Fusion Engineering and Design 112 (2016) 527–534

Contents lists available at ScienceDirect

Fusion Engineering and Design

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

Issues and strategies for DEMO in-vessel component integration

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ARTICLE INFO

Article history:
Received 18 January 2016
Received in revised form 5 April 2016
Accepted 22 May 2016
Available online 2 June 2016

Keywords:
DEMO
Tokamak
In-vessel components
Breeding blanket
Design integration

ABSTRACT

In the frame of the EUROfusion Consortium activities were launched in 2014 to develop a concept of a DEMO reactor including a large R&D program and the integrated design of the tokamak systems. The integration of the in-vessel components (IVCs) must accommodate numerous constraints imposed by their operating environment, the requirements for precise alignment, high performance, reliability, and remote maintainability. This makes the development of any feasible design a major challenge. Although DEMO is defined to be a one-of-a-kind device there needs to be in addition to the development of the IVC design solutions a remarkable emphasis on the optimization of these solutions already at the conceptual level. Their design has a significant impact on the machine layout, complexity, and performance. This paper identifies design and technology limitations of IVCs, their consequences on the integration principles, and introduces strategies currently considered in the DEMO tokamak design approach.

1. Introduction

The EU fusion roadmap Horizon 2020 [1] views a Demonstration Fusion Power Reactor (DEMO) to follow ITER as the remaining crucial step towards the exploitation of fusion power. It advocates a pragmatic approach considering a pulsed tokamak based on mature technologies and reliable regimes of operation, extrapolated as far as possible from the ITER experience. It foresees the development of a conceptual DEMO design by 2020. The three most important requirements for DEMO are defined in [1] as:

1. Achieve tritium self-sufficiency.
2. Demonstrate the production of net electricity.
3. Demonstrate all technologies required for the construction of a commercial fusion power plant, including an adequate level of availability.

Some rationales of these and consequences on the tokamak configuration are described in [2] and [3].

The conceptual design of DEMO must include amongst others a solution for (i) the vacuum vessel supporting the IVCs, (ii) the problem of the heat exhaust on the divertor and the first wall (FW), and (iii) for a breeding blanket achieving tritium self-sufficiency. For risk mitigation four blanket concepts based on different technologies are being developed in parallel throughout the conceptual phase. These are either based on ceramic pebble beds or liquid metal (LiPb) to breed tritium and are water-, helium-, or dual (helium/LiPb) cooled [4].

2. Key design requirements

2.1. Vacuum vessel

The vacuum vessel (VV) confines the radioactive inventory, ensures vacuum, and supports the IVCs. Furthermore it is the main contributor to two functions it shares with the IVCs: to shield the superconducting coils from neutron radiation and to provide passive plasma stability.

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http://dx.doi.org/10.1016/j.fusengdes.2016.05.040
0920-3796/

2.2. In-vessel components

Provide a plasma-facing surface: The surface facing the plasma is composed of the divertor plasma-facing components (PFCs) and the first wall (FW) of the blanket. This surface is exposed to heat loads due to high neutron flux as well as radiation and particle influx. The latter applies in particular to the divertor targets that intersect the scrape-off layer. All PFCs therefore need to be actively cooled and designed for efficient heat removal, be made of irradiation resistant materials, and be armored (in DEMO with tungsten).

Breed tritium: All tritium required to fuel the plasma must be bred within the IVCs. The Tritium breeding ratio (TBR) is the ratio of tritons generated in the blanket to the neutrons generated in the plasma and must be larger than 1.10 to ensure tritium self-sufficiency for DEMO [5]. Currently, tritium is bred only in the blanket. No breeding function is presently implemented in the DEMO divertor to allow for simpler design and integration solutions.

Exhaust heat to allow for efficient energy conversion: Almost all heat in a fusion reactor (>95%) is generated in the IVCs, see Fig. 5. In order to allow for efficient energy conversion into electricity the coolant temperature must be high, >300 °C, and the outlet temperature well controlled. For improved efficiency the coolant temperature should be increased further, the cooling channel hydraulics be optimized to minimize the required pumping power, and the temperature rise in the IVC should be increased to reduce the required mass flow to remove the heat.

Provide neutron shielding: The IVCs must provide sufficient shielding of the VV. The neutron shielding performance of the blanket is not verified against the neutron irradiation limits of the superconducting (SC) coils. Radiation protection of the SC coils is instead ensured by properly defining the VV thickness. Not only is the water/steel composition of the VV a more efficient neutron shield but also the investment and operational costs of increasing the VV thickness are much lower compared to an increase of the breeding blanket thickness, see also Section 3.4.

Contribute to passive vertical stability: Due to their toroidal segmentation the IVCs contribute only little to the plasma vertical stability.

