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Overview of the DEMO Staged Design Approach in Europe

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1. Abstract.

This paper describes the status of the pre-conceptual design activities in Europe to advance the technical basis of the design of a DEMOnstration Fusion Power Plant (DEMO) to come in operation around the middle of this century with the main aims of demonstrating the production of few hundred MWs of net electricity, the feasibility of operation with a closed-tritium fuel cycle, and maintenance systems capable of achieving adequate plant availability. This is expected to benefit as much as possible from the ITER experience, in terms of design, licensing, and construction. Emphasis is on an integrated design approach, based on system engineering, which provides a clear path for urgent R&D and addresses the main design integration issues by taking account critical systems interdependencies and inherent uncertainties of important design assumptions (physics and technology). A design readiness evaluation, together with a technology maturation and down selection strategy are planned through structured and transparent Gate Reviews. By embedding industry experience in the design from the very beginning it will ensure that early attention is given to technology readiness and industrial feasibility, costs, maintenance, power conversion, nuclear safety and licensing aspects.

2. Introduction

As an important part of the Roadmap to Fusion Electricity [1,2], Europe is conducting a pre-conceptual design study of a DEMO Plant due to commence operation around the middle of the century with the main aims of demonstrating the production of few hundred MWs of net electricity, the feasibility of operation with a closed-tritium fuel cycle, and maintenance systems capable of achieving adequate plant availability [3]. This is currently viewed as the final crucial step towards the exploitation of fusion power after ITER, not only in Europe but by many of the nations engaged in the construction of ITER. The DEMO design and R&D activities in Europe are expected to benefit largely from the experience gained from the design, construction and operation of ITER, which remains the crucial machine on which the validation of the DEMO physics and part of the technology basis depends. Nevertheless, there are outstanding physics, materials and engineering challenges, with potentially large gaps beyond ITER that need to be urgently addressed. The main design challenges include: 1) large knowledge gaps in key reactor technologies not fully demonstrated by ITER that require further R&D; 2) design dealing with uncertainties (physics/ technology); 3) high degree of complexity/system interdependencies; and 4) integration of design drivers across different systems.

At present, the EU DEMO design has not been formally selected and detailed operational requirements are not yet available. However, the DEMO plant high-level requirements have been defined following interaction with an external stakeholder group composed of experts from industry, utilities, grids, safety, licensing, etc. The design should be capable of producing electricity (up to ~500 MWe), operating with a closed fuel-cycle and to be a facilitating machine between ITER and a commercial fusion power plant (FPP). The overarching principles

[a] on assignment by Fusion for Energy

of the DEMO development strategy in Europe include: (i) modest extrapolations from the ITER physics and technology basis to bound development risks; (ii) robust design incorporating proven technologies as well as innovations validated through realistic R&D programs; (iii) safety features and design licensability by integrating lessons learned from ITER licensing (and other existing nuclear facilities); (iv) a 'success orientated' approach of DEMO design development taking place in parallel to ITER exploitation, but relying on design and physics validation prior to construction; (v) harnessing the industrial base established in bringing ITER to fruition.

Contacts were also made within the Gen IV fission programme (ASTRID and MYRRHA) and ITER to learn from their experience. Both projects emphasized the following aspects: (i) the plant design should drive R&D and not the other way round. (ii) fusion is a nuclear technology and as such, will be assessed with full nuclear scrutiny by the regulator; (iii) the need for a traceable design process with a rigorous Systems Engineering approach; (iv) the technical solution should be based on maintaining proven design features to minimize technological risks [4].

Emphasis from the initiation of the project has been on the study of main design integration risks that affect the whole DEMO nuclear plant architecture, arising from remote maintenance, power conversion aspects, safety, licensing, and technology feasibility. Such work is essential to develop an understanding of the importance and relative difficulties of various design integration and technological problems to be solved in DEMO. This approach provides a very useful tool to identify and to investigate knowledge gaps in the proper design integration contest and to guide and to streamline the R&D programme towards clear R&D priorities. The lesson learned from ITER clearly shows the consequences of arriving with a low design maturity at the point of launching procurement activities. This has been mainly due to the propagation of design and technology changes imposed by the regulatory body as a result of more stringent nuclear safety regulations after Fukushima, non-safety compliant design solutions or by uncertainties on plasma physics and operation aspects. In addition, low technical readiness of some of the crucial areas such as in-Vessel Components and Remote Handling has led to complex design solutions that require extensive additional R&D and qualification.

