Available online at www.sciencedirect.com
SCIENCE DIRECT®
ELSEVIER
Journal of Nuclear Materials 329–333 (2004) 148–155
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Materials and design of the European DEMO blankets

L.V. Boccaccini [a,*], L. Giancarli [b], G. Janeschitz [a], S. Hermsmeyer [a], Y. Poitevin [b], A. Cardella [c], E. Diegele [c]

[a] *Forschungszentrum Karlsruhe, P.O. Box 3640, D-76021 Karlsruhe, Germany*
[b] *CEA-Saclay, 91191 Gif-sur-Yvette, France*
[c] *EFDA CSU, Boltzmannstr. 2, D-85740 Garching, Germany*

Abstract

The Helium Cooled Lithium Lead (HCLL) and the Helium Cooled Pebble Bed (HCPB) Blanket are the reference concepts in the European Breeding Blanket Programme for the DEMO design and for the related long term R&D. Recently, a similar design for both concepts has been developed, in particular both concepts use helium coolant and RAFM steel EUROFER as structural material. In this paper the interactions between the selected materials and the proposed DEMO designs are discussed. In particular the design features related to the tritium production, power extraction, material compatibility and fabrication processes are addressed. All these features contribute to the definition of DEMO concepts which are attractive for a future fusion power plant in terms of safety, availability and economics.

1. Introduction

As a result of the European Blanket Concept Selection Exercise in 1995, two blanket concepts, the Water Cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB), were selected as the most promising lines for further development towards a DEMO reactor and as candidate for the testing in ITER [1]. The design work for these two concepts has been performed using the DEMONET boundary conditions (see Table 1). The EU R&D Programme and the design of the ITER Test Blanket Module (TBM) since then were based on these two concepts [2]. Both concepts use of a ferritic martensitic (FM) steel as structural material; the target of the R&D programme was the development of a reduce activation (RA) version of the type 8–9CrWVTa able to withstand neutron damage of 70 dpa for the DEMO with target up to 150 dpa for a future fusion power plant (FPP). The development of the RAFM steel for fusion has led to the definition of an alloy composition that is now known as EUROFER 97.

This situation has been strongly modified by two recent important events:

1. In the period 2000–2002 a European power plant conceptual study (PPCS) was undertaken to demonstrate the credibility, the safety and environmental advantages, the economical viability of the FPP and the robustness of the analyses and conclusions [3]. The study has required the definition a comprehensive plant model including a consistent design of plasma, blanket and divertor. Out of the four selected blanket concepts, the WCLL and HCPB blankets were used for the two 'near term' reactor model A and B, respectively. This study has allowed to define new boundary conditions for the blanket development; in particular the adopted maintenance scheme was derived from the ITER experience and therefore based on the use of large modules. Since 2002, it has therefore been decided to adopt these boundary conditions as a basis for the EU DEMO blanket designs (see Table 1).
2. Because of a significant budget reduction on the EU blanket R&D 2002–2006 program, it has been

* Corresponding author. Tel.: +49-7247 82 2415; fax: +49-7247 82 3718.
E-mail address: lorenzo.boccaccini@irs.fzk.de (L.V. Boccaccini).

0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2004.04.125

Table 1
Main DEMO specifications for blanket design

	DEMO 95 [1]	Present HCLL/HCPB assumptions for DEMO
Plasma configuration	Demonet	From PPCS [3]
	Double null	Single null
Neutron wall load (av/peak) (MW/m^2)	2.2/3.5	2.0/2.4
Surface heat flux (av/peak) (MW/m^2)	0.4/0.5	0.4/0.5
Maintenance concept	Segments	Large modules
FW protection	Bare	2 mm W
Blanket lifetime (h)	20 000	20 000

decided to revise the EU strategy concerning blanket development [4]. In order to limit the development risks, the parallel development of two blanket lines, one using ceramic breeder and one using lithium–lead breeder have been reaffirmed. However, cost saving had to be obtained by sharing as much as possible of structures fabrication and coolant circuit technologies (including demonstration mock-ups and testing facilities prior to TBM installation in ITER). Because of incompatibility from the safety point of view between the beryllium present in the HCPB concept and water-coolant, it has been decided to use helium as coolant for EU concepts.

