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Overview over DEMO design integration challenges and their impact on component design concepts

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The EU fusion roadmap defines as one of its goals the development of a Demonstration Fusion Power Reactor (DEMO) to follow ITER. This device shall be tritium self-sufficient, produce net electricity, and acts as a component test facility to demonstrate fusion power plant relevant technologies, e.g. those of the breeding blanket.

This article identifies the main DEMO requirements, introduces the rationales for the selected architecture of the DEMO tokamak, and describes how the configuration of the main tokamak components has been derived. This includes (i) the DEMO shielding concept, (ii) the segmentation of the in-vessel components and their maintenance strategy, (iii) an overview of the vessel and in-vessel component technologies, and (iv) a description of the integrated design of the breeding blanket.

Keywords: DEMO, tokamak, in-vessel components, breeding blanket, design integration

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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 following are the most important mission requirements for DEMO, which are defined in [1] as:

1. Obtain license from nuclear authority
2. Demonstrate reliable plasma operation
3. Operate for several full power years
4. Achieve tritium self-sufficiency
5. Demonstrate the production of net electricity
6. 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 a DEMO plant must include amongst others sound solutions for (i) the problem of the heat exhaust on the divertor and the first wall (FW) plasma-facing components (PFCs), (ii) the breeding blanket (BB) achieving tritium self-sufficiency, (iii) the conversion of power into electricity, and (iv) the integration of in-vessel components (IVCs) inside the vacuum vessel (VV) providing support against all loads and allowing for their replacement via remote handling (RH) tools. This article is focused on the design of the tokamak, plant systems are described elsewhere [4], [5].

2. DEMO Architecture

2.1 Tokamak and Tokamak Building

The DEMO tokamak has the same basic architecture as other superconducting tokamaks, in particular ITER: The magnet system and the VV are structurally independent, i.e. without physical connections, and sit on a ring structure that is part of the cryostat. The cryostat is a large vacuum chamber at room temperature providing a vacuum environment to prevent heat transfer to the cryogenic magnet coils by convection. It is supported by the tokamak building. The TF coils with their intercoil structures provide structural support to the poloidal field and central solenoid coils. The VV provides support to all IVCs and the equipment inside the VV ports, e.g. port plugs and pipe work. The entire plasma surface is covered by a wall made up of blanket, divertor, and port plug PFCs with high heat flux capability. There are a number of small penetrations in the plasma-facing wall to allow access to the plasma for diagnostics and electron or ion cyclotron antennas and medium-size openings to provide conductance for vacuum pumping. In addition there are large openings in the wall (~0.5m²) to allow for injection of neutrals by the neutral beam (NB) systems. The NB duct sidewalls are protected from high heat loads by the NB liner. The large NB openings in the wall are also used to connect the plasma chamber to an expansion tank. In case of a leak into the VV of one of the high pressure cooling loops of the IVCs it is presently planned to use the same scheme as in ITER, i.e. the coolant is evacuated through the NB duct into the expansion tank to protect the VV from over-pressurization.

Feeding pipes for IVCs and auxiliary systems like vacuum pumps, most diagnostics, and heating and current drive systems are integrated in VV ports where the neutron flux and gamma dose rate is lower, accessibility is good and maintainability by remotely controlled tools is far less problematic as compared to the plasma chamber.

2.2 Shielding Concept

DEMO aims at achieving a high lifetime neutron fluence that corresponds to 70 dpa in the FW steel [2], about two orders of magnitude larger than in ITER. Its plasma will generate a neutron wall load of ~1 MW/m². These neutrons cause damage to materials, i.e. degradation of material properties. The high energy neutron irradiation produces helium inside the materials leading to swelling and also produces radioactive isotopes (in component materials and cooling water). These emit alpha, beta, and gamma radiation challenging the function of sensitive equipment and reducing accessibility. Hence there are multiple shielding requirements to be considered in the design of DEMO. These requirements and the currently considered design targets are summarized in Table 1.

The shielding concept of DEMO requires the IVCs to protect the VV from excessive material damage and to decrease its activation. Also the volumetric nuclear heating in the VV needs to be limited in order to avoid excessive thermal stresses in its inner shell. Since a repair of the VV is not foreseen no limit on the amount of helium produced in its stainless steel due to neutron irradiation needs to be considered to allow for re-weldability. In DEMO the most highly irradiated parts requiring re-welding are the cooling pipes of the IVCs inside the ports. However, the option of replacing these pipes together with the IVCs is also considered.