This paper highlights the progress in the DEMO pre-conceptual design activities in Europe carried out by the EUROfusion Consortium. Sect. 2 describes the DEMO staged design approach with Design Phases and Gates and provides some programmatic considerations, including timeline and dependencies with the ITER schedule. Sect. 3 highlights the design choices under consideration in this early design phase and emphasises the criteria and the risks involved in the selection of design parameters and underlying technologies. Sect. 4 describes the progress on the design of the plant systems, including the Tokamak Building and the Balance of Plant (BoP). Sect. 5 describes a design maturation strategy for some key design and technologies for DEMO (i.e., breeding blanket, and ITER test blanket module (TBM), superconducting magnets, remote maintenance, etc.). Sect. 6 describes the role of industry, the technical exchange with the ITER Organization and the role of International collaborations. Finally, some concluding remarks are provided in Sect. 7.

3. Overall timescales and strategy for the DEMO project

2.1 The key role of ITER and dependencies with the DEMO schedule

The European Fusion Roadmap emphasises how crucial ITER is for the validation of the DEMO physics and part of the technology basis. This demonstrates the high degree of schedule dependency between ITER and DEMO, and the 'success-orientated' approach outlined here advocates concurrency between the exploitation of ITER and development of the DEMO design. In this approach, the DEMO design activity proceeds in parallel with the ITER exploitation, but relies on a progressive flow input from ITER for design and physics validation prior to authorisation of DEMO construction.

Fig. 1 shows the main dependencies between the DEMO and ITER schedules. From this figure it can be understood that DEMO design validation from ITER should not be seen as a single discrete event, but rather ongoing and progressive flow of information into the programme. This allows continuous validation of specific aspects of the DEMO design and technologies solutions that are being considered for certain systems that are based on evolutions/ improvements of those used in ITER (e.g., vacuum vessel, superconducting magnets, H&CD systems, etc.) and if necessary, updates to the DEMO design baseline. The most critical and final major validation input for DEMO, is the demonstration of D-T burning plasma scenarios in ITER that are scheduled to start circa2037 (with Q=10 short pulse in 2037 and long pulse in 2039) and the results of the TBM programme.

