Fusion Engineering and Design 141 (2019) 30–42

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

An overview of the EU breeding blanket design strategy as an integral part of the DEMO design effort

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ARTICLE INFO

Keywords:
DEMO
Power plant
Breeding blanket
Design integration
Tritium

ABSTRACT

This paper provides an overview of the newly revised design and development strategy for the DEMO breeding blanket in Europe. This has been defined to take into account the input from the DEMO pre-conceptual design activities, the findings and recommendations of a thorough technical and programmatic assessment of the breeding blanket and the EU ITER Test Blanket Module (TBM) programs, conducted in 2017 by an independent expert panel. This work has led to the identification of (i) the most mature and technically sound breeding blanket concepts to be used as "driver" and "advanced" breeding blankets in DEMO, the latter to be installed and tested in a limited number of properly designed segments, potentially being more attractive for future fusion power plants; (ii) the remaining technical gaps and R&D priorities. A number of urgent steps that are required to better align and strengthen the EU TBM and DEMO Breeding Blanket Program as a whole and to aim at an efficient implementation of the work are described in this paper. These include a proposal to change the EU TBM options to be tested in ITER in order to obtain important and useful information from the two current breeders (solid and liquid) and coolants (helium and water) considered for DEMO.

1. Introduction

As an important part of the Roadmap to Fusion Electricity [1], Europe is conducting a pre-conceptual design study of a Demonstration fusion power plant (DEMO) to come in operation around the middle of this century. The main aims are to demonstrate the production of few hundred MWs of net electricity and to demonstrate feasibility of operation with a closed-tritium fuel cycle, including maintenance systems capable of achieving appropriate plant availability [2]. This is currently viewed as the remaining crucial step towards the exploitation of fusion power after ITER, not only in Europe but by many of the nations engaged in the construction of ITER. The DEMO design and R&D activities in Europe are expected to benefit largely from the experience gained from the design, construction and operation of ITER. Nevertheless, there are still outstanding gaps that need to be overcome, requiring a vigorous physics and technology R&D program beyond ITER.

The DEMO breeding blanket and its ancillary systems (e.g., cooling systems, PbLi circuit, coolant purification systems and tritium extraction systems) must operate safely and reliably from day-one. Achieving tritium self-sufficiency will be an unescapable requirement for any next-step fusion nuclear facility beyond ITER. However, no fusion blanket has ever been built or tested. Hence, its crucial integrated functions and reliability in DEMO and future power plant are by no means assured. However, the program in Europe benefits from many years of design and R&D, primarily carried out in European Fusion Laboratories. In addition, ITER presents a first and unique opportunity to test the response of representative component mock-ups, specifically called Test Blanket Modules (TBMs) at relevant operating conditions, in an actual fusion environment, albeit at very low neutron fluences (see for example [3,4] and references therein).

Recent work on DEMO pre-conceptual design in Europe has brought forward the need to launch a critical re-evaluation of the strategy for breeding blanket design and technology development that minimise the risks that could jeopardize the effort to arrive on time to sound breeding blanket design solutions, addressing both materials and engineering issues and extracting maximum benefit from the ITER TBM program.

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https://doi.org/10.1016/j.fusengdes.2019.01.141
Received 2 October 2018; Received in revised form 13 December 2018; Accepted 29 January 2019
Available online 27 February 2019
0920-3796/

This paper describes the newly revised design and R&D strategy of the breeding blanket in Europe that has been defined to take into account the input from the DEMO pre-conceptual design activities and the findings/recommendations of a thorough technical and programmatic assessment of the EU DEMO breeding blanket and EU ITER TBM programs, conducted in 2017 by an independent expert panel. This was conducted to identify, among the available options, the most mature and technically sound candidates for breeding blanket concepts to be potentially used as the "driver" blanket in DEMO and the remaining technical gaps and to align and strengthen the supporting R&D Program. To ensure a coherent and efficient Program, a change of the EU TBM options to be tested in ITER is proposed in order to obtain important and useful information from the two considered breeders (solid and liquid) and the two coolants (helium and water).

Sect. 2 provides an overview of the adopted EU DEMO staged-design approach and the interdependency and technical input expected from ITER. Sect. 3 describes the main blanket design constraints and integration issues in DEMO and the important role of the ITER TBM Program, Sect. 4 introduces the key aspects of the new DEMO Breeding Blanket/ EU TBM design strategy as Integral Part of the DEMO Design Effort. Sect. 5 provides a brief description of selected and recent achievements about the two most attractive breeding blanket concepts presently considered in Europe for DEMO.

2. DEMO in the EU roadmap

2.1. Programmatic and timeline considerations

At present, the DEMO design has not been formally selected and detailed operational requirements are not yet available. However, the DEMO plant high-level requirements have been defined following interaction with an external stakeholder group composed of experts from industry, utilities, grids, safety, licensing, etc. The design should be capable of producing electricity (up to ~500 MWe), operating with a closed fuel-cycle and to be a facilitating machine between ITER and a future First-of-a-Kind (FoaK) commercial fusion power plant (FPP). The approach advocated by the EU fusion roadmap, is to consider in the early design phase a plant concept that would rely as much as possible on mature design solutions and technologies and the knowledge basis acquired with the design, construction and operation of ITER. It is argued that by delaying the design of DEMO in anticipation of the ultimate advances in plasma physics and technology would postpone the realization of fusion indefinitely [1]. Thus, emphasis has been placed from the very beginning on the study of key design integration issues that are foreseen to affect the whole DEMO nuclear plant architecture, arising from remote maintenance, power conversion, safety, licensing, and technology readiness aspects. The risk of postponing integration, assuming that it restricts innovation and inhibits an attractive DEMO plant, is that designers remain oblivious of integration issues and develop design solutions that cannot be integrated in practice. Thus, an early system integration work is deemed necessary to develop an understanding of the importance and relative difficulties of various design integration and technological problems to be solved in a DEMO plant. It provides the context for further design improvements and contribute to guide future R&D. To this extent, contacts were also made with Gen IV fission and ITER to learn from their experience. Both projects emphasized the following aspects: (i) the plant design should drive R&D and not the other way around. (ii) fusion is a nuclear technology and as such, will be assessed with full nuclear scrutiny by the regulator; (iii) the need for a traceable design process with a rigorous Systems Engineering approach; and (iv) the technical solutions should be based on maintaining proven design features to minimize technological risks [2].

2.2. ITER and DEMO schedule dependencies

The EU Fusion Roadmap emphasizes ITER as the crucial machine on which the validation of the DEMO physics and part of the technology basis depends. There is therefore a high degree of schedule dependency between ITER and DEMO, although the 'success-oriented' approach outlined here advocates concurrency between the exploitation of ITER and development of the DEMO design. In this approach, the DEMO design activity proceeds in parallel with the ITER exploitation, but relies on a progressive flow of input from ITER for design and physics basis validation prior to authorization of DEMO construction. The DEMO design validation from ITER should not be seen as a single discrete event, but rather as an ongoing and progressive flow of information into the program – allowing continuous validation of specific aspects of the DEMO design, and if necessary, updates to the baseline.

Fig. 1 provides an overview of the analysis of dependencies identified between the revised DEMO and ITER schedules. The most critical and final major validation input, is the demonstration of D–T burning

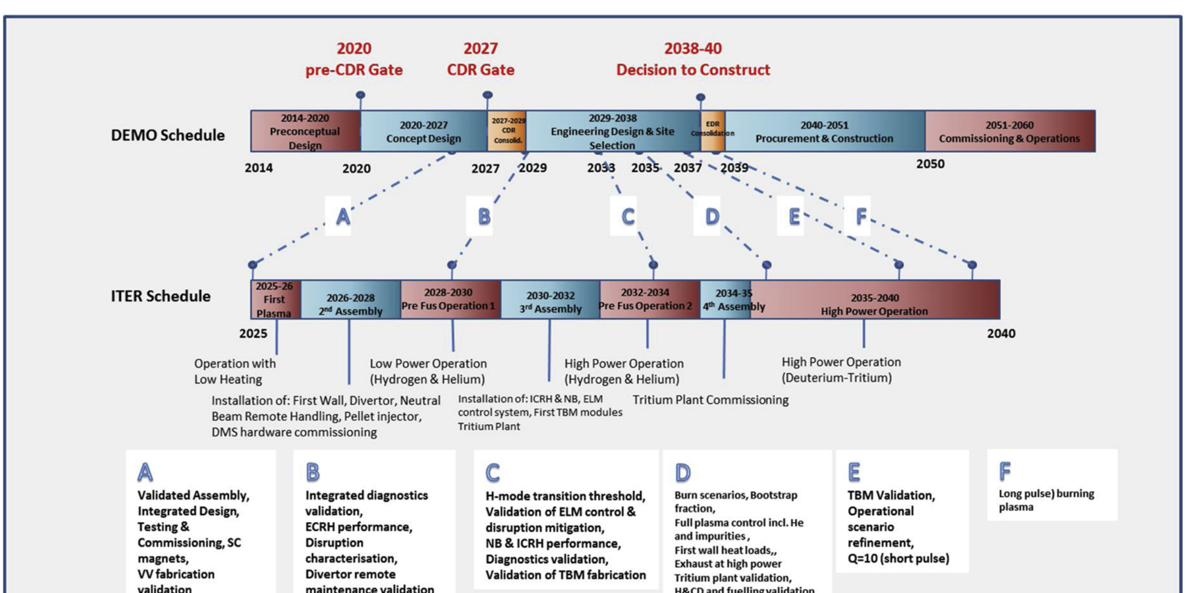

Fig. 1. Overview of phasing and key technical inputs from ITER DEMO Schedule.

plasma scenarios in ITER that are scheduled to start around 2037 (with Q = 10 short pulse in 2037 and long pulse in 2039).

