Assessment of DEMO Challenges in Technology and Physics

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The challenges that DEMO designs encounter in both technology and physics are reviewed. It is shown that it is very important to respect the interlinks between these fields when developing designs for DEMO. Examples for areas where such interlinks put very strict requirements are the development of a steady state tokamak operation scenario and the question of power exhaust taking into account the boundary conditions set by materials questions. Concerning steady state operation, we find that demands on the physics scenario are so high that pulsed operation of a tokamak DEMO should seriously be considered in conservative DEMO designs. Alternatively, the device could foresee a large fraction of externally driven current which calls for optimization of both plasma CD efficiency as well as wall plug efficiency of the CD system. In the exhaust area, a realistic estimate of the admissable time averaged peak heat flux at the target is of the order of 5 MW/m^2, leading to strict requirements for the operational scenario, which has to rely on an unprecedented high level of radiation loss by impurity seeding and the facilitation of partial detachment. Thus, exhaust scenarios along these lines have to be developed which are compatible with the confinement needs and the H-L back transition power for DEMO. In both areas, we discuss possible risk mitigation strategies based on conceptually different approaches.

Keywords: tokamak fusion reactor, pulsed DEMO, conservative DEMO.

1. Introduction

With ITER construction under way, the question of possible designs for a machine after ITER is receiving renewed attention. In many fusion energy roadmaps, such a step, often called DEMO, is a machine that should bridge the gap between ITER and the first commercial Fusion Power Plant (FPP) (see e.g. [1]). Crucial elements for many DEMO designs are hence to

- demonstrate a workable solution for all physics and technology questions
- demonstrate large scale net electricity production with self-sufficient fuel cycle
- prove high reliability and availability over a reasonable time span
- allow the assessment of the economic prospects of an FPP.

For such a DEMO design, a consistent set of physics and technology assumptions has to be made. In many areas, the interlink between the two fields is quite strong and poses boundary conditions that have to be respected from both sides. One example for an area where such interlinks put very strict requirements is the development of an operation scenario that would yield steady state tokamak operation taking into account the controllability of the scenario with the limited sensors and actuators available in a DEMO environment. Another example is the question of exhaust of power through a divertor taking into account the boundary conditions set by materials questions, both in terms of thermal loading as well as the expected level of radiation damage (dpa) necessary to demonstrate a credible route to an FPP with high reliability and availability. For both areas, recent progress is described and implications for both fields are discussed. Finally, risk mitigation strategies in the areas identified as most critical are highlighted.

2. Technology and Physics Challenges

The list of challenges in technology and physics for DEMO has been discussed before on several occasions. Here, we review them based on work by an EU group in 2010 [2]. There, DEMO challenges were defined as 'not necessary for ITER to reach its goals, but absolutely vital for DEMO', consistent with the assumption that ITER will successfully fulfill its mission.

2.1 Technology challenges

According to the criterion given above, these can be roughly grouped under

- development of so-called 'Enabling Technologies' (Remote Handling, Heating and Current Drive systems, Diagnostics and Control, Tritium processing and Superconducting Magnet Technology) concerning the maintenance, the efficiency requirements in terms of energy conversion and the availability that have to apply to a power plant [3];
- radiation resistant materials qualification for designing components with adequate lifetime, up to many dpa for the first wall elements, while keeping the promise of low radiological burden;
- performance and durability of in-vessel components, especially the Breeding Blankets necessary for the tritium self-sufficiency requirements of operation and Divertor/Plasma Facing systems, that will be driven by the extreme heat and neutron loads.

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2.2 Physics challenges

For a tokamak DEMO, the following challenges beyond those encountered for ITER are identified:

- steady-state tokamak operation: under this we summarise the whole challenge of achieving an appropriate scenario, e.g. at high bootstrap fraction and the associated MHD limit(s), ITB formation and control with external knobs (H&CD systems) in an alpha-dominated plasma, etc.;
- operation at high density: due to the unfavourable scaling of Greenwald density n_G with size, it seems unavoidable to operate a DEMO at or above n_G, which is worrying in terms of confinement and disruption danger.
- power exhaust: the PPCS study [4] has shown that pushing a DEMO towards economic attractiveness increases the power exhaust problem into a parameter space where either power handling of PFCs and first wall components is dramatically improved or solutions are found where a very large fraction of power has to be radiated before it reaches the plates. Also, it is not clear if present tools proposed for ELM mitigation in ITER (pellet pacing, in–vessel coils) are also DEMO compatible.
- disruptions: in DEMO, the disruption problem goes beyond machine protection because it can make the whole concept unattractive;
- control: availability of sensors (diagnostics) and actuators (H&CD, fuelling systems) on DEMO will pose strong boundary conditions and should be treated in an integrated manner The DEMO scenario will have to be compatible with the available sensors and actuators.

