Development and Reactor Integration of Helium Cooled In-Vessel Components for DEMO

S. Hermsmeyer, L.V. Boccaccini, U. Fischer, G. Janeschitz, C. Köhly, J. Rey

Forschungszentrum Karlsruhe Euratom Association, Karlsruhe, Germany

e-mail contact of the main author: stephan.hermsmeyer@iket.fzk.de

Abstract. As the physics of a burning plasma are coming closer to being demonstrated in ITER, the focus of fusion research will shift towards the technology needed to build a demonstration reactor. The EU supports the development of its reference breeding blankets mainly within its long-term breeding blanket and materials R&D programmes. The recent European Power Plant Conceptual Study (PPCS) points out that a plant based on Helium-cooled in-vessel components, i.e. breeding blanket and divertor, while assuming limited technological extrapolation, has the potential for producing electricity in a competitive future market. While this outcome is based on careful analysis of key functions of the in-vessel components, the complexity of the task has prevented a comprehensive design of a blanket/divertor system fully integrated within the reactor vessel of a tokamak fusion machine.

The paper presents the current status of a study of design concepts available for reactor integration, based on the recent development in Europe of blankets and divertors, and focussing on the EU modular Helium cooled pebble bed (HCPB) blanket concept. As a geometry relevant for a demonstration reactor, a reactor model of 3300 MW fusion power is assumed. Conceptual design solutions are proposed for (i) blanket segmentation; (ii) curved blanket modules covering the tokamak surface; (iii) replacable shield modules; in-vessel piping including compensation for differential thermal expansion; and, (iv) attachment and remote handling schemes for blanket and shield. The requirements imposed by in-vessel integration have implications on blanket performance, since they affect blanket coverage, the amount of structural steel required, and radial space requirements. The work provides important feedback especially on current breeding blanket design, and input for the definition of a future European DEMO study.

1. Introduction

The breeding blanket programme has been a focus of European fusion research for more than a decade. Recently, it has been driven by the EU Power Plant Conceptual Study (PPCS, [1]), investigating the potential of fusion energy in a future market, and by the goal to develop, build and license Test Blanket Modules that allow testing of reactor-relevant technology in ITER. In particular, the PPCS identified a need to review the Helium Cooled Pebble Bed (HCPB) reference blanket concept. That review was carried out in 2003 [2], with a reduced set of requirements that leaves the main work of integrating in-vessel components for a tokamak demonstration plant to be done.

The work presented in this paper is a continuation of the blanket development. It presents concepts for in-vessel components and their integration in a DEMO-like tokamak geometry. The key issues are (i) coherent design of all components; (ii) quick replacement of the breeding blanket; (iii) possibility of repair of all the in-vessel system preferably with the same remote handling tools; and, (iv) thermal expansion compensation. The work is presented with reference to the HCPB design. The integration of a He cooled divertor is a major subtask that has been started and will be reported soon.

2. Overall concept

The presented blanket integration is guided by the goal of adopting the modular HCPB breeding blanket [2] to the tokamak plant geometry, see Fig. 1, without changing its positive functional properties like Tritium breeding sufficiency, by means of maintaining the overall blanket material composition, and creating an environment consistent with remote handling through vacuum vessel (VV) ports.
The reactor model used for this integration is the original plant model B from the PPCS, see Table I. The distribution of neutron loads inside the blanket system was carried out with a detailed 3D MCNP model [3] that was applied for the evaluation of Tritium breeding during the 2003 review of the HCPB, too [4]. For the present study, those loads were translated to a different blanket segmentation according to the last column of Table II. Also, scaled ITER plasma contours and VV were used to start the design with a realistic blanket geometry.