The consequence of these two events has been the definition of two new blanket design concepts, the modular Helium-Cooled Pebble Bed (HCPB) concept, and the modular Helium-Cooled Lithium–Lead (HCLL) concept which present many common design features. As in the previous designs, the HCPB concept uses as breeder a lithiated ceramic and Be as neutron multiplier in form of pebble beds; the HCLL the eutectic PbLi flowing at low velocity for tritium recovery purposes. Two different ceramic breeders (CB) pebbles have been developed produced and qualified: the lithium orthosilicate (Li_4SiO_4) and metatitanate (Li_2TiO_3). This design activity has successfully ended in September 2003 and it is described in the next section.

2. Design of the new HCLL and HCPB blanket concepts

2.1. Design requirements

For the new design of the HCLL and HCPB blanket, a set of specifications for a typical DEMO module with dimensions at the FW of 2 m×2 m placed at the equatorial outboard region of the reactor has been considered. The thermal loads have been based on the PPCS model B [5]. The design of a typical module includes the first wall (FW), the breeding zone (BZ) and the manifold system; attachments and shielding designs will be considered at a later stage.

The primary design target was to achieve a calculated tritium breeding ratio (TBR) $\geqslant$1.10 in order to ensure a tritium breeding self-sufficiency despite the uncertainties on the obtained results due to uncertainties on nuclear data and to take into account the ^6Li burn-up and the presence of ports without blanket (e.g. plasma heating openings). Also since a decision of the necessity of a FW protection layer has not been taken, an average layer of 2 mm W in the whole blanket FW has been assumed conservatively in the neutronic calculation.

The module is designed to withstand the full coolant pressure in case of an in-box-LOCA; this requirement has been set up to avoid the potential rupture of the module box without relying on performances of rupture disks. The resulting load condition is considered as fault condition; after the accident the reactor should be shut down and the module replaced. A further target of the design is to maximise the coolant temperatures to use the whole temperature windows that EUROFER allows, that is 300–550 °C, where the lower limit is dictated by the DBTT under irradiation and the upper one by creep strength considerations. Hence, both concepts use helium coolant at operational pressure of 8 MPa and inlet/outlet target temperatures of 300–500 °C; the level of temperature selected leads to an efficiency of the power generation system of about 40%.

2.2. Design architecture

The general organisation of the helium cooling proposed for the two concepts makes use of the design principles already adopted in previous helium cooled blankets. In particular, coolant tubes are avoided and the helium flows in channels located inside plate-steel structures to potentially improve the blanket reliability. The cooling is radial; all the large He manifolds are in the rear part of the blanket to maximise the breeder contents in the BZ. The breeder fills the space inside the box that is kept at low pressure; in the HCPB the CB and the Be are in form of a single size pebble beds with a packing factor of about 63% purged by a helium flow at 0.1 MPa. The HCLL uses PbLi at geodetic pressure slowly recirculating throughout the box.

The new version proposed for the HCPB blanket is shown in Fig. 1. The outer shell of the blanket box is made up from a steel plate with internal cooling channels bent into U-shape, the two remaining sides being closed by cooled cap plates. Inlets and outlets of all channels are located at the radial back of the box. Welded into the box is a stiffening grid of radial–toroidal and radial–poloidal plates; each grid plate is cooled by He flowing in internal channels that are fed in the rear part. This grid results in cells open in the rear radial direction with toroidal–poloidal dimensions of about 20 cm $\times$ 20 cm that accommodate the breeder units (BU). The spacing of the grid is determined by the mechanical strength of the box's walls according to the 8 MPa fault condition. The joints of each group of four stiffening plates form a cross that extends into the radial back and is needed for a strong connection of the grid to the module back plate.

The breeder unit for the HCPB concept (Fig. 2) is a base plate that holds two breeder canisters, each providing space for two shallow ceramic breeder pebble beds. The canister walls (one central plate, two side plates) contain a dense array of internal, radially oriented, rectangular cooling channels, fed from headers in the base plate. Each channel of the canister's central plate branches and feeds two channels in the top and bottom plates. The space left by the canisters is filled with beryllium pebble beds and, in case of large volumetric heating, an additional cooling plate. Pipes taking purge gas to the plasma neighbouring front of all pebble beds are welded onto the cooling plates.