2.3. Maintenance

The risk of failure of any in-vessel component cannot be excluded due to the design complexity, the numerous and cyclic loads, and the risk of the plasma damaging plasma-facing components. Therefore all IVCs must be replaceable, in-situ repairable or redundant in their function. In-situ repair is considered impractical for many applications due to the high dose environment that not only prevents human access but also imposes limits in the choice and functionality of remote handling (RH) equipment.

3. Vacuum vessel design concept

3.1. Design and operating condition

The VV is a fully welded toroidally continuous double-wall structure made of a conventional austenitic stainless steel: 316L(N). As in ITER neutron shielding plates are stacked in the interspace between its inner and outer shells. The nuclear heating in the vessel is removed by water serving also as moderator. In order to avoid regular vessel baking cycles at 200 °C (as required for the ITER VV operated at 70 °C) and to reduce thermal expansion relative to the IVCs, the operating conditions of the DEMO VV coolant are 200 °C at 3.15 MPa. The neutron irradiation limit of the DEMO vessel is chosen based on the limit for the negligible irradiation damage applicable to the VV austenitic steel defined in RCC-MRx AFCEN Edition 2012 as 2.75 dpa.

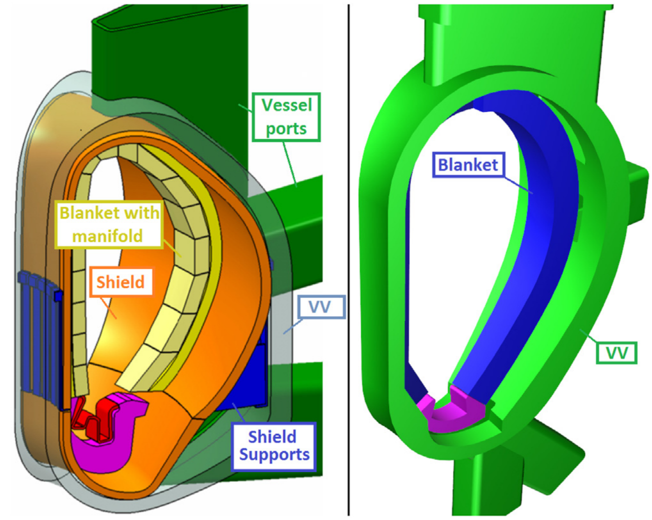

Fig. 1. 2014 DEMO configuration with shield (left) and 2015 configuration without shield (right).

3.2. Inboard wall design limitation

In order to prevent damage to the superconductor in case a quench in a TF coil is detected the coil current is rapidly reduced using a dump resistor. Consequently a poloidal current is induced in the vessel, I_{pol}, the magnitude of which is inversely proportional to the TF coil discharge time constant, τ_{CQ}, since the current decay is slow with respect to the vessel time constant (~1 s). I_{pol} reaches its peak in the initial phase of the TF coil fast discharge (TFCFD) when the toroidal field is still strong. The resultant Lorentz forces $B_{tor} \times I_{pol}$ cause a pressure load on the vessel that is strongest on the inboard. The options to increase the vessel strength are very much limited for two reasons: (i) when increasing the thickness of the vessel shells the current induced in the vessel I_{pol} increases roughly proportionally; hence the stress level remains unaffected. (ii) The pressure causes a hoop stress in the vessel inboard wall, a loading for which the (circular) vessel structure is already optimized. Consequently, a limit had to be defined in DEMO for the minimum TF coil discharge time ($\tau_{CQ} \geq 28$ s [6]), which in turn required the conductor's copper fraction to be increased to serve in case of a quench as temporary thermal buffer for storing magnet energy until the protection circuit is activated.

3.3. Elimination of the in-vessel shield

Previous international DEMO/fusion power plant studies, [7–9], considered a semi-permanent in-vessel shield: a shell structure within the vessel providing support to the IVCs and neutron shielding to the vessel and superconducting coils, actively cooled to a temperature similar to that of the IVCs, see Fig. 1. The impact on the tokamak design complexity of implementing a shield is significant since it must be a toroidally continuous structure in order to support the IVCs against the significant in- and out-of-plane EM loads. As it would be an in-vessel component its design would need to allow for removal and therefore separation into segments. The required mechanical joints would impose significant technological challenges and are currently considered impractical in DEMO.

At the same time the VV provides the required nuclear shielding of the superconducting coils and due to its robust and toroidally continuous design is well suited to support the IVCs. Hence no shield is integrated inside the DEMO plasma chamber, the drawback being that the VV radiation loads are mitigated only by the IVCs.

3.4. Nuclear shielding of the vacuum vessel

Presently the DEMO divertor cassette is a water-cooled steel box as in ITER, which efficiently shields the VV from neutron radiation. The breeding blanket shows poorer shielding performance compared to the divertor as it is designed to minimize neutron absorption to allow for a high TBR [4] and does not contain large amounts of efficient neutron moderators. In spite of the DEMO breeding blanket being about twice as thick compared to the ITER shielding blanket neutron transport assessments [10] indeed found the nuclear heating of the vessel inner shell behind the inboard blanket about one order of magnitude higher than in ITER. A corresponding thermal-structural assessment of the VV inner shell found thermal stresses exceeding the stress limit [11]. Consequently a reduction of the inner shell thickness was recommended to reduce the generated nuclear heat load. This initial result indicates that during the design development of the DEMO vessel the hydraulic conditions providing efficient cooling of the inner shell will play a more important role as compared to the ITER vessel.