It should be noted here that stellarators promise to be advantageous compared to tokamaks in the first two points and also the fourth, but of course based on a data base that is far from the maturity of that of tokamaks.

3. Interlink between technology and physics

The above mentioned challenges have been discussed for quite some time, but often the discussion of a single topic leads to the formulation of requirements for another topic that may be unachievable. We therefore propose to strengthen the link between technology and physics and come to an integrated approach that applies the same level of optimism/realism for each item under discussion. This iteration can lead to a hierarchy of design decisions which then also helps to highlight where progress in a certain field would be especially helpful. In the following, we discuss two examples for such a procedure.

3.1 Implications of steady state tokamak operation

Inductive operation of the tokamak implies pulsed generation of fusion power since the OH solenoid has to be recharged at some point. Since this implies the need for storage systems if, as generally assumed, continuous electricity output is required and also leads to mechanical and thermal cycling, solutions for non-inductive, steady state tokamak operation have been studied since the 1990s. Here, the current should be driven either by external current drive (CD) systems (ECCD, NBCD, LHCD, ICCD) or due to the intrinsic bootstrap effect. However, the CD efficiency of these systems is low, typically of order 0.05 A/W under DEMO conditions, so that the impact on the recirculating power is high and a large fraction of the current should be driven by the bootstrap effect. This in turn means high normalised pressure β, which seriously challenges stability limits. Fig. 1 shows this for an 'ITER-like' DEMO with major radius R=7.5 m, aspect ratio 3.1, magnetic field B_t = 5.2 T and plasma current I_p = 16 MA (q_{95} = 3.5) [5]. Conventional technology has been assumed, i.e. thermodynamic conversion efficiency η_{TD} = 0.3 and wall plug efficiency of the H&CD system η_{CD} = 0.25. As β_N is varied, the fusion power varies from 750 MW at β_N = 2 to 3 GW at β_N = 4.

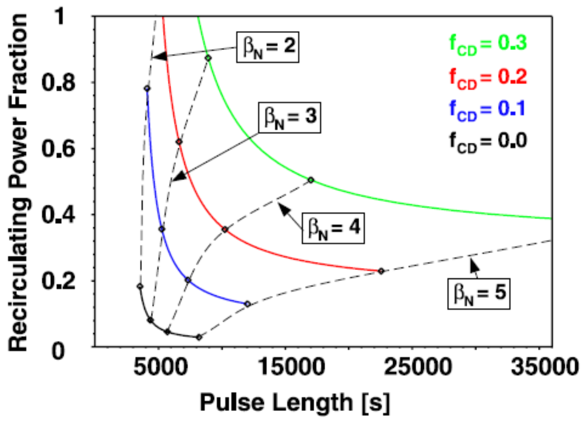

Fig.1 recirculating power fraction versus pulse length for an 'ITER-like' DEMO (parameters given in the text) for a scan of β_N and f_{CD} [5].

It can clearly be seen that under these assumptions, long pulses can only be attained with either large fraction of external CD f_{CD}, at the cost of high recirculating power, or assuming very high normalized pressure β_N, exceeding the values which are routinely achieved in present day tokamaks. A recent survey of experimental results from tokamak discharges at high normalized pressure has indeed shown that the conditions assumed here, namely β_N around 5 at low q_{95} (at least below 4) have not been achieved so far, despite the large experimental effort worldwide, the main obstacle being the MHD stability limits under these conditions [6, 7]. Hence, a 'conservative' DEMO will most likely be a pulsed device and we propose to seriously consider this option in future, which means analyzing the impact of pulsed operation on machine design. Present estimates for the recharge time are of the order of 30 min to 1 hr so that in this time, a storage system would be needed should continuous electricity supply be envisaged. The problem of thermal and mechanical cycling awaits an in-depth analysis and its implications on the design should be clarified in the coming years.

