Fusion Engineering and Design 98–99 (2015) 1423–1426

Contents lists available at ScienceDirect

Fusion Engineering and Design

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

Initial DEMO tokamak design configuration studies

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HIGHLIGHTS

- A definition of main DEMO requirements.
- A description of the DEMO tokamak design configuration.
- A description of issues yet to be solved.

ARTICLE INFO

Article history:
Received 19 September 2014
Received in revised form 9 April 2015
Accepted 13 May 2015
Available online 11 June 2015

Keywords:
Tokamak
DEMO
Power plant

ABSTRACT

To prepare the DEMO conceptual design phase a number of physics and engineering assessments were carried out in recent years in the frame of EFDA concluding in an initial design configuration of a DEMO tokamak. This paper gives an insight into the identified engineering requirements and constraints and describes their impact on the selection of the technologies and design principles of the main tokamak components. The EU DEMO program aims at making best use of the technologies developed for ITER (e.g., magnets, vessel, cryostat, and to some degree also the divertor). However, other systems in particular the breeding blanket require design solutions and advanced technologies that will only partially be tested in ITER. The main differences from ITER include the requirement to breed, to extract, to process and to recycle the tritium needed for plasma operation, the two orders of magnitude larger lifetime neutron fluence, the consequent radiation dose levels, which limit remote maintenance options, and the requirement to use low-activation steel for in-vessel components that also must operate at high temperature for efficient energy conversion.

1. Introduction

The realization of a Demonstration Fusion Power Reactor (DEMO) to follow ITER, with the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle by 2050 is viewed by Europe and many of the nations engaged in the construction of ITER as the remaining crucial step toward the exploitation of fusion power. The recent EU fusion roadmap Horizon 2020 [1] advocates for a pragmatic approach and considers a pulsed "low extrapolation" DEMO. This should be based on mature technologies and reliable regimes of operation, as much as possible extrapolated from the ITER experience [2] and on the use of materials and technologies adequate for the expected level of neutron fluence. In the new Consortium EUROfusion a corresponding conceptual design activity has now been launched.

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http://dx.doi.org/10.1016/j.fusengdes.2015.05.027

During the years 2011–2013 part of the EFDA program was dedicated to initial studies of the physics, technology and the design of the near-term DEMO device described in [3], elsewhere also referred to as DEMO1, to prepare the concept development phase that was launched in 2014. System codes were used to define a plasma configuration with reliable regimes of operation as much as possible extrapolated from the ITER experience [3]. Design and technology studies were carried out across a variety of topics and concluded in an initial DEMO tokamak design configuration that is presented here.

2. DEMO definition

2.1. DEMO requirements

DEMO shall demonstrate that a high availability of a fusion power plant is achievable [1]. DEMO is therefore designed for long plasma pulses (~2 h) and minimized dwell time between two pulses [4].

DEMO shall require tritium supply from external sources only for the plant start-up. Assuming a DEMO-size plasma (~2 GW) being operated during 20% of the time about 22 kg of tritium would be consumed per year – by far more than the annual world tritium production capacity that could be available for fusion rather than military use (currently up to ~2 kg, with a new dedicated facility possibly up to ~10 kg @ ~30–130 M$/kg [6,7]). DEMO shall therefore be tritium self-sufficient [1].

DEMO shall demonstrate the production of several 100s MW of net electricity [1]. The blanket, whose coolant exhausts about 85% of the power from the reactor, needs to be operated at high temperature to allow for efficient energy conversion (coolant inlet at ~300 °C). At the same time technologies and/or systems with high energy demand requiring recirculating power were avoided or kept to a minimum; hence superconducting technology will be used for the magnets and the cooling scheme of the in-vessel components (IVC) is being designed with attention to pressure drops. Also the total power of auxiliary heating and current drive (H&CD) systems was minimized; the DEMO plasma is therefore mainly inductively driven and operated at high plasma to auxiliary power ratio Q (>30) [4].

Reduced activation materials need to be used for some DEMO components to avoid the need for permanent waste repositories [1]. Consequently, Eurofer [5] was chosen as structural material of the blanket.