The dpa damage in the vessel inner shell is predicted for the different blanket concepts to be below ~0.2 dpa/full power year (fpy) [12]. Hence the dpa damage limit of 2.75 dpa is – with the current IVC configuration – not exceeded at the end of the envisaged DEMO lifetime (~6 fpy).

The neutron induced radioactivity in the vessel material (SS-316) produces medium-long life radioactive nuclides, mainly due to the nickel, cobalt, molybdenum and niobium contents. The contribution of the VV to the overall DEMO radioactive waste is therefore significant, in particular more than 100 years after the end of operation when the activity of the IVCs (made of reduced-activation Eurofer) will have decayed significantly [13].

All three issues associated with the nuclear shielding of the VV provided by the breeding blanket: (i) the nuclear heating, (ii) the material damage, and (iii) the activation are critical to the configuration without shield. The nuclear shielding performance of the breeding blanket must therefore be carefully monitored during the conceptual design phase.

4. Design configuration of in-vessel components

4.1. Configuration to ensure tritium self-sufficiency

The blanket is the only system breeding tritium. The design of a breeding blanket aims at the arrangement of breeder and multiplier materials in proximity of the plasma in order to maximize tritium generation. Parasitic losses of neutrons being absorbed by structural materials need to be minimized. The FW protects the breeding zone from plasma heat and particle influx but also absorbs neutrons back-scattered from the breeding zone reducing the tritium breeding rate. The FW design has therefore been optimized towards a minimization of its content of neutron absorbing materials, i.e. steel and tungsten, see also Section 5.3.2. The major fraction of parasitic neutron absorptions occurs in the steel structure of the breeder modules. In current designs ~11–15 vol% of the breeding zone are occupied by steel structures, which deteriorate the breeding performance.

In a dedicated study the Tritium breeding performance potential was investigated regarding the contribution of individual poloidal sections to the TBR (Fig. 2) [14]. This study has concluded in the DEMO divertor to be reduced to a minimum, eliminating the baffle region present in the ITER divertor, see Fig. 4. It shows the potential to achieve tritium self-sufficiency even in a double-null (DN) configuration with a 2nd divertor in the top of the machine. Studies regarding physics issues associated with a DN configuration are underway [15].

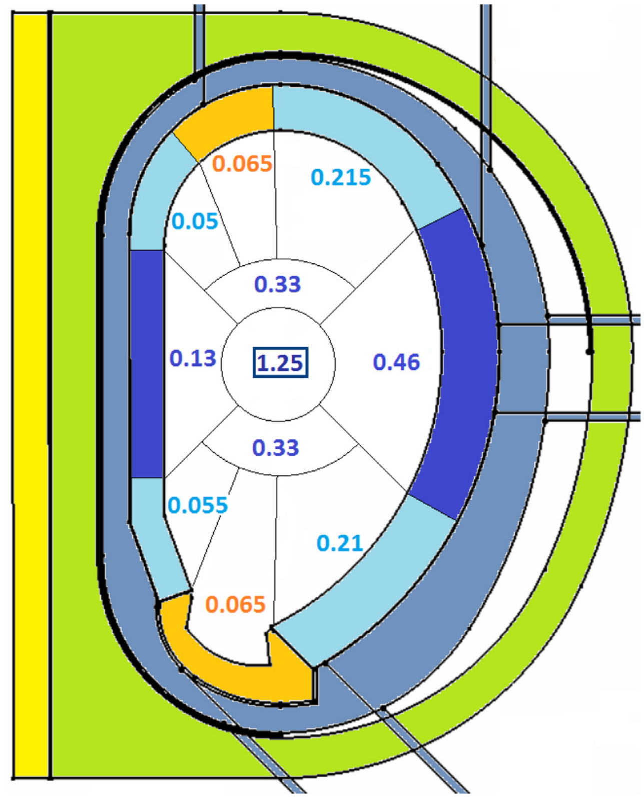

Fig. 2. Tritium breeding ratio potential in four 90° poloidal sections and subdivisions thereof based on DEMO-typical tritium-breeding IVCs and not considering any loss of breeding volume due to the integration of port plugs.

4.2. Size of in-vessel components

4.2.1. Blanket

It is in general desired to reduce the radial thickness of the blanket as this would have a number of advantages, particularly regarding remote maintenance, plasma stability (see Section 3.6), tokamak size, EM loads, and investment cost. The blanket thickness is mainly due to:

1. The size required for the breeding zone: Depending on the blanket concept a breeding zone of typically 40–80 cm thickness is needed to achieve the required TBR [14].
2. The size required for the blanket manifold feeding the FW and the breeding units, which is similar to that required for the breeding zone.

