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Overview of EU DEMO design and R&D activities

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HIGHLIGHTS

- An important objective of the new EU fusion roadmap is to lay the foundation of a DEMO Fusion Power Reactor to follow ITER.
- DEMO should be capable of generating several 100 MW of net electricity and operating with a closed fuel-cycle by 2050.
- The paper outlines the DEMO design and R&D approach recently being undertaken in Europe.
- The paper presents some preliminary design options that are under evaluation as well as the most urgent R&D work needed.
- The R&D on materials for a near-term DEMO is discussed in detail elsewhere.

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ABSTRACT

One important objective of the EU fusion roadmap Horizon 2020 is to lay the foundation of a Demonstration Fusion Power Reactor (DEMO) to follow ITER, with the capability of generating several 100 MW of net electricity to the grid and operating with a closed fuel-cycle by 2050. This is currently viewed by many of the nations engaged in the construction of ITER as the remaining crucial step towards the exploitation of fusion power. This paper outlines the DEMO design and R&D approach that is being adopted in Europe and presents some of the preliminary design options that are under evaluation as well as the most urgent R&D work that is expected to be launched in the near-future. The R&D on materials for a near-term DEMO is discussed in detail elsewhere.

1. Introduction

With the construction of ITER well underway, attention is now turning to the design of a successor device; a Demonstration Fusion Power Plant (DEMO), i.e., the nearest-term reactor design capable of producing electricity, operating with a closed fuel-cycle and to be the single step between ITER and a commercial reactor. Currently, no conceptual design exists for DEMO and work carried out in the past in Europe on fusion reactor design focussed on the assessment of the safety, environmental and socioeconomic aspects of fusion power and less on rigorous technology feasibility assessments [1]. At present, the DEMO reactor design has not been formally selected and detailed operational requirements are not yet available. DEMO is a device which lies between ITER and a power plant, but there is not widespread agreement of where in the range it must lie; motivated in part by the range of options for a power plant design and the timescales on which DEMO should be delivered.

This paper provides an overview of EU DEMO design and R&D activities. Emphasis is on the results of a two-and-a-half year study performed by the EFDA Power Plant Physics and Technology (PPPT) Department focusing on (i) the identification of the DEMO prerequisites, (ii) the main design and technical challenges (physics and technology), (iii) the preliminary assessment of the foreseeable technical solutions and (iv) the prioritization of R&D activities to be launched as part of the new EU fusion roadmap [2,3]. In view of the many uncertainties still involved and recognizing the role of DEMO in fusion development, it is judged undesirable for the initial study effort to focus solely on developing the details of a single design point and there is the need to keep some flexibility in the approach to the conceptual design. However, establishing performance requirements and project development schedules linked to a target start of construction date is expected to be a strong driver in the selection of the technical features of the device;

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favouring more conservative technology choices for near term solutions [2,4].

A system engineering approach is viewed as essential from the early concept design stage [5]: (i) to better understand the problems and evaluate the technical risks of foreseeable technical solutions; (ii) to identify design trade-offs and constraints to address the most urgent issues in physics, technology and design integration; and (iii) to prioritize the R&D needs. Ensuring that R&D is focussed on resolving critical uncertainties in a timely manner and that learning from R&D is used to responsively adapt the technology strategy is crucial to the success of the programme. In general, the progress assessment methodology should be similar to other fields and follow the approach of assigning a technical readiness level (TRL) to the reactor systems and updating the TRL as R&D tasks are completed. There are many examples of TRL scales and their application to systems of varying and evolving maturity. The integration of our expanding physics knowledge into the DEMO conceptual design will also play a crucial role in supporting the design evolution.

Involvement of industry and exploitation of international collaborations on a number of critical technical aspects is highly desirable.

2. Main differences between ITER and DEMO

According to several studies undertaken in Europe in the past (see for example [1,6]) DEMO must:

- Resolve all physics and technical issues foreseen in the plant and demonstrate reactor relevant technologies.
- Demonstrate production of several 100 MW of electricity.
- Achieve *T* self-sufficiency, i.e. DEMO must make its own fuel.
- Achieve adequate availability/reliability operation over a reasonable time span.

The main differences between ITER and DEMO are summarized in Table 1 [7].

Table 1
Main differences between ITER and DEMO [7].

ITER	DEMO
Experimental device with physics and technology development missions	Nearer to a commercial power plant but with some development missions
400 s pulses, long dwell time	Long pulses (>2 h) or steady state
Experimental campaigns. Outages for maintenance, component replacements	Maximize availability
Large number of diagnostics	Only diagnostics required for operation
Multiple H&CD systems	Optimized set of H&CD systems
Large design margins, necessitated by uncertainties and lack of fully appropriate design codes	With ITER (and other) experience, design should have smaller uncertainties
Cooling system optimized for minimum stresses and sized for modest heat rejection	Cooling system optimized for electricity generation efficiency (e.g. much higher temperature)
Unique one-off design optimized for exptl. goals	Move towards design choices suitable for series production
No tritium breeding requirement (except very small quantity in TBMs)	Tritium breeding needed to achieve self-sufficiency
Conventional 316 stainless steel structure for in-vessel components	Nuclear hardened, novel reduced activation materials as structure for breeding blanket
Very modest lifetime n-fluence, low dpa and He production	High fluence, significant in-vessel materials damage
Licensed as nuclear facility, but like a laboratory, not a reactor	Licensing as nuclear reactor more likely
Licensing as experimental facility	Stricter approach may be necessary to avoid large design margins
“Progressive start-up” permits staged approach to licensing	“Progressive start-up” should also be possible (e.g. utilize a “starter” blanket using moderate-performance materials and then switch to blankets with a more advanced-performance material after a few MW-year/m^2)
During design, licensing in any ITER party had to be possible	Fewer constraints

A variety of fusion power plant system designs have been studied in the past across the world, but the underlying physics and technology assumptions were found to be at an early stage of readiness. The recent EU fusion roadmap [2] advocates a pragmatic approach and considers, for the initial design integration studies, a pulsed “low-extrapolation” DEMO that could be delivered in the short to medium term. This should be based on mature technologies and reliable regimes of operation to be, as much as possible, extrapolated from the ITER experience, and on the use of materials adequate for the expected level of neutron fluence [3].