2.2. DEMO plasma configuration

The DEMO plasma is planned to have a major radius of ~9 m, incorporating a conventional H-mode scenario, and generating a fusion power of ~2 GW and is operated in long pulses (~2 h). It is planned to have a high radiation fraction, a high density (Greenwald fraction ~1.2) and a high β_N (~2.4). The DEMO plasma configuration is described in more detail elsewhere [4].

2.3. Tokamak configuration

The DEMO tokamak architecture is that of a typical superconducting tokamak machine: to thermally insulate the magnet coils the tokamak is inside a large vacuum chamber: the cryostat. In addition thermal shields at ~80 K protect the coils from radiation heat. The port structures of a torus shaped water-cooled vacuum vessel penetrate the cage formed by the magnet system (currently based on 16 TF and 6 PF coils) providing access to the plasma, e.g. for systems heating the plasma or driving its current, for diagnostic devices, or to maintain the IVC. Inside the vessel the high heat flux targets of the divertor intersect the scrape-off layer – a narrow band collecting most of the particles that escape the plasma confinement.

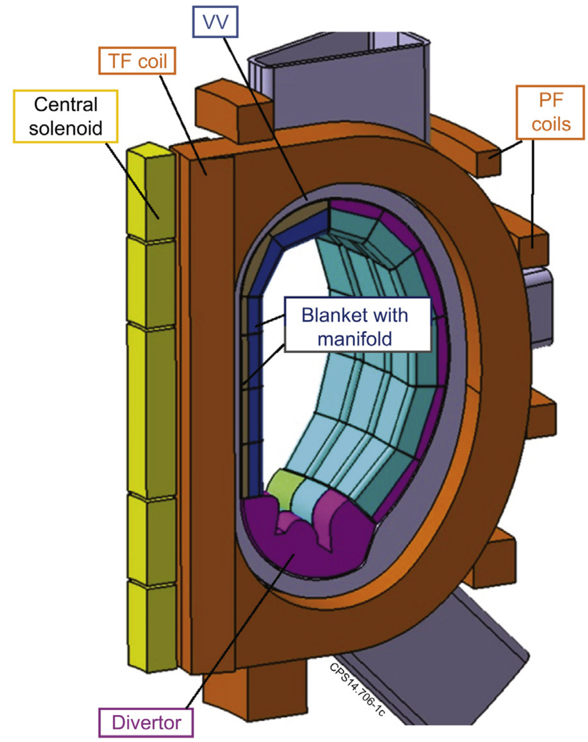

Fig. 1. Configuration of the DEMO tokamak main systems: central solenoid, toroidal field coil, poloidal field coils, vacuum vessel (VV) with port structures, breeding blanket, and divertor.

In DEMO most of the plasma surface (~85%) is surrounded by a blanket containing large amounts of lithium to breed the tritium consumed in the plasma. To compensate for the plasma-generated neutrons not interacting with lithium the blankets also contain a neutron multiplier, either beryllium or led. The high dose rate in the plasma chamber and also in the cryostat due to the activation of the IVC requires all tokamak maintenance to be carried out remotely. A thick concrete bioshield surrounding the cryostat reduces the radiation dose to allow man-access should it be necessary in most of the tokamak building outside plasma operation and remote maintenance phases.

All components inside the plasma chamber are maintainable and can to some degree be inspected. Maintenance of in-vessel components will generally consist of the replacement of components by remote handling (RH) tools. RH equipment will be introduced into the plasma chamber using transfer casks docked to the vacuum vessel port flanges. Due to the activation of the IVC the removed components are transported to the hot cell in transfer casks for disposal as waste or, where possible, refurbishment.

The radial build of the DEMO tokamak, Fig. 1, is composed of the central solenoid, toroidal field (TF) coils, thermal shield, vacuum vessel (VV), and breeding blanket. The space required for the breeding blanket is driven by the tritium breeding requirement; the radial size of the VV is chosen in order to further reduce the neutron flux onto the TF coils to meet the neutron load limits of the conductor [8]; the size of the TF coil is chosen to provide sufficient space for the winding pack and the casing withstanding the significant electromagnetic loads acting on the coil. The build-up of the DEMO tokamak is described elsewhere [9].