The toroidal magnetic field decreases radially with 1/R. A configuration with a slimmer inboard blanket shifts the plasma into the higher field region allowing for a higher plasma current and improved confinement. The consequent enhanced plasma performance would allow the major radius to be reduced. The beneficial effect is however compensated to some degree by the higher required plasma current and the consequent increase of the solenoid that provides the flux swing.

The thickness of the blanket was set in DEMO to about 80/130 cm, inboard/outboard, respectively. This configuration was found to be suitable to achieve the required TBR [12]. It was

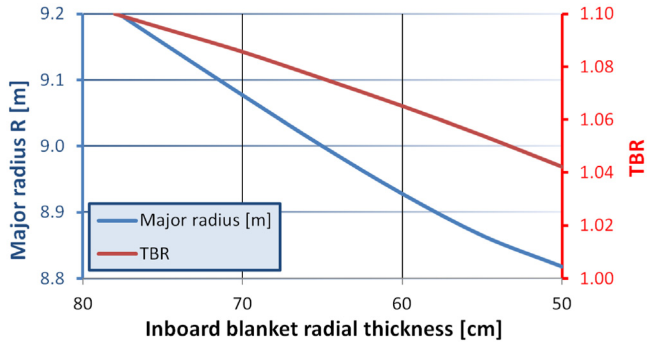

Fig. 3. Effect of the reduction of the inboard blanket radial thickness on the plasma major radius and the TBR assuming an IVC configuration achieving a TBR of 1.1 with an inboard blanket of 78 cm thickness.

accepted that the increased thickness of the outboard blanket compromises the plasma vertical stability, see Section 5.4. A further reduction of the DEMO inboard blanket thickness assuming no increase of the outboard blanket was studied regarding the favorable effect on the major radius and the unfavorable effect on the TBR, see Fig. 3. This modification could therefore be implemented only in case the TBR reduction could be compensated, e.g. by reducing the divertor size and increasing the breeding blanket size. Whereas the dimension of the FW was considered as a constant in this study, the blanket manifold was assumed to shrink proportional to the reduction of the breeding zone. The latter assumption might not be conservative and requires substantiation by design. For the current DEMO configuration it was found that a reduction of the blanket thickness Δt_{BLK} allows a reduction of the major radius $\Delta R \sim 1.35\ \Delta t_{BLK}$.

A reduction of the inboard blanket thickness might require however the integration of comparably small amounts of dedicated shielding material, e.g. tungsten carbide, in the back of the blanket.

4.2.2. Divertor

The design of the divertor cassette aims mainly at providing the following three functions: (i) support inner and outer targets and the dome, (ii) neutron shielding to the VV, and (iii) facilitate vacuum pumping.

The divertor cassette being a water-cooled (and hence water–filled) steel box with internal ribs and possibly internal shielding plates efficiently shields neutrons. It can therefore be significantly thinner than the breeding blanket. To allow the flow of gases from the divertor into the lower port and towards the vacuum pump a cut-out in the cassette body is required, see Fig. 4 The divertor dome must be designed in order to reduce neutron streaming onto the vessel through this duct. As described in Section 3.3 the poloidal extension of the DEMO divertor was reduced to enlarge the blanket. In order to further reduce the size of the divertor the following two approaches offer some potential:

1. In contrast to ITER it is aimed in DEMO to avoid the integration of separate plasma-facing units onto the cassette body, see Fig. 4. Instead the possibility to join the vertical target PFCs directly to the cassette body is being investigated. This approach would simplify the divertor design and save the space required for the plasma-facing units and their supports. The possibility to replace the PFCs in the hot cell and reuse the cassette body would not necessarily be compromised.
2. The vertical and poloidal size of the divertor is driven by the distance between the x-point and the strike points on the target plates. Currently 1.0 and 1.35 m are considered for these distances to the inner and outer target, respectively. A reduction of these distances will however be limited by the required performance of the divertor to facilitate pumping and for power exhaust.

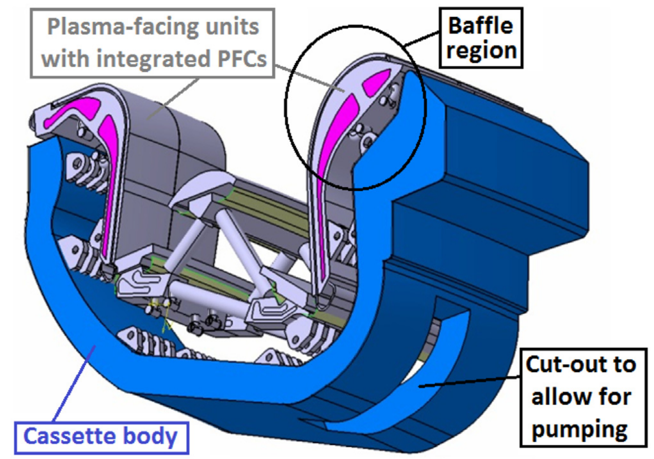

Fig. 4. Previous ITER-like configuration of the DEMO divertor including the baffle region.

