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Definition of the basic DEMO tokamak geometry based on systems code studies

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HIGHLIGHTS

- The definition of the DEMO 2D geometry has been introduced.
- A methodology to derive the DEMO radial and vertical builds from the PROCESS systems code results has been defined.
- Other 2D and 3D geometrical assumptions required to create a sensible 3D configuration model of DEMO have been defined.

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ABSTRACT

This paper describes the methodology that has been developed and applied to derive the principal geometry of the main DEMO tokamak systems, in particular the radial and vertical cross section based on the systems code output parameters, while exact parameters are described elsewhere [1]. This procedure reviews the analysis of the radial and vertical build provided by the system code to verify critical integration interfaces, e.g. missing or too large gaps and/or insufficient thickness of components, and updates these dimensions based on results of more detailed analyses (e.g. neutronics, plasma scenario modelling, etc.) that were carried out outside of the system code in the past years. As well as providing a 3D configuration model of the DEMO tokamak for integrated engineering analysis, the results can also be used to refine the systems code model. This method, subject to continuous refinement, controls the derivation of the main machine parameters and ensures their coherence vis-à-vis a number of agreed controlled physics and engineering assumptions.

1. Introduction

The DEMO design point is a set of parameters characterizing the key features of a DEMO power plant. The evaluation of the various systems shall be based on the design point with confidence that there are no significant conflicts between the requirements for those systems. Systems codes, such as PROCESS [2] representing the full DEMO plant by capturing the interactions between (usually relatively simple) models of all the important plant systems are used to identify such design points based on assumptions regarding the plasma performance and technology. A design point can be chosen by optimizing a figure of merit such as capital cost, major radius, or pulse length. The systems code PROCESS has been used for past pre-conceptual studies such as the European Power Plant Conceptual Study [3], and is now being used for the DEMO concept development. The purpose of using the systems code is to rapidly identify potential solution spaces without having to carry out complex analysis at every point. The design point strategy is outlined in Fig. 1.

The derivation of the geometrical output from the code (see example of the PROCESS geometry output in Table 1) into a geometry fit for further development requires extensive engineering considerations. The present paper describes those considerations and the assumptions taken during this procedure. It also identifies the main uncertainties where further work is required to increase the level of confidence in the results.

2. Definition of the DEMO 2D cross section

Based on various studies, that were carried out during the pre-conceptual development phase to establish a reference DEMO design configuration [4], and applying engineering assumptions wherever no reliable information is available, the thickness of the main components of DEMO are determined. As visualized in Fig. 2,

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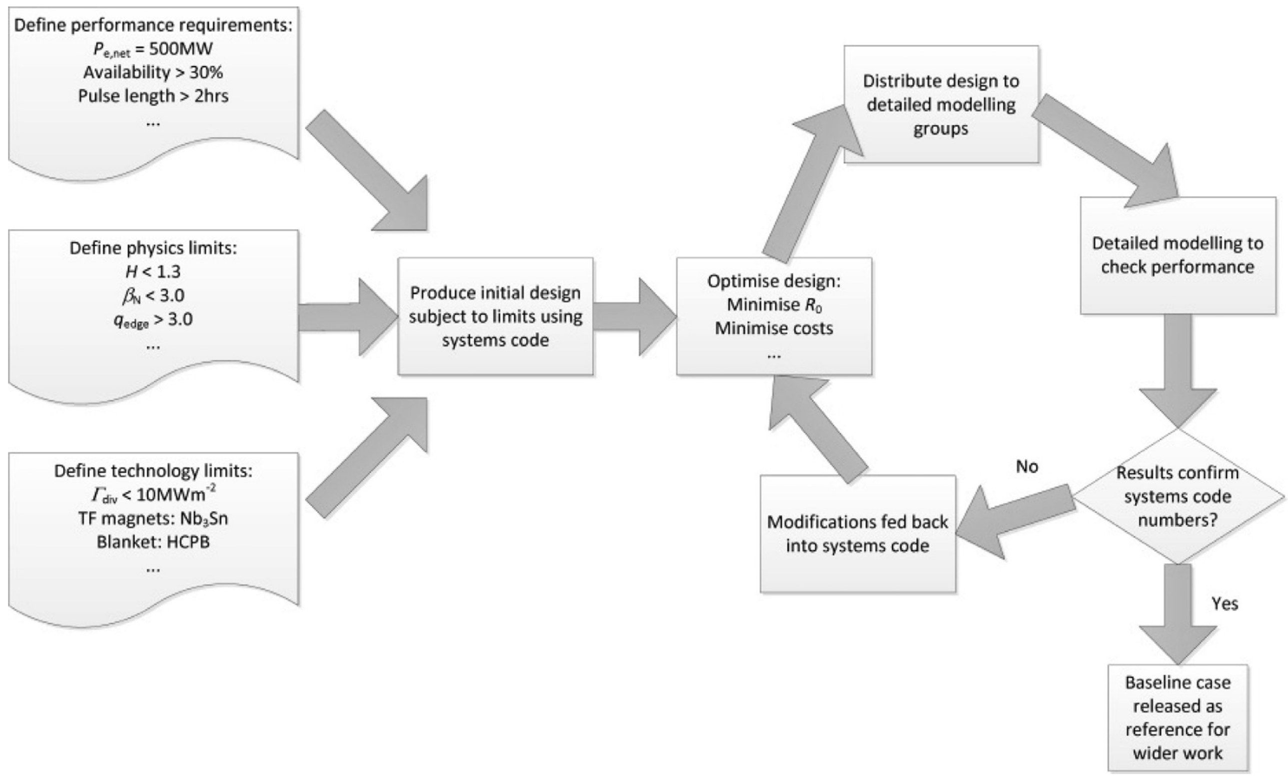