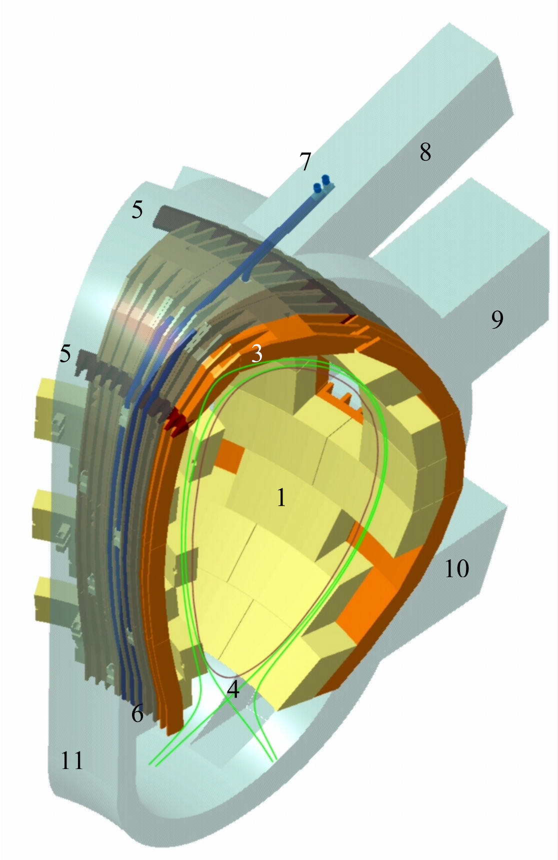


FIG.1. In-vessel assembly. Breeder modules (1), shield modules (2), upper cassette (3), plasma flux lines (4), upper cassette rails (5), He main pipes (continued undera divertor) (6), upper port He outlet (7), upper port (8), mid-plane port (9), divertor port (10), vacuum vessel (11).

TABLE I: KEY PLANT DATA

Fusion power	3300 MW
Major/minor radius	8.6 / 2.87 m
TF coils (i.e. sectors)	18
Blanket thickness (IB/OB)	1 m / 1.3 m
Blanket Helium temp. (in/out)	300 / 500 °C

The blanket system to be integrated is composed of FW, breeding region, shielding and piping. The proposed concept adopts a radial segmentation with, from the plasma to the VV,

1. a high-temperature FW/breeding blanket/first shield ("breeder module") at 300-500°C Helium inlet/outlet temperatures, fixed with a small gap (~20 mm) on
2. a cold shield blanket ("shield module") that is operated at a temperature similar of the VV.

This solution implies that there is no differential thermal expansion between VV and shield; it allows the shield to be bolted to the VV and become part of it with very good thermal conductivity. Large differential expansion exists between breeding and shield modules, where flexible attachment is required. The breeding blanket, operated in DEMO to ~75 dpa within three full power years, will be replaced by cutting the main Helium pipes with in-bore tools, un-locking the mechanical attachment from the plasma side and lifting off the module. This strategy exploits the large dimension of the Helium piping that would pose huge problems to an entirely frontal access.

The prime requirement for the shield is the re-weldability of the VV and the protection of coils from neutron damage. The shield design is modular, with sizes similar to breeder modules; it is designed as permanent component that will remove about 1.5% of the thermal power of the reactor. The shield will be cooled with water (reference solution) or with helium, at temperatures of around 100°C. A key requirement of this system is the option of remote handling in case of repair, with similar tools as used for the breeder modules.

The main Helium pipes are attached to the shield modules; as the temperature of these pipes is up to 400 K higher than the shield, the relative thermal expansion needs to be compensated by the flexibility of the piping system. The cross sections required for Helium limit the choice of solutions; axial and lateral compensators are proposed for each tube attached to the shielding module. For shield module replacement, pipes are cut between neighbouring shield modules.

This general concept is adopted for the modules in the middle region, with the exception of the blanket and shield module in front of the horizontal ports; these modules will be integrated with the port plugs and replaced with them, see Fig. 1.

Another exception will be the upper blanket/shielding region, where the transition from inboard (30° sectors) to outboard (15° sectors) and a tight poloidal curvature complicate an adaptation of the modular breeder and shield. A remote handling strategy similar to that proposed for the divertor region, namely the cassette replacement concept has the potential for fast replacement through the upper ports, independent of breeder module replacement. In addition, the remote handling of the breeder modules through mid-plane ports could take advantage of the rail systems fixed to the VV behind the cassette of divertor and upper blanket. However, the reference is the manipulator system used in ITER [5].