The neutron fluence level at the superconducting coils must be several (~5) orders of magnitude lower as compared to the VV inner shell. In order to meet this requirement steel plates are implemented into the space between the VV inner and outer shells. Given the mix of the VV steel and its cooling water of ~60/40% (Vol.) the VV thickness must be at least ~60 cm. Penetrations are not foreseen in DEMO from the in-cryostat area into the VV or its ports, which would create neutron streaming paths. However, the VV ports offer a potential streaming path for neutrons to escape the plasma chamber and reach the coils through the port walls. Such streaming paths are the main driver of the in-cryostat shutdown dose rate and no notable effect is expected of an increase of the thickness of the main VV body on the outboard beyond 80 cm. Two principal mitigations of this risk are considered: (a) implementing shield plugs within the ports, and (b) increasing the port wall thickness. The implementation of both options however is limited because most equipment integrated in the ports requires access to the plasma or the IVCs. Penetrations of the shield plugs are therefore inevitable and negatively impact on the shielding performance. Increasing the wall thickness of the VV ports would reduce the available space inside the ports. In particular in the cases of the upper and lower ports the space is very tight as these ports are closer to the inboard wall where the distance of the TF coils is small [6].

During plasma operation access for workers inside the entire tokamak building is not possible due to the neutron and gamma radiation. In addition to IVCs, VV, and port plugs/shield plugs other shielding structures are implemented in the VV ports in DEMO to allow man-access and to reduce the radiation level on RH equipment during maintenance phases, and to decrease the neutron fluence on electronic and electrical equipment during operation. These shielding structures are shown in Figure 1 and namely are (i) port closure plates, (ii) bioshield with bioshield plugs, and (iii) port cells with port cell doors. The following gamma dose rate limits are considered depending on the expected need for man-access:

- Frequent access: 10 µSv/h
- Occasional access: 100 µSv/h
- Exceptional access: 500 µSv/h

Due to the low neutron fluence limit of the superconducting coils the activation of all in-cryostat components is relatively low. Hence the radiation dose rate after machine shutdown is low and man-access shortly after shutdown for limited periods could be considered in the yet to-be-developed in-cryostat maintenance strategy for DEMO as in ITER.

Behind the port closure plate in the port interspace, see Figure 1, a relatively high neutron fluence is expected during operation and a relatively high gamma radiation level of several hundred µSv/h 12 days after shutdown. This is expected to be mainly due to penetrations. All regular operations in this area are foreseen to be carried out by RH tools. Man-access without restrictions will be possible during non-active phases anticipated for early operation with reduced plasma power and neutron generation; otherwise in exceptional cases and for short duration only. Electronic and electrical equipment in this area must be neutron and gamma radiation resistant or protected by designated local shielding structures.

The 2m thick bioshield made of borated concrete effectively reduces the neutron and gamma fluence coming from the plasma chamber to levels sufficiently low for frequent man access [7]. However, gaps between the bioshield plugs and the bioshield as well as penetrations through the bioshield plugs cause neutron and gamma streaming that can significantly increase the irradiation level inside the port cell. It has been identified in ITER that in particular penetrations without dog-legs providing a straight path are problematic. To what extent and whether at all maintenance operations in the port cell can be carried out by personnel will depend on the technology and detailed design choices of the bioshield plugs and the design of the individual penetrations (possibly including shield stubs).

It is expected that the neutron fluence and shutdown dose rate levels in the maintenance hall above the tokamak on top of the bioshield roof can be limited to allow for frequent man-access as long as the bioshield plugs are in place.

For most other areas within the tokamak building outside the port cells frequent man-access is currently foreseen. In order to allow for this, pipes carrying radioactive water coolant, LiPb, or liquid nitrogen are planned to be routed through separated corridors.

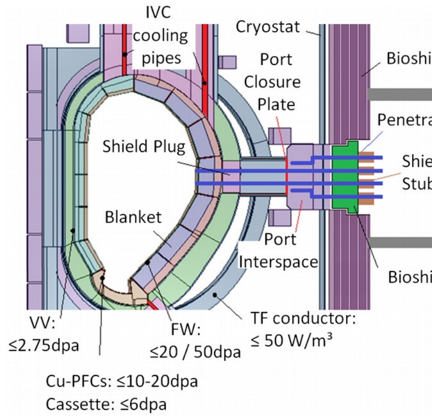

Figure 1 Shielding structures in DEMO and selected neutron irradiation limits

Table 1 Material damage limits, shielding requirements and design targets for shutdown dose rates in DEMO