It is argued that by waiting to design DEMO for the ultimate technical solution in each area would postpone the realization of fusion indefinitely. Since the mission requirements of a near-term DEMO put more emphasis on solutions with high technical readiness levels and realistic performance and component reliability, rather than on high-efficiency, the R&D priorities in the Roadmap are defined to achieve these goals. Nevertheless, these goals remain very ambitious and many technological advances and innovations will be required. More advanced technological solutions will also need be developed but on a longer timescale and as part of a parallel long term R&D programme.

3. DEMO design options

The task of choosing an appropriate set of design parameters and engineering technologies involves trade-offs between the attractiveness and technical risk associated with the various design options. One of the crucial points is the size of the device and the amount of power that can be reliably produced and controlled in it. This is the subject of research and depends upon the assumptions that are made on the readiness of required advances in physics (e.g., the problem of the heat exhaust, choice of regime of operation, efficiency of non-inductive Heating and Current Drive (H&CD) systems, etc.), technology and materials developments. An overview of the results of initial analyses is provided elsewhere [4]. Here, we summarize some of the key results on the main device design drivers and how they have been explored in these studies.

Two different DEMO design options are currently investigated, in an attempt to identify a realistic range of possibilities:

- A near-term DEMO (DEMO1) is a rather “conservative baseline design”, i.e. a DEMO concept deliverable in the short to medium term (e.g., construction possibly starting ∼20 years from now), based on the expected performance of ITER (Q = 10) with reasonable improvements in science and technology; i.e., a large, modest power density, long-pulse inductively supported plasma in a conventional plasma scenario. The design of Balance of Plant (BoP) for a near-term DEMO must also make use of mature and reliable technology.
- a more advanced, DEMO design concept (DEMO2) based around more optimistic (but “less mature”) physics assumptions, which are at the upper limit of what may be achieved in ITER phase-2, i.e., an advanced higher power density high current drive steady-state plasma scenario. It is clear that this can only be delivered on a longer term (e.g., construction to be started on a much longer time scale assuming that the required significant advances in the

physics basis be demonstrated using ITER and the limited number of satellite fusion devices available in the next 10–20 years).

It is not to be inferred that two DEMOs should be built but rather that there is a need to incorporate some flexibility to mitigate the uncertainty in the design requirements for DEMO.

The approach to select realistic DEMO machine parameters being followed in Europe consists of:

- defining a set of main requirements, constraints and assumptions, captured in a set of DEMO design guidelines, both for physics and technology [8];
- integrating these into a coherent conceptual design through implementation in 0-D systems codes [9];
- using the results of systems code to spin off more detailed assessments, using higher dimensional integrated modelling codes, e.g., codes that solve the time-dependent plasma transport equations (1-D) self-consistently with the magnetic equilibrium (2-D).

Clearly, it is necessary to iterate between these steps, for instance finding that some of the assumptions lead to inconsistencies in the integrated design, or that the results of more detailed assessments lead to changes in some of the assumptions. Therefore, the definition and assessment of plasma scenarios for DEMO, an essential step for the design of the machine, requires a careful iteration of the two approaches, systems codes and specific codes modelling relevant aspects of DEMO physics with a higher level of detail.

3.1. Sensitivity of design to assumptions

A highly important aspect of systems studies is how to choose a working point from the 'landscape' of available options, having defined the constraints and assumptions. One attractive figure of merit may be the minimum device size, which produces the required net electric power within the imposed constraints. In a steady-state device it may be decided instead to minimize the current drive power, on the assumption that this is the most important factor, however we can easily show that this is not a robust approach.

Our effort at the moment is concentrating on sensitivity studies, around initial reference design options (discussed in [4]), to identify the key limiting parameters, to explore the robustness of the reference design to key assumptions, analyze the trends and improve early design concept optimization.

3.1.1. Aspect ratio

An important related design parameter is the aspect ratio. The majority of data from tokamaks is available around an aspect ratio of 3 but, if the systems code is allowed, it may suggest designs with larger, or smaller, aspect ratio. A larger aspect ratio will allow a larger machine bore, which can be occupied by a larger central solenoid and therefore give greater flux swing, but this may also change the allowable elongation and certainly will change the allowable plasma current and therefore affect energy confinement. As was pointed out in Ref. [10], varying the aspect ratio in a systems code approach clarifies why intuition is not a good guide to such complex systems, since there is found to be an optimum aspect ratio producing a maximum pulse length (see Fig. 1).

In this case, the target $P_{e,net}$ and R_0 were fixed, so as the aspect ratio increases the plasma volume falls and better confinement is required to maintain power output. This requires higher B_T and hence thicker TF coils, eventually sharply reducing the machine bore and resulting flux swing available for current drive. Fixing different parameters will change the shape of this curve. In general, the elongation for which the plasma is vertically stable reduces as aspect ratio increases, worsening the problem. This graph illustrates the importance of modelling all aspects of a power plant self-consistently. Fig. 2 shows two design configurations with different values of aspect ratios.