Fig. 1. DEMO design point development strategy. The detailed modelling stages include physics and engineering modelling to confirm that the design meets performance requirements and limits [4].

only the radial and vertical thicknesses are considered at this stage, while the rest of the 2D geometry is assumed based on best judgement. In the following a brief description is provided on the method how each thickness is concluded. The radial and vertical builds are arranged into four areas to allow the consideration of different features associated with the location, i.e. radial inboard, vertical top, radial outboard and vertical bottom. The component thicknesses at these four locations are also summarized and arranged after the description in Table 2.

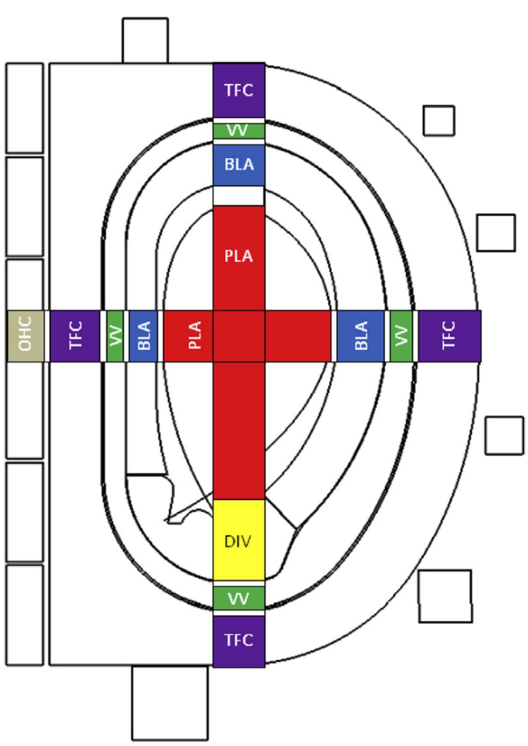

Fig. 2. Radial and vertical builds of the DEMO tokamak.

The below abbreviations are applied throughout the current paper:

OHC	ohmic coil (Central Solenoid)
TFC	toroidal field coil
VV	vacuum vessel
BLA	blanket
PLA	plasma
DIV	divertor

2.1. Bore

The thickness calculated by PROCESS is taken into account without further modification.

2.2. Ohmic coil

The thickness calculated by PROCESS is taken into account without further modification.

2.3. Gap (between OHC-TFC)

The gap is defined 50 mm constant as an engineering assumption to allow sufficient space for manufacturing and thermal expansion tolerances.

2.4. Toroidal field coil

The adjustment of the toroidal field coil thickness on the radial inboard side is required since the gap between toroidal field coil and vessel considered by PROCESS is insufficient and does e.g. not include provisions for the integration of the thermal shield (see the

Table 1
PROCESS geometry output example.