As pointed out above, integrated plasma operation scenarios demonstrating the stability requirements outlined in stationary conditions above do not exist. While research in this 'Advanced Tokamak' area is an active field, we have analysed the prospects for long-pulse scenarios along the lines discussed above. These rely on a more modest extrapolation of the ITER Q=10 H-mode scenario and are known as 'hybrid', 'improved H-mode' or 'advanced inductive' scenarios. They rely on the fact that H-mode operation at β_N around 3 (ITER Q=10: β_N = 1.8) has been demonstrated in several devices in these scenarios [6] and confinement quality exceeding the ITER scaling is usually observed under these conditions at a level of H = τ_E / $\tau_{E,ITER}$ =1.2 - 1.5. This would allow a simultaneous decrease of plasma current that leads on the one hand to longer pulses and on the other hand to more operational stability by increasing the safety factor q_{95}. DEMO devices optimized along these lines would employ external CD and could achieve pulse lengths of several hours [5].

On the technology side, this line of developments calls for optimization of the wall plug efficiency η_{CD} of CD systems since the recirculating power is determined by the product η_{CD} γ_{CD}, where γ_{CD} is the plasma CD efficiency. We note here that recent optimisation shows that ECCD could reach values of γ_{CD} of 80 % of those predicted for NBCD [7], while previous estimates had predicted 50 % as the maximum achievable. With this finding, and noting that the high η_{CD} of gyrotron is likely to remain above that of NBI, ECCD is an attractive candidate for DEMO.

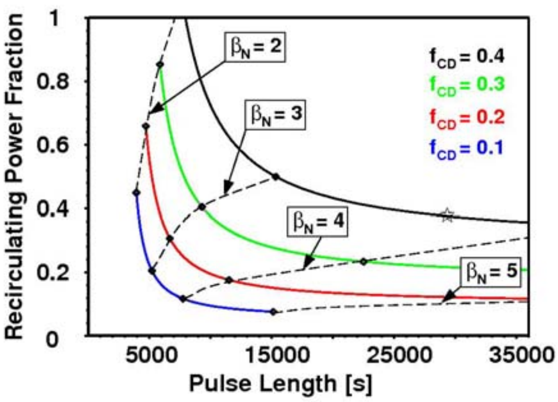

Fig.2 same as Fig. 1, but under the assumption of 'improved H-mode' operation (H=1.2) and improved efficiency of the CD system, both in plasma physics and in technology (η_{CD} x γ_{CD} = 0.2). The operational point discussed in the text at β_N=3.5 is marked as a star on the f_{CD}=0.4 line.

To illustrate this, Fig. 2 shows a plot similar to Fig. 1, but with H = 1.2, allowing to lower the current to 14 MA. We also have assumed more efficient current drive (γ_{CD}=0.4 and η_{CD} = 0.5). An operation point at 8 hrs pulse length and 37 % recirculating power, indicated by the star symbol in Fig. 2, can be found for f_{CD} = 0.4 and β_N = 3.5, consistent with 'hybrid' operation. Here, the net electrical power is of the order of 500 MW.

3.2 Implications of the exhaust problem

To our present understanding, exhaust of power and particles in a DEMO tokamak will be handled by a poloidal divertor. Contrary to present day experiments, also the main chamber first wall heat and particle load will require careful consideration. In fact, our present knowledge about these is poor and does not allow reliable extrapolation, so it is an area where more research on present day devices, together with adequate modeling, should be conducted. Another important aspect for the main chamber wall is that the dpa will have to be quite high (at least an order of magnitude higher than for the divertor) for reasonable life time. It will have to be assessed in light of these boundary conditions if all first wall component will have to be covered with a sacrificial W layer (which might in turn affect the breeding) or if areas in the main chamber can be left uncovered as bare steel (EUROFER) wall.

In the exhaust area, the main interface between technology and physics is the tolerable heat and particle flux (which, together with the divertor plasma temperature, determines the target erosion) that arrives at the divertor target. The problems in designing a divertor target element that can take a reasonable heat load in the DEMO environment are large. A recent analysis indicates that, depending on the coolant, a time averaged peak heat flux of 5-10 MW/m^2 must not be exceeded if a life time of about 2 full power years (fpy) is envisaged, which will lead to neutron damage of the order of 5 dpa [9], depending on the thermal power and the material choice. We note that this number is consistent with the expected erosion limits only if the divertor plasma temperature is below 5 eV [10]. Hence, an increase of the power handling capability cannot easily be exploited since the lifetime will then be dominated by erosion and shorter than the dpa limitation. This means that a significant rise in the power handling capability can only be expected from genuinely different technological approaches, such as the use of liquid metal as target material (see Section 4). Thus, for 'conventional' PFCs, we are left with a strict boundary condition of the target heat load of 5-10 MW/m^2 and a divertor plasma temperature not exceeding 5 eV.