Component/ location	Requirement
Starter blanket FW	Displacement damag Eurofer
2nd blanket FW	
Divertor cassette body (@ 180°C), [8]	
Divertor PFCs, [9]	Displacement damag CuCrZr
VV	Displacement damag
	Nuclear heating, [10
	Activation, [2]
Cutting/re-welding location in IVC cooling pipes	Helium production
Superconductors, [11], [12]	Total neutron fluenc epoxy insulator
	Fast neutron fluence the Nb3Sn
	Neutron fluence to C stabilizer between Tl warm ups

	Nuclear he winding pa
Port interspace	Shutdown days after s
Port cells (occasional access)	
Maintenance hall above tokamak	
In-cryostat area, [7]	
Tokamak building areas beyond port cells requiring frequent access, [7]	Shutdown day after sh
Critical electronic equipment	Neutron flu operation
Non-critical electronic equipment	

2.3 IVC maintenance and segmentation

Failure of IVCs cannot be excluded, see section 3.2. Some IVCs might be designed with redundancy as to allow continuation of plant operation without the need for replacement. All other IVCs must however be replaceable, which requires a corresponding remote maintenance plan. The shutdown dose rate in the plasma chamber in front of the PFCs is high, 8 weeks after plasma shutdown ~1000 Sv/h [13], see Figure 2. This corresponds to an absorption rate in silicon of ~900 Gy/h. For comparison the dose rate in the ITER plasma chamber during remote maintenance is lower by a factor of about 3 (several hundred Gy/h, [7]). The dose rate in the containment vessel of the Fukushima reactor was found in early 2017 to be 530 Gy/h, [14].

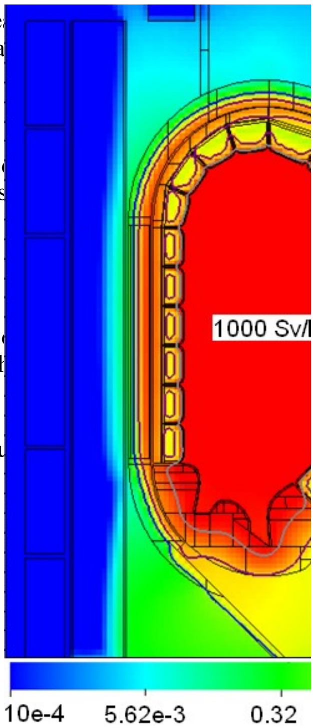

Figure 2 Gamma radiation level (shut-down dose rate) 8 weeks after plant shutdown. Note that the insufficient shielding in the lower port area is recognized and currently being addressed

The high dose rate level imposes limits in the choice and functionality of RH equipment. Visual cameras and photogrammetry weld inspection equipment are especially critical and have a predicted lifetime in the DEMO plasma chamber conditions of only 3h [15]. For the damaged Fukushima reactor RH tools equipped with visual cameras have been developed with a lifetime of 10,000 Sv, [16], i.e. only 10h in the DEMO plasma chamber. ITER blanket RH tools are required to survive 1 MGy (~0.9 MSv), i.e. ~900h in the DEMO plasma chamber, [17].

As also explained in [6] the segmentation of the DEMO IVCs is based on the aim to avoid RH tools having to operate in areas with very high gamma radiation. Part of the backside of each IVC is therefore directly accessible via at least one VV port. This allows the feeding pipes of all IVCs to be cut / re-welded with all IVCs installed. Hence during the cutting and rejoining operations the IVCs shield the in-port area reducing the gamma radiation level by several orders of magnitude, Figure 2. However, during removal of the IVCs naturally the radiation dose rate inside the port is expected to be of a similar level as in the plasma chamber.

In order to comply with this approach the DEMO BB is divided into vertical segments. A large upper vertical port has been implemented to allow their extraction. The divertor is toroidally divided into cassettes that can be handled through lower ports adopting the approach selected in ITER. The four main remaining issues that could require a modification of the blanket configuration are: (i) structural feasibility and RH compatibility of blanket attachment structures, which at the same time guarantee the required precision of the FW position, see paragraph 4.2 (ii) integration of auxiliary equipment requiring access to the plasma in port plugs that penetrate but do not divide blanket segments, see paragraph 4.3 (iii) selection of the divertor configuration and requirement of FW protection with high heat flux limiters to cope with high thermal loads during unavoidable thermal transients, especially in the upper region [2].

3. Vessel and in-vessel Technologies

3.1 Vacuum Vessel Technology

The VV provides confinement of the radioactive inventory inside the plasma chamber and its annexes.