In light of the above, the present DEMO development plan consists of the following three phases: (i) a Pre-Concept Design Phase to explore a number of DEMO plant concepts and develop system requirements up to 2020 (ii) a Concept Design Phase to mature and validate the baseline concept up to 2027[2] ; and (iii) an Engineering Design Phase beginning roughly around 2030 to develop the detailed design and prepare for the launch of major procurement activities around 2040′s, after ITER's nuclear operation has confirmed the robustness of the underlying assumptions.

2.3. The role of the ITER TBM program

The design, R&D and testing of TBMs in ITER is viewed as an essential step to reduce the remaining technical risks and uncertainties associated with the demonstration of power extraction and tritium breeding technologies essential for a DEMO fusion power plant. This is required for: (i) developing and validating the scientific understanding and predictive capabilities; (ii) demonstrating the principles of tritium self-sufficiency in practical systems; (iii) developing and qualifying the breeding technologies to be used in next-step machines (i.e., DEMO); (iv) providing the first integrated experimental results on safety, environmental impact, and efficiency of tritium extraction systems; and (v) providing initial components and operational reliability data for different ancillary systems (e.g, PbLi circuit, cooling systems, coolant purification systems and tritium extraction systems). The lesson to be learnt by the design and R&D of the ITER TBMs (both breeding boxes and ancillary systems) is viewed to be particularly valuable to aid the development and the down selection of the DEMO breeding blanket concept and will be discussed later in this paper.

However, to enable a consistent DEMO construction decision in time, the TBM Program must cover the best combination of design options that are considered to be the most promising candidate for the blanket to effectively minimize the main technical risks for DEMO. Thus, the results of the TBM Program, during all the phases (i.e., R&D, qualification, procurement and testing), are expected to provide important input.

It is nevertheless clear that risks and gaps will remain after ITER and, therefore, a sound and complementary R&D Program for DEMO to address long time performance at higher neutron fluence and high reliability is needed. In particular, vigorous materials irradiation in the limited number of existing fission research Material Test Reactors (MTRs) and ultimately in a DEMO-Oriented Neutron Source like IFMIF-DONES [5] is urgently required together with the construction of a limited number of dedicated non-nuclear blanket test facilities (or upgrade of the existing ones) for testing integrated multi-effect blanket behaviour.

3. DEMO breeding blanket design approach

3.1. DEMO design constraints and integration issues

In DEMO, the breeding blanket must perform a number of essential functions: (i) first, it must absorb the largest (~80%) part of the fusion energy transported by neutrons from the plasma and deposited volumetrically in the surrounding in-vessel structures. The remaining part (~20%) of the fusion power (fusion alpha particles) with the addition of the auxiliary heating power (~100 MW) constitutes the so called "power exhaust", and is deposited as surface heat onto the plasma-facing-components (PFCs), i.e. the first wall (integrated in the front-side of the blanket), the divertor and possible limiters. Taking into account the exothermal heat produced by nuclear reactions (about 1.2–1.3 energy multiplication factor depending on the neutron multiplier materials adopted in the breeding blanket), in a reactor of about 2 GW of fusion power, the blanket system has to extract about 1900 MW of nuclear power. Conversion of this energy at adequate thermodynamic efficiencies requires that the coolants are at high temperature and pressure. This has a strong influence on reactor engineering. (ii) Second, it must breed sufficient amount of tritium by capturing fusion neutrons in lithium-bearing materials (in solid or liquid form). Just as an example, a 2 GW fusion power DEMO is expected to consume around 111 kg of tritium per full power year (fpy), and this clearly underscore the indispensable requirement for the breeding blanket to produce and enable extraction of the bred tritium to achieve tritium self-sufficiency (i.e., it must produce its own fuel). The implications of the tritium breeding requirements on the design and integration of the tokamak in-vessel components that compete for space usage that is needed for breeding (i.e. divertor, protection limiters, auxiliary heating systems, etc.) are briefly discussed below (see also [6]). In addition, (iii) together with the vacuum vessel, the blanket must effectively contribute to shield various components from nuclear radiation (e.g., superconducting magnets and other equipment outside the reactor). Finally, (iv) the breeding blanket must be designed to enable efficient extraction of tritium and minimize losses of tritium. Considerations in this paper are limited to aspects of design and R&D of the breeding blanket. Further information on the tritium fuel cycle can be found elsewhere [7].

Fig. 2 [2] shows: a vertical cross section of the current EU DEMO and the physical interfaces between the blanket and the other systems like vacuum vessel and superconducting coils. The tritium breeding performance competes with the shielding performance in space restricted regions such as the mid-section of the inboard region.

The utilization of the space on the inner side of the torus represents a crucial design aspect in tokamak design and deserves some further considerations.

The power density in a tokamak can be written as

$$P_F \propto \beta_t^2 \, B_{t,o}^4$$

where β_t *(beta)* is the plasma kinetic-to-magnetic pressure ratio and $B_{t,o}$ is the toroidal field strength at the centre of the plasma. By increasing $B_{t,o}$ and/or *beta* one clearly obtains a significant increase in power output. However, *beta* is limited by plasma stability and $B_{t,o}$ is limited by technological constraints on the maximum practical magnetic field, $B_{t,m}$, at the magnet windings that is limited by technological constraints e.g., ~13 T for Nb_3Sn.

The relationship between $B_{t,m}$ and $B_{t,o}$ (see Fig. 3) [8,9] is given by:

$$B_{t,0} = \left(1 - \frac{1}{A} - \frac{\Delta_{pw} + \Delta_{BS}^{i}}{R}\right) B_{t,m} \tag{2}$$

where A is the aspect ratio (R/a) (typically in DEMO A ~ 2.5-3.5), *R, a* are the major and minor radius of the plasma, respectively, and $\Delta_{p,w}$ is the clearance between the plasma and the first wall (in DEMO ~0.2 m). The parameter Δ_{BS}^{i} is the thickness of the region occupied at the inboard by the breeding blanket and the vacuum vessel and includes also maintenance clearance and the thermal shield (i.e, the distance in mid-plane from the first wall to the TF coil windings). The cost of the TF coils, typically, increases as $B_{t,m}^2$.

Eq. (2) clearly shows that by reducing Δ_{BS}^{i}, for a given $B_{t,m}$, one can increase the value of the toroidal field strength at the center of the plasma, and, thus, the reactor power, or for a given reactor power can reduce the machine size (i.e. *R*).

Similarly, the flux core radius, r_{OH}, for the OH coil (equivalent to the central solenoid) is given by (see Fig. 3):

$$r_{OH} = R - (a + \Delta_{pw}) - \Delta_{BS}^{i} - \Delta_{m}^{i} - \Delta_{OH} \tag{3}$$

[2] It should be noted that a transition phase of 2-3 years is expected for the concept design review consolidation and preparation of the Engineering Design Phase

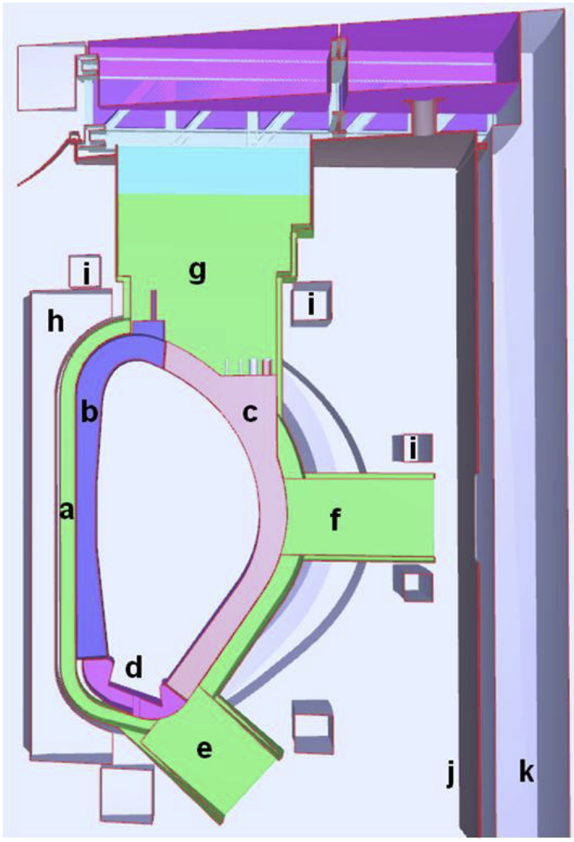

Fig. 2. Elevation view of the tokamak as generated by PROCESS; a) vacuum-vessel; b) breeding blanket (inboard); c) breeding blanket (outboard); d) divertor; e) lower port; f) equatorial port; g) upper port; h) toroidal field coils; i) poloidal field coils; j) cryostat; k) bioshield.