The technology used for a divertor design has to be chosen according to the cooling concept: if water cooling is assumed (as a follow-up on the ITER technology), the coolant temperature is relatively low (below ~350 ^oC) and W-based materials cannot be used as structural material due to the severe radiation embrittlement in this temperature window (the use of W as plasma facing material is still assumed in order to meet the erosion requirements). Hence, the choice of Cu-based heat sinks would be a logical step, but the established CuCrZr as structural material seems not applicable due to embrittlement and strength issues. Hence, development is needed here. We also note that the use of Cu implies high activation and the use of water a potential safety issue with the detriation of the cooling water, so this approach might not be suitable to exploit the full benefits of fusion. Alternatively, if He is used as a coolant, the coolant temperature can be much higher and W-based materials could be used. This approach has been followed up at KIT and a prototype of a He-jet cooled finger concept, illustrated in Fig. 3, was successfully tested for 3000 cycles at 10 MW/m^2, however under un-irradiated conditions. Recently, progress has been made in developing tungsten composite materials with higher ductility [11] applicable for designing larger components. In summary, both lines will require further developments and a consistent solution does not yet exist.

Fig.3 concept and individual parts of the 9-finger module of He-cooled divertor module developed at KIT [12].

On the physics side, recent experimental results indicate that under attached divertor conditions, the power decay length does not scale with major radius but rather with the poloidal ion gyroradius [13], which means that the wetted area in the divertor will only scale linearly with machine size. Extrapolation using the scaling for attached divertors leads to a power decay length of the order of 1 mm in the outside midplane for ITER and DEMO conditions. By flux expansion and target inclination, this number will be of the order of 1 cm at the target, but cannot easily be increased further since the angle of incidence for field lines in the divertor is limited to above 1-2 ^{o} because of alignment issues.

The linear increase of the wetted area with machine size at q_{95} = const. means that then, the previously advocated figure of merit for divertor similarity P_{sep}/R [14], where P_{sep} is the power crossing the separatrix, is indeed a measure of the severity of the exhaust problem from the plasma physics side. In DEMO, the total P/R is typically a factor of 3 higher than in ITER, which itself is about a factor of 1.5 - 2 higher than present day experiments. A simple estimate indicates that under these conditions, attached divertor operation is impossible in a DEMO (as is the case for ITER). Hence, the divertor has to be operated in an at least partially detached state where the plasma pressure along field lines in the Scrape Off Layer (SOL) is no longer constant and the neutral pressure starts playing a role in the pressure balance in front of the target plates. Here, the term ‘partial’ detachment refers to detachment at the strike line where the peak heat flux occurs, while in the region of less the plasma may still be attached. These conditions, in which recombination plays a role, are characterized by a low divertor temperature of less than 3 eV, compatible with the requirements outlined above. Under detached conditions, also reduced heat flux together with a broadening of the wetted area, thought to be due to increased radial transport, is found.

Detached divertor conditions are usually achieved by strong puffing of the working gas in the divertor that increases substantially the density there. Achieving the detached state can on the one hand be eased by closing the divertor geometry w.r.t. the main chamber, on the other hand by increasing the losses in the SOL and the divertor through additional power loss by electromagnetic radiation from externally introduced seed impurities. Here, typically noble gases are used due to their favorable recycling properties. In present day experiments, usually Ne or N (which, in a full-metal environment, also shows good recycling properties) are used due to their radiation characteristics.

We note that present modeling capabilities are not yet adequate to quantitatively predict the value of P_{sep} that can be handled for a given divertor design in accordance with the postulated target conditions. Experimentally, values of P_{sep}/R up to 7 MW/m have been demonstrated with partially detached divertor conditions compatible with DEMO requirements [15]. Estimates for ITER assume that around 15 MW/m can be extinguished, so further progress in this area is needed in experiment and theory to validate this assumption.

For DEMO, this number indicates that P_{sep} values of around 100 MW can be tolerated. Assuming at least 2 GW of fusion power, this implies that at least 75 % of the total power has to be radiated from inside the separatrix to be compatible with the allowable P_{sep}. Such high values have not been demonstrated experimentally with good H-mode confinement, but close to 70% was recently demonstrated in ASDEX Upgrade.