3. Major design choices

3.1. Overview

An overview of the main DEMO features and open choices is given in Table 1.

Table 1
DEMO features and open choices.

DEMO features	Open choices
Major radius ~9 m	Aspect ratio
Long pulses ~2 h	Breeder concept
Full tungsten armor	Blanket coolant
Starter blanket (20 dpa), 2nd blanket (50 dpa)	FW/limiter configuration
Materials:	Divertor cassette cooling scheme
Nb3SN (TF conductor)	H&CD mix, technologies and integration
Eurofer (blanket)	Confinement barriers
AISI 316 (vessel)	Diagnostic technologies and integration
Cu-alloy (divertor target heat sink)	

3.2. In-vessel components maintenance scheme

During the replacement of actively cooled in-vessel components their cooling circuit needs to be cut and re-welded. In DEMO in-vessel re-welding is avoided given the high helium production due to the high neutron fluence but also to simplify the in-vessel RH operations. Many previously adopted RH technologies are challenged in the high dose rate environment expected in the DEMO plasma chamber. The blanket and divertor cooling pipes are foreseen to be cut and re-welded inside the vessel ports instead where sufficient neutron shielding can be provided. Whereas this limitation allowed the adoption of the ITER divertor RH strategy a different RH strategy needed to be developed for the blanket. The blanket is divided into large inboard and outboard vertical segments, see Fig. 1, and vertically handled through large upper ports. This concept had also been adopted in the European Power Plant Conceptual Study [10] and has been further developed for DEMO [11]. Cantilevered RH tools carrying the blanket weight as e.g. in ITER are avoided. This is considered a necessity given the weight of one single blanket segment of up to ~60 t.

3.3. Blanket design

An individual blanket segment is composed of a solid back-structure onto which a number of individual breeding modules are mounted providing sufficient strength to withstand the significant EM loads acting on the modules. An individual module has a box structure integrating the module-internal manifold, the breeding units, and the FW channels. The back-structure integrates all feeding pipes supplying the primary coolant to the modules as well as either helium purge gas or LiPb, depending on the breeder concept. The feeding pipes provide active cooling to the back-structure of the blanket segment whose temperature gradients depend on the temperature conditions of the different feeding pipes and their integration in the manifold. Temperature gradients in the back-structure of the blanket segment complicate the design of the blanket attachment structures. It is therefore an aim to integrate the feeding pipes in such a way that the back-structure temperature is driven mainly by the inlet pipes, whose temperature is well controlled and constant during transients.

3.4. First wall design and technology

The preliminary choice was made in DEMO to integrate the first wall (FW) in the breeding module in order to avoid re-welding and/or connections in the high neutron fluence area. Hence, it cannot be repaired or exchanged separately (as in ITER). This requires high component reliability and a robust FW protection concept.

The DEMO FW concept – whether helium- or water-cooled – is based on an array of parallel (Eurofer) cooling channels protected from plasma particles by tungsten armor. Copper was excluded as heat sink material due to its severe material degradation under irradiation, its upper temperature limit, and its activation under neutron irradiation.

3.5. First wall protection/limiter configuration

The thermal conductivity of Eurofer is about one tenth that of copper. Even using high velocity coolant and accepting the consequent pressure drop the heat load capacity is limited to ~1.5 MW/m^2 (water-cooled) [12] or ~1 MW/m^2 (helium-cooled), respectively. It is therefore conceivable that limiters will be implemented in DEMO to protect the FW from contact with the plasma.

The design of limiters as well as their integration in the DEMO tokamak will soon be initiated. Limiters are likely to have water-cooled plasma-facing components (PFC) similar to the divertor targets and not to be required to contribute to the tritium breeding.

3.6. In-vessel components integration

Previous worldwide fusion DEMO/power plant studies [10,13,14], considered a semi-permanent in-vessel shield: a toroidally continuous shell structure actively cooled to a temperature similar to that of the IVCs, providing support to the IVCs and neutron shielding to the superconducting coils. Instead in the current DEMO configuration both blanket and divertor are directly supported by the vessel, which also takes over the neutron shielding function of the shield. Hence a shield component inside the plasma chamber is not required in DEMO, which simplifies significantly the in-vessel integration and reduces the number of in-vessel interfaces. In order to reduce relative thermal expansion issues the DEMO VV is cooled to 200 °C, which is close to the operating temperatures of the divertor cassette (~220 °C) and the blanket (~300 °C). Shut-downs to bake the vessel at 200 °C as in ITER are therefore not required in DEMO.