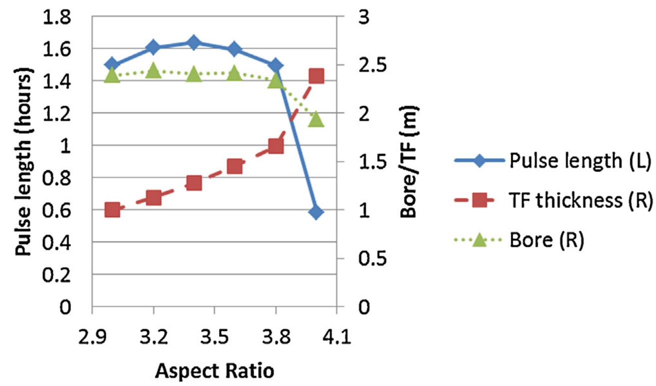

Fig. 1. Plot of aspect ratio against pulse length (left hand scale) and machine bore and TF coil thickness (right hand scale). The major radius was fixed at 9 m, and the net electric output at 500 MW. Power to the divertor was also fixed at 150 MW.

3.1.2. Divertor heat load

The peak power load on the divertor target is a key constraint on the design of DEMO and power exhaust will ultimately determine the size and choice of the operating scenario for DEMO. At present there is little confidence in divertor heat load models, typically based on extended 2-point models, so an approach is taken that the majority of the power coming from the plasma, primarily from the alpha heating, is to be radiated away onto the first-wall thereby protecting the divertor. This needs either high temperature operation, to give a high synchrotron power, or impurity seeding, and we favour the latter in these studies; as a result the DEMO designs discussed here tend to have a relatively high Z_{eff}. This has the effect of driving up machine size and plasma current, both to maintain the energy confinement despite large radiation losses and to increase the divertor footprint. The former could be relaxed, of course, if the underlying confinement, captured by the assumed *H*-factor, were higher. Table 2 includes, for example, a "radiation-corrected" *H*-factor in which the core radiation has been subtracted from the loss power, and an H^*, for which this correction has not been done. Typically radiation losses (usually low in the empirical dataset) are neglected in the derivation of confinement scaling laws. Fig. 3 illustrates the role of divertor-protection as a size driver: if we wish to limit divertor heat loading to less than 20 MW/m (normalizing the exhaust power to the major radius), which is a quantification of the divertor challenge, whilst remaining in *H*-mode for a 500 MW electrical output, 2 h pulse length plant, there is a small window between ∼9 and ∼9.5 m major radius allowing this operating point. Of course, many other assumptions regarding confinement, H&CD efficiencies, plant power balance, radiation control etc. lie under these calculations and work is ongoing to assess the sensitivity of the solutions to these assumptions.

Many of the same issues affect DEMO2 but with different emphasis, for instance large current drive power will exacerbate the divertor heat load, as well as increasing the fusion power required to supply the electricity demand of the current drive system. This drives a greater need for radiation, challenging the confinement even more, but at the same time the flux swing needs are reduced. This highlights some issues of key importance. Ref. [4] describes some preliminary design options for DEMO2.

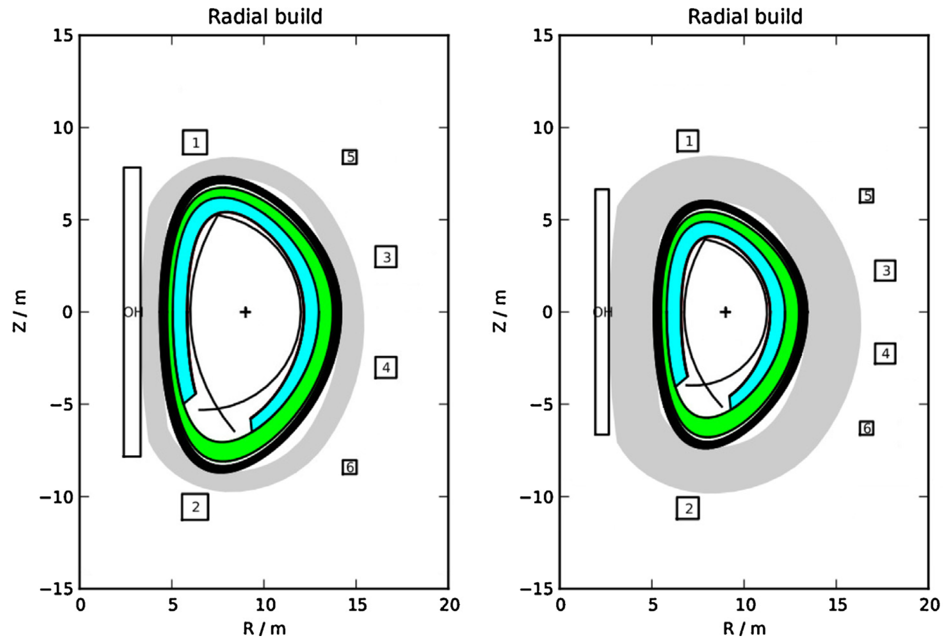

Fig. 2. Pulsed DEMO cross-section for $A = 3$ (left) and $A = 4$ (right). As the aspect ratio increases, the magnetic field increases to maintain confinement, requiring larger coils. Eventually the magnetic field at the TF coil becomes too large and no solution is found.

3.1.3. Net power output versus pulse length

Fig. 4 shows the relationship between burn time and net power output when a plant is operated at maximum power with machine size and geometry fixed ($R_0 = 9$ m, $A = 3.6$) and the power across the separatrix fixed at 150 MW. Hence P_{sep}/R, which is a quantification of the divertor challenge [12], is about 17–20 MW/m. It seems too optimistic to further raise this value considering that the highest value demonstrated in an existing device is about 7 MW/m. Ref. [4] provides a full description of the machine parameters used in this analysis.