4.3. Configuration to allow maintenance

The risk of failure of any in-vessel component cannot be excluded due to the design complexity, the numerous and cyclic loads, and the risk of the plasma damaging plasma-facing components. Therefore all IVCs must be replaceable or in-situ repairable. The latter is considered impractical due to the high dose environment that not only prevents human access but also imposes limits in the choice and functionality of remote handling (RH) technologies.

To allow extraction through the vessel ports all IVCs are divided into parts with individual feeding pipes. In order to reduce plant downtime when IVCs require exchange it is in general aimed at dividing the IVCs into large segments or cassettes. The maximum sizes of the ports are however defined by the cage formed by toroidal and poloidal field coils. Furthermore the backside of all IVCs should be accessible through at least one port to facilitate cutting and rejoining operations of feeding pipes. These principles have led to the division of the blanket in three outboard and two inboard segments, see Fig. 7 and three divertor cassettes per TF coil. These are vertically handled through large upper ports [16]. This approach had also been adopted in 1988 in the Next European Torus (NET), [18].

Due to the segmentation of IVCs disruptions typically cause eddy current loops within single IVCs. The consequent EM loads, e.g. the radial moments, could in principle be reacted amongst the IVCs. The required structural connections between IVCs would however obstruct the kinematics required for their removal during maintenance. Therefore the main supporting function of each IVC is individually provided by the VV from the back and even equal opposite forces acting on adjacent IVCs are transferred through the vessel.

4.4. Configuration for heat exhaust

4.4.1. Power distribution in DEMO

Exhausting the power generated in the plasma is the ultimate goal of operating a fusion machine. To allow for efficient conversion of the heat into electricity it is aimed at removing the heat with high temperature coolant. The technologies selected for the IVCs however impose upper limits on the coolant temperature:

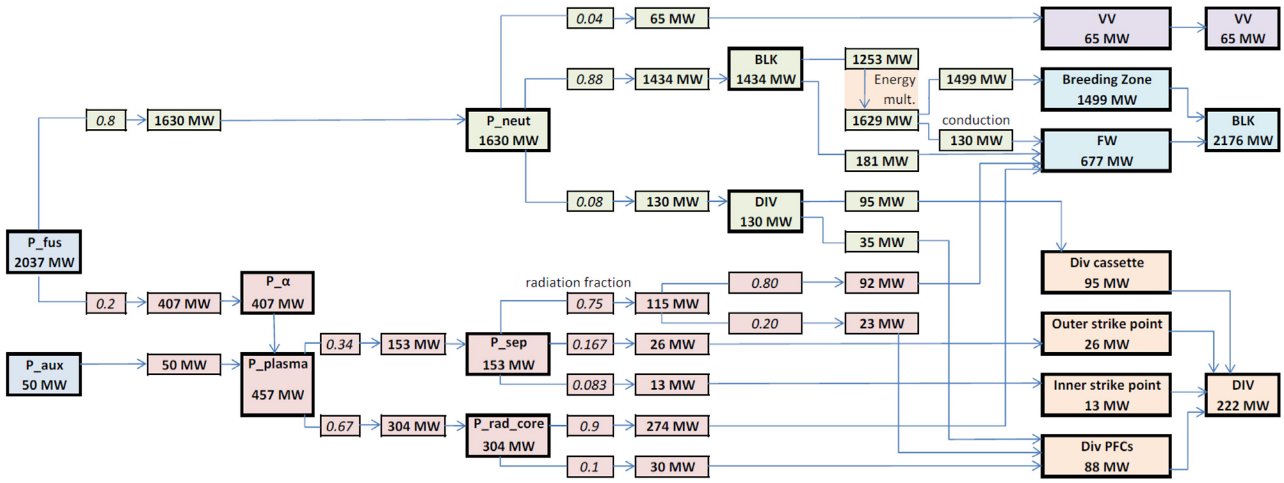

Fig. 5. Power generation during flat-top operation in the DEMO plasma and its division into the three forms by which energy leaves the plasma: as neutrons (green boxes), particles (P_sep), and radiation (P_rad_core). Furthermore: conversion of particle energy into radiation in the divertor (radiation fraction), energy multiplication due to interaction of neutrons with multiplier materials, and possible energy distribution to VV, blanket and FW, divertor cassette and divertor PFCs. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