The HCLL concept uses the same stiffening box design, leading to the same open cells (20 cm $\times$ 20 cm); however, the requirements for the cooling of PbLi allow a simpler design of the BU (see Fig. 3). Few cooling plates (5 plates are required for the region affected by the peak power density) with meander channels for the helium are connected to a back plate. The PbLi fills entirely the free space in the cell. The structure of the box and grid is adapted to realise the flow of the PbLi inside the module; the vertical grid delimits the flow in vertical columns. The PbLi is fed at the top of each column and is collected at the bottom; a meandering flow in the poloidal–radial plane is realised through the cells (see Fig. 4); alternative flow holes at front/back of horizontal grid plates allow this circulation.

The blanket back region is built up from several large steel plates (see Fig. 1). The thicker outer (D) and the inner plates (B) are connected by radial ribs to provide the mechanical strength needed for contain the high He pressure; the intermediate plate C with radial collectors has only the function to divide the space realising the separate manifolds necessary for the radial cooling scheme. In the HCPB concept, the space between the

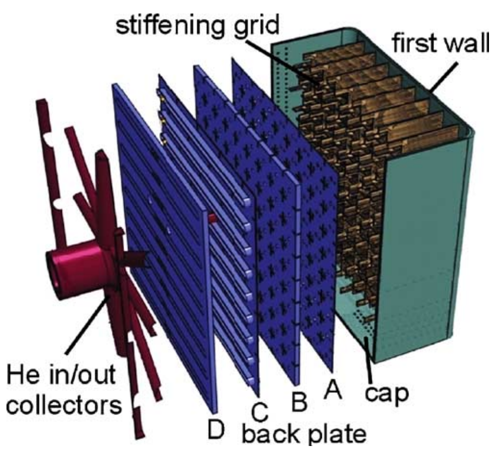

Fig. 1. Blanket box with stiffening grid and exploded back wall (HCPB variant).

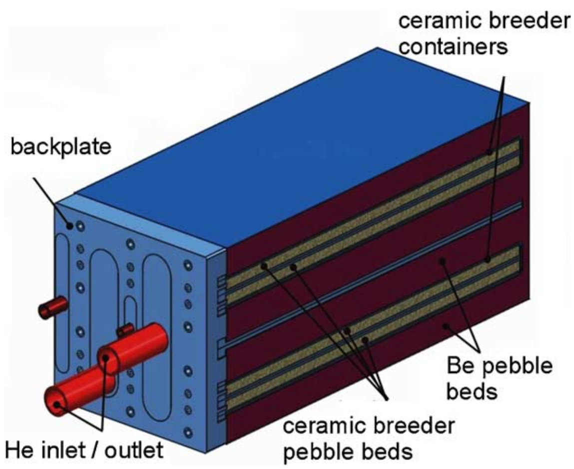

Fig. 2. Side view of HCPB blanket breeder unit.

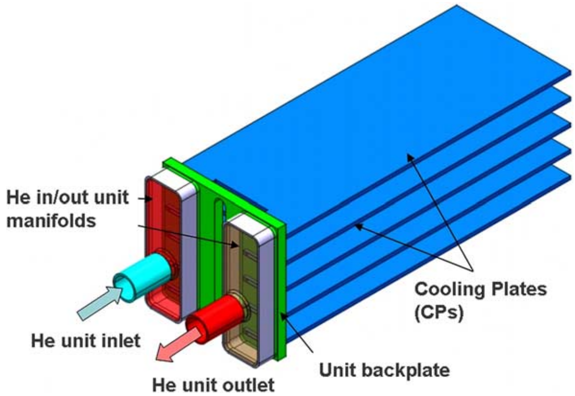

Fig. 3. Side view of HCLL blanket breeder unit.