It can be seen that if we design a device to generate a net P_{el} of 500 MW with a minimum pulse duration of 2 h and then extend the purely inductive pulse duration by auxiliary H&CD systems to be installed at a later stage, we may become over-constrained because any attempt to expand the pulse length will be off-set by the need to increase the recirculation power (depending on the ultimate efficiency of the H&CD systems) and therefore reduce net electricity output. This confirms that we must design from the beginning a system that can accommodate for this otherwise we have a loss of performance (see Fig. 4).

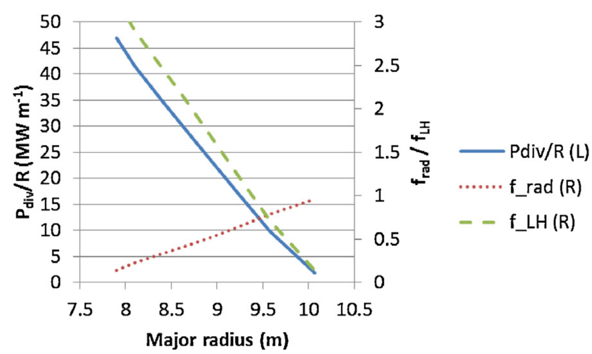

Fig. 3. DEMO1 power across the separatrix (normalized to major radius) as a size driver. Here the target net electrical output power (500 MW) and aspect ratio (3.5) were fixed, and pulse length fixed at 2 h. Major radius was minimized subject to these parameters. The divertor is protected by increasing impurity doping, resulting in increasing radiation fraction and lower conducted power to the divertor. Here all radiation including from the mantle and SOL is included, and so $f_{LH} = (P_{heat} - P_{rad})/(P_{LH})$ is a worst-case value [11].

Table 2
Example of output from cases of pulsed DEMO. The difference between H and H^* is discussed in the text.

Parameter	$A = 3$	$A = 4$	Comment
R, a (m)	9.0, 3.0	9.0, 2.25	
Plasma current (MA)	20.3	15.2	$f_{bs} = 34\%$ ($A = 3$), 36% ($A = 4$), $q = 3.0$
Toroidal field (T)	5.24	7.36	B_{max} cond. = 10.8 T ($A = 3$), 13.0 T ($A = 4$)
Fusion power	1794	1807	
$\langle n \rangle$ (10^{20} m^{-3})	0.77	1.03	
T_e (keV)	12.8	12.9	
H factor	1.1	1.1	$H^* = 1.0$
Pulse length (h)	1.5	0.6	See text for extended duration pulses
Z_{eff}	2.62	2.7	Ar and He impurities
β_N, $\beta_{N,th}$ (% MA m^{-1} T^{-1})	2.57, 2.24	2.50, 2.19	
P_{aux} (MW)	50	50	
P_{div} (MW)	150	151	
P_{net}, P_{rec} (MW)	500, 385	500, 391	He cooled blanket
P_{NW} (MW/m^2)	0.9	1.2	
Lifetime starter blanket (dpa)	20	20	See text

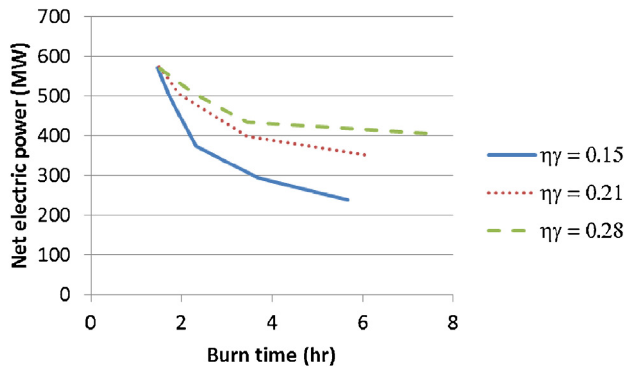

Fig. 4. DEMO1: Trade-off between net electrical power and pulse length for different values of $\eta\gamma$, the product of H&CD wallplug efficiency and current-drive efficiency. Machine geometry and size was fixed, as was the power to the divertor. Injected power varies from 0 MW (on the left – starting point of the graphs) to 175 MW for the longest-pulse low efficiency system (continuous line). The longest pulse in the high-efficiency system used 100 MW (upper dashed line). It should be recognized the values foreseen for $\eta\gamma$ in ITER are $<0.15 \times 10^{20}$ AW^{-1} m^{-2}.

3.2. Prospects and implications of a credible design and/or operation phasing approach for DEMO

A key consideration in the design and operation of DEMO is the degree to which a "staged operation" approach would be possible, aimed at developing the plasma physics, materials science and technology over the operational lifetime of the power plant to realize performance improvements through feasible system or component upgrades. This approach could be achieved, at least in principle, by learning from the experience gained from operating such a device and by extending its nuclear capability step-by-step (e.g. upgrade of blanket, divertor, heating and current drive systems, diagnostic systems, etc.) in a similar fashion to past approaches taken with most magnetic confinement experiments.

Existing devices have been successfully operated and their performance substantially improved through significant design modifications and machine upgrades throughout their lifetimes. A good example of where a staged operational approach has been applied is JET. Substantial increases in auxiliary power, an internal poloidal divertor and significant remote handling capability have been implemented at various points during JET's operational lifetime. JET was designed with sufficient margins to prepare for the envisaged upgrade paths and, notably, the inclusion of D-shape TF coils proved to be a very important design decision that allowed the subsequent inclusion of a divertor. In addition, RH tools have been instrumental in the refurbishment/repair of components in conditions where machine access was very limited.

Indeed, the need to include sufficient flexibility in the design of DEMO to accommodate improvements in plasma performance and design improvements of core components is generally supported. However, for a nuclear fusion reactor (like DEMO), this flexibility is likely to be much more limited and it is not currently clear to what extent this can be achieved. Due to many interrelated aspects, it will be necessary to investigate in detail the feasibility and impact on design margins and machine performance.

