Progress of the conceptual design of the European DEMO Breeding Blanket, Tritium Extraction and Coolant Purification Systems

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In the frame of the EUROfusion consortium activities the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium Lead (WCLL) concepts are being developed as possible candidates to become driver Breeding Blanket (BB) for the EU DEMO, which aims at the tritium self-sufficiency and net electricity production. The two BB design options encompass water or helium as coolants and solid ceramic with beryllium/beryllides or PbLi as tritium breeder and neutron multipliers. The BB segments have evolved towards a more stable conceptual design taking into account multiple feasibility aspects and requirements imposed by interfacing systems. Possible solutions to improve shielding capabilities of Helium cooled BB are investigated and the impact of water coolant activation is assessed by studying the spatial distribution of ^{16}N and ^{17}N isotopes dose rates, in particular in proximity of isolation valves. The reference and back-up technologies for the Tritium Extraction and Removal (TER) from the helium purge gas and the PbLi are developed addressing key feasibility aspects and implications on the tokamak layout and with considerable R&D efforts. As the BB internals offer an ideal environment (high temperatures, thin structural material) to promote the tritium permeation, studies are devoted to establish a tritium balance in the different systems during operation, with special care to the permeation rate and inventory in the coolant. Those are the key parameters for the feasibility assessment and technology selection for the Coolant Purification Systems (CPS).

Keywords: Breeding Blanket, Coolant Purification, Tritium Extraction and Removal.

1. Introduction

The component that in the EU DEMOnstration Fusion Reactor (DEMO) will accomplish the function of breeding the tritium is the Breeding Blanket (BB) [1]. Taking into account the ambitious schedule of the DEMO roadmap and the novelty and feasibility concerns of most of the technologies used in the BB design a sustained program of R&D is implemented in EUROfusion, in the Work Package Breeding Blanket (WPBB), to accompany the development and selection of the reference BB concept. An essential element of the Breeding Blanket development for DEMO is the alignment with the EU TBM program for ITER [2][3]. As a matter of fact, the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium Lead (WCLL) concepts are the two TBM concepts to be tested in ITER and are being developed as possible candidates to become driver Breeding Blanket (BB) [4].

The design and integration work conducted to date shows clearly that some technical features of the BB (the type of coolant, the type of breeder, the technology used for the tritium extraction, the tritium control, the technology used for the coolant purification) impact not only the design of the BB system but also the design of the interfacing systems and as a consequence the overall tokamak layout. The coordination of design activities and studies among the different areas involved allow to gain confidence that the design and architectures being developed are consistent and the major issues are addressed and tackled. In this paper, after a brief recall of the last evolution of the HCPB and WCLL designs, we focus on: the studies carried out on the BB shielding performance and the water activation, the design of Tritium Extraction and Removal Systems (TER) and the accompanying R&D, the tritium transport analyses and the design of the Coolant Purification Systems (CPS). The topics presented are not meant to be exhaustive of the design and integration efforts, but summarize the most relevant recent achievements.

2. Breeding Blanket and Tritium Extraction Systems design and performances

The HCPB and the WCLL BBs have a number of common architecture and design features, imposed by the use of a common DEMO baseline. They are both characterized by 16 sectors (dictated by the number of TF coils), each including two inboard and three outboard BB segments. They also make use of the same reduced activation ferritic-martensitic steel as structural material, EUROFER97. Both designs are developed on the basis of the Single Module Segment (SMS) approach. Each

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segment consists of a long continuous EUROFER97 external box, composed by the first wall, the side walls, the bottom and top caps and the Back Plate (BP). Each segment is supported by a Back Supporting Structure (BSS), which connects the breeding blanket to the VV.

2.1 HCPB design

Historically the EU HCPB made use of Be as neutron multiplier and had the internals design based on Cooling Plates [5][6]. The reduced operational temperature of Be (<650°C in order to avoid excessive swelling and thus risking the pebble's integrity) poses a safety risk of excessive tritium inventory at the blanket End-of-Life due to the large tritium retention in Be (40%) at 600 °C. Also, the use of cooling plates results in the presence of numerous cooling channels, making problematic the reduction of pressure drops and negatively affecting the blanket reliability. Motivated by the will to tackle these issues, recent research activities have led to make use of $Be_{12}Ti$ as neutron multiplier in an enhanced design configuration based on single-module segments with a hexagonal arrangement of fuel-breeder pins [7][8][9] (see **Figure 1**). The use of $Be_{12}Ti$ allows operating at larger temperatures, mitigating tritium release issues. Moreover, the use of pins greatly simplifies the blanket internals and reduces the number of cooling channels about an order of magnitude, allowing a significant reduction of plant circulating power (about the half of the current sandwich design), as well as improving the blanket reliability. Analyses show the advanced maturity of this design [5][8][10], fulfilling basic nuclear, thermohydraulic and thermomechanical requirements.

The HCPB design is based on the use of Li_4SiO_4+35mol%Li_2TiO_3 pebbles as tritium breeder material, hexagonal prismatic blocks of $Be_{12}Ti$ as neutron multiplier and He (inlet 300°C, outlet 520°C, 8 MPa) as coolant. A low pressure purge gas (0.2 MPa) of He (carrier gas) with an addition of 0.1% wt. H_2 (doping agent) sweeps both the $Be_{12}Ti$ blocks and the ceramic breeder pebble beds (in series). The doping agent is of special importance, as it enhances the tritium desorption rate from the ceramic breeder keep a low inventory. Moreover, H_2 is the promoter of the isotopic exchange to form HT, which will be processed in the Tritium Extraction and Removal (TER) system (see sec 2.5).

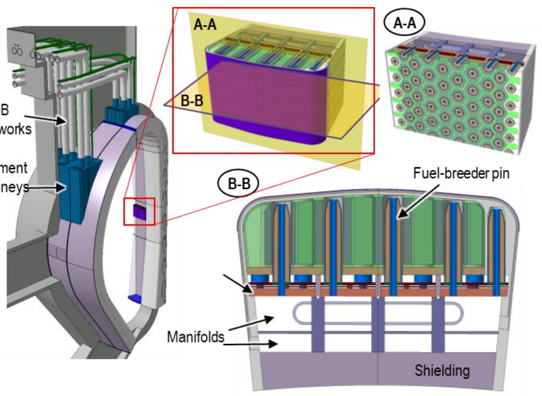

Figure 1: HCPB design with fuel-breeder pin hexagonal assembly.