The VV integrity must therefore be verified against criteria defined for mechanical components of nuclear installations by using internationally accepted codes. This limits the material choice to materials qualified for such applications. As in ITER, the austenitic stainless steel 316L(N) was selected for the DEMO VV, i.e. X2CrNiMo17-12-2 austenitic stainless steel with controlled nitrogen as defined in RCC-MR Edition 2007 and in later Editions of RCC-MRx. RCC-MRx AFCEN Edition 2012 (or later edition 2015) defines neutron irradiation limits applicable to 316L(N) austenitic stainless steel; the same is valid for some European grades similar to 316L e.g. X2CrNiMo17-12-2. The material damage level due to neutron irradiation damage corresponding to 2.75 dpa is defined as negligible based on the assumption that the loss of ductility is lower than 30%. For fracture mechanics analysis a reduced fracture toughness of J_{IC} = 350 kJ/m² is recommended compared to unirradiated material (500 kJ/m²). Previous studies have shown that this neutron irradiation limit of the

VV material (2.75 dpa) can be met, [18].

3.2 In-vessel components Technology

The IVCs provide a plasma-facing surface and reduce the neutron flux onto the VV. The DEMO BB contains Lithium-bearing materials and breeds Tritium in the presence of a neutron-multiplier, such as Be and Pb. The large amount of heat generated in the IVCs and on their PFCs is being removed with active cooling loops and transferred through heat exchangers to a power conversion system including a steam turbine. The coolant of the IVCs is operated at high temperature to allow for an efficient conversion of heat into electricity:

- Blanket: depending on the coolant being either water or Helium: ~300°C/ ~450-500°C [19].
- Divertor: coolant water at 130-180°C [9].

In case of the blanket the selection of the coolant and other technical features is still open [19].

It is currently expected that no safety function will be assigned to IVCs other than neutron shielding. The structural integrity of IVCs cannot be guaranteed to a level that would allow crediting these components with other safety functions due to remaining uncertainties regarding the occurrence of heat load transients, the complexity of the IVC design, and poor in-service inspectability. However, a break of an IVC is the initiator of an accident, e.g. in-vessel loss of coolant event (LOCA), and the failure of a blanket box containing the breeder materials is an aggravating failure, e.g. after an internal cooling channel leak. Hence the likelihood of the failure of IVCs will be part of the safety case requiring approval from the regulatory body and the structural integrity of the IVCs must be guaranteed to a level as high as possible, even if not with the scrutiny of a nuclear design code.

The structural material choice is therefore not limited to materials listed in nuclear design codes. For DEMO IVCs Eurofer was selected, a ferritic-martensitic steel with reduced contents of alloys such as nickel, cobalt, molybdenum and niobium that produce medium-long life radioactive nuclides as a consequence of neutron irradiation [20]. Due to their coarse microstructure ferritic-martensitic steels show, other than austenitic steels or CuCrZr, low swelling rates under neutron irradiation, [21]. Given the reported swelling rates Eurofer has the potential to be suitable for high neutron irradiation levels corresponding to 10 or more DEMO full power years. Hence, should Eurofer be used as IVC structural material in a fusion power plant, few replacements of all IVCs would be required in its lifetime from a material lifetime point of view.

The low thermal conductivity of Eurofer (~30 W/(m²K) limits however the power handling capability of PFCs with Eurofer as heat sinks to ~1-1.5 MW/m² depending on the coolant, [3]. Since in the strike point regions on the divertor targets the expected heat loads are expected to be significantly higher (10-20 MW/m²) ITER-like technologies are being developed based on Cu-alloys as heat sink material [22]. The high thermal conductivity of Cu-alloys (~350 W/(m²K) ensures a temperature gradient as low as 86 K through a 1.5 mm thick heat sink wall at a heat load of 20 MW/m². Making use of subcooled flow boiling [23] a very high convection heat transfer coefficient of more than 200 kW/(m²K) is achieved [24]. This limits the temperature increase in the convection zone to no more than 100 K at 20 MW/m². The convection coefficient achievable with gas coolants, which cannot rely on subcooled boiling, is significantly lower, at least by a factor of ~5 [25]. Also for high heat flux panels or limiters required in certain zones, e.g. to allow for plasma ramp-up, water-cooled Cu-alloy-based PFC technology is foreseen. It must be noted that the irradiation lifetime of Cu-alloys is significantly shorter, corresponding to about one DEMO full power year. This requires the divertor and other IVCs using this technology to be more frequently replaced, which negatively affects the reactor availability.