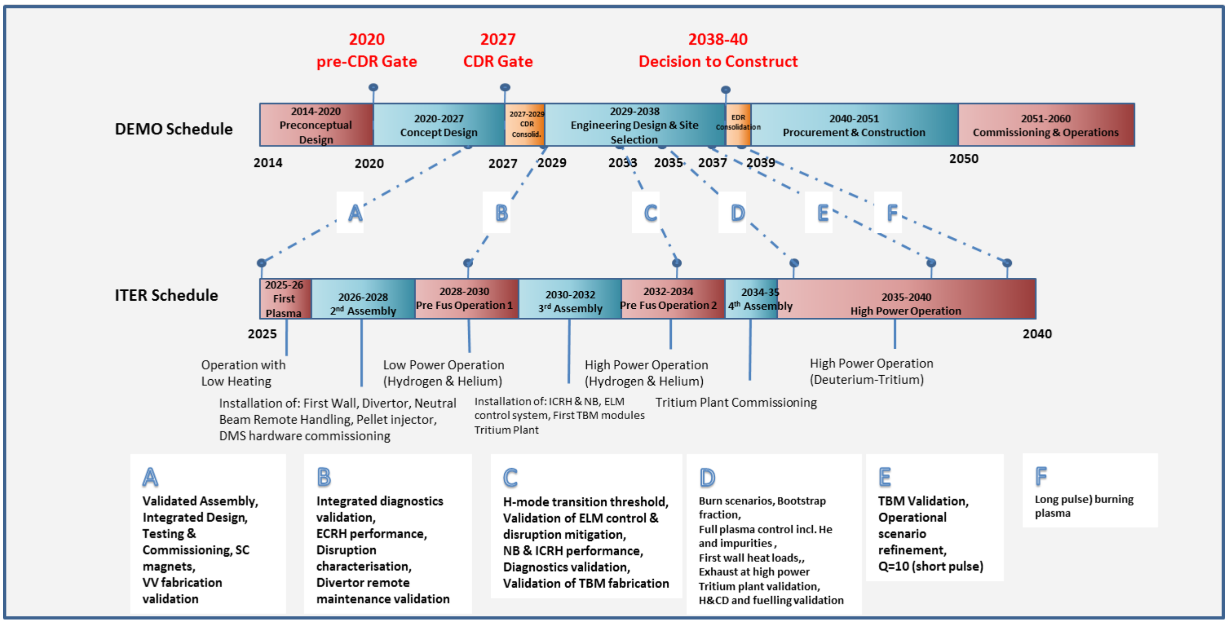


Figure 1 - Overview of phasing and key technical inputs from ITER DEMO Schedule

2.2 DEMO Design Phases and Gates

The DEMO staged design approach consists of three main technical phases:

(i) *a Pre-Concept Design Phase (PCD) to explore a number of DEMO design options (i.e., optioneering[b]) and system requirements up to 2020;*

(ii) *a Concept Design Phase (CD) to mature and validate the baseline concept up to 2027[c], by down selecting key design technology solutions for the DEMO plant, on the basis of the results of a sound R&D program; and*

(iii) *an Engineering Design Phase (ED) to follow and develop the detailed design and to conduct extensive testing of the concepts and technologies required and prepare for the launch of major procurement activities around 2040's, after ITER nuclear operation has confirmed the robustness of the underlying assumptions.*

A schematic of the phases and gates preceding Procurement and Construction Phase is outlined in Figure 2 [5]. In the present early phase of the design, strong emphasis is on the PCD Gate (G1), where main design integration risks, and corresponding design and technology options are evaluated by using a structured and traceable assessment methodology. In parallel, the technical maturation plan adopted for each of the major tokamak systems will be evaluated, with the aim of ensuring realistic down selection of the most promising technologies during the concept design phase. Given the level of readiness and some major uncertainties concerning the DEMO physics basis, the pre-concept DEMO baseline (output G1) will be a "set of design solutions and technologies" baseline (i.e. a set of candidate design and technology solutions to be further investigated). It is still uncertain whether the main machine parameters and plasma configuration can be frozen in 2020.

An intermediate gate (G2) has been introduced in the middle of the CD (~2024) to select the design solution(s) for critical systems (i.e, breeding blanket, divertor configuration, remote maintenance scheme, heating and current drive (H&CD) mix, etc.) together with the main machine parameters and reference plasma scenario to arrive to a consistent and verified DEMO conceptual design by 2027.

[b] Optioneering is a structured evaluation of options in support of decision-making. Such an evaluation may take the form of an Option Study that collates information on the options and the different attributes that will influence the decision to be made and may also consider how the decision is influenced by different value judgements.

[c] A transition phase of about two years is expected for the concept design review consolidation and preparation of the Engineering Design Phase.

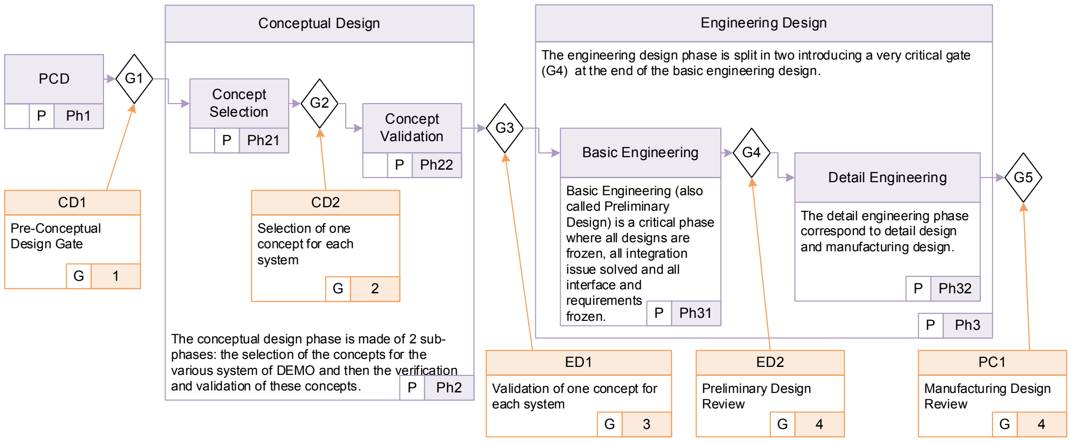


Figure 2: Phases and Gates preceding Procurement and Construction assumed for DEMO in Europe [5]