2.2. WCLL design

The DEMO WCLL BB concept relies on the adoption of Lithium-Lead (PbLi) as breeder, neutron multiplier and tritium carrier, and water in Pressurised-Water-Reactor (PWR) conditions as coolant [11].

Each BB segment can be thought as a poloidal stack of elementary cells having a poloidal height of 135 mm (see **Figure 2**). The elementary cell consists of an actively cooled C-shaped plate (First Wall - FW and Side Walls - SWs) closed by two toroidal-radial (i.e. horizontal) Stiffening Plates (SPs) on top and bottom sides and by a back-plate in the rear zone. The internal part of the elementary cell, the Breeder Zone (BZ), is further divided in six channels by means of poloidal-radial (i.e. vertical) SPs. The region behind the back-plate houses the newly designed PbLi and water manifold system, where a series of spinal collectors allows the fluid distribution along the poloidal abscissa.

SW-FW system and BZ are cooled by two independent circuits, both adopting water in PWR conditions at the pressure of 15.5 MPa, with inlet and outlet temperatures of 295 °C and 328 °C, respectively. Square channels are envisaged for the SW-FW complex while Double-Walled Tubes (DWTs) are foreseen to be employed in the BZ. The 2019 WCLL-BB reference configuration foresees 22 horizontal DWTs, even though alternative layouts adopting both horizontal and vertical tubes, as well as the possibility of water recirculation within the BZ are being assessed to improve its cooling performances [12].

Concerning the thermo-mechanical behaviour of the WCLL-BB, particular effort has been put on the design of the Top and Bottom Caps. In fact, a dedicated design has been carried out for these regions, mainly consisting in the local and gradual thickening of the FW, from 25 mm up to 42 mm [13]. The 40 mm thick Caps are cooled by square channels very similar to those present in the FW.

Finally, the PbLi-water interaction as a consequence of a small in-box LOCA due to a DWT rupture is being deeply assessed. An experimental campaign has been carried out at ENEA Brasimone to fully characterize this phenomenon [14].

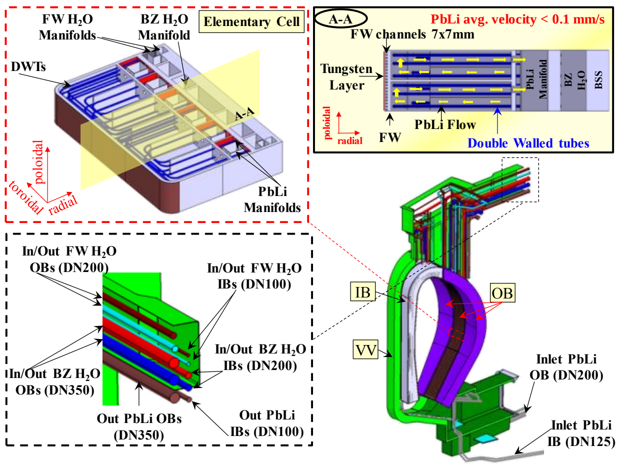

Figure 2: WCLL design. View of Outboard segment and the elementary cell.

2.3. Breeding Blanket shielding performance studies

The BB has not as primary function to be the main shielding barrier protecting Toroidal Field Coil (TFC) from nuclear irradiation. But being the largest and only nuclear component between the neutron source and the TF, it must contribute as much as possible to meet the nuclear heating requirement of the TF coils and the material activation limits of the Vacuum Vessel (VV). As a matter of fact an arrangement of efficient materials mitigating accumulation of nuclear damage in the VV can be considered as a design option improving the shielding performances of the BBs [15]. The primary function of the BB is to achieve a sufficient TBR: U. Fischer et al. set the TBR design target, which includes margins for calculation uncertainties and incomplete models, to TBR $\geq$ 1.15 [16]. HCPB can notoriously achieve higher TBR [16] while WCLL activate the VV at a lower level [17][18]. Neutronic studies to screen and assess shielding capabilities of BB configurations have been carried out to understand under which conditions an HCPB BB could reach similar shielding performance capabilities as the WCLL.

New 3D MCNP DEMO reactor models of both the HCPB and WCLL BBs have been prepared [8]: for the first time the same approach has been adopted to build step by step fully heterogeneous models of the HCPB and WCLL. This approach enables to assess plenty of nuclear responses including the large variety of geometry heterogeneous effects. Starting from the WCLL shielding performances computed by this model, a number of different possible candidates shielding materials have been included in the HCPB design with the aim of achieving similar level of nuclear damage accumulation in the inner shell of the VV as those of the WCLL, i.e. 0.013 dpa/FPY. A shortlist of realistic shielding materials have been considered; i) Boron carbide (commonly used in the nuclear technology) fair moderator and absorber of fast neutrons; ii) WC not as efficient as hydrates as neutron moderator but very efficient absorber of low energy neutrons, but with very high material density (i.e. high weight), and high nuclear power generation; iii) Y/Zr/Li/Ti-hydrates: dissociation takes place in all hydrates at different temperatures but hydrogen is by far the most efficient moderator and those hydrates mimic or even are superior to water in terms of hydrogen density. In hydrates, the presence of hydrogen obliges to use them in the BSS only enclosed into steel claddings; but because of the relatively low specific nuclear heat generation, shield blocks made of hydrates could be attached to the back wall of the BB (assuming an effective radiation heat exchange in place with the VV).

If the shield blocks are integrated inside the BSS of the HCPB, the most efficient shielding materials are 18cm inserts of $ZrH_{1.6}$, TiH_2 and WB, and they can provide shielding performance comparable to the WCLL ones: 0.015 dpa/FPY for $ZrH_{1.6}$, 0.014 dpa/FPY for TiH_2, 0.019 dpa/FPY for WB. Reducing the shielding thickness down to 12cm the results are not acceptable. If those 18cm shield block are attached to the back wall of the HCPB blanket, $ZrH_{1.6}$, WB and TiH_2 can potentially provide less dpa accumulation in the VV compared to the WCLL blanket: 0.010 dpa/FPY for $ZrH_{1.6}$,0.008 dpa/FPY for TiH_2, 0.012 dpa/FPY for WB. Integrating the shield inside the BSS provides a worse protection due to the neutron streaming through the gaps between blankets and manifolds in the BSS.