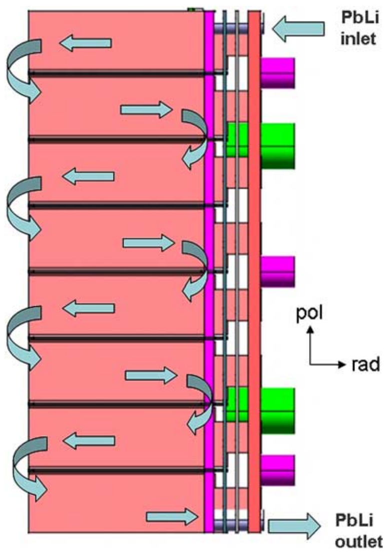

Fig. 4. Flow schema for the PbLi in the HCLL concept.

breeder units and the high pressure manifolds is used by the purge system with a thin plate (A) creating two headers for the inlet and outlet of the purge flow.

Supplying all structures with sufficient cooling is one of the challenging tasks of designing the blanket. In the present HCPB design, helium coolant passes the major blanket parts in series: The first pass is through the U-shape first wall/side walls, the second pass is the stiffening grid plates (75%) and the caps (25%) in parallel and the third pass is the breeder units. An alternative cooling scheme has been adopted in the HCLL, reducing the cooling to two flows in series combining in parallel the stage 1 and 2. Trade-off studies are ongoing to evaluate advantage and disadvantage related with the two proposals.

Neutronic calculations based on PPCS model B [6] have shown the achievement of a TBR of 1.14 with a breeder zone thickness of about 46 cm for the HCPB and 55 cm for the HCLL.

3. Material–design interactions

The interactions between materials and design for reactors using RAFM with solid or liquid breeder have been described in several papers during the last years [1,7]. The new concepts do not change much about the basic issues connected to the materials; in general, the new designs are more demanding in term of TBR due to the presence of a large reinforcement structure.

The switch from WCLL to HCLL has led to an increase of the mean temperature of the blanket structures because of the higher coolant temperature. This has consequences in the design; from one side the concept has a better efficiency of the power conversion system and increases the margin to cope with the increase of DBTT during irradiation. On the other side, the higher maximum interface temperature between PbLi and structures will increase the steel corrosion level. Also tritium permeation towards the coolant is significantly increased, although if at least partially compensated by the more efficient detritiation techniques available for He compared to water.

Additional consideration should be made about the manufacturing technologies related to the new design. As an implicit requirement used during the design work was to try to use fabrication technologies already under discussion in the previous designs, almost the whole R&D programme implemented for the helium cooled structures of the HCPB can be kept; at the present the main effort is to identify and include in the R&D programme additional fabrication technologies (or heat treatments) that could become necessary for the manufacturing of the new blankets [8].

3.1. RAFM steel EUROFER

Many results concerning the EUROFER R&D are already available in literature [9,10]. In the definition of EUROFER the focus was on the improvement of the ductile-to-brittle transition temperature. New results of irradiations in excess of 10 dpa show encouraging results [11].

For a design point of view the most important information about the material are collected in the corresponding material database and the design rules [12]. This information has been used for the new HCLL–HCPB blanket designs. They allow the design at the blanket BOL conditions; extension to EOL conditions is only estimated and will need confirmation in the next years with data from the irradiation programme. The irradiations in fission reactors will be able to qualify the material for ITER application (max 3 dpa at the FW): The qualification of this material for DEMO application will need in addition a verification with a neutron facility (IFMIF is the EU candidate) for simulating the effects of the 14 MeV neutrons at high neutron fluence (with resulting damage >70 dpa and He production). Additional irradiation programmes are requested and already underway for low fluences to characterise the joint region (welded and HIP-ed components).

Neutronic performances have been already taken into account in the definition of the EUROFER composition; for instance, the relatively low amount of W (about 1%) was a compromise between the opposite requirements to increase the mechanical strength and not reduce too much the TBR.

EUROFER shows a good chemical compatibility with Li_4SiO_4, Li_2TiO_3 and Be at least up to 550 °C at the reference gas composition of the purge flow. Corrosion rates observed in PbLi with EUROFER are lowest compared to all other investigated RAFM; however, the acceptability of the corrosion rate at temperatures above 500 °C needs to be checked.

Also the ferromagnetic properties of the EUROFER play an important role in the design of DEMO blankets and ITER TBM. In ITER, the presence of the ferromagnetic TBM in a non-magnetic environment needs to be accounted for the plasma control to compensate local distortion of the magnetic confinement. In DEMO, the effect of a complete ferromagnetic blanket has to be assessed. Mechanical loads caused by disruptions are also affected by these properties as well.