3. Design description

3.1. Breeding blanket modules

Fig. 1 displays the modular blanket segmentation, with 108 inboard modules, 240 outboard modules, and 54 upper blanket cassettes covering the roof of the VV. The module size is determined by the assumption of a 10t weight limit assumed for remote handling, and by the limited piping space on the inboard, see Table II. The segmentation is guided by the goal of having modules with similar toroidal extension, both inboard and outboard, to unify design and fabrication.

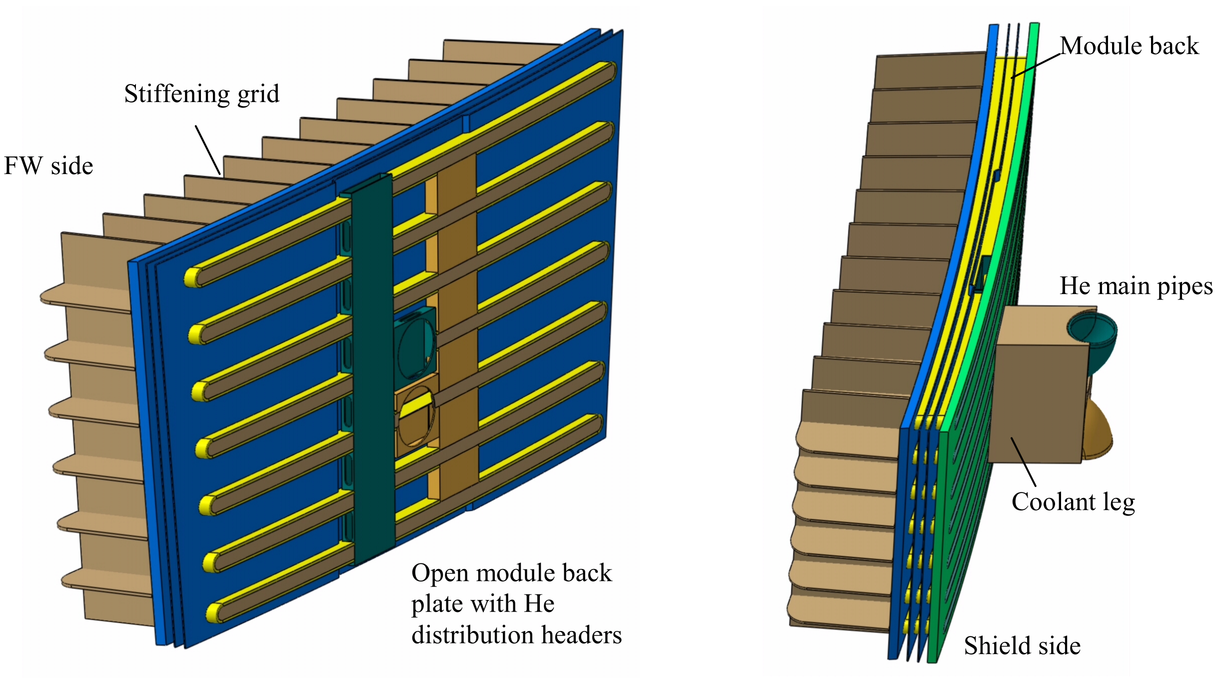


FIG. 2. Blanket module (lower vessel, outboard). Module w/o FW and with open back plate (left), with complete back plate and coolant leg (right).

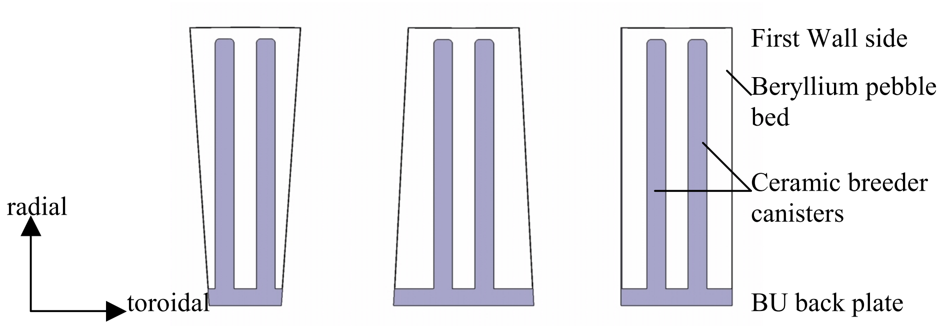


FIG. 3. Types of breeder units. Inboard (left), outboard (centre), and outboard key module (right).