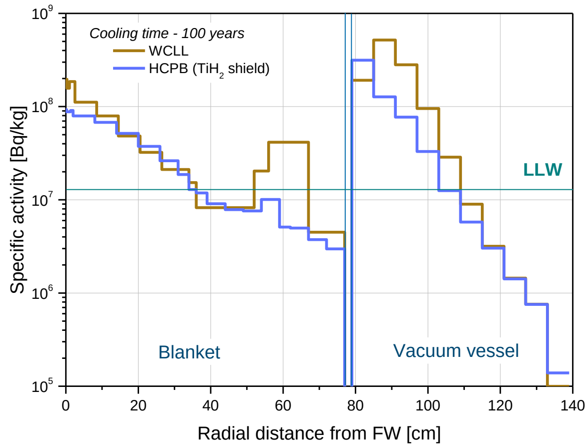

Figure 3: Radial profiles of the specific activities of the steel in the HCPB with the 18cm TiH_2 shield and WCLL DEMOs

Activation calculations were performed for the 18cm TiH_2 shield and showed a reduction of the specific activity of the SS-316L(N)-IG steel in the VV also compared to the WCLL design, see **Figure 3**. Nevertheless, the additional shield insert cannot fully prevent the accumulation of the activity in the VV above LLW level (referring to the current UK limits). The dominant nuclide 100 years after the reactor shutdown is ^{63}Ni (~98%) produced mainly in $^{62}Ni(n,\gamma)^{63}Ni$ reaction due to the neutron intensive slowing down in the VV.

Integration of shielding structure up to 18cm thick implies a redesign of the BSS manifolds which has not been done in this context. But preliminary numbers show that correct coolant distribution is feasible and the pressure drop would remain acceptable.

2.4. Water activation

Nowadays, in the design of DEMO fusion reactor and in particular of the BB, it is common practice to take into account, since the early stage, all the safety aspects related to both normal operation and accidental cases. For instance, a particular issue can be represented by mobilized water activated products (e.g. ^{3}H, ^{19}O, ^{16}N and ^{17}N) that are transported within the primary heat transfer system (PHTS) in WCLL BB. In particular, the nitrogen isotopes represent a distributed photon and neutron source that, also under normal operation, can jeopardize safety equipment like valves. For this reason a dedicated procedure has been developed for the assessment of the spatial distribution of dose rates, due to the decay of ^{16}N and ^{17}N produced by coolant activation, around some key components of WCLL BB cooling circuit and/or nearby their locations.

To this purpose, a coupled neutronic/fluid-dynamic problem is solved following a theoretical-numerical approach and adopting an integrated computational tool mainly relying on the use of MCNP6 and ANSYS CFX codes [19]. The operative procedure adopted foresees the assessment of spatial distribution of nitrogen production rate within FW and BZ BB cooling channels and tubes by means of complete heterogeneous neutronic models. A fully 3-D approach is, then, used to compute the nitrogen concentration within the In-Vessel complex flow domain, while a 1-D lumped parameters approach is adopted to calculate the nitrogen concentration spatial distribution along the Ex-Vessel BB PHTS [20]. Once the mapping of the radioactive nitrogen isotopes inside the PHTSs is completed, dedicated neutronic and photonic transport analyses are foreseen to assess dose rates in the target locations [21].

The spatial distribution of these dose rates has been assessed in the DEMO Upper Pipe Chase (UPC), focusing the attention on the space neighboring a typical isolation valve of the PHTS. To this end, a computational approach has been followed adopting MCNP5 Monte Carlo code. In particular, a totally heterogeneous neutronic model of a portion of the UPC has been set up, including the valve and the main FW and BZ PHTS piping, and the spatial distribution of nitrogen isotopes concentrations, previously assessed, have been used to model the photonic and neutronic sources. In **Figure 4**, the dose for 7 fpy in the UPC due to the γ ^{16}N decay is reported as example. The full-scale of 2 MGy has been selected as it represents the end of life dose value for the typology of valves taken into account.

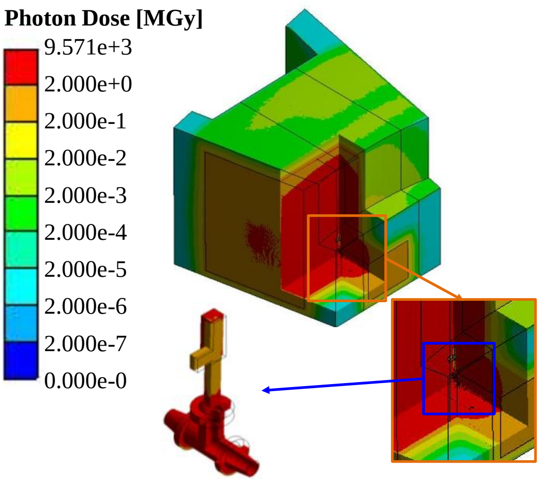

Figure 4: Dose due to γ ^{16}N decay in with completely heterogeneous UPC equipped with valve.

Preliminary results on DEMO PHTS seem coherent and promising therefore, it is assumed that the application of the method could provide useful information for the development of the WCLL BB design. Further activities are foreseen to minimize the dose to the valve introducing design modifications both in the BB design and PHTS layout.

2.5. TER HCPB design

In the case of tritium extraction from the solid breeders, the reference technology is based on tritium transfer from the solid breeder into a purge gas stream followed by tritium recovery from the purge gas. In the HCPB the breeding zone containing the pebbles beds is purged with helium, in a closed loop, and the current reference technology for tritium trapping from the purge gas includes cryogenic trapping. The helium purge contains 0.1 mol. % H2/ D2 at a rate of 10,000 Nm3h-1. The backup solution considered for the HCPB TER relies on the zeolite membrane continuous approach.

Parameter [content]	**Estimated mass flow [g .day^{-1}]**	**Estimated flow [mol .day^{-1}]**	**Mol Fraction [%] with He**	**Mol Fraction [%] w/o He**
He	4.21 x 10^{7}	1.05 x 10^{7}	99.9	N/A
HT/DT	309	76.8	7.3 x 10^{-4}	0.728
HTO/DTO	55.8	2.79	2.65 x 10^{-5}	2.64 x 10^{-2}
H_2O/D_2O	6.60 x 10^{3}	367	3.48 x 10^{-4}	3.47
H_2/D_2	2.04 x 10^{4}	1.01 x 10^{4}	0.096	95.8

Table 1: The reference composition of the output gas stream from the HCPB BB.