Design staging is not an improvisation or a one-off modification but must be carefully thought out, planned and continuously managed. In addition, there is a need to understand how the embedded flexibility affects other system attributes such as performance, safety, risk and cost. Particular emphasis must be given to achieving a reasonable plant availability i.e. reliability and maintainability of components (design and manufacturing simplification, cost minimization). Furthermore, consideration of the impact that upgrades may have on established interfaces must be given attention e.g. the blanket remote maintenance tooling and procedures will be designed to suit a particular configuration of cooling pipes, structural attachments and other service connections.

Defining the required system performance in the initial operational period must be accomplished through trading-off what can reasonably be achieved with current state-of-art technology and what performance improvements could be provided by feasible future upgrade paths.

In Section 3.1 we have discussed the possibility to extend the purely inductive pulse duration by auxiliary H&CD systems to be installed at a later stage, notwithstanding the impact on the electrical output, which will be reduce if not properly designed for. In addition, it is currently proposed to utilize a "starter" blanket with a 20 dpa damage limit in the first-wall steel using moderate performance structural materials and conservative design margins and then switch to a second set of blankets with a 50 dpa damage limit and an optimized design, and if available, improved structural materials. This type of approach has been used for the fuel cladding in fission reactors for many years; by limiting the maximum exposure level of the replaceable cladding to below the regulatory limit, while data for higher exposure operation is generated in test reactors or load test assemblies. This approach benefits from the multiple-barrier safety approach in fission reactors, including the pressure vessel as a key safety boundary for regulatory approval. Licensing approval for operation up to moderate exposures could be obtained for the "starter" blanket, while high-dose engineering data for a more advanced materials blanket is being generated. In addition, the benefit of this "progressive" approach would also include the possibility to start with a less optimized thermo-hydraulic or mechanics design (larger safety margin) to cope with large uncertainties in the overall reactor loadings and performances.

3.3. In-vessel lifetime design requirements and materials challenges for DEMO

A crucial point is that the structural materials of the in-vessel components used in existing fusion machines and also in ITER are not considered to be relevant for DEMO or will be exposed to radically different operating conditions. For example, due to considerations of reduced activation, thermodynamic efficiency and neutron irradiation effects, the austenitic stainless steel used for ITER and several other large-scale plasma machines is expected to be replaced in DEMO for in-vessel components structural applications by reduced activation ferritic/martensitic steel or another structural material that can operate at much higher temperatures and displacement damage doses. Significant uncertainties exist regarding allowable component fabrication procedures, operating temperature windows, and anticipated lifetimes. Near-term research activities that would help to reduce some of these uncertainties are summarized in [3]. There are important lessons to be learned from fission reactor material development, especially in safety and licensing, fabrication/joining techniques and development of manufacturing and supply-chain.

The main materials relevant features of the current near-term DEMO are listed below [3,4]:

High divertor power handling, i.e. the ability to withstand power loads larger than 10 MW/m^2. To cope with this, use of water and copper alloys as in ITER is being considered. The radiation damage from the neutronics simulations of the divertor show that the predicted damage for the tungsten divertor armour would be ~3 dpa/fpy, whilst if copper were the coolant interface material in the high-heat-flux components of the divertor, the radiation damage would be a maximum of about 5 dpa/fpy, but would be as low as ~3 dpa/fpy in the strike zone areas [13]. It is important to note that particle sputtering is going to be very likely limiting the lifetime of the divertor armour in DEMO and current rough estimates [14], that need to be further verified, predict required armour replacements approximately every 2 fpy. The present divertor RH strategy, assumes a whole divertor replacement because, in order to minimize the maintenance downtime, it is currently not proposed to perform complex robotic handling to remove the armour from the substructure and replace with new armour before re-installing the whole structure. This operation time span roughly matches with the irradiation limit of about ~5 dpa that the copper–alloy structures could withstand with tolerable degradation of mechanical strength.

Near-term DEMO should act (at least) in its first phase of operation as a “component test facility”. For example, it will utilize a “starter” blanket configuration using moderate-performance materials (which will not affect regulatory approval) and then switch to blankets with a more advanced-performance material after a limited accumulated MW year/m^2. A similar philosophy might be applied to the divertor. A “starter” blanket should be designed using moderate-performance materials (which will not affect regulatory approval) and capable of withstanding ~20 dpa damage in the blanket front-wall steel. The second blanket should be capable of lasting up to 50 dpa.

The replacement of blankets or divertors cannot be accompanied by a complete change of the Balance of Plant (BoP), as this is clearly unfeasible. Thus, the series of blanket concepts and divertor concepts must each assume the same coolant for the entire lifetime (although the divertor and blanket coolants could, in principle, be different).

Lessons from the fission programme highlight the use of precedents in licencing a new reactor. For fusion, ITER licensing experience can be used to refine the issues in nuclear testing of materials. As with ITER, if the DEMO safety case focuses on identifying the lightly-irradiated vacuum vessel as the primary safety boundary, then materials qualification prior to DEMO licensing will change radically. At the vacuum vessel wall irradiation damage levels ~0.1 dpa will be encountered over the lifetime of DEMO with levels of helium transmutation also very low (below 1 appm, resulting from the much softer neutron spectrum at the shielded vessel wall). This would limit the scope of the required 14 MeV neutron spectrum testing for licensing the DEMO design to very modest levels, and allow a more limited scope 14 MeV testing programme to be focused on yielding results for engineering code development and design support for the materials of the heavily-irradiated blanket first wall structure. Nevertheless testing with 14 MeV neutrons is essential to fusion materials development. To do this in a timely manner requires deployment of a ≥30 dpa (steels) 14 MeV testing capability by the middle of the next decade. The optimization of the testing programme by the pre-testing with fission neutrons on isotopically- or chemically-doped steels and with ion-beams is discussed along with the minimum requirements of the 14 MeV testing programme itself.