	Thickness (m)	Total size (m)
Radial build		
Device centreline	0.000	0.000
Machine bore	2.606	2.606
OH coil	1.034	3.640
Gap	0.050	3.690
Bucking cylinder	0.000	3.690
TF coil inner leg	1.412	5.103
Vacuum vessel	0.350	5.453
Gap	0.050	5.503
Inboard shield	0.300	5.803
Inboard blanket	0.775	6.578
Inboard first wall	0.022	6.600
Inboard scrape-off	0.150	6.750
Plasma geometric centre	2.250	9.000
Plasma outer edge	2.250	11.250
Outboard scrape-off	0.150	11.400
Outboard first wall	0.022	11.422
Outboard blanket	0.800	12.222
Outboard shield	0.760	12.982
Gap	0.100	13.082
Vacuum vessel	0.350	13.432
TF coil outer leg	1.695	15.127
Vertical build		
TF coil	1.412	7.526
Vacuum vessel	0.350	6.113
Gap	0.250	5.763
Top shield	0.530	5.513
Top blanket	0.788	4.983
Top first wall	0.033	4.196
Top scrape-off	0.225	4.163
Plasma top	3.938	3.938
Midplane	0.000	0.000
Plasma bottom	3.938	−3.937
Lower scrape-off	1.600	−5.537
Divertor structure	0.200	−5.737
Lower shield	1.050	−6.787
Gap	0.250	−7.037
Vacuum vessel	0.350	−7.387
TF coil	1.412	−8.800

gap (TFC-VV) definition in Section 2.5). Therefore the gap increase is deducted from the toroidal field coil thickness.

On the radial outboard the thickness is calculated in the following way: the radial dimension of the inboard is split up to the following parts: nose, winding pack and case. The nose thickness is calculated assuming that similarly to ITER 65% of the hoop stress is taken by the nose. The case thickness is assumed 70 mm constant. Therefore the winding pack thickness can be calculated deducting the nose and case thickness from the total. Taking the winding pack and adding sufficient case thicknesses on both sides determine the outboard thickness of the toroidal field coil. The assumed inboard and outboard cross sections are shown in Fig. 3.

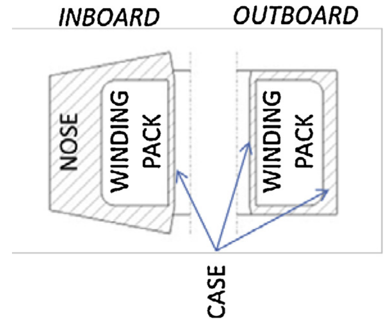

Fig. 3. TFC cross section assumption.

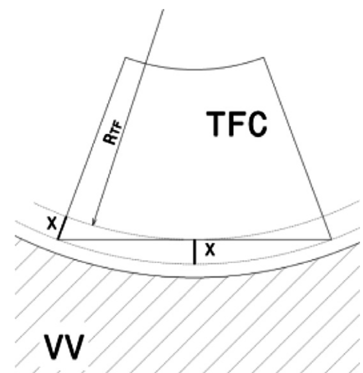

Fig. 4. Sketch of the TFC section to show the issue of the radius given by PROCESS in contradiction to the effective radius of the coil edge.

On the toroidal field coil vertical top and vertical bottom the radial build outboard thickness is applied for simplicity. Nevertheless the D shape of the coil is optimized to be close to in-plane bending free.

2.5. Gap (between TFC-VV)

On the radial inboard a significant increase of the gap is estimated compared to the PROCESS output to allow sufficient space to the following:

- 120 mm is allocated to reserve space for (i) a thermal shield based on the ITER value of 56 mm; (ii) the manufacturing tolerances (∼30 mm) of the toroidal field coil, thermal shield and vacuum vessel; and (iii) the thermal expansion between operation and shut-down states, assuming the operating temperature of the vacuum vessel of 200 °C (∼31 mm).
- An additional increase of the gap is required to compensate the incorrect calculation of the toroidal field coil radial thickness in PROCESS. As shown in Fig. 4, the radial build is defined in the

Table 2
Definition of the radial and vertical thickness of the major components [mm].

Volume category	Radial inboard	Vertical top	Radial outboard	Vertical bottom
BORE	Process	N/A	N/A	N/A
Ohmic coil – OHC	Process	N/A	N/A	N/A
GAP (OHC-TFC)	50	N/A	N/A	N/A
Toroidal field coil – TFC	Calculated	Calculated	Calculated	Calculated
GAP (TFC-VV) including thermal shielding	120+	120	120	120
Vacuum vessel – VV including shielding	600	600	600	600
GAP (VV-BLA)	20	20	20	N/A
Blanket – BLA (breeding, manifold, FW)	780	Progressive increase	1300	N/A
GAP distance between FW and plasma	150	450	150	N/A
Plasma – PLA	Process			
Divertor – DIV	N/A	N/A	N/A	2221

middle of the coil, so assuming a flat surface instead of curved the x distance needs to be added. It must be noted, that this gap heavily depends on the assumed toroidal field cross section and it needs to be calculated in case of each new set of radial build parameters.