The output stream expected from a HCPB BB is displayed in Table 1: after purging the BB, the stream is diverted through the TER to remove the tritium and to recycle helium back to the BB. The current TER design features a reactive and a cryogenic molecular sieve beds (RMSB and CMSB respectively). The gas stream is first directed towards one of two RMSB (one bed is absorbing while the other is regenerated to maintain an almost continuous flow). It is expected that the entire Q_2O content (6.1kg/day) is retained by the RMSB and the remaining (dry) gas stream is cooled before it enters one of two CMSBs. After 24 hours the RMSB's absorbtion capacity is expected to be met. A H_2/D_2 stream is used to flush the RMSB loop while it is heated to promote isotopic exchange. In the CMSB, 10Nm^{3}h^{-1} of Q_2 is trapped over a 5 hour absorption period. In order to release this gas, a subatmospheric pressure is applied and the vessel is heated. The helium stream is returned to the purge loop and the extracted Q_2 (including 5-10mol% helium) is sent to the tritium plant over a 3.5 hour period at 14.29m^{3}h^{-1}. Thus the TER design results in multiple streams being sent to the tritium plant [22]: i) Q_2O absorbed by the RMSB can be directly sent to the DEMO WDS for processing: 6.1kg/d of water will contain 6.7g/d of tritium; ii) D_2 (with trace tritium) following the regeneration of the RMSB can be sent for isotope separation, where tritium and hydrogen/deuterium are extracted for reinjection into the fuel cycle (the tritium plant can accept flows up to 25Nm^{3}h^{-1}); iii) Q_2 from the regeneration of the CMSB is expected to contain low levels (5-10 mol%) of helium that must be separated before the Q_2 can be returned via exhaust processing to the isotope rebalancing loop in the fuel cycle [23].

The preliminary evaluations concerning the tritium distribution on molecular species at the outlet of the BB shows that HTO is almost 20 times lower compared with HT form. There is a real benefit as far as tritium permeation in the cooling gas is concerned to minimize the HT tritium form and to increase the HTO form in the purge gas. Therefore, helium purge gas containing steam may be an option in view of mitigating the tritium permeation in the BB cooling stream. Preliminary evaluations have been carried out in view of identifying technologies for tritium removing from purge gas with increased content in HTO. For tritium recovery from the helium loop containing steam two steps would be required (removal of the tritiated water followed by tritium recovery from the tritiated water) and four technologies could be considered: for removing the tritiated steam the adsorption on the RMSB and the steam condensation; for tritium recovery from tritiated water electrolysis and PERMCAT processes. A selection of the most promising combination of technologies will be done in coordination with the developers of the Tritium and Fuelling Cycles and, among others, should take into account the energy consumption, the amount of generated tritiated water and the load for the Tritium Plant. The increase of the amount of steam in the purge gas should take into account the allowable limits that avoid intolerable corrosion of the BB components: indeed historically water has been excluded as a doping agent because of its incompatibility with pure Be and the associated safety issues and the increased corrosion of Reduced-Activation Ferritic Martensitic steels.

2.5. TER WCLL design

Two technologies are prioritized in EU as candidates for the WCLL TER: the Permeator Against Vacuum (PAV) and the Gas Liquid Contactor (GLC), the latter being the technology used in the EU ITER TBM programme [24]. Vacuum Sieve Tray (VST) technology is not anymore considered as a realistic short term option for EU DEMO as recent works (to be published) have shown unrealistic size and burdens on the TER plant when small scale experiments are extrapolated to DEMO size. The PAV allows to ideally separate in one step the tritium from the PbLi: routing the PbLi into specially designed components made by thin permeable material to Q_2 and facing a vacuum environment. The tritium concentration gradient forces the tritium to permeate and be available as pure Q_2 on the vacuum side. Technological difficulties have to be solved for the engineering design of components including those of thin permeable barriers (made by V or Nb), but also the sometimes very large uncertainties in few key properties of PbLi (like the Sievert constant) could lead to unrealistic size of the component [25]. Therefore an aggressive R&D programme is in place and aims at providing experiments results of mock-ups based on the PAV technology by mid-end 2020 [26]. A layout of the TER loop based on PAV technology for the WCLL DEMO has been completely, including the sizing of the main components, and has been made available for integration into the tokamak complex. Such TER would have a PbLi mass flow of 956kg/s and would be split in 6 loops: IB loops are served by 55kg/s, OB loops by 264kg/s. A daily production of tritium of 320gr (assuming a full power day at continuous 2GW fusion power burn and with a TBR of 1.05) would lead to a tritium pressure of 20-200Pa into the PbLi. In order to increase the TER efficiency the PbLi is heated up to 500°C, even if inlet/outlet temperature of the PbLi in the BB is 326°C. It is assumed that TER efficiency needs to be as high as 80-90%. to minimize the tritium inventory in the PbLi loop since TER is placed in the outlet of the BB segments. PAV using V membrane is designed by CIEMAT and for the WCLL DEMO TER would consist of single rectangular permeators (1.4x1.2m front section x10m) in which 148kg/s of PbLi flows at 1-4cm/s and into which layers of V membranes would provide a contact area of 2800m^2 with PbLi requiring 7m^3 of volume for the vacuum section. PAV using Nb is designed by ENEA in the form of cylindrical permeators with a shell-tube configuration (diameter 4m and height 7m). PbLi is on the U-shape tube side and vacuum is on the shell side. PbLi velocity would be 0.5m/s with 250kg/s manageable per cylinder.

The back solution is based on GLC technology, where a gas and a liquid phase are brought into contact for a diffusion interchange. Tritium moves from liquid phase to gas phase thanks to its low solubility in PbLi.The TER concept based on GLC technology was qualified in TRIEX-II (TRItium EXtraction) facility at C.R. ENEA Brasimone in 2018-2019, instead a mock-up of PAV manufactured in V will be characterized in CLIPPER (CIEMAT lithium-lead permeation experiment) facility 2020 and a mock-up of PAV with Nb membrane will be qualified in TRIEX-II in 2020.

2.5.1 R&D in support of the PAV technology

A new facility, CLIPPER has been constructed to investigate tritium extraction from PbLi, see Figure 5 [27]. In particular, the PAV technology will be tested. The installation consists of a forced circulation loop operating up to 500 ℃, but designed to cover a wide operational range to fully characterize the PAV with enough flexibility to be representative for any PbLi blanket concept. One of the peculiarities of CLIPPER is that the liquid metal is heated, melted and cleaned outside the loop, in a tank integrated in a dedicated glove box under argon atmosphere. This allows the cleaning and preinspection of the melted PbLi previously to be injected in the closed circuit. PbLi inventory in CLIPPER is around 25 lt. One of the main components of the loop is a permanent magnets pump which moves the PbLi and is capable of operating (short time periods) up to 550 ℃. It provides a mass flow range between 2 and 39 kg/s.