4 Blanket configuration and integration

4.1 Breeding blanket poloidal configuration

a) Tritium breeding rate

The design concepts of several essential DEMO systems and tokamak performance parameters are affected by the configuration of the BB radial thickness. In the previously conducted Power Plant Conceptual Studies [26] the BB radial thickness was defined based on mainly two considerations: (i) in order to maximize the Tritium-breeding performance the size of the breeding unit (BU) was increased to a level close to saturation, and (ii) on the inboard side the BU size was reduced trading-off the maximization of the T-breeding performance in favor of an optimized machine radial build. The reduction of the inboard blanket thickness Δt_{BLK} was found to effectively reduce the DEMO major radius by ~1.35 Δt_{BLK} [3]. This trade-off was confirmed also in early DEMO studies [18] considering the fact that the contribution to the total generation of Tritium in the inboard side of the BB is with ~25-30% only moderate, [27]. Based on these considerations the radial thickness of the BB had been defined in [26] and [18] as 80 cm and 130 cm on the inboard and outboard, respectively.

It must be noted that an increase of the blanket coverage fraction of the plasma surface is much more efficient to increase the breeding ratio compared to an increase of the breeding zone thickness. The reason is the progressively decreasing neutron flux towards the backside of the blanket. The Tritium generation rate in the BU therefore decreases with the radial distance from the FW. Compared to BU parts close to the FW the same BU volume in the rear of the BB contributes significantly less. Its cost however is the same, in terms of investment but also in terms of increased complexity of the entire tokamak (see discussion below). The relative contribution of radial sections of the BU was studied on the basis of the water-cooled lithium led (WCLL) blanket concept [28]. For this purpose a Monte-Carlo N-Particle Transport (MCNP) model of a half DEMO sector has been employed including a detailed heterogeneous model of the BB FW and BU. The results were normalized with respect to the default FW+BU thickness on the outboard (80 cm in front of a 50 cm back-supporting structure with integrated manifold), see Figure 3. It can be seen that the breeding rate practically saturates at a BU radial size of about 100 cm, and that the rear quarter of an 80 cm BU only contributes about 4.5% to the Tritium generation. Since the BUs on the inboard are thinner (~40-50 cm) their decrease would cause more significant losses in Tritium generation. But the opposite is equally true. Hence a reduction of outboard breeding units can be compensated by the increase of significantly fewer inboard breeding units. Figure 3 is roughly independent on the neutron wall load.

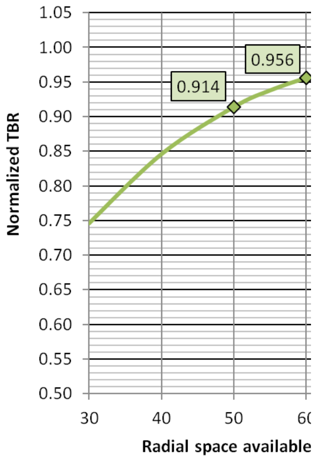

Figure 3 Normalized tritium breeding ratio (TBR) in a WCLL breeding module depending on the radial space available for the FW and the breeding units with respect to a space of 80cm, calculated for the equatorial outboard position

b) Neutron shielding

The minimum radial size of the BB is derived from the requirement to shield the VV, see section 2.2, and the TBR requirement. A BB with ~80 cm thickness was found to provide sufficient shielding in DEMO [18], both in terms of material damage and VV nuclear heating. A further reduction might be possible but has not been considered in DEMO for the more significant impact on the TBR, see Figure 3. It is recognized but not yet quantified and also not considered that components behind the blanket would benefit from the better neutron shielding provided by a thicker blanket, e.g. in-vessel diagnostics or IVC support structures.

Considering a helium-cooled BB the level of VV activation is not significantly affected by the BB thickness. Even the substantial increase of the thickness of helium-cooled blankets from 80 to 130cm does not reduce notably the radial depth of VV material that needs to be stored as intermediate-level waste for more than 100 years [29]. It has to be noted that in case of a water-cooled BB the blanket thickness does affect the VV activation level. Also the implementation of materials with better neutron shielding properties into the blanket would certainly reduce the VV activation level, which is a major contributor to the total activation waste of DEMO [30]. However, the complexity of the BB design and its integration can be expected to increase with an increase of its thickness and hence size. Therefore, increasing the blanket thickness for the purpose of reducing the VV activation is not currently considered.