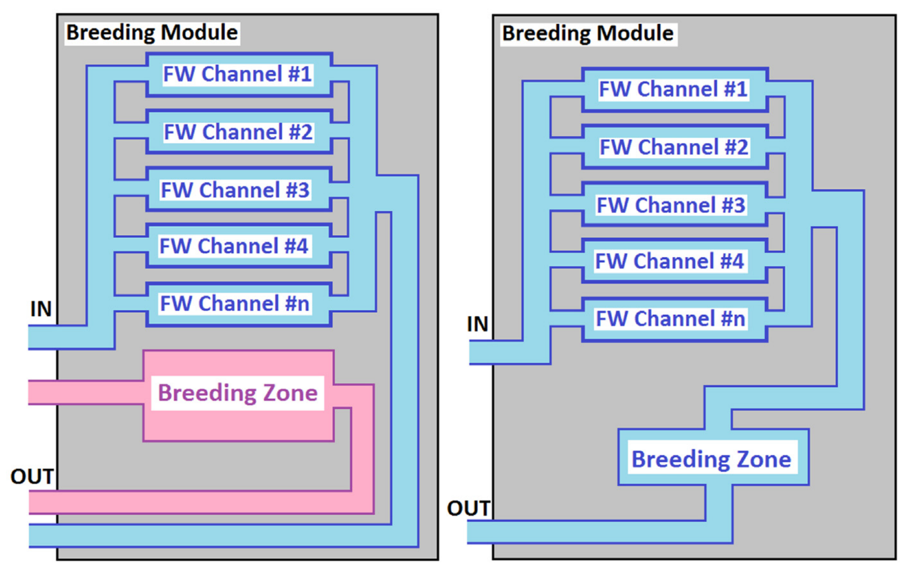

Fig. 6. Breeding blanket cooling scheme with two separate cooling loops for breeding zone and FW (left), and a single cooling loop (right).

Copper alloys soften at high temperatures. CuCrZr, used in ITER, softens at temperatures >300–350 °C [19]. Eurofer instead maintains sufficient strength up to >500–550 °C [20]. Since large temperature gradients occur in PFCs due to the high heat loads the temperature of their structural material may be ~100 °C higher than the coolant, requiring the coolant temperature of copper-based PFCs to be below ~250 °C. In contrast a Eurofer-based FW – the standard technology considered in DEMO – allows coolant temperatures of up to ~380 °C. In the breeding zone the energy density and therefore the temperature gradients in the cooling channels are comparably low. Here the coolant temperature can be close to the material's limit temperature, hence ~500 °C.

In case liquid water is chosen as coolant it must be increasingly pressurized by raising temperatures to prevent boiling. This is possible up to 374 °C. For the DEMO water-cooled blanket the same operating condition was chosen as in pressurized water reactors (PWR): up to 325 °C at a pressure of 15.5 MPa. Gaseous coolants like helium naturally do not have an upper temperature limit but have much lower heat removal efficiency.

The challenging heat loads acting in particular on the PFCs require trade-offs being made between heat exhaust efficiency and heat load capacity. The choice of technologies for the different components is described in Section 4 below. These trade-offs must also consider the amount of heat removed by each individual component compared to the total heat generated in the reactor. Fig. 5 shows one possible scenario for the distribution of heat to the individual components. For some of the indicated distributions and in particular for the distribution of radiation and particle heat across the PFCs (red boxes) there are significant uncertainties and arbitrary choices have been made. The values provided in Fig. 5 should therefore not be mistaken for the heat load maxima to be considered in the design of the individual components. Despite the uncertainties the figure highlights the primary significance of the breeding zone, the secondary significance of the FW and the moderate importance of the divertor in the overall power balance. Finding a solution for the power flux density at the strike points is an extreme challenge, [21]. It is hence logical to accept the most significant compromises in the cooling of these zones regarding the achievable conversion efficiency.

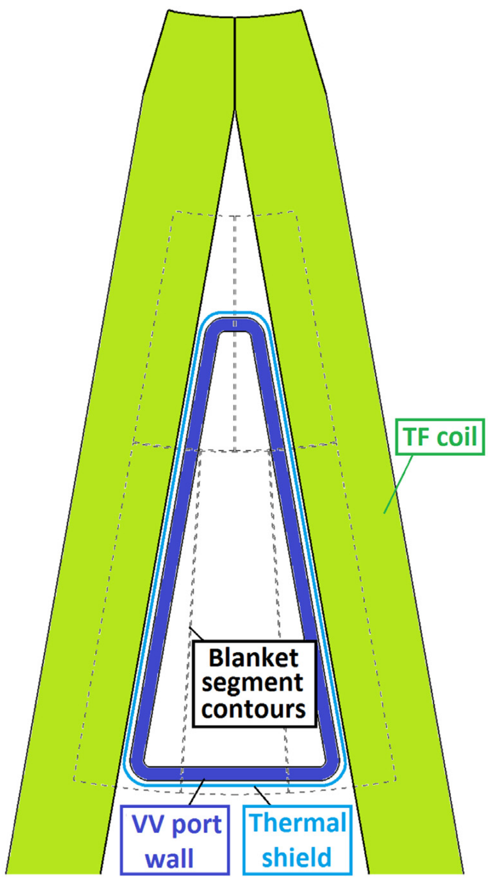

Fig. 7. Horizontal cut through one DEMO upper port with adjacent TF coils limiting the port size and contours of the 3 outboard and 2 inboard blanket segments.