At a laboratory scale, hydrogen isotopes need to be solubilized in the liquid PbLi at a known concentration since it is not possible to reproduce the chemical reactions occurring in a real fusion reactor. This task is accomplished by a novel gas injection system specifically designed for CLIPPER. This system, based on a multi-tube component with permeable niobium membranes, is able to solubilize the hydrogen in the PbLi up to the required concentration.

Figure 5: CLIPPER facility and glovebox for PbLi melting.

The test section includes a prototype of permeator (TRITON) and its auxiliary systems, such as its associated vacuum system and instrumentation. TRITON is a rectangular multi-channel permeator with vanadium membranes and SS structure. The prototype assembly was completed and first tests in gas phase were performed. However, due to the complexity of the manufacturing process, and in particular in the welding V-SS, important leakages have been found. For that reason it was decided to produce a new design, for which two new approaches are being considered. The first one is the use of electron-beam welding, since preliminary tests have shown a good performance for V. The second one, and trying to avoid a new leakage problem, is the use of special graphite gaskets instead of welding.

The activities in CLIPPER are planned as follows: in a first phase the experiments will be oriented just to demonstrate the technique. For that reason the permeator will operate in Diffusion Limited Regime (that is, at high H/D partial pressures), high temperature (up to 550 ºC) and a reduced mass flow rate. This allows the most favourable scenario for permeation. A second phase will be concentrated on a test matrix which covers most of BB parameters, firstly focused on WCLL conditions. In parallel the validation of codes will be produced, including a new development in Ecosimpro needed for intermediate regimes.

In parallel a mock-up using a Nb membrane PAV in a cylindrical tube-shell-like configuration is being designed to be tested at ENEA Brasimone in TRIEX_II facility (see **figure 6**).

Figure 6: TRIEX-II facility with installed GLC test section.

The experimental facility TRIEX-II, designed and installed in 2018, is used to test the extraction efficiency of a mock-up of TER based on the packed column technology (GLC) or PAV.

The facility is a ring isothermal loop (see **figure 7**) with a by-pass for mass flow regulation that can operate in the range of temperature between 300 and 530°C. 200lt of PbLi were charged in the facility with a separate melting furnace under inert atmosphere. The Liquid metal is pumped in the facility with a Permanent magnet pump that, with the support of two regulation valves, can allow the control the PbLi mass flow rate in the range between 0.1 and 5kg/s. The Q_2 is solubilised inside the PbLi with a GLC saturator S200, the Q_2 is injected through a gas injector, in a mixture with helium. The continuously injected gas in the tank leaves from the top with appropriate flow control so as to adjust the pressure inside the tank and consequently the level of PbLi.. A Thermal mass flow allow to measure the liquid metal mas flow rate, while the pressure drops is measured with dedicated relative pressure meter. The tritium concentration in the liquid metal is measured with the support of three hydrogen permeation sensors, calibrated in a dedicated facility. The chemical composition of the tritium saturation/extraction stripping gas were analysed with a mass spectrometer able to detect the Deuterium concentration in helium stripping gas. The liquid metal is stored in a separate tank S100, isolated with an automatic actuated valve. The facility has completed in 2019 the characterization of GLC test section at the TBM operative conditions.

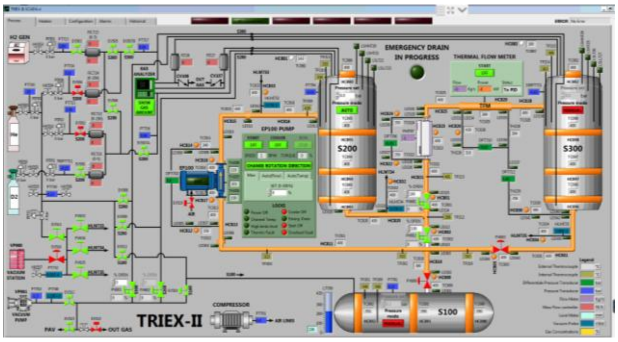

Figure 7: TRIEX-II DACS, the main component are the Saturator (S200), tritium extractor (S300), storage tank (S100) and the pumping system.

2.5.2 GLC technology

In GLCs, a gas and a liquid phase are brought into contact to promote diffusion interchange between them. In standard GLC applications packed vertical columns are filled with a medium providing a large interfacial surface between liquid and gas phase in both counter-current and co-current flow. In GLC applied to the WCLL, an helium stripping gas extracts tritium/hydrogen when forced in counter-current through the PbLi flow. At the bottom of each column He enters with addition of H_2; at the outlet the stripping gas will contain T_2, HT and H_2. Also in case of GCL a dimensioning of the WCLL TER has been carried out.

Two different kind of GLC units are designed for the IB (50kg/s) and OB (250kg/s) PbLi loops. In both cases 5 units are needed. In the IB loop each unit would be 1,5m high and 1,5m in diameter or 0.8m high and 2m in diameter, with a stripping gas flow of 1100Nl/h. Of the OB loops the diameter is larger and GLC could be as total high as 7,9m and 1,5m in diameter or 4,3m and 2m in diameter, with a stripping gas flow of 5800Nl/h. The Q_2 extracted by the stripping gas has to be sent to a second stage of TER similar to the one of the HCPB BB before being routed to the tritium processing system.

GLC mock-ups have been qualified in TRIEX-II using helium with hydrogen as tripping gas. The experimental campaign proved a tritium extraction efficiency up to 44%.

3. Tritium transport and permeation

3.1. Tritium transport analyses

Tritium transport analyses at system level are carried out in EU in support of the TBM and of the DEMO BB development programs by EcosimPro [18]. In particular recent efforts in EcosimPro development have focused on the improvement of the models of the two candidate driver blankets and on the integration of the ancillary systems impacting the tritium transport and inventories [28][29]. Results from the last update of the HCPB and WCLL models including the BoP are summarized in Table 2 and Table 3. Great attention is paid to the estimated tritium permeation rate from the BB breeding zone into the coolant, as this impact strongly the tritium management strategy and the feasibility of the coolant purification. The system level results will have to be incorporated in the ongoing efforts to describe the whole fuel cycle at plant level [30].