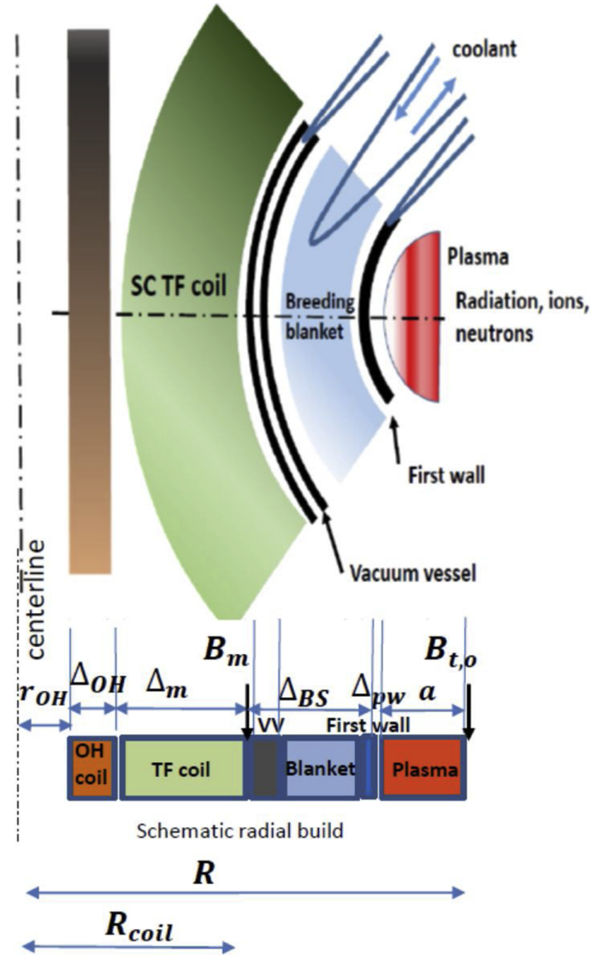

Fig. 3. Schematic showing the radial build at the inboard.

where Δ_m^i is the thickness of the inner TF coil leg and its support structures and Δ_{OH} is the thickness of the OH support cylinder. For a given R, a, $\Delta_{p,w}$, $B_{t,0}$, and P_F, reducing Δ_{BS}^i reduces also $B_{t,m}$ and Δ_m and r_{OH} increases. Increasing r_{OH} reduces the ohmic heating field, B_{OH} ($B_{OH} \sim 1/r_{OH}^2$). Besides the technological constraints on B_{OH}, the cost of the OH coils, and more importantly the cost of the power supply increases rapidly with B_{OH}.

All these factors, provide a strong incentive to reduce Δ_{BS}^i. However, satisfying the energy conversion and tritium breeding requirements in the blanket and providing the radiation attenuation in the blanket/shield necessary for magnet protection favours a relatively large Δ_{BS}^i.

In the current DEMO design the space utilization on the inner side of the torus and the required fractional coverage the breeding blanket needed to achieve tritium self sufficiency has been set on the basis extensive neutronics calculations [10–12]. They are used to define the basic geometric configuration, in particular, the radial reactor build. The main adopted design guidelines and criteria are described in Table 1 [13].

Due to the numerous penetrations (see Fig. 2) neutron streaming across penetrations on the outboard also represents a serious design issue. A biological shield is necessary to reduce the radiation biological dose outside the reactor to the maximum permissible dose for occupationally exposed individuals. It is conceived that the walls of the reactor building can serve the dual purpose of providing the necessary containment as well as biological shielding.

On the basis of the results of design effort conducted to date, it was found that about 1.40 m (60 cm vacuum vessel and ~80 cm inboard breeding blanket thickness was found to provide sufficient shielding, both in terms of material damage and vacuum vessel nuclear heating. To ensure tritium self-sufficiency, thin PFCs are required and, in addition, the number of penetrations must be minimised (about 85% of the plasma must be covered by the breeding blanket) and there are constraints on the space occupied by the divertor [14]. It should be noted that the definition of the effective radial build depends on the specific type of blanket design. For example, a blanket design with water as coolant (i.e. the water-cooled lithium lead concept (WCLL) is expected to shield better than the case of helium (moderation of water and reduced streaming in the manifolds/header) but breed worse [15,16]. On the contrary, a design concept based on He cooling that use Be multiplier (i.e. the helium-cooled pebble bed (HCPB) concept) can achieve better TBR values and, therefore, can be thinner from the breeding efficiency point of view, but the n-shielding performance is worse. Analyses are in progress to calculate the radial build of water-cooled and helium-cooled concepts aiming at determining the optimum thickness from the standpoint of tritium breeding, n-shielding and minimization of activation of the surrounding vacuum vessel [16,17].

The modularity of the blankets is given by the magnet structures (i.e., the number of toroidal field coils) which, in the current design configuration, leaves 16 toroidal interspaces to give access to the blankets for remote maintenance purposes from the top of the machine [18]. Each of the 16 blanket sectors is divided into three segments at the outboard and two segments at the inboard. The estimated average neutron wall load in DEMO is ~ 1 MW/m^2. Based on current operation considerations, up to 30,000 pulses (as in ITER) with a burn-time per cycle of 2 h each (much longer than ITER) are required to attain a total cumulative limiting fluence of 7 MW a/m^2 during the machine lifetime, which corresponds to 70 dpa in EUROFER steel components of the plasma near structures (Table 2) [2].

Table 1
Breeding and shielding requirements and design targets used in the present design phase of DEMO.

Tritium breeding	A TBR ≥ 1.05 requires: ● thin PFCs ● limited penetrations, e.g.,about 85% of the plasma must be covered by the breeding blanket ● constraints on occupied divertor space
n-shielding	● Max displacement damage in Vacuum Vessel (2.75 dpa) ● Cutting/re-welding location in In-Vessel Component (IVC) cooling pipes helium production 1 appm ● Total neutron fluence to epoxy insulator $10^{22}/m^2$ ● Fast neutron fluence to the Nb^3Sn $10^{22}/m^2$ ● Neutron fluence to Cu stabilizer between TF coil warm ups $1–2{\cdot}10^{21}/m^2$ ● Volumetric nuclear heating in winding pack 50 W/m^3 ● Port interspace: Shutdown dose rate 12 days after shutdown ~500 μSv/h (target) ● Port cells (occasional access) 100 μSv/h (target) ● In-cryostat area, 100 μSv/h (target) ● Tokamak building areas beyond port cells requiring frequent access, Shutdown dose rate 1 day after shutdown 10 μSv/h (target)

Table 2
Current EU DEMO design assumptions.

Main design assumptions
- Pfus ~ 2000 MW ~ 500 MWe Pulses: 2 h
- Single-null water cooled divertor; PFC armour: W
- Low Temperature Super Conducting magnets Nb3Sn (grading)
- 16 TF coils; Bmax conductor ~12 T
- EUROFER for IVCs, AISI ITER-grade 316 for Vacuum Vessel
- In-vessel RH: vertical (blanket)/ horizontal (divertor)
- DEMO plant lifetime (design) ~7-8 fpy
- Neutron wall loading (average) ~ 1 MW/m2
- Thermal conversion efficiency > 30%
- Tritium fuel cycle: self sufficient
- Blanket lifetime
• Starter blanket: 20 dpa
• Second blanket : 50 dpa
- Reactor availability: it is assumed that the availability of a DEMO plant during its initial years of operation (starter blanket) is relatively low and increases to about 30% or more.

A more detailed description can be found in [2].

3.2. A progressive approach for blanket operation in DEMO

It is currently foreseen that DEMO will utilise a first blanket with a 20 dpa damage limit in the first-wall steel (EUROFER) and conservative design margins and then switch to a second set of blankets with a 50 dpa damage limit with an optimized design (i.e., with somewhat reduced design margins), and if available, improved structural materials that need to be qualified in advance. As it is unfeasible to change the Balance of Plant (BoP), the same coolant must be used while switching from the first set to the second set of blanket.This type of approach has been used for the fuel cladding in fission reactors for many years; limiting the maximum exposure level of the replaceable cladding to below the regulatory limit, while data for higher exposure operation is generated in test reactors or load test assemblies [17]. The selection of the 20 dpa value as a target for the 'starter' blanket is discussed in [19]. Irradiation of structural materials up to 20 dpa can be simulated, with sufficient accuracy, in existing MTRs, because the level of the He production up to this fluence foreseen in a 14 MeV fusion spectrum is deemed to be still relative modest (~few hundred appm) to significantly affect material properties [19]. Fusion irradiation data to be provided in IFMIF-DONES [5] foreseen to become operative by the end of the decade will be important to validate data collected in MTRs and extend irradiation data at higher fluences, relevant for the second set of blanket. It is also currently envisaged that DEMO act as a Component Test Facility for the breeding blanket. While operating with a near-full coverage "driver" blanket, which must be installed by day-1 to achieve tritium self-sufficiency and extract the thermal power and convert this in electricity), it must be used to test and validate in a limited number of dedicated segments of more advanced breeding blanket concept(s) that have the potential to be deployed in a future FoaK FPP. The idea to test advanced blanket concepts in a reactor operating with a conservative breeding blanket design is not new. Early considerations were already given to this in the 80′s (see for example [20,21].