Fig. 2 displays the internal design of an outboard blanket module. The notable changes in comparison to the generic design [2] lie in the breeder units (BUs), see Fig. 3. All inboard modules and the outboard non-key modules (see below) provide breeder unit space in their stiffening grid that has constant poloidal height but changing width between BU back plate to the First Wall. From a fabrication point of view, vertical (i.e. radial-poloidal) breeder canisters offer the simplest design, as the deviation from rectangularity is simply filled by a Beryllium pebble bed; however, R&D is needed to show that thermal ratcheting can be avoided in vertical breeder beds. In contrast to [2], the module main He headers have been integrated into the back plate, as proposed in [6].

Fig. 4 displays the assembly of an inboard breeder module with a shield module. The mechanical attachment of the blanket box to the shield is by a system of shear keys for forces in poloidal-toroidal direction, and by 4 hinged rods for radial forces. These rods that are developed by CIEMAT for the HCLL model in the EU PPCS [7] have a degree of freedom in the direction of thermal expansion of the blanket module. The module toroidal fix point is at the foot of the Helium inlet/outlet leg, the poloidal one is defined by the shear keys.

The attachment system has not been designed any further; after the box has been positioned on the shear keys, it is required to lock and pre-stress the hinged rods by front side access through 30 mm diameter holes between the breeder modules, see Fig. 4.
While in-bore tools in the in-vessel Helium supply and purge lines are operated for cutting and joining, front access is used for mounting and unlocking the mechanical attachment. Parallel poloidal caps mean that the removal of modules by a manipulator towards the opening plasma space is straightforward on the inboard. On the outboard, every second module in toroidal direction is a key module with parallel toroidal faces that may be removed to the plasma chamber at any time; the remaining modules fill the radially opening space and need their toroidal key neighbours to be taken out before them.

TABLE III: SCALED POWERS AND OTHER DATA FOR THE BREEDER MODULES

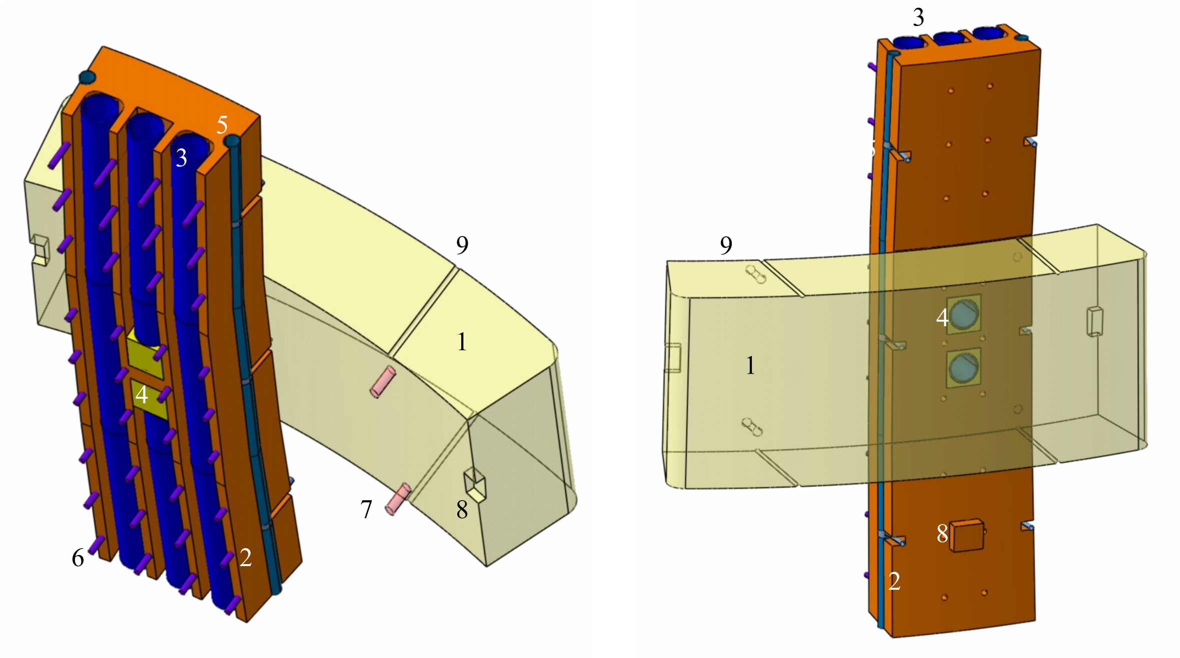
			In-vessel pipes		
Modules	n	P_{mod} [MW]	d_i [mm]	$0.5\rho c^2$ [Pa]	PPCS [#]
Inboard	12 x 9	8.8	150	22930	1- 4
Upper cassette	54	10.1	200	9560	5, 6
Upper outboard	24 x 3	9.7	200	8814	7, $\frac{2}{3}$ 8
Mid-plane outboard	36 x 2	9.1	170	14860	$\frac{1}{3}$ 8, 9
Lower outboard	24 x 4	10	200	9370	10, 11