Two blanket cooling scheme principles are currently considered, Fig. 6. In helium-cooled concept the in-series cooling scheme allows heating the coolant up first in the FW to ∼380 °C and then in the breeding zone to ∼500 °C. This has the potential to reach the desired outlet temperature in the entire blanket coolant. On the downside the mass flow must be equal in FW and breeding zone, a constraint that might be difficult to meet since the ratio between FW heat load and breeding unit heat load might vary strongly in different poloidal locations. If on the other hand the FW is cooled by a separate cooling loop the cooling conditions can be defined with greater flexibility, however the outlet temperature of the FW loop is limited. This limitation mainly affects the helium-cooled blankets that aim at outlet temperatures of ∼500 °C, which cannot be reached with water as a coolant, see above.

5. Key issues

5.1. Remote access to in-vessel components

Due to the high energy neutrons produced in D-T fusion plasmas and the high neutron fluence in DEMO, helium production and material neutron damage is high in the IVCs. Reliable re-welding of pipes could therefore only be achieved at the backside of the IVCs. In the ITER blanket the feeding pipes are accessed from the front through the FW, which also requires space around the pipes to allow for orbital cutting and joining. Front-side access would be practical also to maintain diagnostic sensors, or to release IVC support structures or electrical straps at the back of IVCs. However, in DEMO the IVCs and the vessel become activated due to the high neutron fluence, and cause a dose rate in the plasma chamber that would severely decrease the lifetimes of current viewing and weld inspection tools to some 100 s of hours [22]. In addition front-side access would require access holes penetrating the FW panel. The FW cooling channels in DEMO are however relatively small and their spacing was reduced to a few millimeters to avoid temperature peaks in the FW. The size of any penetrations that could be integrated between two adjacent cooling channels is therefore very limited. Solutions requiring pipe cutting and re-welding operations carried out at the back of the IVCs with front-side access are therefore considered impractical [17].

Instead all IVC pipework is designed for cutting and re-welding either inside the better shielded VV ports or at the back-side of the IVCs using in-bore tools operating in a lower dose environment. Replaceable FW panels as in ITER are therefore not foreseen in DEMO.

5.2. Space in the upper port

Much of the toroidal space in the area of the upper port is taken up by the TF coils. In addition a thermal shield must be integrated between the port and the coils with the necessary clearances. Independent of the poloidal location, the sizes of the TF coil, thermal shield, and port wall are constant and towards the inboard increasingly narrow down the remaining space inside the port. In ITER the vessel ports are on the outboard where the distance between the TF coils is large. The DEMO upper port though is rather narrow in particular on its inboard side and space is tight for the removal of the large DEMO blankets. However, reducing the toroidal size of the TF coils and hence of the winding pack is not considered. To maintain the winding pack's cross section its 2nd dimension would need to be increased. On the inboard this would require an increase of the machine radial build. In an attempt to enlarge the port internal space the feasibility of implementing a single plate of steel as toroidal port sidewall instead of the standard double-wall structure is being studied.

5.3. Plasma-facing components

5.3.1. Divertor plasma-facing components

In DEMO an ITER-like divertor target technology is being considered as a reference based on a water-cooled tungsten mono-block with a copper alloy heat sink. To enhance the performance and to comply with the DEMO conditions advances of this the target concept are studied making use of advanced materials [23].

This concept was chosen given its superior performance under high heat loads thanks to the high thermal capacity of liquid water and the small temperature gradient through the pipe wall due to the very high thermal conductivity of copper. These PFCs must be operated at a pressure significantly higher than the saturation pressure corresponding to the water temperature. This precaution prevents local water boiling and the resultant loss of heat removal in the event of heat load excursions, e.g. due to loss of detachment [24]. In order to realize a reasonable margin against this burn-out phenomenon and at the same time keep the operational pressure reasonably low the coolant temperature of the divertor PFCs was defined relatively low (150 °C). This choice also prevents excessive corrosion [25] and material softening. Advanced composite Cu-alloys are expected not to soften up to higher temperatures. On the downside the relatively low coolant temperature does not allow for an efficient conversion of the removed heat into electricity. Given the relatively small amount of heat removed by the divertor PFCs (∼5% of the total power generated in DEMO, see Fig. 5) the poor electricity conversion efficiency was accepted in favor of performance and technology readiness. The loss of ductility of CuCrZr irradiated a low temperature [26] is a concern but the material degradation of CuCrZr in the DEMO target plates is not expected to differ substantially from that in ITER.