c) Plasma vertical stability

The VV inner shell is the closest toroidally conductive structure to the plasma and therefore the main contributor to the plasma vertical stability. In DEMO it is especially effective in the outboard areas above the equatorial and above the lower port. Compared to ITER, which has a shield blanket with a thickness of ~45 cm, in DEMO the BB requires the VV to be relatively far from the plasma. The consequent poor passive vertical stability has been a main factor to limit the elongation in DEMO to κ_{95} = 1.59 [31]. Considering the TBR-sensitivity described above the outboard blanket thickness was reduced from 130 to 100 cm in order to improve the passive plasma vertical stability and also to reduce the BB volume and hence its cost. Considering this change and an optimization of the 2D FW profile the plasma elongation could be increased to κ_{95} = 1.65 maintaining the vertical stability margins [2], [32]. A system code benchmark study of the isolated impact of the increased plasma elongation (1.59 → 1.65) found a considerable reduction of the DEMO major radius by 0.31m.

d) Selected breeding blanket configuration

The poloidal shape of the blanket FW was defined with an automated procedure following the isoflux lines wherever possible to minimize the particle heat loads on the wall, and elsewhere aiming at distributing the particle heat loads across a large FW area [33]. The FW is at a defined distance of 22.5 cm [32] at the mid-plane and increases towards the top and bottom of the machine as the flux expansion increases.

On the entire outboard the blanket thickness is approximately 100 cm, see Figure 4. On the inboard side the blanket thickness has been defined as slightly less than 80 cm at the mid-plane, which will be the critical area from a neutron shielding point of view.

The inboard leg of the TF coil is straight whereas the D-shaped plasma naturally is curved. Between plasma and TF coil the thermal shield, the VV and the blanket are integrated. The thermal shield and the VV have a constant thickness on the inboard wall since the shielding requirements are constant. Although towards the top and bottom of the inboard wall the distance between the FW and the plasma increases somewhat, effectively more space is available for the BB. The chosen blanket configuration takes advantage of this fact aiming at an increase of breeding unit volume.

On the top of the plasma a decrease of the blanket thickness would allow a reduction of the tokamak vertical size. In particular the reduction of the TF coil height would be beneficial as the top poloidal field coils would be closer to the plasma reducing the required active control power. Also the stored magnetic energy as well as the total length of the TF coil conductors would reduce. However, the design is aimed at implementing a small number of TF coils to decrease the number of tokamak components and to increase the upper port toroidal width. Hence the size of the TF coils cannot be reduced to the minimum matching the outer contour of the VV in order not to exceed the TF ripple requirement.

It must be noted that the currently selected blanket configuration is based on a rather detailed design of the FW and the BU and a rough sizing of the back-supporting structure (BSS). Once the design of the BSS that integrates also the blanket manifold is developed to a higher level of detail further adjustments of the blanket thickness will likely become necessary / possible.

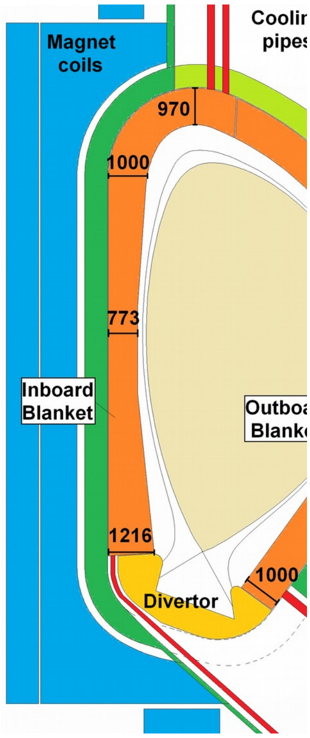

Figure 4 DEMO poloidal cross-section including plasma separatrix, magnetic isoflux line at 5 cm offset from the separatrix at the outer mid-plane, and optimized breeding blanket configuration, all dimensions in [mm]

4.2 Breeding blanket integration

a) Blanket attachment requirements

The vertical maintenance approach has been introduced for the breeding blanket in 1988 in the Next European Torus (NET) [34] and has since been considered as the most suitable concept in DEMO and fusion power plant studies [26], [35], [36]. The design of the blanket attachment structures is critical to this concept and have to meet the follolewing requirements: (i) provide a sufficiently precise positioning to the FW that is integrated in the blanket, (ii) withstand all relevant loading conditions [37], (iii) compatible with the blanket removal kinematics, and (iv) suitable for engagement and release in the in-vessel environment by RH tools.