Such flexibility and capabilities, however, have to be properly investigated early in the conceptual design phase and formalized as high level requirements, since they have major implications on the plant architecture, and systems requirements. This implies that adequate equipment external to the DEMO basic device (test loops) must also be installed at the beginning, or provision made for its later installation. The design features of the test elements should be compatible, reliable and safe enough not to jeopardize the operation of the DEMO Plant. The detailed design of the test elements will be done during the conceptual design phase.

The final decision on the type of "driver" blanket cannot be made today, because of the existing performance uncertainties and feasibility concerns even for the most mature design concepts. A down-selection, however, is deemed possible by the middle of the next decade, taking into account design and R&D input obtained not only in the area of blanket and TBM, but safety, materials, BoP and remote maintenance, etc. [22]. This will enable a DEMO plant concept to be coherently designed for a design review by 2027 (see Sect. 2).

3.3. Main breeding blanket design concepts in Europe

Breeder blanket systems have been under development since the start of civil fusion investigations in the early 1950′s. In Europe, major design studies were performed in 1990–1999 under the NET Program [20,21,23], in 2000–2004 under the Power Plant Conceptual Studies [24] and 2005–2007 under the DEMO studies [25]. Major comparative studies have also been carried out in the US (see for example [26,27]). An excellent review of the main technical issues in developing the blanket/first wall and the key R&D needs in non-fusion and fusion facilities on the path to DEMO can be found in [28].

Two main breeding blanket concepts had been investigated in Europe up to 2002: a HCPB concept and a WCLL concept. Mainly due to budget limitations, a decision was made in 2003 to narrow down the related R&D and to limit the work on two helium-cooled design concepts: the helium-cooled lithium-lead (HCLL) and the HCPB. In 2008, when the decision of the European TBM concepts to be tested in ITER had to be confirmed, and in the absence of a comprehensive DEMO design study, the choice was made to consider for the TBM program the same type of breeding blanket concepts as developed up to that point (i.e, HCLL and HCPB). In addition, it was assumed that parallel advanced development in areas of BoP of nuclear systems and structural materials were to be expected from fission industry and in particular from the development of advance fission systems (i.e., Gen. IV).

Because of the numerous remaining uncertainties and feasibility concerns, four blanket concepts were originally considered in the DEMO pre-conceptual design phase conducted by the EUROfusion Consortium in the work package Breeding Blanket (WPBB) since 2014, covering all the possible technologies that are believed promising for a

DEMO with a development time compatible with the EU fusion roadmap goal. These included (i) the two Helium cooled concepts (HCPB and HCLL) that were part of the EU ITER TBM Program and make use of solid and liquid breeder respectively; (ii) a WCLL, which makes use of Lithium Lead as a breeder; and a (iii) a dual coolant concept (DCLL) using helium and liquid breeder/coolant. Technical details of these concepts can be found elsewhere [29].

However, the awareness of the importance of the integration aspects, especially those related to the choice of the breeding blanket coolant, which affect the whole DEMO Plant [22], along with the perception of the technology gaps still to be overcome in some areas, have recently motivated a critical re-evaluation of the technical choices for the DEMO breeding blanket and the TBM concepts to be tested by Europe in ITER.

As we said, the choice of the breeding blanket coolant provides a clear example of a design issue that pervasively affects the overall design layout of the nuclear plant, and bear a strong impact on design integration, maintenance, safety because of his interfaces with all key nuclear systems. It is generally agreed that water should be considered as the divertor coolant for a near-term DEMO design as the divertor surface heat flux conditions prove to be beyond present helium power handling capabilities [19]. However, the choice of the breeding blanket coolant is still open. Technical issues influencing the choice include: (i) thermal power conversion efficiency; (ii) pumping power requirements; (iii) required power handling capabilities of the blanket first-wall; (iv) n-irradiation structural material mechanical properties; (v) n-shielding requirements (e.g., reduce the blanket thickness that is critical at the inboard side); (vi) achievable tritium breeding ratio; (vii) breeder tritium extraction; (viii) tritium permeation and tritium inventory control and purification; (ix) chemical reactivity, coolant leakages and chronic release; (x) design integration and feasibility of BoP; and (xi) design of safety system like the Vacuum vessel Pressure suppression System (VVPSS) that shall contain and confine the primary coolant in case of in-vessel Loss of Coolant Accident (LOCA) keeping the Vacuum Vessel (VV) pressure below the limit presently set to 2 bar (as in ITER).

4. Revised programmatic strategy

4.1. Drivers for a new proposal

The rationale of the new strategy is based on the following main considerations:

- First, in the European Fusion Roadmap emphasis is given to the objective to develop DEMO design solutions that will allow fast deployment of fusion energy based as much as possible on mature technologies. Thus, the capability to develop breeding blanket solutions that can be delivered in the miD–Term must be secured now so as to decrease delays on the demonstration and deployment of fusion power. This intrinsically pushes for conservative solutions with high reliability and design margins.
- Second, the input from the DEMO pre-conceptual design activities, in particular, the importance of the design integration aspects, clearly shows that the selection of the breeding blanket for DEMO must not be solely based on performance criteria of the breeding blanket. It shall account for the interfacing systems, the tokamak integration and the safety approach. Investigating these fully represents one of the major goals and drives the design and R&D efforts during the DEMO Pre-Conceptual Design Phase [30].
- Third, ITER will be the only opportunity to test relevant concepts for the breeding blanket for DEMO. This implies that the TBM design must be carefully defined to confirm and validate the most promising concepts for the DEMO "driver" breeding blanket. Previous design options of the EU TBM (HCPB and HCLL) were made in the absence of a comprehensive DEMO design study and assuming that important parallel advanced development in areas of the BoP of nuclear systems and structural materials were to be expected from fission industry and in particular from the development of advanced fission systems (i.e. Gen. IV), especially in the area of helium cooling. Unfortunately, this progress has proven to be much slower than expected.
- Fourth, as DEMO is foreseen to play the role of a "component test facility" for the breeding blanket [2,31], its design must incorporate the ability and the flexibility to accommodate for testing at least one type of advanced tritium breeding blanket concept, with the potential to be deployed in a FoaK FPP.

In light of the above, a technical and programmatic assessment of the DEMO breeding blanket program and the EU TBM program, has been made in 2017 to study the feasibility and the technical coherence of a change of the EU TBM program and to identify: (i) the best and most cost-effective strategy including the necessary preparatory R&D activities and take into account ITER construction schedule slippage; (ii) the required preparatory work for the operation and scientific exploitation of the EU ITER TBM, including the required output for DEMO; (iii) the consolidation of the overall technical rationale and programmatic needs to validate a technically-coherent and financially-sound program, which harmonizes the DEMO breeding blanket and EU ITER TBM and the associated R&D program. The key recommendations are summarised below.

4.2. Key recommendations

- Focus should be given from now, in the EUROfusion Work Package Breeding Blanket (WPBB), on the two most promising blanket concepts for DEMO, the HCPB and the WCLL. Nevertheless, a limited R &D activity should also be maintained on the other concepts, i.e., HCLL (e.g., if both water cooling and the solid breeder turn out unfeasible) and DCLL (as a potentially very attractive long-term option). Work in the latter concepts should be restricted to the aspects of R&D not already covered by HCPB and WCLL, and should not include design integration activities at least in the near-term. The two lines with title "Breeding Blanket design" in Fig. 4 show the steps and milestones of the WCLL and HCPB breeding blanket design to be aligned with the top level DEMO schedule: DEMO Gate Review, Driver Blanket selection in 2024, CDR in 2027 up to the engineering design for both driver and advanced blanket concepts.
- Accordingly, two TBM concepts to be tested by Europe in ITER are a HCPB concept and a WCLL concept, the latter to replace a HCLL. This strategy, will enable testing both coolants (helium and water) and both breeder materials (PbLi and ceramic/Be) and is perceived to be the best to minimize the technical risks and gaps to arrive, in the time frame foreseen by the EU Fusion Roadmap, to a consolidated design for the driver breeding blanket for DEMO. Due to the tight ITER schedule, which requires the completion of the conceptual design review for the WCLL TBM by 2020, the process to replace the HCLL TBM by a WCLL TBM has been initiated at the time of writing of this paper. On top of Fig. 4 the two lines for the development of HCPB and WCLL Tritium Blanket Systems (TBSs) show the main milestone to be achieved for their delivery to ITER, i.e. Concept Design Review (CDR), Preliminary Design Review (PDR), Final Design Review (FDR), along with the delivery of the experimental results in the latest nuclear high-duty DT phase.
- The selection of the "driver" breeding blanket and most promising advanced blankets is now impossible because of the existing uncertainties. However, in view of the DEMO schedule (Sect. 2) and interdependency with the ITER schedule, a decision of the DEMO driver breeding blanket is deemed to be possible and should be made at the latest by the first half of the next decade taking into account design and R&D input obtained not only in the area of blanket, but also for the topics related to safety, materials, design integration, Primary Heat Transfer System (PHTS), BoP and remote