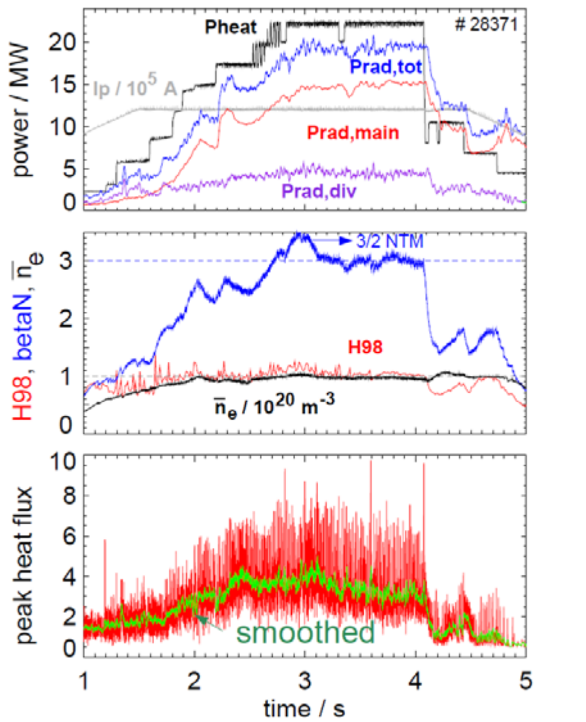

Fig.4 demonstration of operation at P_{tot}/R = 14 MW/m with both divertor conditions and plasma performance compatible with DEMO requirements on ASDEX Upgrade [15]. Here $P_{rad,main}/R$ = 9 MW/m is achieved, comparable to DEMO conditions. In a similar experiment, P_{sep}/R = 7 MW/m has been achieved.

Fig. 4 shows an example of such an experiment. This was enabled by applying double radiation feedback, controlling simultaneously divertor and main chamber radiation by injecting two different impurity species, namely Ar and N, in ASDEX Upgrade [15].

Another boundary condition to be taken into account is the H-L transition threshold power $P_{sep} > P_{sep,HL}$, which, in DEMO, will be of the order of the 100 MW mentioned above assuming the ITER scaling [16] also holds under highly radiative conditions. Thus, in this area, the technology limits drive the plasma physics solutions to be applied into a corner where the conventional divertor approach will at least be close to its limits.

Finally, we note that an increase of the thermodynamic efficiency η_{TD}, which could be achieved using the He cooling concept that allows higher coolant temperatures, would reduce the exhaust problem since less fusion power has to be generated for given electrical output power, but this implies a larger auxiliary power need due to the substantial pumping power needed for the He coolant so that it is also subject to further optimization studies.

4. Risk Mitigation

From the analysis presented above, it is clear that many areas require substantial progress to meet the ambitious targets. Detailed analysis of areas where progress is needed has been presented elsewhere and will not be repeated here. Rather, in this section, we will focus on some areas in which a conceptually different solution may offer a back-up solution to the ‘conventional’ path outlined above. The alternatives mentioned here are at present not developed to a level of maturity comparable to the ‘conventional’ solutions. Their development may hence be regarded as a ‘risk mitigation’ strategy on the path to a fusion power plant, in line with the recent assessment done by EFDA for the EU Fusion Roadmap.

4.1 Steady State operation

As outlined above, the achievement of steady state tokamak operation is not guaranteed and pulsed operation is considered an alternative, with its own drawbacks in terms of storage requirements and increased cyclic fatigue. An alternative to the tokamak is the stellarator, which does not need an intrinsic plasma current to generate the confining magnetic configuration. Stellarators hence are inherently steady state and do not have the disruption problem. In addition, present stellarator experiments allow operation at much higher plasma densities than the tokamak. The development of the stellarator line can hence be seen as a risk mitigation measure in the area of steady state operation.

The absence of an intrinsic plasma current means that stellarators cannot be axisymmetric, and this is the reason why their design and construction poses additional challenges. Designing the magnetic configuration needs computer-aided optimization, which has only become possible in the last two decades. The resulting 3-d structure of the machine leads to a more complex shape of individual components (including the coils) and also to a larger variety of individual component shape. For example, it is estimated that an advanced stellarator power plant would have about 25 different blanket module shapes, while in a tokamak, toroidal symmetry will reduce the number of different elements [17]. Nevertheless, stellarator reactor studies indicate the technical feasibility and maintenance concepts are proposed that can be compatible with DEMO and FPP requirements.