3.2. Shield modules

Fig. 4 shows that shield modules are bolted to the VV along poloidal webs that fill the space between the He pipes. For a detailed design it is necessary to integrate shield attachment, shear keys and flexible supports for the breeder module, and penetrations for coolant/purge supply, with neutron moderator and cooling systems of the shield. Using borated water for cooling and neutron moderation is likely to be the most simple and reliable choice, and regarded as the reference. The proposed design of the shield as an extension of the VV gives reason to expect that the design could achieve safety properties little different from the PPCS case where the VV is water cooled. As a back-up solution, a He cooled shield using zirconium hydride or tungsten carbide is viable but more complex to design.

3.3. In-vessel piping

The in-vessel piping is attached to the shield modules' back side; it supplies breeder modules with Helium coolant at 300°C inlet and removes it at 500°C outlet temperature. Helium inlet and outlet pipes stick out of the module as legs: they penetrate the shield and are welded to supply lines by in-bore tools in the well-protected region behind the shield; the material choice of austenitic steel at the weld joint, with a transition from Eurofer steel on the blanket side, results in reliable welds with simple heat treatment.
At a shield temperature of around 100°C, the differential thermal expansion could reach up to 14mm for a 3m-high shield module. In the non-corrosive Helium coolant atmosphere, multi-layered metal compensators appear as the most compact design option. A preliminary lay-out by Witzenmann GmbH, Germany suggest that a stainless steel compensator system of



seen from vacuum vessel seen from the plasma

FIG. 4. Breeder module (1), shield module (2), main He pipe (3), inlet/outlet legs (4), purge manifold with flexible breeder module connections (5), shield-to-VV bolts (6), breeder module stub keys (7), shear key system (8), front side access holes (9).

diameter DN 150 can be designed for 8 MPa, 550°C for e.g. 27000 load cycles. At an outer diameter of less than 200mm, this component can be integrated into the current shield design.
A main concern of the piping design is its fitness for using in-bore tools, which demands that all bending radii support this technology and that pipes allow maintenance access close to their exit from the VV. The reference design of the Helium purge lines is similar to the coolant, with in-bore cutting/joining behind the shield.
The high stagnation pressure for the inboard piping, see Table III, suggests that much pressure drop could be saved by increasing the pipe diameter there.