4. Design Choices under Considerations

3.1 DEMO design points studies

The process to define an appropriate set of plant design parameters and technical features starts with the definition of the plant requirements (e.g., net electricity output, tritium self-sufficiency, plant availability, operation mode, etc.) and always involves trade-offs between the attractiveness and technical risk associated with the various design options considered. It should be noted that some of the physics assumptions (e.g., energy confinement, plasma pressure, H-mode access threshold, bootstrap current fraction, etc.), and technology assumptions (e.g., allowable divertor heat loads, n-load limits on the structural materials, maximum field in the superconducting magnets, plant thermodynamic efficiency, wall-plug efficiency of H&CD systems, etc.) play a major role in the tokamak dimensioning process. As such the readiness and experimental/operational basis of some of the invoked technologies remain highly uncertain.

System codes (see for example [6,7,8]) representing the full DEMO power plant, are currently being used in Europe to underpin DEMO design studies to find meaningful design points [9]. For DEMO, these codes have been used to find solutions with a minimum tokamak size. In arriving at these solutions, the three overarching limitations preventing further reductions are: (1) the divertor protection, (2) the access to the H-mode, and (3) the maximum field in the conductor of the toroidal field (TF) coils and the stress in the coil casing. The divertor power handling has been found to be an important size-driver in DEMO from the very beginning [4] and is going to be discussed further in Sect. 3.2.

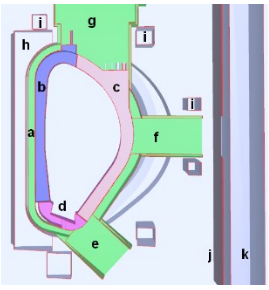
At present, work in Europe continues to be focused on the design of a pulsed DEMO plant concept (the so-called “DEMO 1”) based on modest extrapolations from the ITER physics and technology basis to bound development risks. This is not intended to represent an exclusive design choice but rather a “proxy” to be used to identify and resolve crucial design integration problems (see Sect 4.2). Considerations are also given to a design based on the latter-stage ITER Scenario (i.e., Q = 5, Ip=9 MA) capable of operating in a short pulse mode (e.g., 1 hr) for nominal extrapolated performance (H98=1.0) and capable of moving to steady-state operation while maintaining the same fusion power and net electrical production in the case of a better confinement being feasible (see Table 1). However, this option requires a much higher confidence in physics extrapolation and highly reliable and efficient current-drive and control systems, which need to be deployed by day-1 and still need to be developed.

A schematic cross sections of the current DEMO 1design and a list of parameters for the design option being considered are shown in Table 1, together with the main design parameters. Table 2 shows the preliminary design features adopted in the design.

Table 2. DEMO design options under study

Tokamak radial-build: a) vacuum-vessel;	**DEMO1**	**Parameters**	**Flexi-DEMO**

breeding blanket (inboard); c) breeding blanket (outboard); d) divertor; e) lower port; f) (equatorial port; g) upper port; h) toroidal field coils; i) poloidal field coils; j) cryostat; k) bioshield



		lop$_{(ind)}$ [a]	hop$_{(ss)}$ [b]
9, 2.9	R_0, a (m, m)	8.4, 2.71	8.4, 2.71
3.1	A	3.1	3.1
5.9	B_T(T)	5.8	5.8
18, 3.6	I_p (MA), q	16.63, 4	14.17, 4.7
1.6, 0.33	k_{95} / δ_{95}	1.69, 0.33	1.69, 0.33
12.6	$<T_e>$ (keV)	12.1	15.1
0.73	$<n_{e,vol}>$ (10^{20}m-3)	0.88	0.75
2.2	Z_{eff}	2.23	2.86
1.1	H	1.13	1.48
2	t_{burn} (hrs)	1	St. State
39	f_{bs} (%)	47	66
<10	P^*_{CD} (MW)	>100	>100
161	P_{div} (MW)	165	194
120	P_{LH} (MW)	123	109
2014/500	P_{fus} / $P_{e,net}$ (MW)	2000/395	2000/399
1.0	AvNWL(MW/m2)	1.15	1.15