The 3-d nature does not only lead to additional complexity, but also to a larger freedom in configuration space. This is the reason why presently stellarators are less mature than tokamaks and quite different concepts exist, namely the HELIAS line (with W-7X the most prominent representative), the compact stellarator (e.g. NCSX) and the Heliotron line (e.g. the Force Free Helical Reactor FFHR). The plasma parameters in devices of this size are comparable to medium sized tokamaks, and hence the question arises how a possible roadmap towards an FPP based on the stellarator principle could look like.

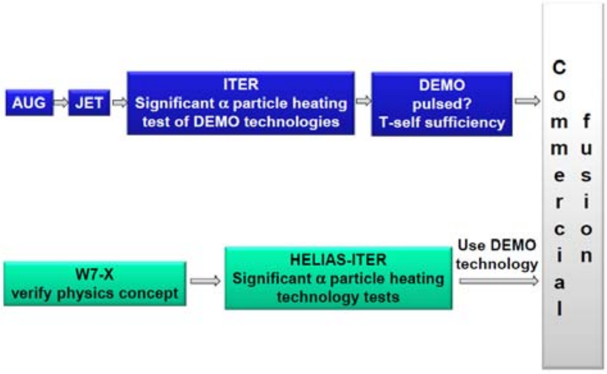

Fig.5 a possible roadmap to a commercial fusion reactor based on the stellarator principle.

A possible answer to this question is given in Fig. 5 for the HELIAS line, as discussed in [18]. It is based on the assumption that burning plasma physics in a stellarator cannot be predicted with a level of confidence that would allow to proceed directly from W7-X to an FPP. In particular, the physics of α-particle driven instabilities and their back-effect on the α-particle distribution in 6-dimensional space is a strongly nonlinear problem that should be addressed in a dedicated experiment with dominant α-heating. Due to the similarity with the goals of the ITER step for the tokamak line, this step is dubbed 'HELIAS-ITER' in Fig. 5. On the other hand, the technology goals of DEMO (e.g. the demonstration of T-self sufficiency with adequate blanket design) might be reached in a tokamak DEMO in a manner that would allow proceeding directly. Hence, the assumption has been made in Fig. 5 that the combination of results from a tokamak DEMO and a HELIAS-ITER will allow the construction of the first generation FPP as a stellarator.

4.2 Exhaust

As outlined above, present day materials limit the allowable time averaged target heat flux at the target to 5 MW/m^2, together with a divertor plasma temperature of less than 5 eV and the impurity content is small enough to be compatible with erosion life time. A significant increase of these numbers is expected for the use of liquid metals as target material. If these can be operated in a mode where the liquid is circulated in a closed loop, both problems could be addresseed simultaneously. However, present experiments with liquid Li in a porous system rely on the evaporation energy as heat sink and have not been demonstrated in closed loop so far. A credible closed loop design must be proposed before this can be considered a viable alternative. Another complication is the high uptake of T in Li which means that the coolant will effectively represent a second pump and ways will have to be found to detritiate the coolant in the inner loop. This problem might be addressed by using a different liquid metal, such as Ga, as coolant.

Another area where different concepts might constitute a risk mitigation is the use of alternative divertor magnetic geometries, allowing for a broadening of the SOL which should reduce the heat loads and ease detachment. Here, two concepts have been proposed, namely the Super-X divertor [19] and the Snowflake divertor [20] which use additional PF coil currents to increase the flux expansion. However, the currents required to form and control these may be excessive in a DEMO environment [21] and credible designs have to be demonstrated before these alternatives can be taken further.

5. Conclusions

We have argued that the challenges that DEMO designs encounter in both technology and physics should be analysed in an integrated manner respecting the boundary conditions set by both fields. We have analysed two areas in which the interlinks between the two fields are quite tight. Concerning steady state operation, demands on the physics scenario are such that pulsed operation of a tokamak DEMO should seriously be considered in conservative DEMO designs. Alternatively, the device could foresee a large fraction of externally driven current which calls for optimization of both plasma CD efficiency as well as wall plug efficiency of the CD system. In the exhaust area, a realistic estimate of the allowable time averaged peak heat flux at the target is of the order of 5 MW/m^2, leading to strict requirements for the operational scenario, including an unprecedented high level of radiation loss. Thus, exhaust scenarios along these lines have to be developed which are compatible with the confinement needs and the H-L back transition power for DEMO. Possible risk mitigation strategies in the two fields include the development of stellarators for inherent steady-state as well as the study of liquid metals as PFC and alternative magnetic divertor geometries in the area of exhaust.

Acknowledgements

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