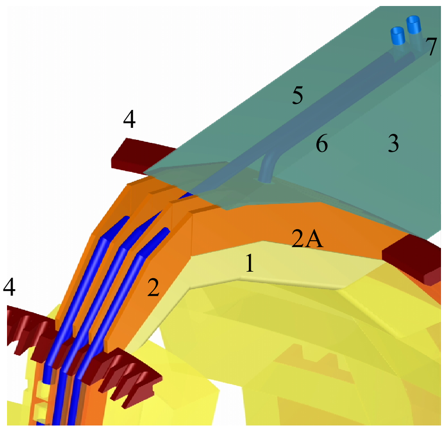


FIG. 5. Upper blanket cassette. Breeder part (1), shield part (2, 2A), upper port (3), rail system (4), inboard He supply (5), cassette He supply (6), access for in-bore tools (7)

3.4. Upper vessel zone with blanket cassette

Fig. 5 shows the main elements of the cassette system; again, it is separated into a consumable breeder part and a permanent shield. The toroidal rail system takes the breeder cassette from the maintenance port to its final position. For the key cassette, the shield 2A is part of the port plug. The figure indicates that the upper vessel is critical for routing the inboard Helium main pipes to their exit through the upper ports. The location for inserting in-bore tools is outside the VV; it imposes restrictions on the pipe bending radii up to the access point.

3.5. Lower vessel zone with divertor cassette

The divertor region has so far not been a concern of the integration exercise. However, the large cross section of inboard He supply lines has led to the requirement that these pipes enter under the divertor and leave the VV through the upper ports. This is a major boundary condition for a future integration of the divertor.

4. Blanket replacement considerations

The downtime required for replacement is one key for the potential of a blanket concept to support high plant availability. In the proposed concept, the interfaces of a module are 2

TABLE IV: POSSIBLE SCENARIO FOR REPLACEMENT OF BLANKET MODULES

	Working in parallel on different modules	
	Plasma-side access	In-bore access
Access paths	4 mid-plane ports	2 x 348 coolant pipe, 3 x 60 purge manifolds, outside VV
Assumed remote handling	ITER-like rail system	In-bore train
Operation	Detach/attach module, grip module, take from position to port	Cut/join and inspect hydrau-lic connections
Parallelism	2 manipulators (up to 4)	Nearly arbitrary, determined by logistic situation
Estimated removal time	1 h (after cutting pipes)	2 h per cut, e.g. 5 cuts per module in parallel
Removal of blanket	174 h for 2 manipulators	174 h for 20 trains (5/quadrant)
Overall removal time	174 h (8 d)	
Estimated joining time	2 h	6 h per joint
Installation of all blanket modules	384 h for 2 manipulators	522 h for 20 trains
Overall installation time	522 h (22 d)	

coolant lines; 2, possibly 3, purge lines; 1 positioning key and 4 rods for mechanical attachment; 2 shear keys for disruption loads; and, a strong electrical connection to the shield.
The use of in-bore tools for the hydraulic lines opens an attractive path for remote handling: cutting and welding operations of these lines can be done independently of the in-vessel manipulator, and with a large degree of parallelism. This is in contrast to ITER, where all

remote handling uses front side access. Table IV illustrates how beneficial parallelism is in cutting down on times required for the welding and cutting of pipes, without going to its limits. Similar considerations hold for upper cassettes and divertor cassettes.
This concept was not considered in the PPCS; it puts into perspective the categorical rejection of using several hundred mid-size blanket modules expressed in that study.

5. Conclusions and further work

This paper has aimed at proposing realistic, near-term concepts for in-vessel components, and their integration in a tokamak reactor. Even though it describes an intermediate state of the integration exercise, it has addressed key questions like segmentation, thermal compensation, and potential for high availability. Possibly most noteable is the fact that a combination of front-side and pipe in-bore handling offers a significant potential for parallel operation and thus good availability for a blanket system with several hundred modules. It seems safe to conclude that a continued effort will lead to a consistent and attractive Helium-cooled next step reactor.
Open questions that are currently addressed are the mechanical attachment, a detailed remote handling concept and the integration of a Helium-cooled divertor. The design of the complete piping of a sector including compensators, and of the upper cassette are further examples for pressing issues. Once these design issues have been addressed, the focus will move to analyses of the design regarding its capability of sustaining operating conditions in the reactor.
The design is believed to be a valuable input for a DEMO study planned in the EU.

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