Table 2: Preliminary DEMO design features

- Single-null water cooled divertor; PFC armour: W
- LTSC magnets Nb_3Sn (grading)
- Bmax conductor ~12 T
- EUROFER as blanket structure and AISI 316 for VV
- Maintenance: blanket vertical RH / divertor cassettes
- Lifetime: starter blanket: 20 dpa (200 appm He); 2[nd] blanket 50 dpa; first divertor: 5 dpa (Cu)

The main assumptions and guidelines that have been used to determine the radial build and thus the machine size, are described in Table 3.

Table 3 Rational physics/ design assumptions in the system codes

TF Coils: thickness determined by the strength of the peak field and the need to (i) resist large mechanical forces and keep stress below allowable (ITER-consistent) limits and (ii) ensure TF coil protection and temperature margins in the conductor. Ensure TF coil protection during current quench by limiting the maximum permissible temperature rise during a quench to limit the current density. The n-power deposited in the TF coils is estimated but is only used in the calculation for cooling power (see under blanket; neutron budget is set by insulator properties rather than superconductor).	
• *Peak magnetic field in the superconductor:*	12.1 T
• *Stress limit in the TF coil structures/ stress criterion:*	660 MPa/ Tresca
• *Min. temperature margin in the conductor:*	1.5 K
• *Max. permissible temperature rise during a quench to limit the current density*	150 K
• *Current per turn for TF coil*	60-90 kA
• *Copper fraction of TF conductor*	50-94 %
• *Number of TF coils*	16
• *Critical parameterization*	Nb3Sn WST
• *Maximum allowable TF ripple at plasma edge*	0.6 %
Central Solenoid: Thickness determined by the strength of the peak field in the CS and flux swing requirement, i.e., pulse length.	
• *Peak magnetic field in the superconductor:*	13 T
• *Stress calculation*	Only hoop stress considered
• *Copper fraction in CS conductor strand*	70 %
• *Critical parameterization*	Nb3Sn WST
• *Stress limit in the TF coil structures:*	660 MPa
• *Flux swing required for start-up*	~380 Wb
• *Flux swing required for burn (determined by pulse length requirement)*	~340 Wb
• *Estimated contribution of flux from PF system*	~320 Wb (44 %)
• *Flux target for CS*	~400 Wb (56 %)
Divertor protection	
• *Peak heat flux (for attached plasma conditions): Uses condition $P_{sep}B/qAR$*	9.2 MW.T/m (see Sect. 3.2)
Breeding Blanket: Thickness determined primarily by the requirements to: (i) produce sufficient tritium in the breeding blanket; (ii) maintain the tolerable radiation damage in the TF coils below an agreed limit over the lifetime of the device; (iii) keep the nuclear heating below a limit (in the TF coils).	

• *Tritium breeding ratio (TBR)*	≥1.05 across whole machine, locally higher. *This requires thin PFCs, ~85% of plasma coverage by breeding elements; constraints on divertor space*
• *Shielding (together with Vacuum Vessel)*	
• *Peak volumetric nuclear heating in TF winding pack*	50 W/m^3
• *Peak fast neutron fluence to the Nb_3Sn superconductor*	1×10^{22} n/m^2
• *Neutron fluence to Cu stabilizer between TFC warm ups*	1–2·10^{21} n/m^2
• *Total neutron fluence to epoxy insulator 10^{22}/m^2*	10^{22} n/m^2 , equivalent to 1x10^7 Gray
• *Max displacement damage in VV*	< 3 dpa
• *Cutting/re-welding location in IVC cooling pipes helium production*	1 appm
• *Allowable neutron wall load*	8 MW/m^2
• *Inboard blanket thickness (fixed)*	0.755 m
• *Outboard blanket thickness (fixed)*	0.982 m
• *Inboard shield thickness (fixed, including VV)*	0.600 m
• *Outboard shield thickness (fixed, including VV)*	1.100 m
Other Build Items	
• *Inboard gap between CS and TF (variable with lower bound 5 cm)*	0.050 m
• *Inboard gap between VV and TF coil (variable with lower bound 2 cm)*	0.020 m
• *Plasma-wall nominal spacing (fixed)*	0.225 m