The tritium permeation from the breeding zone of the BB into the primary coolant loops can be mitigated by using dedicated coatings (i.e. anti-permeation barriers and/or natural oxide layers) and/or by processing a fraction of the primary coolant flow rate inside the Coolant Purification System (CPS). The efficiency of a coating in reducing the Q_2 permeation can be expressed by the Permeation Reduction Factor (PRF) indicating the ratio of Q_2 permeation flux between bare and coated material. The CPS uses to be characterized by the coolant fraction routed through the CPS (α_{CPS}) and its efficiency in removing the tritium (η_{CPS}). According to last EcosimPro results the tritium permeation rate into primary coolant, for the case of $PRF_{BB}=1$, is largely different for HCPB, where it corresponds to the 0.035-0.35% (i.e. ~0.10-1.10 g d^{-1}) of the tritium generated, compared to the WCLL, where is amounts to the 11.71% (i.e. ~38 g d^{-1}). In the water coolant the isotopic exchange reaction between HT and H_2O will trap most of the permeated tritium as HTO: the gas phase HT partial pressure remains always very low promoting the permeation from the blanket. Consequently, the WCLL will need anti-permeation barriers into the blanket region to reduce the tritium permeation rate. On the other hand the water represents a “sink” for tritium, from which it cannot escape in normal operation into working area or into external environment. In HCPB the very large helium purge gas flow promotes a low Q_2 pressure difference between blanket and coolant reducing the permeation. A CPS unit will be therefore necessary for both, water and helium coolant, to avoid the continuous increase of tritium inventory into the coolant.

Input	Value
Tritium generation rate, g/d	320
Purge gas mass flow rate, kg/s	4.97×10^{-1}
H_2 addition in purge gas, % wt.	0.1
TES efficiency, %	80
Coolant mass flow rate, kg/s	1841
H_2 addition in coolant, % vol.	0.1
CPS efficiency, %	95
CPS bypass, %	0.083

Table 2: Main input for the HCPB simulation.

Input	Value
Tritium generation rate, g/d	320
TES efficiency, %	80
Coolant mass flow rate, kg/s	5330
H_2 amount in coolant, ppm	8
CPS efficiency, %	90
CPS bypass, %	0.01

Table 3: Main input for the WCLL simulation.

The efforts in the conceptual development of anti-permeation barriers and CPS have focused on the identification of viable technologies in particular in terms of PRF and α_{CPS} that realistic solutions could guarantee.

3.2. Experiments in support of tritium modelling

It is of vital importance to support the development of the tritium transport-modelling tool by a number of experiments dedicated to validate the assumption on parameters used to simulate Q_2 species permeation. Despite the codes/phenomena are largely validated, a number of phenomena need indeed to be fully understood and quantified, like the co-permeation of Q_2 species. A number of experimental set-ups is being developed at KIT (for the actual experiments) and CIEMAT (for the numerical support) to include various measuring cells designed to measure the permeation parameters that characterise WCLL and HCPB blankets [31].

3.2.1 Experimental rig relevant for the HCPB

The experimental rig is meant to simulate the tritium permeation from purge gas to cooling gas and will be based on co-axial cylinders to be operated with Helium and tritium on one side and Helium with hydrogen/deuterium at various partial pressure on the other side (at pressures and temperatures relevant for DEMO permeation case). Great care is paid to incorporate enough sampling points and the required instrumentation in order to quantify the dynamic of the tritium permeation from one side to another. The measuring cell is modular and designed to allow easy dismantling: this will allow the characterisation of the permeation surface that is believed to influence the dissociation and recombination coefficients. The plan is to perform long operation with hydrogen and deuterium in order to understand the dynamic of the surface and to give references concerning the possibilities for its chemical treatment. The dimensioning of the permeation cell has been made using EcosimPro: this allowed to define the most suitable thickness of the permeation plate, dimensions of the cylinder, amount of tritium etc, and to develop the optimal experimental procedure in order to measure the parameters needed for the calibration of the modelling tool.

3.2.2 Experimental rig relevant for the WCLL

For the WCLL two different set-ups are investigated to measure the tritium permeation from the PbLi to the primary coolant and from the primary coolant to the secondary coolant.

For the WCLL permeation case in a coaxial configuration the use of Helium containing tritium on one side and water on the other side will provide valuable information; the atomic tritium permeation can be indeed investigated using a tritium partial pressure as high as the one expected in the PbLi on the Helium side and avoid the use of PbLi. Currently three experimental rigs with volumes of the order of ~10ml are being tested for the Pbli to water permeation case. The procedure consists in feeding the inner Helium gas volume with 1,20E+10 Bq of tritium and the outer volume with 6 g of demineralized water; after keeping the three cells for 10 days at 350^0 C the volumes are sampled measuring the tritium amount by Liquid Scintillation.

Additionally measurements having tritiated water on one side and non tritiated water on the other side will contribute to characterise the tritium permeation dynamic. The measurements shall provide information on possible tritiated water dissociation on one side, followed by tritium atomic permeation through the separation plate and the isotopic exchange with water on the opposite face. The water is to be sampled at certain time intervals in order to provide information on the dynamic of the process. The experiments are on-going and will be benchmarked against the tritium permeation predicted by the code.

3.3. Anti-permeation coatings

In liquid metal blankets coatings of inner metallic surfaces have the functions to mitigate the corrosion of the structural material and the tritium permeation through the structural material. Al_2O_3 has proven to be a suitable reference coating material to fulfill those functions and three techniques have been identified to create dense and adherent Al_2O_3 coatings onto ferritic martensitic steel like the EUROFER97: the Pulsed Layer Deposition (PLD), the Atomic Layer Deposition (ALD) and the electrodeposition of aluminium from an ionic liquid (ECX). Along with high adhesive strength the capability to cover complex 3D shapes remains to be proven at relevant scale.

Several H_2 permeation tests have been carried out at different temperatures on bare EUROFER97 disks and disks coated by PLD, ALD and ECX techniques. Performances are assessed in terms of PRF. Measurements show that H_2 flux permeating EUROFER97 samples is reduced with higher PRF when the PLD background pressure is lower (i.e. 0.1Pa vs. standard pressure of 2-100Pa). The PLD coatings appear to be dense, homogeneous and without open porosity with a thickness at least of 2μm, while reducing the thickness the open porosity under irradiation increases [32]. Preliminary results show that the anti-permeation capability of ALD films is even better than the one measured with PLD samples [33].