Four additional requirements for all plasma-facing IVCs are related to their electrical integration: (a) all IVCs must be electrically grounded to the VV, (b) electrical connections between IVC and VV are required in vicinity of all IVC cooling pipes in order to avoid halo currents to flow through cooling pipes generating large electromagnetic forces, (c) attachments where electrical contact cannot be guaranteed in all operational conditions shall be electrically insulated to ensure the predictability of currents induced in EM events, (d) (electrical) contact between adjacent IVCs due to their deformation must be prevented, either by defining an appropriate gap size or by electrically insulating the contact area.

b) Blanket attachment concept

Recently, a preliminary concept for the BB attachment has been developed. This concept is based on taking advantage in several ways of the Ferromagnetic force that acts on the ferromagnetic blanket material Eurofer. This is caused by the radial decay of the toroidal magnetic field and pulls each blanket towards the inboard with a large radial force of ~7 MN on each segment. These are additional loads acting on the blanket attachments. However, the fact that the toroidal field is constant during and in-between pulses allows relying on them. The guaranteed physical contact at the bottom and the top supports of the blanket segments allows relying on an electrical contact, too. Hence, as in the attachment of the ITER divertor cassette, no electrical straps are foreseen. This results in a significant reduction of in-vessel design complexity and RH operations.

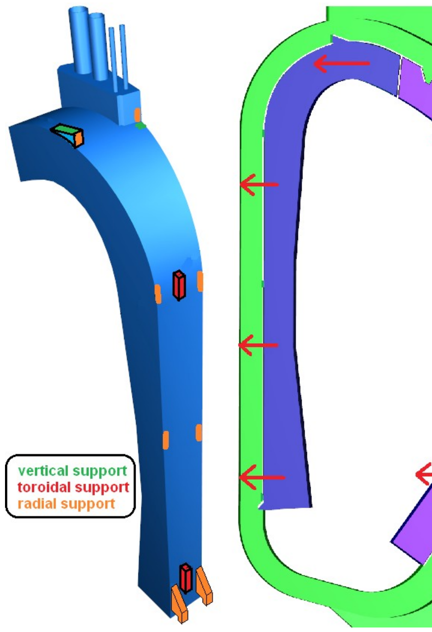

Figure 5 Attachment concept of the vertical blanket segments left: inboard blanket segment, red arrows: Ferromagnetic force

The attachment concept of the *inboard* segment, see Figure 5, is based on the following four features: (i) A vertical support at the bottom supporting downward loads, e.g. dead weight, (ii) a series of radial supports on both lateral sides of the segment support also against twisting about the vertical axis, (iii) two toroidal supports on the bottom and on the top providing support against the large radial moment that occurs during a plasma current quench, and (iv) a 2nd vertical support on the top mainly to control the vertical position of the FW at the top.

The basic structural concept of the *outboard* segment attachment, see Figure 5, is that of an arch bridge where the radial Ferromagnetic force acting on the blanket corresponds to the vertical gravity force acting on the bridge, see Figure 6. Translations of both end points of the arch are constrained. Hence the arch of the bridge will rise in case of a temperature increase.

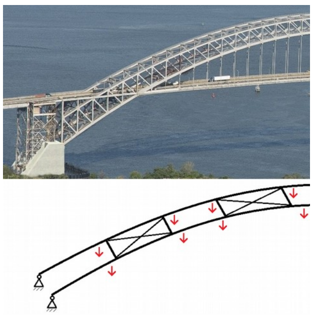

Figure 6 Arch bridge (New Jersey Bayonne bay bridge) and its structural principle adopted to the outboard blanket with forces due to dead weight symbolized by red arrows

c) Accommodation of thermal expansions

Accommodating different thermal expansions of the blanket segments is an important requirement in the design of the support structures. Three temperature conditions have been considered (i) assembly at room temperature, (ii) steady state operation, and (iii) ex-vessel loss of coolant incident (ex-vessel LOCA). Note that baking of the blanket is not foreseen due to its high operating temperature. During operation the blanket BSS is controlled by the inlet coolant at ~300°C to ensure that its temperature profile does not vary much during the pulse. The blanket temperature profile during plasma operation causes the blanket to expand and straighten. It can currently not be defined precisely as it varies amongst the different blanket concepts and also depends on the blanket architecture being multi-module or single-module [19]. In an ex-vessel LOCA the blanket is assumed to lose active cooling instantly whereas heat loads decrease relatively slowly within ~60s (→ soft plasma shut down). Radioactive decay heat in the blanket's materials will be generated for a much longer time [38], [39]. In this scenario the blanket temperature profile is expected to be most extreme.

