. Design **81** (2006) 393-405.

[2] L.V. Boccaccini, et al., Overview of Solid Breeder TBM Concepts and Programmes, this conference (2007).

[3] Y. Poitevin, et al., The European Breeding Blankets Development and the Test Strategy in ITER," Fusion Eng. Design **75-79** (2005) 741-749.

[4] Y. Wu, FDS Team et al., Conceptual Design Activities of FDS Series Fusion Power Plants in China, Fusion Eng. Design **81** (2006) 2713-2718.

[5] Y. Wu, FDS Team, Design Status and Development Strategy of China Liquid Lithium-Lead Blankets and Related Material Technology, J. Nucl. Mater. **367-370** (2007) 1410-1415.

[6] C.P.C. Wong, et al., An Overview of Dual Coolant PbLi Breeder First Wall and Blanket Concept Development for the US ITER-TBM Design, Fusion Eng. Design **81** (2006) 461-467.

[7] C.P.C. Wong, et al., Assessment of Liquid Breeder First Wall and Blanket Options, Fusion Sci. Technol. **47** (2005) 502-509.

[8] I.R. Kirillov, RF DEMO Team, Lithium Cooled Blanket of RF DEMO Reactor, Fusion Eng. Design **49-50** (2000) 457.

[9] J.F. Salavy, et al., The HCLL TBM: Present Reference Design, System Integration in ITER and R&D Needs, this conference 2007.

[10] I.R. Kirillov, et al., RF TBMs for ITER tests, Fusion Eng. Design **81** (2006) 425-432. [11] D.W. Lee and KO TBM Team, Preliminary Design of a Helium Cooled Molten Lithium Test Blanket Module for the ITER Test in Korea, Fusion Eng. Design **82** (2007) 381-388.

[12] D.L. Smith, J. Konys, T. Muroga, V. Evitkhin, Development of Coatings for Fusion Power Applications, J. Nucl. Mater. **307-311** (2002) 1314-1322.

[13] N. B. Morley, Y. Katoh, S. Malang, et al., Recent Research and Development for the Dual Coolant Blanket Concept in the US, this conference 2007.

[14] K. Abe, et al., Development of Advanced Blanket Performance Under Irradiation and System Integration Through JUPITER-II Project, this conference 2007.

[15] I.V. Vitkovsky et al., Development and Testing of Electro-insulating Barriers for Lithium-Vanadium Fusion Blanket, Fusion Eng. Design **61-62** (2002) 739.

[16] N.B. Morley et al., Liquid Magnetohydrodynamics – Recent Progress and Future Directions for Fusion, Fusion Eng Design **51-52** (2000) 701.

[17] F. Gabriel, et al., A 2D Finite Element Modeling of Tritium Permeation for the HCLL DEMO Blanket Module, Fusion Eng. Design **82** (2007) 2204-2211.

[18] L. Barleon, U. Burr, et al., "Magnetohydrodynamic Heat Transfer Research Related to the Design of Fusion Blanket," Fusin Technol. **39** (2001) 127-156.[19] C. Mistrangelo and L. Buhler, Electric Flow Coupling in HCLL Blanket Modules, this conference 2007.

[20] K. Starke and L. Buhler, et al, Experimental Investigations of Liquid-Metal MHD Flows in a Mock-up of a HCLL Blanket, this conference 2007

[21] J. Li, EAST and its Technical Program in Preparation to ITER, this conference 2007.

[22] S. Smolentsev, N.B. Morley, C.P.C. Wong, M. Abdou, MHD/heat Transfer Considerations for the DCLL Blanket for DEMO and ITER TBM, this conference 2007.

[23] N.B. Morley, M.-J. Nir, R. Munipalli, and M.A. Abdou, MHD Simulations of Liquid Metal Flow Through a Toroidally-oriented Manifold, this conference 2007.

[24] B. Pint, Oak Ridge National Laboratory, private communication, 2007.

[25] W. Krauss, et al., Analyses of EUROFER Samples Exposed to Pb-17Li Flow Without and With Magnetic Field of 1.7 T at a Flowrate of v=5 cm/s and T=550ºC for 2000 h, Forschungszentrum Karlsruhe Internal Report, Fusion 253, October 2005.

[26] B. Can Der Schaaf, et al., High Dose, up to 80 dpa, Mechanical Properties of Eurofer 97," ICFRM-13 Conference, 2007.

[27] R.J. Kurtz et al. Recent Progress on Development of Vanadium Alloys for Fusion, J. Nucl. Mater. **329-333** (2004) 47.

[28] M. Kikoshita et al., Experimental Study of Tritium Recovery from Liquid Lithium by Tritium, Fusion Eng. Design **81** (2006) 567.

Acknowledgment

This work was supported in part by the US Department of Energy under DE-FC02-04ER54698 and DE-FG03-86ER53266.