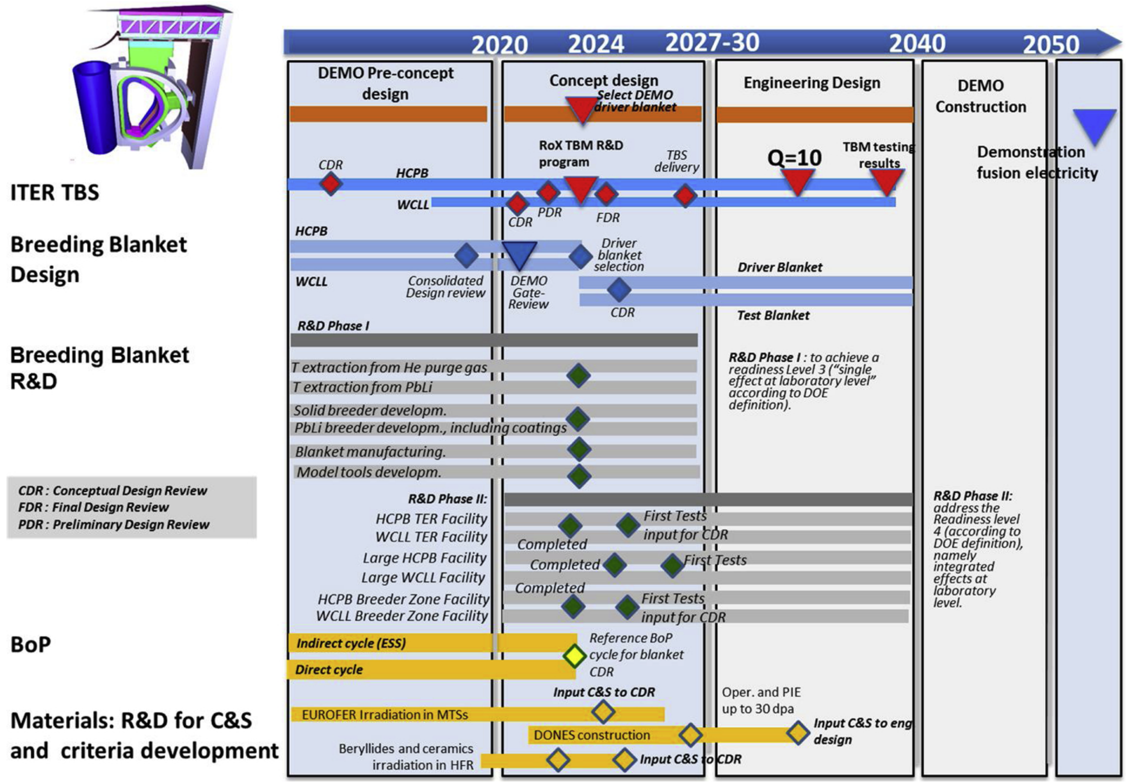

Fig. 4. New R&D strategy proposed to re-align the ITER TBM and DEMO breeding blanket.

maintenance etc. This will enable a DEMO plant concept design review by 2027. It should be noted that the choice of the driver blanket now planned in 2024 will not further affect the TBM program. In Fig.4 the decision point for the selection of the driver blanket in 2024 is aligned with a number of expected outcomes that are expected to affect this decision, e.g, the development of the BoP design associated to water or helium coolants, the development of the design code for the structural material, etc.

- The completion of the TBM R&D program is mandatory for the verification of the choice of the "driver" blanket, with validation being completed before starting DEMO construction. The RoX and technical information foreseen by the EU TBM R&D Program (see Table 3) will play an important role in the down-selection of the driver blanket. In addition, the results of the TBM tests during the nuclear phase will validate the interpretative/predictive codes required to confirm design choices for the DEMO breeding blanket. DEMO breeding blanket design will benefit by then also from the result of materials irradiation tests to be achieved on a well suited test bed such as DONES. [5]

All the financial, administrative and governance aspects of the new programmatic strategy have been addressed and are being implemented in a co-funded collaboration program by the EUROfusion Consortium and its laboratories and the European Union's Joint Undertaking for ITER and the Development of Fusion Energy (F4E). While the responsibility for the design and R&D of the DEMO breeding blanket remains entirely under EUROfusion, the roles and responsibilities for the EU ITER TBM, are defined as follows. F4E is the prime contractor in front of the Nuclear Operator, i.e., the ITER International Organization (IO), with the responsibility to deliver the two ITER Test Blanket Systems (TBS) according to the ITER schedule. F4E is also the manufacturer of the two TBS as per the ESPN Order, i.e, the regulation of the Nuclear Pressure Equipment in France. EUROfusion is primarily responsible to

Table 3
Return of Experience (RoX) and technical information provided by the EU TBM R&D Program.

◊ EUROFER97 development & qualification. Database on base material and welded joints, including properties of irradiated EUROFER97 (low dose up to 2 dpa); introduction in the RCC-MRx code.Experience from the EUROFER97 finished products fabrication/procurement: development of advanced (non-conventional) fusion welding (laser, TIG, EB) & diffusion welding (Hot Isostatic Pressure - HIP) technologies.
◊ Development and qualification of Functional Materials (FM) (i.e., ceramic beeders, Be, Pb-16Li alloy): FM fabrication routes for advanced CB, Be/beryllide materials, Pb-16Li eutectic alloy; Li-6 enrichment issue, including regulatory & export control aspects, market availability; FM characterization results, including neutron irradiation response, Be/air & steam interactions; capitalization of FM data in the Material Assessment Report (MAR) & Material Database Report (MDBR).
◊ PbLi technology; Pb-16Li purification (experimental validation); Magneto-hydrodynamics (MHD) experimental validation of the predictions; Safety aspects, i.e. impurities limitation (e.g. Bi, Hg, Tl), polonium issue, Pb-16Li/water interaction (pressurization and H2 generation).
◊ Tritium technology; Tritium extraction from Pb-16Li, i.e technologies experimental validation (gas/liquid contactor, PAV), efficiency (TRIEX experiments), Tritium Accountancy.
◊ Molecular sieves technology for tritium extraction from He flow (experimental validation).
◊ Validation of Tritium database (Sievert's constant, permeation, diffusivity,…).
◊ Pressurized helium technology: thermal-hydraulics of high-pressure Helium (prediction & validation); helium circulator – design, performance (experimental validation), failure rate (at level of R&D – HeFUS, HELOKA)
◊ Pressurized water technology
◊ Instrumentation / sensors development & qualification
◊ Predictive tools development & validation
◊ Complete licensing process covering the complete consultation cycle with an Agreed Notified Body and ASN (French Nuclear Regulator)
◊ Implementation of the French/European Regulation on (Nuclear) Pressure Equipment ESP(N), PED Directive (Directive 2014/68/EU), ESPN Order (Arrêté du 30 décembre 2015)