The ITER divertor attachment *concept* can in principle be considered also in DEMO; its *design* will however require several developments to be suitable to the DEMO configuration and requirements. The attachment of the DEMO blanket segments on the other hand requires a new development at the conceptual level that has been initiated recently [15]. This needs to be compliant with the very large forces predicted in preliminary EM assessments, e.g. a radial moment acting on a blanket segment during disruptions of the order of 30 MNm [16]. The need is recognized to protect the supports from direct neutron irradiation while at the same time ensure adequate accessibility by RH tools.

4. Design options and issues

4.1. Design options

4.1.1. Divertor configuration

An ITER-like single-null divertor configuration with the divertor cassette at the bottom of the VV is initially considered in DEMO. Strong emphasis is given in the roadmap [1], to identify alternative divertor configurations that alleviate the problem of excessive heat loads on the DEMO divertor targets.

4.1.2. Blanket concept

Four blanket concepts are being developed in parallel throughout the DEMO conceptual design phase within the same tokamak configuration to allow a direct comparison and eventually a down-selection based on technical rationales. These concepts use two different breeder concepts (PbLi or ceramic pebble beds), are either helium-, water- or dual cooled (PbLi also used as a coolant) and are described elsewhere [17].

4.2. Unsolved issues

4.2.1. Vertical stability

The current DEMO design does not provide adequate passive plasma vertical stability. The main reason is the large distance (up to 1.3 m) between the plasma and the (toroidally conductive) vessel inner shell due to the presence of the breeding blanket. The fact that the upper vertical port interrupts the toroidal continuity of the vessel is an additional drawback. Active in-vessel coils, passive conductors electrically connecting adjacent blanket segments, electrically bridging the upper port, as well as a local reduction of blanket thickness to allow the vessel inner shell to locally approach the plasma are options currently considered to mitigate the issue. It is also recognized that a reduction of the plasma elongation significantly improves the vertical stability.

4.2.2. Space in upper port

In ITER the vessel ports are on the outboard side 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. In addition a neutron shield plug is required in this port penetrated by the numerous blanket feeding pipes and possibly diagnostic or H&CD systems complicating the design integration. To enlarge the port internal space the feasibility of single-walled toroidal port sidewalls is being studied.

4.2.3. VVPSS for in-vessel LOCA

In order to limit the pressure in the plasma chamber in case of an IVC coolant leak (LOCA) the Vacuum Vessel Pressure Suppression System (VVPSS) provides a connection from the plasma chamber to an expansion volume that is normally closed by a rupture disk. In case of water coolant the required size of the expansion volume is significantly reduced by condensing the steam. This technique is however not applicable in the case of helium coolant and initial studies point to the need of a very large expansion volume.

4.2.4. Blanket thermohydraulic design

In DEMO the FW channels are cooled in parallel and will for practicality have a uniform design. Due to the torus shape of the device the FW channel lengths will vary to some degree; at the same time the FW heat load varies in poloidal direction. These factors cause a variation in the outlet temperatures of the FW channels. To enable the FW to withstand peak heat fluxes that cannot be excluded the FW coolant flow rate must be beyond what would be required to remove the normally occurring heat load. These uncertainties make a thermohydraulic optimization of the blanket cooling loop difficult.

4.2.5. BoP for pulsed heat load

A Rankine cycle is foreseen for the secondary loop in DEMO (since the operating temperature of the IVC is limited due to the softening of the structural material). For the hot start of a steam turbine durations of at least ~30 min are reported in literature [18], much longer compared to the plasma ramp-up (~1 min). It may therefore be required to temporarily dump part of the steam during the initial phase of a pulse. Industry support is sought to advice on this issue and to identify operating schemes of the balance of plant (BoP) optimized for pulsed operation.

Acknowledgements

This project has received funding from the European Union's Horizon 2020 Research and Innovation Programme under grant agreement number 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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