5.3.2. First wall

The default FW technology in DEMO is a tungsten-coated Eurofer plate with parallel cooling channels. The low thermal conductivity of Eurofer limits the heat load capability of this technology to ~1–1.5 MW/m^2 [3]. For comparison the entire ITER FW is based on CuCrZr and is designed for significantly higher heat loads (up to 5 MW/m^2 [26]). The reasons are (i) incentive to retain sufficient operation flexibility being ITER an experimental device, (ii) penalty factors to account for misalignment and non-uniform heat load distribution due to shaping of the FW, and (iii) sufficient margins to accommodate for off-normal thermal transients.

The DEMO FW must be configured aiming at distributing the heat loads as uniformly as possible across the FW and along each single FW channel to reduce local peaks. The FW should therefore be precisely aligned and its shape follow the plasma as smoothly as possible. In addition the distance between FW and plasma was, compared to ITER, increased to 22.5 cm to reduce the FW heat flux variations during changes of plasma shape and position within the control limits and in off-normal thermal transients. Nonetheless, areas with particularly high heat loads or erosion rates may exist. Hence plasma limiters with Cu-alloy based PFC technologies and designed for a more frequent replacement than the blanket might be required [28].

Compared to ITER the DEMO FW must fulfill the additional function to efficiently exhaust the heat for electricity production. This requires on one hand the cooling condition at every point of the FW channels to be close to the optimum and therefore – again – a uniform heat load distribution. Moreover possible temporary heat load excursions must be minimized since these would require raising the nominal coolant velocity to increase the heat load the FW can tolerate. This would increase the required pumping and hence recirculating power significantly. Furthermore the nominal FW outlet temperature would be below what is aimed at to enhance electricity conversion efficiency.

5.4. Passive plasma vertical stability

Good passive stability of the plasma in general terms is achieved by providing a toroidally continuous conductive shell in vicinity of the plasma. Most effective are those parts of such a shell where the radial magnetic field generated by the plasma current is large, i.e. on the outboard at both top and bottom. A good passive stability relaxes the requirements of the active control system regarding both the necessary power and the reaction time. A conductive shell is naturally provided by the vessel inner shell, even if locally discontinuous due to the presence of the vessel ports. In a tritium breeding reactor the vessel inner shell is relatively far from the plasma due to the large radial size of the breeding blanket. The consequent poor vertical stability [29] needs to be compensated by one or more of the following options:

i Small plasma elongation,
ii Small aspect ratio,
iii Implementation of toroidally continuous vertical stability coils in the plasma chamber,
iv Integration of active or passive saddle coils in IVCs,
v Poloidally continuous first wall panel of the outboard blanket segments, or
vi Toroidal electrical connections between adjacent IVCs.

It is also recognized but not yet quantified that a double-null configuration might have a much improved vertical stability. The first two options have a significant impact on the tokamak design. In the current single-null configuration in order to guarantee adequate vertical stability the plasma elongation at the surface of 95% poloidal flux was reduced to k95 = 1.59, previously and the aspect ratio to A = 3.1, previously A = 4. As a drawback in order to maintain the fusion power an increase of the plasma volume was required. This again required an increase of the tokamak size and cost.

The integration of coils in the plasma chamber or IVCs is possible but increases the design complexity.

6. Summary and outlook

The additional requirements in DEMO compared to ITER for higher lifetime neutron fluence, to breed tritium and produce electricity require different solutions for IVCs. The DEMO blanket must be a breeding rather than a shielding blanket and cannot contain large amounts of efficient neutron moderators. Consequently the radial thickness of the DEMO VV must be about twice that of the ITER VV to sufficiently reduce the neutron fluence on the TF coils (on the inboard). Nonetheless and also due to the DEMO blanket being about twice as thick as the ITER blanket, early studies allowed the conclusion that the VV does not require an additional neutron shield behind the blanket for its neutronic limitations to be met, i.e. (i) material damage <2.75 dpa, (ii) thermal stresses in the inner shell due to nuclear heating within the design criteria, and (iii) minimized activation level (currently not quantified).

The high gamma dose rate in front of the DEMO PFCs severely reduces the lifetime of critical RH equipment, e.g. welding tools, to some 100 s of hours. This fact together with the difficulty to integrate access holes in FW and breeding modules has led to the exclusion of cutting and re-welding operations of FW, blanket or divertor service connections requiring front-side access. The backside of each IVC should therefore be accessible through at least one port to facilitate access to the feeding pipes. These principles have led to the division of the blanket into vertical segments and the adoption of the ITER approach, separating the divertor into three cassettes per TF coil.

Whereas the ITER Cu-alloy based technology of the divertor target might be feasible also in DEMO, the FW is based on Eurofer to demonstrate a power-plant-relevant technology with high irradiation lifetime, reduced activation, and allowing for high-temperature coolants. The low thermal conductivity of Eurofer and the limited heat flux capability of the DEMO FW are a primary concern in DEMO and the main risk to the currently adopted approach for the FW.

Acknowledgments

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 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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