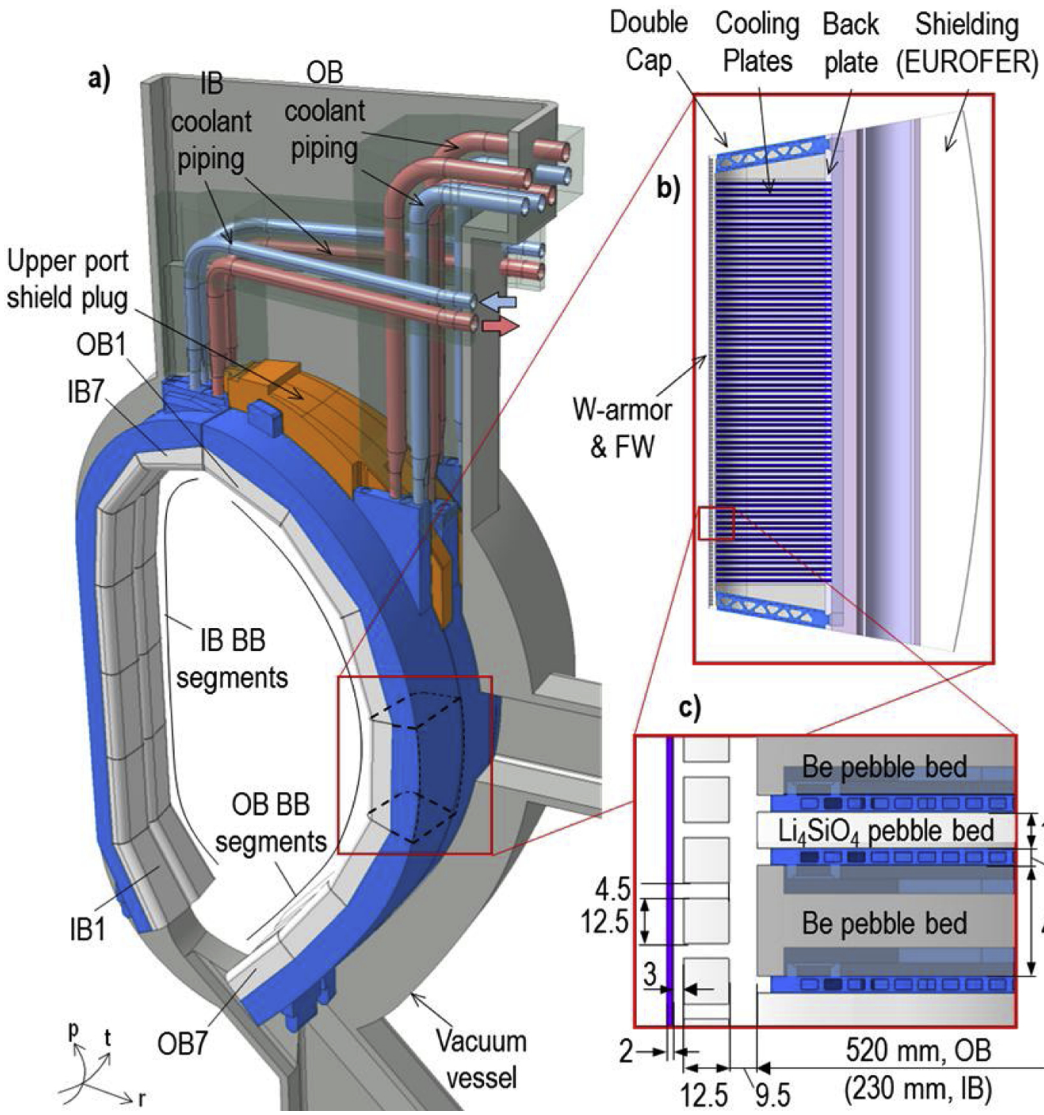

Fig. 5. DEMO HCPB Design: (a) elevation view showing 2 inboard and 3 outboard segments per sector; (b) blanket toroidal cross-section, showing the breeding (front) and shielding (back) regions; and (c) detail ceramic breeder and Be multiplier elements in the breeding blanket region.

manage R&D Program for DEMO Blankets and TBM and TBS in the fields of: functional materials development, EUROFER qualification and design rules, sensor technologies development, predictive tools development and qualification, instrumentation / sensors development & qualification. A new EU TBM Project Team is being established with the aim to support/improve interactions with safety, nuclear licencing authorities and to strengthen project oversight and control, design and system integration ensuring best RoX implementation from TBM project into the DEMO breeding blanket. The first EUROfusion contract for the conceptual design of a WCLL TBM has been launched. It covers the conceptual design of the WCLL TBM and of its ancillary systems in order to meet the ITER TBM WCLL Concept Design Review date set by mid of 2020.

The expert panel also endorsed some additional important elements of the breeding blanket development strategy that were already included in the original roadmap. This includes:

- The role of DEMO as a Component Test Facility for the breeding blanket, as described above.
- The phased-operation strategy and progressive licencing to utilise a “starter” breeding blanket with a 20 dpa damage limit in the first-wall steel (EUROFER) and conservative design margins and then switch to a second set of blankets with a 50 dpa damage limit with an optimized design, and if available, improved structural materials.
- An early engagement with a licensing consultant is needed to understand and tackle potential safety implications through design amelioration.

The Expert Panel has recognized the criticality of the issue of tritium availability for operating fusion power plants after ITER. Based on the results of a study conducted in 2017 [32] and the forecasts of tritium production in Heavy Water Reactors (HWRs) of Canadian Deuterium Uranium (CANDU) type-reactors in countries where tritium extraction is carried out, or planned to be carried out, worst-case scenarios were identified where it would appear that there is insufficient tritium to satisfy the fusion demand after ITER.

Clearly there is a need to better understand and monitor the future availability of tritium and understand the impact of limited resources on the timeline of DEMO. However, there is essentially very little that the fusion community can do to exert an effect on the supply side, as tritium is a by-product of the operation of these reactors and not the primary economic incentive. Defense stockpiles of tritium are unlikely ever to be shared, and commercial CANDU operators will not alter their plans just to sell more tritium for the start-up of the first fusion power plants. In the short-term it is recommended to monitor the production of tritium in HWRs and estimate the available supply commercially. If, at some point in the future, it looks as though the demand for DEMO will exceed the supply from CANDUs, then action would have to be taken. It is likely that production of significant amounts of tritium from a dedicated source would be very expensive and take a long time. The "tritium window" as it was once defined by Paul Rutherford [33] is not open indefinitely. Based on current estimates, we believe it would be open until around 2050, after which it closes quite rapidly, unless the future of the CANDU reactor program turns out much more favorably than could presently be expected. The most advantageous way to fit fusion development into the tritium window would be to timely construct DEMO after ITER on the presently current timetable in Europe. Any program strategy that substantially delays substantially the DEMO step places fusion at risk, by allowing the unique and effectively irreplaceable tritium resource to decay to levels, which may be insufficient to complete fusion's technological development.

5. Example of selected achievements

Due to the limitation in space, only some brief technical information and main recent achievements related to the two breeding blanket concepts currently considered in Europe for DEMO are summarized here. References to further relevant work published in these proceedings or elsewhere are provided.

5.1. HCPB design and R&D

The current DEMO HCPB design is based on the use of Li_4SiO_4 as tritium breeder material, Be as neutron multiplier and He (inlet 300 °C, outlet 500 °C, 8 MPa) as coolant. The HCPB blanket system is formed by 18 sectors with 5 segments each (3 outboard and 2 inboard), in which each features 7 poloidal blanket modules (see Fig. 5(a)). In particular, each blanket is a sandwich-like structure of parallel, actively cooled cooling plates (CPs) that subdivide the breeder zone in slices of the breeder and multiplier pebble beds (see Fig.5(b) and (c)) [34,35].

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 ceramic breeder and the Be pebble beds, independently. The doping agent is of special importance, as it enhances the tritium desorption rate from the ceramic breeder, maintaining 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. The purge gas chemistry and flow control is currently an area of R&D. In particular, the possibility to use H_2O instead of H_2 as doping agent would allow the isotopic exchange of the tritium with H_2O to form tritiated water species, instead of HT. Such tritiated water species are non-permeating, which would significantly reduce the tritium permeation in the high-pressure He primary coolant by orders of magnitude. However, careful analysis is needed to show that such a change isn't simply trading one problem for another. Historically, H_2O is excluded as a doping agent because of its incompatibility with pure Be and the associated safety issues. In addition, the increased corrosion of Reduced-Activation Ferritic Martensitic (RAFM) such as EUROFER, that could be caused by the H_2O addition must also be addressed. As a matter of fact these steels contain only ~9%Cr, which is too low to provide robust oxidation resistance.

EUROFER97 is foreseen as blanket structural material, which is especially suited for the use of He coolant at the given temperatures. Due to the moderate heat transfer in the He coolant, the temperature difference between the fluid bulk temperature and the coolant walls is large enough to maintain the EUROFER97 temperature beyond its lower limit (~350 °C) to avoid the Ductile-to-Brittle-Transition-Temperature (DBTT) shift, especially at high dpa regions of the blanket like the first wall. Recent R&D points to the possibility to extend the high temperature operating window of EUROFER97 [36]. Albeit preliminary, these results if confirmed by the R&D program in place, would allow an increase in the outlet temperature of the coolant up to ~600-650 °C, improving the reactor net efficiency. Such a temperature outlet will allow to increase the temperature difference along the reactor core, reducing the coolant mass flow and thus the circulating power, and also in the steam generators / intermediate heat exchangers, reducing the size of these components, thus minimizing the coolant inventory.

Emphasis on the HCPB design concept has been on improving the understanding of the nuclear performance (i.e. tritium breeding and nuclear shielding), as well as simplifying the design in order to minimize the amount of steel (i.e. reduced parasitic absorption for better neutron economy). Also, the simpler blanket internals based on a single component (CPs) have reduced the pressure drops from former designs, achieving a plant circulating power of ~130 MW. A comprehensive set of thermomechanical analyses show a correct overall behaviour of the blanket against normal and accidental conditions.

Despite the aforementioned progress, some key issues still need to be solved. In particular, the current focus on the use of mature technologies for DEMO, especially for the BoP, poses a strict restriction on the upper limit for the plant circulating power for a He-cooled DEMO. On the other side, 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 [35]. Also, a concept architecture based on cooling plates results in the presence of numerous cooling channels, making more problematic a further reduction of pressure drops and negatively affecting the blanket reliability.