On the radial outboard, vertical top and vertical bottom locations, however, the above described increase of the gap does not apply due to geometrical reason, therefore it is set to 120 mm constant. It also must be noted, that on the radial outboard the gap might be further increased to reduce the ripple to an optimum value.

2.6. Vacuum vessel

It must be noted that the total thickness of the following components – i.e. vacuum vessel, thermal shield (not discussed in the paper), breeding blanket and first wall – are given by the space allocated between the toroidal field coil and the front surface of the first wall and set in PROCESS to about 1500 mm based on previous calculation [5].

The thickness of the VV is steered by the requirement of providing sufficient neutron shielding to the toroidal field coils. It is also assumed to fill the space in the radial build remaining after allocating to the other components mentioned above. Currently 2 mm × 60 mm wall thickness plus shielding in between totaling to 600 mm is assumed all around the tokamak. However on the outboard a progressive increase of the thickness is considered below the equatorial mid-plane to provide a thicker vacuum vessel in the bottom outboard region to replace the function of the triangular support and therefore simplify the design while still enhance the vertical stability of the plasma (Fig. 2).

2.7. Gap (between VV-BLA)

The gap is defined 20 mm constant as an engineering assumption to allow enough space for manufacturing and thermal expansion tolerances all around the tokamak.

2.8. Blanket

The blanket in this context includes the breeding region, the supporting and feeding manifold and the first wall. 2013 tritium breeding and neutron shielding calculations of the different blanket concepts considered in the DEMO development concluded in the common recommendation of applying 780 mm on the inboard and 1300 mm thickness on the outboard for each blanket concept [6]. On the vertical top, however, a progressive increase of the blanket thickness between the inboard and the outboard is assumed resulting in a thickness in the range of 1000 and 1100 mm.

2.9. Gap (between first wall and plasma)

On the inboard and outboard radial locations 150 mm is considered until further evaluation.

The first wall profile, however, is an adaptation of the nominal plasma boundary contour shape. It is obtained by picking the magnetic poloidal flux line at the 150 mm distance from plasma edge in the outer mid-plane and following it to the top of the plasma. This choice is aimed to avoid concentration of charged particle flux on the first wall.

2.10. Plasma

The minor radius and the elongation parameters of the plasma calculated by PROCESS determine its radial and vertical dimensions that are considered without further modification.

2.11. Divertor

The development of the divertor cassettes is rather premature, therefore a very preliminary assumption is taken to define the divertor height in the vertical bottom location of the tokamak. Namely the distance between the PROCESS defined plasma bottom point and the vacuum vessel bottom inner contour point is based on the corresponding distance in ITER and is set to 2221 mm.

3. Further 2D and 3D geometrical assumptions

In addition to the definition of the radial and vertical builds further assumptions need to be applied to create a self-consistent 2D cross section. Additional considerations are also required for a simplified 3D configuration model of the DEMO tokamak. These considerations are as follows:

- The toroidal field coil vertical symmetry axis is at a different elevation compared to that of the other symmetric components.
- Toroidal field coils cross section assumed as in Fig. 4.
- Poloidal field coils cross sections and locations are defined by plasma equilibrium calculations [7].
- The vertical position of the equatorial port is aligned to the plasma centre line.
- Both the radial position and size of the vertical upper port are defined following the assumption, that the same portion of the inboard and the outboard blanket are directly accessible through the port as considered and verified through the remote maintenance studies carried out in 2012. The reference middle vertical line is defined by the blanket segmentation, which on the other hand is assumed to be on the vertical middle line of the plasma.
- The divertor port is considered to be inclined to 45° and its size to follow the size of the divertor which is currently scaled up from the ITER divertor proportional to the major radius.
- Separation between the inboard and the outboard blanket segments is assumed at the top tangential point of the blanket so that the cut is perpendicular to the wall of the blanket.

4. Conclusion

Based on the above considerations the PROCESS systems code output can be transformed into a full 3D CAD configuration model. Naturally most of the above assumptions can be fed back to the systems code following the cycle of Fig. 1 – this activity is recently under way – but others are resulting from either more detailed calculations or concerns geometry which is not included in the very simplistic view of the code. The paper represents the current knowledge only, since the assumptions shall be continuously improved and extended based on the new development results in the future.

Acknowledgements

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