For the control of tritium, data of permeation of the H isotopes are necessary. Tritium permeation coefficients in EUROFER have been measured [13]. The permeation is dependent on the surface status, in presence of an oxidation potential in helium the EUROFER surface is coated by a natural oxide layer with an estimated tritium permeation reduction factor (TPRF) at least of 10; this value has been measured for MANET, experiments are on going to confirm this data for EUROFER.

3.2. Materials for the HCPB concept

During the past years a considerable effort has been made in the EU programme for the development of CB pebbles; the Li_4SiO_4 pebbles produced by melt-spraying and the Li_2TiO_3 pebble produced by extrusion–spheronisation–sintering process. Fabrication [14,15] and recycling [16,17] routes have been defined and demonstrated, specifications for DEMO (control of impurities) have been formulated and the production of both CB has reached a semi-industrial level with the maximum capacity of about 150–300 kg/year. The characterisation of these ceramics in out-of-pile conditions have been already concluded including chemical compatibility with EUROFER and purge gas, thermo-mechanical properties, high-temperature long term behaviour, and tritium release characteristic; the preparation of a database that will collect all the available information for the design is ongoing.

During several years of investigation [18], no killing issues have been identified for any of these two materials. The T residence time is sufficiently low and not significantly degraded under irradiation, resulting in low tritium inventory. As Li_2TiO_3 is not sensitive to moisture, its handling is simple; a possible advantage for the design point of view can be its lower thermal expansion that reduces the build-up of stresses during the heating phase and the possible gap formation during cooling. Li_4SiO_4 is convincing for the simpler fabrication and recycle processes. Its higher Li-density could be an advantage for neutronic design; however, up to now any proposed design could reach the required TBR using both CB with the condition to increase the ^6Li enrichment for the Li_2TiO_3 by about 20–30%.

It is common opinion among the experts that the most important criteria for a selection of one of these materials for DEMO application will be the demonstration of its resistance under relevant irradiation. In the EU programme dedicate irradiations (EXOTIC-8) of samples of Li_4SiO_4 (11% Li burn-up as anticipated in DEMO) and Li_2TiO_3 (17% due to the lower content of Li) have been already completed; results of the post irradiation examination will presented at this conference [19]. A new one-year-irradiation is planned in Petten in the frame of the HICU project [20]. Characteristics of this irradiation are high temperature and ratio dpa/% Li burn-up relevant for DEMO conditions.

As far as the beryllium is concerned, since 1999 the reference material grade has been considered the 1-mm pebbles produced by NGK with electrode rotating methods. The specifications of this pebble for DEMO application include a general control of impurities for activation reason (in particular U) and a control of Al, Mg, Fe and Si impurities to avoid low melting phases. These beds have been investigated in detail for several years [21]; steam–air reaction, and electrical conductivity were included in the programme. The thermo-mechanic characterisation should be completed in the next year; in the data base under preparation in FZK the properties of the stress–strain curve including creep effects in uniaxial test and thermal conductivity for beryllium at high temperature (up to 650 °C) are still missing.

The major design issue connected with the use of Be is its behaviour under irradiation, mainly swelling and tritium inventory. Lack in the database and in the modelling give large uncertainties in the design calculation of the EOL tritium inventory in Be in FPP conditions. Also possible solutions by design to reduce this inventory (e.g. increasing the operational temperatures of the beds or baking of the blanket in the reactor at about 650 °C) cannot be supported by calculations. In spite of the progress made to better understanding the physic of the phenomena [22], the goal of producing a reliable code to support the designer in these choices, has not been achieved. An irradiation campaign to obtain data of Be at 3000 appm of helium in 2006 and 6000 appm helium in 2008 with temperatures in the range 500–700 °C will start the next year in Petten in the frame of HIDOBE task. With these data the modelling should be improved and complementary an empirical extrapolation to the DEMO condition (18 000 appm) could be attempted.

Alternatives Be grades have recently been proposed such as the Be–Ti intermetallic compound [23]; this material promises better tritium release, reduced swelling, and better mechanical properties at high temperatures. However, the neutronic performances will be slightly decreased and fabrication technologies for the pebbles should be demonstrated.