A vertical gap is foreseen at assembly at the top between the blanket and its supports of about 100 mm at the outboard segments and about 60 mm at the inboard segments. It was chosen to provide space for most but not all of the blanket thermal expansion during operation. This prevents a large bowing of the outboard segments and at the same time ensures physical contact in order to better control the FW position. During maintenance these gaps provide clearance and allow for a small initial lift to clear the bottom supports.

d) Control of asymmetric blanket deformation

Out-of-plane deformation of the blanket segments occurs due to asymmetries. Generally these can stem from the blanket design and its temperature profile, the support conditions, or the loads acting on the blanket. It is aimed to minimize such deformation in order to distribute well the loads to the supports, but mainly to avoid misalignment of the FW and non-uniform heat loads during operation, e.g. due to twisting about the vertical axis. The current design of the BB and its cooling concept [19] is not expected to cause large asymmetries. However, in the area of the upper port the supports of the inboard and the lateral outboard blankets are partly inside, partly outside the port. It was found that a difference in stiffness of these support structures causes asymmetric out-of-plane twisting of the blanket segments. In the on-going detailed design of these support structures particular attention is being paid to ensuring similar stiffness and initial gap sizes. The main asymmetric loads occur due to plasma disruptions. It cannot be avoided that the reaction forces in these events are very non-uniform and the blanket segments twist asymmetrically.

e) Verification

The most significant reaction forces on the blanket supports that have been calculated are the following: (i) Up to ~4 MN radial force on the main supports of the outboard segments due to the Ferromagnetic force and electromagnetic loads during a fast disruption, (ii) ~4 MN vertical force on the bottom supports of the outboard segments designed with single-module architecture in an ex-vessel LOCA event. Although these are significant forces initial design studies indicate that attachment structures with reasonable dimensions can withstand them. On the other hand the blanket structure and its deformation will need to be verified, too. Initial studies indicate the feasibility of the attachment concept in this regard [40].

4.3 Port plug integration with breeding blanket

The integration of equatorial port equipment with the BB will require large penetrations in the BB. Equatorial port plugs of similar size as those in ITER (2.2 m x 1.75 m cross-section) would likely require dividing the central outboard segment into an upper and a lower part. Not only would this require the development of additional, new and possibly complex support structures. The integration of the feeding pipes from the lower blanket part alongside the equatorial port plug and upwards through the upper port would be difficult and possibly not be compliant with the approach to cut and re-weld the pipes with all IVCs in-situ, see paragraph 2.3. Also the blanket removal kinematics would be less coherent. Hence the development of the DEMO equatorial port configuration aims at maintaining the outboard blanket segment poloidal integrity. Instead cut-outs are considered affecting no more than about one third of the BSS toroidal size. The configuration of the equatorial ports is therefore adapted to the shape of the blanket segments, i.e. poloidally high (~3 m) and toroidally slim (~1 m). Toroidally inclined ports, first of all NB ports, are configured to penetrate between two adjacent outboard segments in order to minimize the impact on the BSS of either of them, see Figure 7. The reduced thickness of the outboard blanket, see paragraph 4, simplifies the integration of the equatorial ports.

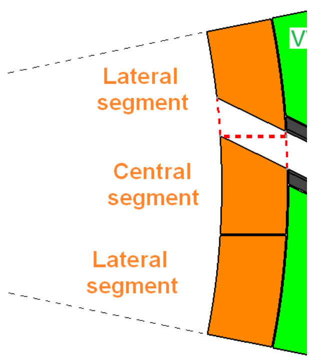

Figure 7 Principal configuration of the NB port with liner and cut-outs in breeding blanket segments for the beam

5 Summary and outlook

An overview over the DEMO mission requirements has been provided and their impact on the DEMO architecture and technologies has been discussed:

- The DEMO shielding concept has been presented including the shielding structures necessary to meet the limits of tokamak components and the requirements for manual and remote maintenance.
- An introduction was given of the chosen RH approach that aims at minimizing operations in the high gamma dose rate environment in front of the PFCs. The consequent required segmentation of the IVCs has been discussed.
- The DEMO requirement to demonstrate IVCs with high irradiation lifetime has led to the choice of Eurofer as IVC structural material. At the same time the choice of austenitic steel for the VV has been confirmed to be suitable also in DEMO as in ITER.
- Various aspects to be considered in the poloidal configuration of the breeding blanket thickness have been discussed by means of the selected configuration in DEMO. It was noted that the VV activation is not currently considered in the design of the BB.
- An attachment concept for the vertical blanket segments has been developed taking advantage of the Ferromagnetic force acting on the ferromagnetic blanket material. This concept requires no connections in the plasma chamber requiring access by RH tools for release/fastening and requires few if any electrical straps.

In the future development of DEMO the shielding structures including their penetrations must be designed in greater detail to verify the shielding concept.

A more in-depth verification of the BB attachment concept will be required. Details of the attachment structures need to be developed, their RH compatibility must be assessed, more precise temperature conditions of the BB need to be defined and be considered, and the BB has to be designed and verified based on these boundary conditions.

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