At present there are many discussions about making fusion power producing devices smaller, cheaper, and faster, but there is no magic bullet to solve the integrated design problems. The approach, advocated by the EU fusion roadmap, is to investigate designs based as much as possible on modest extrapolations from the ITER physics basis, on robust design features, incorporating either proven technologies or innovations that can be validated through realistic R&D programs to bound development risks, and on safety features and design licensing by integrating lessons learned from ITER licensing (and other existing nuclear facilities). The present designs of EU DEMO (either DEMO 1 or flexi-DEMO) are the logical consequence of the most mature knowledge in physics– i.e. the H-mode scaling and exhaust and technology, not an a priori desire to be big [3]. These designs also provide a sound and detailed basis for investigating the engineering integration issues, which are considerable.

The size of DEMO is currently limited by the ability to handle the divertor exhaust power for a given machine size, represented in systems code terms as $P_{sep}\,B/qA\,R_0$, where Psep is the power crossing the separatrix, B is the toroidal field in the plasma, q is the safety factor at the plasma edge, A is the aspect ratio, and R0 is the major radius. A limit of 9.2 MW-T/m is currently assumed for DEMO, based on similarity with the ITER divertor, and can be considered as a divertor protection limit. A machine achieving the same fusion power with a higher toroidal field, and thereby smaller major radius, would effectively require a divertor solution capable of exceeding the present performance limit, or high radiative impurity levels in the plasma to reduce Psep, probably impacting on plasma control and access to H-mode. At present there is no clear evidence that the SOL/ divertor power handling capability in a standard divertor configuration can be significantly higher than assumed for ITER. In fact, there are big uncertainties on the plasma side due to the lack of real predictive capability. Also we need to assume that the target may have lower power handling limits than ITER, considering that the materials impact of neutron damage increases the challenge. Investigating the effects on plant design of higher limits is straightforward but it is not reasonable to base a design on speculative extrapolations. Alternative divertor configurations are proposed but the plasma performance is unproven and there are considerable problems with integrating them into a practical power plant design, not least managing the remote handling access (see Sect. 4.2).

A second limit on the size of DEMO is the magnet performance. In the models used, the field available is principally limited by the stresses reached in the coils, rather than the superconductor performance. The forces vary with B^2, and since the coil cannot expand toroidally it must become radially larger rapidly limiting how small the machine can become. With an aspect ratio of 3.1, space for a breeding blanket, and stress limits of <700MPa in the structural coil materials, targeting a field of 5T in the plasma leads to a device with R_0>7m without considering other limitations. A growth in the coil allowing higher fields representative of high-temperature superconductors (HTS) without a corresponding increase in the stress limit results in a larger machine (albeit, one with improved plasma confinement). To an extent this can be overcome by, for example, excluding tritium breeding from the inboard side to reduce the plasma-magnet distance, but this seriously compromises the ability to breed fuel. Also limiting the benefits of increasing the field, in order to access H-mode it is assumed that the amount of power crossing a flux surface just inside the separatrix must exceed the L-H transition threshold power P_{LH}. At present, it is assumed that P_{sep}> P_{LH} for DEMO, as it is likely that P_{sep} will need to be higher than P_{LH} in order to achieve sufficient controllability and confinement quality. Using the

Martin 2008 scaling for P_{LH} and Greenwald scaling for density limit, it can be found that, for a fixed P_{sep} / R_0 (i.e. divertor protection figure of merit) increasing the ratio of P_{sep} / P_{LH} can only be done by reducing the magnetic field [[10]]). The option of operation in I-mode is under investigation to explore the consequences of its use, but the current related physics basis (e.g. extrapolation of the LI-threshold) is so weak that it does not fulfil conservative criteria for DEMO1 by a considerable margin.