For ceramics, a maximum allowable temperature of about 920 °C is commonly accepted. The maximum temperature of the beryllium is limited to 650 °C due to swelling, degradation of mechanical properties and safety (steam reaction) concerns; however, a clear limit has not been set and should be supported by new investigations of the real impact on the proposed design. The temperature control in Be and CB pebble beds affect strongly their tritium release characteristics, swelling and thermo-mechanical behaviour (stresses, bed thermal conductivity, gap formation, etc.). Prediction methods and codes are necessary for the design; as mentioned, an almost complete data base of thermo-mechanical properties of the materials is already available, but specific tools for the calculations, especially in presence of creep, have not yet been completely developed [24].

3.3. Materials for the HCLL concept

The lithium lead is used in its eutectic composition ($PbLi_{eu} = 15.8$ mol.% Li) with 90% ^{6}Li enrichment to ensure the target tritium breeding ratio. It is slowly recirculated to allow composition control, tritium extraction and its purification from oxides, corrosion products and transmutation products (Bi, Po, Tl and Hg). Process techniques required for a FPP needs to be developed and/or improved.

PbLi purification may be necessary for extracting corrosion and activation products. In particular, radioisotopes like the α-emitter ^{210}Po, characterised by high ingestion and inhalation hazard potential in case of accidental release, are of special concern. The production of the ^{210}Po isotope is unavoidable because it is produced by a sequential reaction on Pb passing through the production of Bi (beside initial Bi impurities). Experimental results shows that the release of Po is determined by the vapour pressure of an intermetallic PbPo compound which is orders of magnitude lower than that of Po and then the presence of ^{210}Po may not be so critical depending on temperature rise scenarios during accidental situations. However, it may be needed to develop on-line Bi-removal techniques (a Bi concentration below 1 ppm would limit the ^{210}Po concentration below 0.1 ppb). The purification (e.g. by magnetic or cold trapping) may occur in-line or by batches depending on the design criteria to be considered in a FPP. The purification system should be placed after the T-extractor in series with it. Part of this system can be used also for ^{6}Li refurbishing.

To mitigate the problem of corrosion and optimise tritium recovery cycle, barriers/coatings have been proposed to protect the EUROFER from direct PbLi exposure. However, the question of the need of high-performance tritium permeation barriers between PbLi and EUROFER structures in HCLL concept is not fully fixed today. It has been previously said indeed that the increase of the steel temperature in contact with PbLi significantly increases the tritium permeation towards the He-coolant. Thus, the possibility and interest of over-dimensioning the former He-purification system up to the dimension of a true extraction system has to be evaluated in term of economical and safety criteria. In particular safety limits have to be assessed related to the maximum acceptable total losses of T both by permeation to the secondary water-steam loop through the heat exchanger and by leakage through the He-coolant circuit components (pumps, etc.). In case of attractive results in this approach, high-performance tritium permeation barriers might not be needed, although coatings could remain of first interest for corrosion issues. In parallel, the R&D program on tritium permeation barrier needs to be continued with some adaptations, at least to better understand the limitations of such technologies and evaluate the limits of the above mentioned approach.

In spite of a development of several years, a breakthrough in the fabrication and reproducibility of suitable Al-based permeation barriers from the PbLi-side, based on steel surface aluminisation and Al-oxide, has not been yet achieved [25]. Although permeation barriers tested in gas phase showed very promising results, the values of the TPRF reached in out-of-pile condition with coated EUROFER in contact with PbLi were largely lower than the design target (as expressed in the past for the WCLL concept). The low results in term of TPRF obtained by chemical vapour deposition Al-based coating demonstrated the high sensitivity to deposition process parameters. Micrographs showed a low quality of coating and poor adherence to the bulk EUROFER material that were less observed in former experiment with coated MANET. Also hot-dip aluminising coating gave unexpected results. In this case, the problem was identified in thermal stresses induced in the welded area of specimen leading to crack initiation. Not uniformity in coating thickness can also induce the formation of brittle intermetallic Fe–Al phase, corrodible in liquid metal. In addition, self-healing of these barriers could not be significantly demonstrated in those experiments.