For these reasons, alternative design concepts, which could potentially minimize these issues are explored. This research activities have led to an enhanced HCPB design [37,38] based on single-module segments with a hexagonal arrangement of fuel-breeder pins. This enhanced HCPB concept uses $Be_{12}Ti$ as neutron multiplier, which permit operation at larger temperatures in this material, 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 [37], fulfilling basic nuclear, thermohydraulic and thermomechanical requirements.

Due to some inherent issues of Be (toxicity, limited resources, industrialization difficulties and high production costs), additional research has been conducted to find back-up alternatives to Be/Be-alloys as neutron multiplier. Starting from the same fuel-breeder pin architecture, an alternative solid breeder blanket based on molten lead as neutron multiplier has been developed [39,40]. This alternative has shown to potentially fulfil the basic blanket requirements while eliminating the issues associated with the use of Be.

5.2. WCLL design and R&D

The current DEMO WCLL breeding blanket design is characterized, as the HCPB, by 16 sectors (dictated by the number of TF coils), each including two inboard and three outboard segments. This breeding blanket uses reduced activation ferritic-martensitic steel, EUROFER97, as structural material, Lithium-Lead (PbLi) as breeder, neutron multiplier and tritium carrier, and water as coolant at Pressurised-Water-Reactor (PWR) conditions: 295–328 °C @ 15.5 MPa [41,42].

The design is developed on the basis of the Single Module Segment (SMS) [42,43] approach (Fig. 6(a)). Each segment consists of an 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.

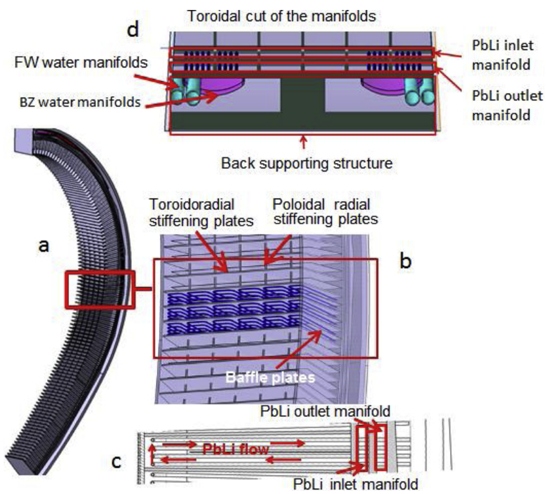

Fig. 6. WCLL BB SMS and BZ stiffening approach, cooling tubes, PbLi flow path and calculated EUROFER T field in outer central segment equatorial zone.

The WCLL breeding blanket segments are equipped with internal stiffening plates (Fig. 6(b)), placed along the poloidal-radial and toroidal-radial planes, in order to guarantee that the performance be compliant with the structural design criteria for in-vessel components in normal operation (Normal operating condition, correspond to category I and Level A criteria Level to be satisfied in the structural design code SDC-IC) and in over-pressurization loading scenario (Level D rules prescribed by the SDC-IC structural design code are to be satisfied in case of an event classified as Category IV corresponding to an extremely unlikely event). For the second condition the design pressure is 15.5MPa + 20% [44,45]. The overall segment is formed of breeding units. Radial-toroidal baffle plates, placed between two horizontal stiffening plates, ensures the PbLi circulation in radial-poloidal direction.

The WCLL breeding blanket is cooled by two independent systems: the first-wall and the breeding zone cooling systems [41,45]. They are operated at the nominal pressure of 15.5 MPa and the temperatures 295–328 °C. The first wall cooling system is an integrated part of the blanket segment directly exposed to the plasma, constituted by a EUROFER U-shaped plate 25 mm thick, bent in a radial direction. The first-wall plasma facing area is covered by a tungsten layer of 2 mm. The water flows in square channels (7 × 7 mm) in counter–current directions. The integrated system can safely remove the maximum average plasma heat flux (1.17 MW/m^2) and power nuclear heat depositions of DEMO, delivering coolant to the first-wall PHTS heat exchanger at the design temperature [46].

The breeding zone cooling system removes the power deposited in the PbLi and EUROFER structures. It relies on a Double-Wall Tubes (DWT) technology to reduce the probability of “in-box-LOCA” occurrence [41,42]. The reference design 2017 foresees 4 U-shaped DWT, installed in one breeding unit (Fig. 6(b)). This configuration simplifies the manufacturing procedure and allows a temperature field in the breeding zone symmetric to that of the poloidal plane passing for the center of the segment. Moreover, the thermal gradient in the radial direction of the structure is limited, being the minimum and the maximum temperatures calculated with ANSYS CFX 18.1 equal to 300 °C and 410 °C, respectively.

The PbLi enters in the breeding unit from the bottom of the breeding cell (which is formed by the breeding units of the segment at the same poloidal elevation), flows in radial direction, from the back plate to the first-wall and, then, recirculates towards the back plate, as shown in Fig. 6(c), with an average velocity lower than 0.1 mm/s. The liquid metal hydraulic path in the breeding zone was studied considering the effects of the magnetic field that include, but are not limited to, flow pattern variation, turbulence suppression, and additional MHD-related pressure drops [46,47].

The water manifolds are integrated in the region between the BSS and the BP. The inlet and outlet of the breeding zone system are defined in a way that the symmetry is ensured and the number of structures is minimized. The PbLi manifolds are constituted by two gaps of 40 mm, delimited by three plates of 30 mm. The PbLi is distributed (collected) in the six channels of the breeding units through holes (see Fig. 6.d).

The BSS is based on a plate 100 mm thick with ribs welded in the convex part connecting the breeding zone back plate. The thermo-mechanical performances of the BSS and the SMS were assessed in normal operation conditions and in case of central major disruption loading scenarios, by means of the ABAQUS v. 6.14 commercial FEM code (steady state analysis). The modelling approach and the analyses addressed both the segment structure stand alone and the overall 20° sector including three outboard and two inboard segments and the VV attachment scheme. The preliminary verifications, according to ITER Structural Design Criteria for In-Vessel Components (SDC-IC), were satisfied [46,47].

Recent studies have been devoted to analyse a number of design alternatives to optimize the breeding blanket internals, the PbLi distribution, draining capability and minimize the associated MHD issues [46]: the goal is to explore a large number of options and minimize the technical risks of the current WCLL configuration. In parallel a number of key R&D activities are carried out. This primarily includes the study of accidental scenarios with water/steam ingress into the PbLi [48] and the development of anti-corrosion/anti-permeation barriers, to minimize the corrosion of the EUROFER surface in contact with the flowing PbLi and minimize the tritium permeation through the structural material and consequent capture under the form of HTO in the primary coolant. Progress has been made in the production and characterization (also under irradiation) of Al_2O_3 coating developed by Pulsed Laser Deposition (PLD) and Atomic Layer Deposition (ALD) [49]. In addition, work is progressing on the development of TER technologies for the PbLi. The solution envisaged for DEMO is the Permator Against Vacuum (PAV) and a dedicated R&D program is in place to qualify the components at a proper level of maturity (note that PAV is not used in ITER TBMs) [50].

Computational tools developed for studying the PbLi flow under magneto-hydrodynamic (MHD) flows [47] are being validated with properly designed experiments. In particular, a dedicated experiment is being conducted in the Magnetohydrodynamic PbLi Experiment (MAPLE) at the University of California Los Angeles (UCLA) [51] through an EU/ US collaboration. A team of experts from Europe and UCLA has recently completed the upgrade of the facility, which is equipped with a rotating magnet and will be dedicated to studying the mixed-convection phenomena of volumetrically heated liquid metals flowing in variable magnetic fields [52]. First experiments on MHD mixed convection were performed at the time of the preparation of this paper

The WCLL breeding blanket coolant system is integrated with the water coolant PHTS. Connecting pipelines are routed through the upper ports. The largest inlet/outlet coolant pipelines are DN-200 and are connected with the outer segment breeding zone system. The PbLi loops feed the breeding blanket form the lower ports and receive the breeder from the upper ports. Current pipes are DN-200, based on fluid velocity, but analyses are in progress to quantify the pressure drop induced by the magnetic field.

Due to the low tritium solubility in Pb-16Li, an important matter of investigation is the tritium management in the WCLL breeding blanket and, particularly, an accurate and reliable prediction of the tritium permeation into the coolant and into different reactor areas as well as tritium inventory in the structural and functional breeding blanket materials. To tackle this fundamental topic a significant effort has been spent and is still ongoing to develop suitable tritium transport modelling tools, incorporating advanced physics (i.e. co-diffusion of hydrogen isotopes, surface dissociation and recombination, etc.) and implementing an updated tritium-materials interaction database.

Synergies between what is planned within the TBM and DEMO breeding blanket Programs have been identified so that a common effort in this area will be optimized and better implemented in the next years [53].