Efforts to measure PRF performances of samples loaded by H_2 before and after thermal cycling and ion irradiation have been carried out. Thermal cycling has not impact on coating anti-permeation capabilities, instead the ion irradiation seems to increase the open porosity size on the coating and therefore increase the permeation flux, even if the PRF still remain higher than 1000 [34]. The relevancy of ion irradiation test for assessing coatings performances under a relevant environment remains to be proven. Coatings will be tested in 2019 in CVR reactor under neutron flux in PbLi in order to check their capability to resist the harsh environment of the breeding blanket portions. To that purpose and optimization of PLD coating was performed to be qualified in LVR-15 reactor [35].

3.4. CPS design

The main function of CPS is to extract Q_2. Furthermore, together with BOP systems, it shall also remove solid, liquid or gaseous impurities, being a requirement on BB and BoP to minimize the formation of electrochemical and corrosion products. Based on the tritium mass balance as computed by the tritium dynamic simulations, a range of desired η_{CPS} and α_{CPS} are identified for the CPS. Based on those values, suitable technologies and process are identified and the first rough dimensioning to the DEMO scale CPS is carried out.

3.4.1 Helium CPS

Assuming an allowable HT partial pressure in the Helium coolant of 4×10^{-3}-4×10^{-2} Pa [36] and a realistically achievable CPS efficiency of η_{CPS}=0.9, the permeation rate calculated as per Table 2 leads to a flow rate inside the CPS of about 3 kg s^{-1}. With such a mass flow to be processed the ITER CPS technologies proposed for the EU HCPB TBM could be extrapolated to the DEMO CPS size (see **Figure 8**). This system relies on regenerable getter beds (CuO beds for oxidation, zeolite molecular sieves for Q_2O removal and metallic bed for Q_2O reduction). Nevertheless the difficulties of the metal oxide and of the reducing beds regeneration processes have motivated the investigation of the possible use of Non-Evaporable Getter (NEG) materials (see **Figure 9**). Use of NEG alloys is motivated by the possibility to avoid the formation of tritiated water that has to be necessarily treated afterwards. NEG indeed sorb by chemical reactions most of active gas molecules with particular efficacy for hydrogen isotopes. As described in [37][38] ZAO alloy (Zr-V-Ti-Al) exhibits the best properties at the foreseen operational temperature of 573K and represents the most promising material to be used in Helium CPS.

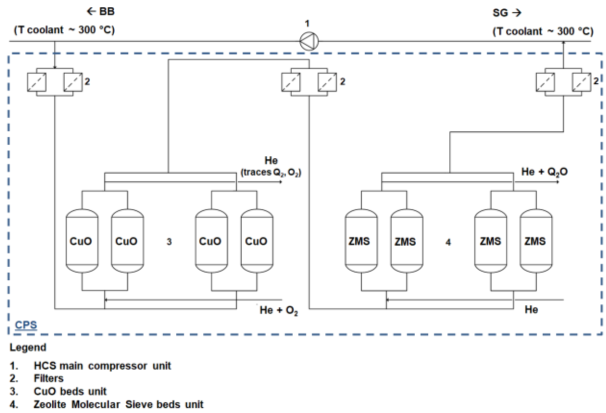

Figure 8: Helium CPS pre-conceptual design based on the ITER scale-up process.

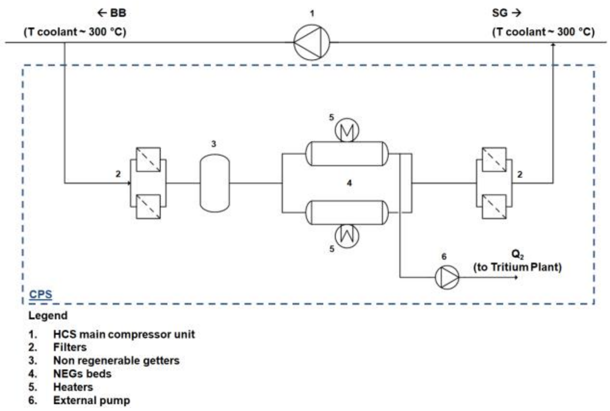

Figure 9: Helium CPS pre-conceptual design based on the use of NEGs technology.

A pre-conceptual layout for the HCPB CPS has been prepared based on both technological options: as showed in Figure 8 and Figure 9 the CPS unit is located downstream the steam generators along the Helium cold leg where the Helium temperature of ~573 K maximizes the efficiency of the selected technologies. The CPS dimensioning is strongly influenced by the H_2 content in Helium coolant. The pre-conceptual dimensioning of the He CPS components for both technological choices has been carried out for three different H_2 contents: 80 Pa, 300 Pa and 1000 Pa [37]. The obtained dimensions of CuO and ZMS beds on the one hand (ITER-like solution) and of the ZAO alloy on the other hand confirm the technological feasibility of the Helium CPS of both technologies for an Helium by-pass flow rate of 3 kg s^{-1}. As showed in [39] the H_2 content impacts strongly the regeneration time of both technologies and the amount of the required getter material. In particular it is stressed that the H_2 addition into the He coolant should not exceed the value of 300 Pa. Despite the NEG solution being very attractive with several advantages, its capability to work at the expected efficiency under relevant CPS conditions remains to be proven.

3.4.2 Water CPS

In water coolant the allowable tritium concentration has been assumed to be of 1.85×10^{11} Bq kg $^{-1}$ as in CANDU experience [40]. Achievable CPS efficiency are assumed to be of η_{CPS}=0.9. Considering the permeation rate as per Table 2 the coating of the blanket inner volume is required or the water flow rate to be processed in the CPS for keeping the tritium concentration below 1.85×10^{11} Bq kg^{-1} would be thousands of kg h^{-1}. As of today technologies able to handle such mass flow do not exist and do not seem to be energetically viable.