Concerns have been also expressed for the use of Al based coatings from the point of view of waste management. In order to keep the benefit of reduced-activation characteristics of the irradiated EUROFER structures, a suitable process may be required to remove the coatings layer from the supporting EUROFER structures during blanket dismounting. Further investigations considering other kinds of barriers at the PbLi contact, namely oxide- and carbide-type coatings (e.g. Cr_2O_3, Y_2O_3, Cr_3C_2) and tungsten layers, have been started. On the other side, promising results of permeation barriers in contact with helium could open the possibility of using permeation barriers on the He-side.

Detailed design studies are ongoing to investigate the HCLL configuration and, at the end, to precise the requirements of barriers and purification/extraction systems for both PbLi and He coolant. The development of permeation/corrosion barriers is today still a concern for the R&D programme oriented to the demonstration of the attractiveness of this concept for a future FPP.

Finally, the low circulating PbLi makes the MHD pressure drops acceptable for the design without using any insulating coatings; in any case the design should be checked against MHD to avoid not uniform distribution of flow in parallel channels, or the formation of stagnant region inside the blanket with T accumulation [26].

3.4. Materials for shield/manifolds and joints

The study of the design and the materials necessary for the shielding/manifold region behind the FW/BZ proposed for the new HCLL/HCPB design is at the beginning. For the HCPB design a first definition of this region has been presented in the frame of PPCS; a concept of reactor integration based on a high temperature shielding zone integrated to the large modules and a low temperature shielding zone connected to the manifolds has been presented in that study [5]. For the HCLL, an inclusion in the PPCS is foreseen in 2004.

As far as the composition of the shielding is concerned few studies have been issued. The only comparative study has been presented recently by Taylor [27] in the frame of the PPCS; he compares steel shields with water (EUROFER, OPSTAB, or a combination of these two materials) and alternative compositions if water should be entirely excluded; mixtures of steel or tungsten with hydrides and He cooling have been also investigated as well tungsten–carbide. The study has been performed under the point of view of the waste managements. Further work is necessary in this field considering a trade-off among other design criteria, e.g. reduction of the in-vessel thickness of the blanket, effect of the temperature increase during accident (after-heat and effect of decomposition of hydrides), etc. with the goal to define an optimised composition of the shield for each concept.

For the assembling of the modules and the shielding/manifold components suitable materials and joint technologies (e.g. friction welding) should be defined.

4. Conclusions

A new design for the DEMO blanket has been proposed in this year, with a common architecture for the HCPB and HCLL concepts. These designs are the reference for the testing in ITER and for the definition of the R&D programme in Europe.

As far as the materials is concerned, the successful development of the HCPB blanket depends almost entirely by the behaviour of the breeder and multiplier materials in term of capacity to withstand the fusion environmental without unacceptable degradation of their properties for a sufficient lifetime.

In the case of HCLL, the most important issue is the tritium control which will have to be assessed in the near future in order to establish the requirements in terms of permeation barriers, T-extraction efficiency from both PbLi and He, and potential of reduction of He-leakage in the He-cooling circuit components.

A reassessment of the DEMO specifications will be launched in Europe in the next few years as a continuation of the work started with the PPCS. It will address the definition of a revised EU Demo reactor and complete the integration of the proposed Blanket systems in this reactor. Aspects such as the mechanical and hydraulic connections of the modules, the shield and the manifolds region, but also the integration of divertor components compatible with the blanket systems needs to be part of this revision. As far as the materials are concerned, new classes of material have to be included in the EU long term Programme e.g. shielding and joint materials.