4. Design readiness and strategy for critical technologies for DEMO

A number of outstanding technology and physics integration issues must be resolved before a DEMO plant concept selection is made. Each of them has very strong interdependencies. They include the selection of (i) the breeding blanket concept and, in particular, the selection of blanket coolant and the BoP (ii) the divertor concept and layout configuration (iii) the first-wall design and integration to the blanket (mechanical and hydraulic) taking into account that the first-wall might see higher heat loads than assumed in previous studies (iv) the H&CD mix including minimum pulse duration and (v) the remote maintenance scheme. The impact on the overall plant reliability and availability of the various system design options will be analyzed in an integrated approach.

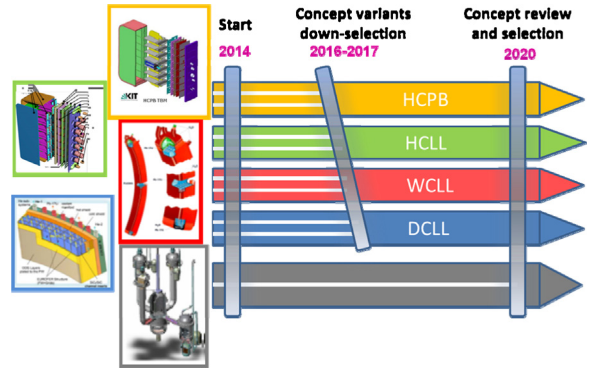

Fig. 5. Breeding blanket concepts to be developed: Helium-cooled Pebble Bed HCPB/Helium-cooled Lithium Lead (HCLL blanket concepts complementary to the TBM Programme. Water-cooled Lithium Lead (WCLL). Dual-Cooled Lithium Lead (DCLL) blanket concept.

ITER construction and operation will represent a huge accomplishment and a significant step forward in many fusion engineering areas. ITER will demonstrate integrated technologies for confining, heating, fuelling, and burn control of the plasma. These include low-temperature superconducting magnets (LTSC); plasma heating technologies like Neutral Beam Injection (NBI), Electron Cyclotron Heating (ECH), Ion Cyclotron Heating (ICH); pellet injection, vacuum and tritium processing (except tritium recovery from the blanket). In most of these areas, modest R&D is foreseen in the conceptual design phase of DEMO, aiming mainly at design and performance optimization (e.g., see [15]).

We briefly discuss here the areas with largest uncertainties that have a large impact on the concept definition and that require extensive development in parallel to and beyond ITER.

4.1. Key design drivers

4.1.1. Breeding blanket technology and coolant

The choice of the coolant has substantial impact on the design, operation, maintenance, safety and economics of DEMO. It is generally agreed that water should be considered as the divertor coolant for a near-term DEMO design as the divertor surface heat flux conditions prove to be beyond present helium power handling capabilities. However, the choice of the breeding blanket coolants is still open. Technical issues influencing the choice include: (i) thermal power conversion efficiency; (ii) pumping power requirements; (iii) power handling requirements of the first-wall; (iv) *n*-shielding requirements (e.g., reduce the blanket thickness that is critical at the inboard side); (v) achievable tritium breeding ratio; (vi) breeder tritium extraction; (vii) tritium permeation and primary coolant tritium purification and control; (viii) chemical reactivity, coolant leakage; and (ix) design integration and feasibility of BoP.

The breeding blanket is one of the most important and novel parts of DEMO. Large gaps would exist even with a successful TBM programme. In view of the existing performance uncertainties and feasibility concerns, R&D must be strengthened and a selection now is premature, without conducting the required R&D [2]. A sustained programme of technology R&D is required to reduce the risks to the DEMO blanket development that cannot be fully explored in ITER, and/or to develop adequate knowledge to evaluate alternatives to the mainline concepts. R&D and design activities foreseen in Europe on breeding blanket are summarized in Fig. 5 and ref. [16]. The ambitious goal is to achieve a down selection to the best concept by 2020 that includes design integration and feasibility of BoP.

If, as a result of the design and R&D work, a different breeding blanket module needs to be tested in the ITER TBM programme, this will have to be done during ITER phase-2 with a delay of the DEMO blanket development programme. Possible risk mitigation may arise from some sharing of information on the TBM programme among the ITER parties. In addition, China is designing a Chinese Fusion Engineering Test Reactor (CFETR) and this facility should start tritium operation around 2030. Options for potential participation in the exploitation of such a facility, for example, by testing alternative blanket concepts should be seriously pursued.

4.1.2. Divertor design configuration and technology

Significant progress has been made during the last two decades (thanks to the ITER programme) in the development of technologies for divertor high-heat-flux-components cooled with either water or helium. Water-cooled prototypes fabricated both with carbon and tungsten have been successfully tested under cyclic loads of up to 20 MW/m^2; whilst helium-cooled solutions have been found to withstand 10 MW/m^2 for a large number of cycles. It should be recognized that these represent upper technological limits set by the intrinsic limitations of the thermo mechanical properties of the limited number of suitable materials. In addition, in the case of exposure under large *n*-irradiation fluences, such as those expected in DEMO, the power handling limits above will be reduced.

In a tokamak power plant the loss power would be a factor of 5–10 greater than that in ITER, while the area of the divertor high heat flux components will probably differ by less than a factor of 2. It will therefore be necessary to radiate the majority of the loss power, whilst still maintaining sufficient confinement and plasma purity. At present, the most favoured approach is to use impurity seeding to establish highly radiating edge and divertor plasmas, combined with core radiation, to distribute the exhaust power over a large wall area. This would be combined with tungsten plasma-facing-components (PFCs), to ensure an acceptable interval between PFC replacements.