Water detritiation facilities are based on two stage processes: a front-end process to transfer the tritium from the aqueous into an hydrogen gas stream and a back-end process for the separation of the Q_2 forms. The back-end process relies on cryogenic distillation. Depending on the flow rate to be managed, existing facilities adopt different front-end processes: JET WDS with 3.7-5.6 kg h^{-1}, uses Direct Electrolysis; ITER WDS with 20 kg h^{-1} uses Combined Electrolysis and Catalytic Exchange, Wolsong with 100 kg h^{-1} uses Liquid Phase Catalytic Exchange and Darlington with 360 kg h^{-1} uses Vapor Phase Catalytic Exchange. For the DEMO case a parametric scan has been made to check if any viable technological solution might be used as a function of the achievable PRF and the mass flow to be processed by the CPS. The results (see Figure 10) show that: without anti-permeation barriers (i.e. PRF_{BB}=1) no existing water detritiation technology enables processing the amount of water required to maintain the tritium concentration below the threshold; considering a PRF_{BB}=10 the water coolant mass flow in the CPS would be ~100 kg h^{-1} (Wolsong-like); assuming a PRF_{BB}=100 the water coolant mass flow in the CPS would be ~20 kg h^{-1} (ITER-like).

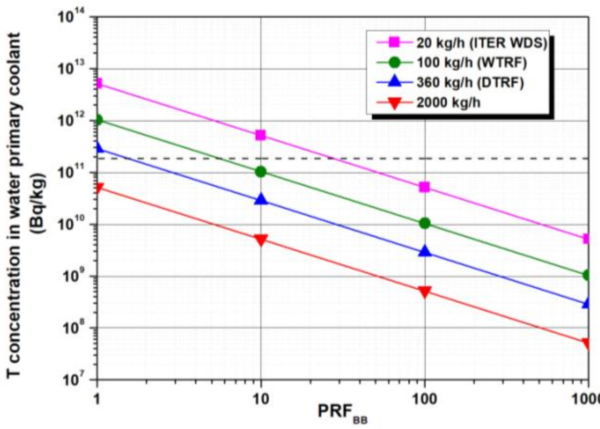

Figure 10: Tritium concentration in water primary coolant vs. PRF_{BB} by assuming several CPS size corresponding to some existing tritium removal facilities.

An alternative being considered in DEMO CPS conceptual design (in line with CANDU approach) is to process, after a certain time of DEMO operation, the primary water coolant off-line in an external facility (CPS off-line). For example considering a PRF_{BB}=100 the tritium concentration in the coolant after one year of operation would be ~1.62×10^{11} Bq kg^{-1} and the water inventory (300 m^3) could be completely replaced by fresh water discharging the tritiated water with 35 kg h^{-1} to an ITER-like WDS.

4. Conclusions

HCPB and WCLL BB are the two candidate driver BB concepts for the current EU DEMO. Both concepts have significantly evolved to improve their functional performances, solve some potential killing issues and improve their interfaces with other systems. The HCPB in particular is now based on fuel-breeder pins assemblies making use of $Be_{12}Ti$ as neutron multiplier. A number of studies have been carried out to gain awareness on issues that are BB design dependent but may have a strong impact on interfacing systems and ultimately on the tokamak layout. At BB level potential solution to improve the poor shielding performances of helium cooled BB have been studied: 18cm blocks of hydrates like $ZrH_{1.6}$ or TiH_2 can potentially lead to less dpa accumulation in the VV compared to the WCLL blanket. For the water cooled BB the decay of ^{16}N and ^{17}N produced by coolant activation has been studied developing a dedicated procedure by coupling neutronic/fluid-dynamic tools in an integrated approach (relying on MCNP6 and ANSYS CFX). Cumulated doses after 7 fpy in the UPC due to the γ ^{16}N decay have been calculated referring to the current end of life dose of 2 MGy of valves used in PHTS. The results obtained are coherent and promising and the efforts are now focused on minimizing the dose to the valve introducing design modifications both in the BB design and PHTS layout.

At BB system level, i.e. BB segments and TER, the current HCPB TER relies on consolidated technological choices and features a two-step separation based on reactive and a cryogenic molecular sieve beds (RMSB and CMSB respectively). Focus point of the on-going TER developments is the increase of the amount of steam in the purge gas up to the allowable limits that avoid intolerable corrosion of the BB components. The WCLL TER features PAV as prime candidate and GLC as second solution, the latter being the reference choice for the EU TBM program. The PAV would allow a less invasive tritium extraction as it relies on the permeation of Q_2 from flowing PbLi against vacuum through thin V or Nb permeable barriers. Impressive R&D efforts are deployed in CIEMAT and ENEA to provide experimental evidence, extrapolable at DEMO scale, of the PAV technology by early 2020.

The tritium balance is not only driven by the performances of the TER system, but we must rely on computational tools for the dynamic tritium inventory simulation and control. In EU present reference values of tritium inventories and permeation are provided by EcosimPro. The code development is supported by a number of dedicated small-scale experiments aiming at quantifying some key parameters that may largely affect the estimated permeation fluxes: in particular two dedicated experimental rigs at TLK aim at quantifying co-permeation of Q_2 species in He/He or He/water set-ups. According to Ecosimpro last results the tritium permeation rate into primary coolant, for the case of PRF_{BB}=1, is ~0.10-1.10 g d^{-1} for HCPB and ~38 g d^{-1} for the WCLL (where the isotopic exchange reaction between HT and H_2O trap most of the permeated tritium as HTO). The tritium permeation can be mitigated by using dedicated coatings (i.e. anti-permeation barriers and/or natural oxide layers) and/or by processing a fraction of the primary coolant flow rate inside the Coolant Purification System (CPS). In case of WCLL BB the combined use of coatings and coolant purification seems mandatory. On-going R&D efforts aim at characterizing the two candidate coatings, PLD and ALD, in terms of fabrication and capability to coat complex 3D structures, and performances under relevant operational conditions (i.e. under neutron irradiation). Assuming for the water coolant an allowable tritium concentration to be kept by the CPS as the one used in CANDU reactors, i.e. 1.85×10^{11} $Bqkg^{-1}$, and considering a PRF_{BB}=10, the water coolant mass flow in a Wolsong-like CPS would be ~100 kg h^{-1}; assuming a PRF_{BB}=100 the water coolant mass flow in a ITER-like CPS would be ~20 kg h^{-1}. A combination of anti-permeation coatings and CPS allows then to rely on existing technology for the water detritiation. In case of helium coolant, without anti-permeation coating (PRF=1), a flow rate inside the CPS of about 3 kg s^{-1} would have to be processed and ITER CPS technologies proposed for the EU HCPB TBM could be extrapolated to the DEMO CPS size. Nevertheless the difficulties in the use of regenerable getter have motivated the investigation of the possible use of Non-Evaporable Getter (NEG) materials, which is the current focus of CPS development for He cooled concepts.

Acknowledgement

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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