References

[1] L. Giancarli, M. Dalle Donne, W. Dietz, Fusion Eng. Des. 36 (1997) 57.
[2] M. Gasparotto, L.V. Boccaccini, L. Giancarli, S. Malang, Y. Poitevin, Fusion Eng. Des. 61&62 (2002) 263.
[3] D. Maisonnier et al., in: Third IAEA Tech. Comm. Meeting on Steady State Operation of Magnetic Fusion Devices, Greifswald, Germany, May 2002.
[4] M. Gasparotto, R. Andreani, L.V. Boccaccini, A. Cardella, D. Maisonnier, G. Federici, L. Giancarli, G. Le Marois, S. Malang, A. Moeslang, Y. Poitevin, B. van der Schaaf, M. Victoria, Fusion Eng. Des. 66–68 (2003) 129.
[5] S. Hermsmeyer, S. Malang, U. Fischer, S. Gordeev, Fusion Eng. Des. 69 (2003) 281.
[6] Y. Chen, U. Fischer, P. Pereslavtsev, F. Wasastjerna, The EU power plant conceptual study – neutronic design analyses for near term and advanced reactor models, FZKA 6763, April 2003.
[7] A.R. Raffray, M. Akiba, V. Chuyanov, L. Giancarli, S. Malang, J. Nucl. Mater. 307–311 (2002) 21.
[8] A. Cardella, E. Rigal, L. Bedel, Ph. Bucci, J. Fiek, L. Forrest, L.V. Boccaccini, E. Diegele, L. Giancarli, S. Hermsmeyer, G. Janeschitz, R. Laesser, A. Li Puma, J.D. Lulewicz, A. Moeslang, Y. Poitevin, E. Rabaglino, this conference.
[9] T. Muroga, M. Gasparotto, S.J. Zinkle, Fusion Eng. Des. 61&62 (2002) 13.
[10] B. van der Schaaf, F. Tavassol, C. Fazio, E. Rigal, E. Diegele, R. Lindau, G. Lemarois, Fusion Eng. Des. 69 (2003) 197.
[11] Rensman et al., this conference. doi:10.1016/j.jnucmat.2004.04.033.
[12] A.-A.F. Tavassoli et al., this conference. doi:10.1016/j.jnucmat.2004.04.020.

[13] A. Aiello et al., Fusion Sci. Technol. 41 (2002) 872.
[14] J.D. Lulewicz, N. Roux, J. Nucl. Mater. 307–311 (2002) 803.
[15] W. Pannhorst, V. Geiler, G. Räke, B. Speit, D. Sprenger, in: Proceeding of 20th Symposium of Fusion Technology, 7–11 September 1998, Marseille, France.
[16] C. Alvani, S. Casadio, V. Contini, A. Di Bartolomeo, J.D. Lulewicz, N. Roux, J. Nucl. Mater. 307–311 (2002) 837.
[17] R. Knitter, G. Piazza, J. Reimann, P. Risthaus, L.V. Boccaccini, CBBI-11, Tokyo, in press.
[18] G. Piazza, J. Reimann, E. Günther, R. Knitter, N. Roux, J.D. Lulewicz, J. Nucl. Mater. 307–311 (2002) 811.
[19] G. Piazza, A. Erbe, R. Rolli, O. Romer, this conference. doi:10.1016/j.jnucmat.2004.04.235.
[20] J.B.J. Hegeman, E.D.L. van Essen, M. Jong, J.G. van der Laan, J. Reimann, Fusion Eng. Des. 69 (2003) 425.
[21] A. Möslang, E. Rabaglino, L.V. Boccaccini, A. Cardella, E. Alves, F. Druyts, U. Fischer, J.B.J.W. Hegeman, D. Leichtle, G. Piazza, J. Reimann, L. Sannen, M. Scibetta, J. van der Laan, BeWS-6, 2–5 December 2003, Miyazaki, Japan, in press.
[22] E. Rabaglino, J.P. Hiernaut, C. Ronchi, F. Scaffidi-Argentina, J. Nucl. Mater. 307–311 (2002) 1424.
[23] M. Uchida, E. Ishitsuka, H. Kawamura, J. Nucl. Mater. 307–311 (2002) 653.
[24] J. Reimann, L.V. Boccaccini, M. Enoeda, A.Y. Ying, Fusion Eng. Des. 61&62 (2002) 319.
[25] D.L. Smith, J. Konys, T. Muroga, V. Evitkhin, J. Nucl. Mater. 307–311 (2002) 1314.
[26] J. Reimann, G. Benamati, R. Moreau, Report on working group MHD for the blanket concept selection exercise (BCSE), FZKA 5652, September 1995.
[27] N.P. Taylor, Shield design choices for optimum waste management performance, WMS/UKAEA/TSW1D4/1, August 2001.