Allowing a variation in aspect ratio may appear to overcome some of these limits. As the aspect ratio falls elongation can be increased and higher β_N is achievable; however the increased minor radius means that the field in the plasma is lower and the actual plasma pressure does not change much. Overall, for the same achievable field at the TF coil, there is no significant change in power density, although lower aspect ratio designs can deliver higher absolute power due to increased plasma volume (but must still respect power exhaust constraints). This increased power comes at the cost of much bigger in size but thinner TF coils to accommodate the increased plasma volume.

Taking all these elements into account using more detailed models, and allowing for some conservatism, leads to a device of R_0~9.0m.To significantly reduce the size would require confidence in advances in plasma physics (particularly control and diagnostics in a fusion environment, plasma scenarios that reduce the power density to the divertor target, and highly reliable techniques to mitigate the effects of ELMs or plasma scenarios without ELMs); materials and design solutions to handle higher power densities in multiple parts of the machine during steady-state operation and transients; remote handling approaches that maintain high availability with restricted access; and improved magnets capable of generating higher fields and handling the resulting structural stresses. All of this must be achieved using systems capable of reliable and safe performance in a fusion environment, which can be remotely maintained. In general assuming improved performance in only one system results in a transfer of loads to other systems and only a minor reduction in overall size. In order to have confidence in achieving the high-level goals for in the given timescales, such alternative speculative solutions are excluded. This does not mean that EU-DEMO is low-risk, but the approach is chosen to minimize the risk in extrapolation.

If the tolerance for risk is increased, there are potential approaches allowing design changes, which may ultimately reduce the size of DEMO. The first is a reduction in conservatism – a more complete scientific and technical basis allows a reduction in safety margins on extrapolation and increased confidence in plasma control at high radiative fraction, PFC surface erosion rates, or higher β_N. This may be offset by the need to operate in e.g. ELM-free regimes. It is anticipated that the ITER and DEMO research programmes will naturally improve matters here over time before the DEMO design point is finalised.

If the high-level goals of DEMO are relaxed (e.g., through a reduction in target electricity production or tritium self-sufficiency, or less targeted technology transfer to a fusion power plant) then size savings can be achieved. Pulse length could also be shortened (to save solenoid space) or lower aspect ratio explored (lower A can generally achieve higher bootstrap current fraction, supporting longer pulse length without additional auxiliary current drive). In the first case, the DEMO mission is compromised and in the second, the design is based on a reduced scientific basis.

3.2 Divertor Protection and Plasma Power Exhaust Scenarios

One of the crucial points in the dimensioning of a power producing fusion plant, remains the size of the device and the amount of power that can be reliably produced and controlled within it. This heavily depends, amongst other things, on the heat load that can be tolerated by the divertor under normal and –off-normal operation. The reference plasma scenario adopted so far for the EU-DEMO is the ELMy-H mode [[11]], which is known to exhibit a lower threshold on the charged particle power P_{sep} crossing the last closed magnetic surface, below which the confinement capability of the machine is significantly reduced as the L-mode is recovered.

Our design is based on the assumption to operate with at least a partially-detached divertor, implying thus that a significant fraction of P_{sep} shall be dissipated in the scrape-off layer before actually reaching the target plate. Otherwise the power striking on the plates would be too high to deal with via the currently available technology. The necessary high dissipation is planned to be obtained with the use of seeded, radiative impurities, such as Ar or Kr [[12]], which re-distribute the necessary fraction of the exhaust power onto the first wall in form of photons. The deployment of these impurities is however not without consequences for the machine operation. A certain fraction of the seeded atoms, in fact, is expected to migrate into the plasma core, where, depending on the edge profile characteristics can cause either a reduction of the fusion power via fuel dilution or trigger some radiative instability [10]. It is therefore necessary to find an adequate balance between the radiation level in the SOL and the impurity content in the core, but it is not a-priori obvious whether this is feasible for every machine configuration.