Although present day machines have demonstrated highly radiating scenarios (e.g. ASDEX-Upgrade), several differences compared to DEMO make the results difficult to extrapolate. Firstly, to achieve a similar radiated power fraction in the edge for DEMO parameters requires a much higher radiation power density, and associated steep temperature gradients. Secondly, power handling in DEMO is achieved by reducing the target power to an acceptable level which is in the order of 10 MW/m^2, not close to zero as in present experiments. This means that effects such as thermal instabilities and changes in confinement cannot be directly extrapolated to DEMO.

In addition to developing high radiation scenarios in standard X-point configurations, it is also prudent to look into solutions using advanced magnetic configurations, where issues such as thermal stability and radiating volume may be different, or advanced plasma facing materials targets, such as liquid metals, which could reduce the radiation requirements. Emphasis should be on the heat exhaust capability of the solutions proposed during normal conditions but also with a view to the occurrence of possible non-controlled transients. More specifically, feasibility studies aimed at objectively and rationally determining and comparing the strengths and weaknesses of the various configurations (e.g. snowflake, X-divertor, super-X divertors, etc.) are urgently needed. In particular, an analysis of the implications of the target configuration on the mechanical integration and maintenance scenarios, and an investigation of the design integration and engineering constraints arising from the use of large current coils in the divertor region, or of liquid metal, in the case of LM targets.

It is recognized that the final concept selection of the divertor bears a strong impact on the machine design, parameter selection and operation scenario development. Hence, this problem must be tackled from the outset and until uncertainties are significantly reduced through a well-focused and vigorous R&D program, any conceptual design proposals remain questionable.

Activities in Europe are oriented along three main lines: (1) conventional radiative divertors (2) divertors with advanced magnetics configurations (e.g. snowflake and super-X) (3) novel divertor ideas, e.g. liquid metal divertors. Progress in this area is described elsewhere (e.g., see [4])

4.1.3. First-wall protection and integration to the blanket

An important decision in the evolution of the first-wall/blanket design is whether or not to make the first-wall hydraulically and mechanically integrated with the blanket or whether to make it, as ITER plans, separable from the blanket for replacement or repair operations. The first-wall and the blanket have a number of fundamentally different design requirements and functions. Also, previous design studies have predicted first-wall design lifetimes considerably lower than the breeding blankets. A first-wall blanket design with an easily separable first-wall is, in general, more complex than the one in which the first-wall and blanket are integrated both mechanically and hydraulically. The design of a first-wall that can be separated from the blanket represents a very complex challenge (i.e. large amount of structural material and coolant will be detrimental for the breeding capability; will require more radial space and access to permit simple first-wall removal operation (increase reactor size for a given neutron wall loading).

4.1.4. H&CD systems capabilities for DEMO

In a DEMO based on long-pulse/inductive regimes of operation and not on fully steady-state operation, H&CD systems primarily need to provide heating power for H-mode access, suppression of MHD instabilities and increase of the pulse length; whereas the detailed control of the equilibrium current density profile will not be the primary requirement. This choice will reduce the risks on the DEMO H&CD systems and is in line with the pragmatic approach advocated in the EU Roadmap. ITER will test the potential of different heating systems (ICH, NBI and ECH, with Lower Hybrid CD as a possible upgrade to be decided at a later stage) from the point of view of their application to DEMO regimes of operation. The approach is to pursue only specific developments to comply with the parameters of DEMO (e.g. higher magnetic field) and to ensure high system availability (e.g. by minimizing the need of maintenance outside the scheduled periods), reliability (e.g. by ensuring the modularity of the systems) and plant efficiency (e.g. by minimizing the re-circulating power). This will involve mainly R&D activities in the area of NB and EC technologies without excluding design integration assessments for all the systems. These activities will be complemented by an analysis of the plasma regimes in DEMO to guide the final decision on the H&CD systems to be taken on the basis of the ITER experience. An assessment was recently conducted to determine the CD efficiency of ICCD, LHCD, ECCD and NBCD for DEMO case studies and the results are discussed elsewhere [17]).

4.1.5. Remote maintenance

The anticipated high neutron flux and the consequent damage to in-vessel components in DEMO will result in the need to regularly replace the breeding blankets and divertor components. Characterization of the shutdown radiation environment has shown that in-vessel absorbed dose rates will range from around 2 kGy/h at 1 week following shutdown to 0.5 kGy/h at 1 year following shutdown. This is an important input in the feasibility assessment and design of RH equipment and prohibits the use of dexterous RH equipment inside the vessel.

A number of proposals to implement in-vessel component removal and replacement have been proposed, for example:

large port maintenance systems—such as DREAM, CREST, SlimCS, and the US ARIES-RS and ARIES-AT, and vertical maintenance systems—namely NET and other European concepts—where vertically arranged multi module blanket (MMS) segments are extracted through a vertical access port.

There are many merits to the vertical maintenance system, including: smaller toroidal field coils, simplified coil structural supports, smaller vacuum vessel closure plates, and reduced size and scope of containment cask handling and hot cell activity. This system, however, does require complex in-vessel and ex-vessel remote handling equipment.

Conceptual designs for a set of remote handling equipment for the vertical maintenance scheme are being developed incorporating designs for in-vessel transporters, to detach and attach the blanket segments and divertor cassettes, and cask-housed vertical maintenance devices to open and close access ports, cut and weld service connections, and extract blanket segments from the vessel [18]. Further study is needed in order to comprehensively establish the feasibility and evaluate the efficiency of the proposed maintenance system however it is clear from the outset that the design of in-vessel components and the vessel itself is intrinsically linked the design of the remote maintenance system. It is of critical importance that these design activities are progressed at the same rate and with close integration and collaboration.