5.3. Balance of plant

As it was mentioned above, considerations related to the characteristics of the BoP play an important role in the selection of the 'driver' breeding blanket. Emphasis at this early design phase has been on a few important aspects of BoP, particularly the PHTS [54,55] and the relevant Power Conversion System (PCS) [56] because of their technical complexity and strong impact on design integration, maintenance, safety [57]. The main issues for the blanket coolants have been described in [2] together with the results from preliminary design work. The requirements of the DEMO BoP are very demanding in comparison with the similar systems of a fission power plant (NPP). Different cooling fluids, different temperatures and pressures and pulsed operation represent significant challenges to the design of the heat transfer and conversion system as well as the very large and, in part, pulsed electrical power requested by the different electrical loads necessary for the fusion reactor (several times bigger than the electrical power requested in a nuclear or conventional power plant) [58]. Any effort to reduce the complexity of the DEMO BoP, through simplification and a rationalization of the design and operation of the main reactor systems are expected to have beneficial returns on the design of BoP systems, the safety the operation of the plant and ultimately of the costs.

Work is ongoing with the strong support of relevant industry, for both options of helium and water as coolants for the breeding blanket to advance the design of PHTSs, Intermediate Heat Transfer System (IHTS) and PCS and to assess the readiness of the technologies postulated for a plant that operate with an Energy Storage System (ESS) [59–62]. Fig. 7 shows the layout and Table 4 summarises the main characteristics of main BoP equipment for the case with helium and water, respectively. Such work is useful to: (i) assess dimensions of main components (e.g. HEX, circulators/ pumps, pipes, collectors); (ii) identify technical feasibility issues; (iii) understand commercial availability and R&D needs; and (iv) establish layout requirements and evaluate integration implications with other systems.

An attractive alternative design option is being investigated

Table 4
Representative characteristics of main BoP equipment.

	HCPB	WCLL
BoP main equipment		
- BB IHXs/	8	N.A.
- BZ OTSGs	N.A.	2
- FW IHXs	N.A.	2
- IHTS HCSGs	4	4
- VV HXs	2	2
- Div PFCs HXs	2	2
- Div Cassettes HXs	2	2
- Pressurizer	6	8
- MS tank	2 (3000 m^3 each)	2 (11,000 m^3 each)
Overall piping length (km)		
- BB PHTSs (both BZ and FW)	~3	~2
- Div PHTSs (both PFCs and Cassettes)	2.3	2.3
- VV PHTS	1.5	1.5
- IHTS	0.75	1.2

Acronyms. BB: Breeding Blanket; Div: Divertor; VV: Vacuum Vessel; PFCs: Plasma Facing Components; PHTS: Primary Heat Transfer System; IHTS: Intermediate Heat Transfer System; BZ: Breeding Zone; FW: First Wall; IHX: Intermediate Heat eXchanger; OTSG: Once-Through Steam Generator; HCSG: Helical Coil Steam Generator; MS: Molten Salt.

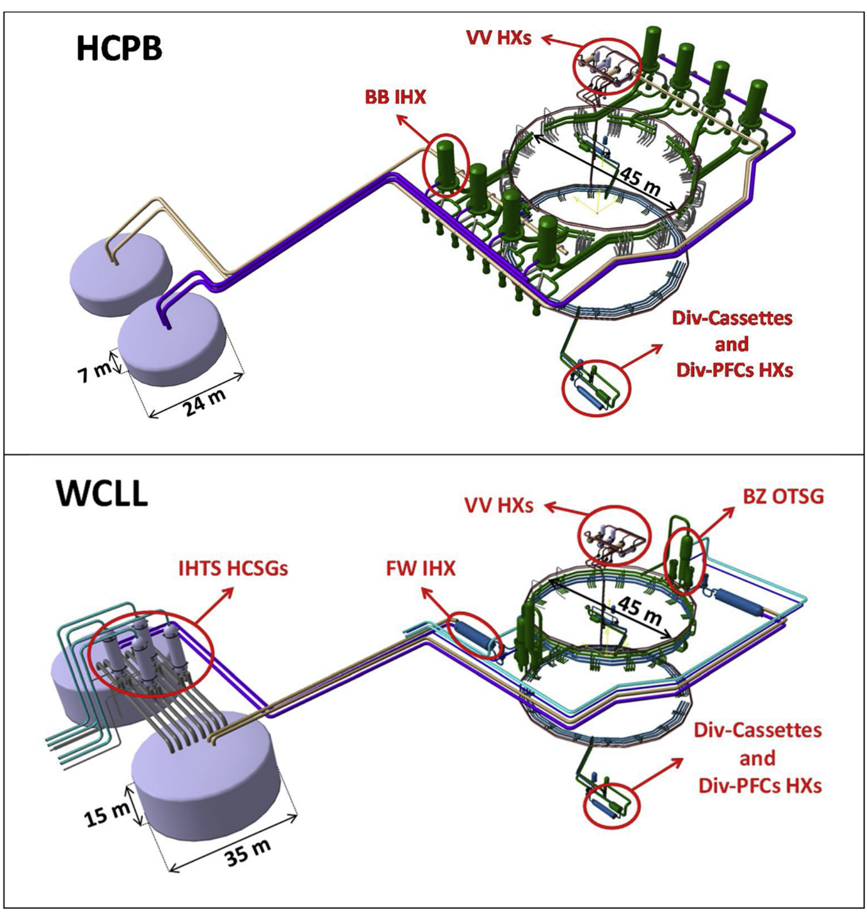

Fig. 7. Layout of PHTS and IHTS for HCPB and WCLL BB: Breeding Blanket, Div: Divertor, VV: Vacuum Vessel, PFCs: Plasma Facing Components, IHTS: Intermediate Heat Transfer System, BZ: Breeding Zone, FW: First Wall, IHX: Intermediate Heat eXchanger, OTSG: Once-Through Steam Generator, HCSG: Helical Coil Steam Generator.

providing a more direct coupling of the PHTS to the PCS with a small ESS. In this case, only about 10% of nominal flow would be used by the steam turbine during the dwell period and a much smaller storage of molten salt (HITEC) would be required.

6. Concluding remarks

The performance and reliability of breeding blanket systems represent the foremost considerations in the successful development and deployment any future fusion devices using a DT fuel cycle after ITER. DEMO or any other future conventional aspect ratio nuclear fusion device after ITER would need a breeding blanket system able to produce and recover reliably its own fuel from the very beginning of operation. Despite its criticality to the penetration of fusion in the electricity market, no breeding blanket has ever been built or tested. Hence, its integrated functions and reliability represent a very challenging development. This paper described the recently revised design and development strategy of the DEMO breeding blanket in Europe. This has been defined taking into account: i) the ultimate objective of the EU Fusion Program to demonstrate, and subsequently deploy, fusion electricity to the grid as soon as is feasible, with a target date for demonstration around the middle of this century; ii) the input from the DEMO pre-conceptual design activities, which has shed light on the importance of integration aspects associated to the breeding blanket and the interfacing BoP systems, often neglected in the past; iii) the findings and recommendations of a thorough technical and programmatic assessment of the breeding blanket program and the EU ITER TBM program, conducted in 2017 by an independent expert panel.

The testing of the TBM represents a critical step toward validating the principles and technologies of T self-sufficiency. However, even with a successful exploitation of the TBM program in ITER, gaps are expected to remain to validate the DEMO blanket design, particularly due to the difference in the neutron loads as well as in tritium generated, extracted and processed so that a parallel vigorous R&D is required especially on neutron material irradiation. But without the testing experience and knowledge gained from the full deployment of the ITER TBM program, the risk of proceeding to a prototype device like DEMO requiring tritium breeding is unacceptably high. Thus, the maximum technological risk minimization on a consistent and timely decision on the DEMO construction is obtained if the TBM program covers a wide enough combination of coolants, breeding materials and technology design options that are considered attractive breeding blanket design for DEMO.

In addition, it is important to recognize the importance of the gradual increase of the involvement of industry in the design and monitoring process from the early stage to ensure that early attention is given to industrial feasibility, costs, nuclear safety and licensing aspects, and the strengthening of international collaboration to better exploit synergies and minimize duplications.

Acknowledgements

Most of the strategy aspects described in this paper are the result of an effort that has involved a large number of individuals. The authors wish to thank and acknowledge, on behalf of the EUROfusion Consortium and Fusion for Energy, the outstanding effort of the member of the independent expert panel composed of M. Gasparotto (Chair), H. Ait Abderrahim, G. De Antoni, W. Korn, E. Salpietro, F. Wagner, S. Zinkle. We are also very thankful to A. Grosman (CEA), K. Hesch (KIT), A. Ibarra (CIEMAT), A. Pizzuto (ENEA), M. Shannon (CCFE), G. Veres (Wigner), L. Vala (CV Rez), D. Maisonnier (EU Commission DG RTD), L. Petrizzi (EU Commission DG ENER) for supporting the work of the expert panel and endorsing their recommendations, as well as for setting the foundation to address important management and governance aspects that support this proposal and ultimately will contribute to the efficient implementation of the program. The quality of their work and the important recommendations that they have provided has played an important role in reshaping and strengthening the DEMO breeding blanket / TBM Programs in Europe.

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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