4.1.6. Balance of plant

Exploratory power cycle modelling and assessment of technology maturity highlight the water/steam-based Rankine cycle as an appropriate choice for DEMO [19]. For example, cycle simulations with a water-based divertor (<250 °C) and helium-cooled blanket (300–500 °C) indicate water/steam-based Rankine cycles are able to meet the required net plant efficiency target of 25% via a cycle incorporating use of divertor heat, reheat and feed heating. Such a cycle offers substantial operational precedence and low levels of technical risk for key components. Further work is currently underway to assess the most applicable variation of this cycle for use with the WCLL blanket.

The performance benefits, risks and recent precedence for novel cycles such as supercritical carbon dioxide Brayton cycles, are also being assessed to understand the viability of options beyond water/steam-based Rankine cycles. However, it is clear that high temperature options such as helium-based Brayton cycles are not applicable for the current DEMO blankets due to the necessity for primary coolant temperatures well in excess of 500 °C in order to meet the net plant efficiency target.

5. Summary

The demonstration of production of electricity before 2050 in a Demonstration Fusion Power Reactor (DEMO) that produces its own fuel represents the primary objective of the fusion development programme in Europe. The approach followed in Europe to achieve this goal is outlined in this paper together with a preliminary description of the design solutions being considered and the R&D strategy required to resolve outstanding challenges that still lie ahead. ITER is the key facility in this strategy and the DEMO design is expected to benefit largely from the experience that is being gained with the ITER construction. Nevertheless, there are still outstanding gaps that need to be overcome requiring a pragmatic approach especially to evaluate and improve through dedicated physics and technology R&D the readiness of the foreseeable technical solutions. A system engineering approach is needed and industry must be involved early in the DEMO definition and design. Availability of sufficient resources and an adequate implementing organization are prerequisite to success.

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References

[1] D. Maisonnier, D. Campbell, I. Cook, L. Di Pace, L. Giancarli, J. Hayward, et al., Power plant conceptual studies in Europe, Nucl. Fusion 47 (2007) 1524.
[2] F. Romanelli, Roadmap to the Realization of Fusion Energy, 2014, Submitted for publication.
[3] D. Stork, The EU Roadmap Materials Assessment Group, Materials R&D for a timely DEMO: Key Findings and Recommendations of the EU Roadmap Materials Assessment Group, 2014, Submitted for publication.
[4] G. Federici, G. Giruzzi, C. Lowry, R. Kemp, D. Ward, R. Wenninger, et al., EU DEMO Design and R&D Studies, 25th Symp, Fus. Eng. (2013), June 10–14, San Francisco USA.
[5] J. Harman, G. Federici, R. Kemp, The application of systems engineering principles to the European DEMO design and R&D studies, in: 25th Symposium on Fusion Engineering, 10–14 June 2013, San Francisco, USA, 2013.
[6] Report of the Ad hoc Group on DEMO Activities, 3/2010.
[7] N. Taylor, P. Cortes, Lessons Learnt from ITER Safety and Licensing for DEMO and Future Nuclear Fusion Facilities, 2014, Submitted for publication.
[8] H. Zohm1, C. Angioni, E. Fable, G. Federici, G. Gantenbein, T. Hartmann, et al., On the Physics Guidelines for a Tokamak DEMO, Nucl. Fusion 53 (2013) 073019 (6pp).
[9] P.J. Knight, A User's Guide to the PROCESS Systems Code, 2.1.0. ed., UKAEA Fusion, 1996.
[10] J.L. Duchateau, Influence of the magnetic toroidal field on the design of magnet systems for future fusion reactors, Fusion Eng. Des. 81 (2006) 2351–2359.
[11] Y.R. Martin, T. Takizuka, ITPA CDBM H-mode Threshold Database Working Group, Power requirement for accessing the H-mode in ITER, J. Phys. Confer. Ser. 123 (2008) 012033.
[12] A. Kallenbach, M. Bernert, R. Dux, L. Casali, T. Eich, L. Giannone, et al., Impurity seeding for tokamak power exhaust: from present devices via ITER to DEMO Plasma Phys, Control. Fusion 55 (2013) 124041, http://dx.doi.org/10.1088/0741-3335/55/12/124041.
[13] M.R. Gilbert, S.L. Dudarev, S. Zheng, L.W. Packer, J.-Ch. Sublet, An integrated model for materials in a fusion power plant: transmutation, gas production, and helium embrittlement under neutron irradiation, Nucl. Fusion 52 (2012) 083019 (12 pp.).
[14] R.P. Wenninger, M. Bernert, T. Eich, E. Fable, G. Federici, A. Kallenbach, et al., DEMO divertor limitations during and in between ELMs, submitted in Nucl Fusion.
[15] P. Bruzzone, Pre-conceptual studies and R&D for DEMO Superconducting Magnets, Submitted for publication.
[16] A. Li-Puma, L.V. Boccaccini, C. Bachmann, P. Noriajitra, P. Sardain, Design and Development of DEMO Blanket Concepts in Europe, Submitted for publication.
[17] H. Zohm, E. Barbato, I. Jenkins, R. Kemp, E. Lerche, E. Poli, et al., Assessment of H&CD System Capabilities for DEMO, EPS Finland (2013).
[18] A. Loving, O. Crofts, J. Harmon, U. Fischer, J. Sanz, M. Siukko, et al., Pre-conceptual Design Assessment of DEMO Remote Maintenance, Submitted for publication.
[19] M. Porton, H. Latham, Z. Vizvary, E. Surrey, Balance of plant challenges for a near-term EU demonstration power plant, in: 25th Symposium on Fusion Engineering, 10–14 June 2013, San Francisco, USA, 2013.