Ref. [13] discusses the criteria to be employed in the preliminary phases of a tokamak fusion reactor dimensioning to ensure the integrity of the divertor for sufficienty long operating times, without at the same time compromising the stability of core plasma or the fusion power generation. There, it is shown that two high-level requirements are necessary to be fulfilled, namely 1) the concentration of seeded impurities in the SOL has to be lower than some critical value in order not to compromise the fusion plasma performance or stability, and 2) the design of the divertor target must be able to withstand accidental re-attachment of the plasma for a sufficiently long time to recover detachment or to ensure a safe, controlled termination of the plasma discharge. The main conclusion of [9] is that, for a given fusion power level, the contemporary fulfilment of both requirements limits the viable reactor size both in terms of major radius R and in terms of toroidal magnetic field B.

Deliberate periodic movement of the divertor strike points (sweeping) of the x-point (wobbling) by external coils is being considered as a measure to distribute the heat loads over a larger surface area in the case of re-attachment. Clearly, this strategy and the definition of the sweeping parameters (i.e., sweeping amplitude and frequency) depends on how large the flux on the target plate is in the case of divertor re-attachment and this depends on the value of the scrape-off width which is still uncertain [14]. Studies have been carried out [15,16] to determine relevant sweeping parameters and to determine the impact of thermal fatigue on the high-heat-flux components, AC losses, etc.

In addition, concerns exist on the consequences of unmitigated type-I ELMs and disruptions. ELMs are a well-known plasma instability, which characterises the pedestal of H-mode discharges in tokamaks, leading to a periodic release from the confined plasma region of particles and energy mainly directed onto the divertor. In present experimental devices, the ELMs do not represent a particular threat for the regular operation. However, extrapolations to larger machines, as for example ITER and DEMO, suggest that even an actively cooled divertor could withstand only a very low number of ELM events (below few tens, or even less) before being severely damaged, this occurrence is clearly incompatible with a long term operation of the machine, also in view of the high natural frequency at which ELMs occur (~1 event per second) [17].

Currently, many active methods for the mitigation, or even the suppression of the ELMs (e.g. resonant magnetic perturbation (RMP) coils, ELM triggering via pellets, vertical "kicks"), are under investigation in many laboratories, both in Europe and overseas. However, the possibility of recurring to such methods in a future, high power nuclear fusion reactor for the production of electricity like DEMO is debatable for a number of reasons. Primarily, it is unclear whether these methods are effective enough in reducing the ELM size to an acceptable level at reactor relevant parameters. Secondly, because their impact on the plasma pedestal, and thus on the confinement, could excessively compromise the plasma performance in terms of fusion power outcome. Thirdly, because the reliability required to these systems (at most few tens of events allowed during the foreseen divertor lifetime) might be impossible to meet from an engineering point of view, especially in a harsh environment like the DEMO burning plasma chamber.

For these reasons, plasma configurations which are naturally ELM-free are a particularly attractive solution for a nuclear fusion power plant, where the integrity of the machine must be ensured over long time. Among these, two candidates are of particular interest: (i) *the Quiescent H-mode (short: QH-mode)*; and (ii) t*he Improved L-mode (short: I-mode)*. Both regimes exhibit the noteworthy advantage of being naturally ELM free. However, both of them are quite poorly explored and understood in comparison to the standard ELMy H-mode configuration, which represents the ITER reference scenario. As such, there are still many open points which need to be carefully evaluated, both in terms of experiments and in terms of modelling, before anything can be concluded about their suitability for electricity producing tokamak reactors. R&D in present devices must be focussed towards building knowledge on such regimes (e.g. I-Mode or QH-Mode), namely to find operating boundaries, confinement and transition power scaling.

3.3 Systems code sensitivity analyses and trade-off studies

A power-producing tokamak reactor is a highly complex device embodying the results of innumerable assumptions and decisions. In view of the several highly complex system interdependencies in a power-producing tokamak there is a need to conduct trade-off studies to understand the trends arising from the variation of some design assumptions and improve early design concept optimisation. Similarly, because of the many uncertainties still affecting some of the physics and technology assumptions, sensitivity analyses are necessary to identify the key limiting parameters and explore the robustness of the reference design points to key assumptions.

To date DEMO trade-off studies have been conducted for the aspect ratio, the reduction of the thickness of the outboard breeding blanket, the number of TF coils, the impact of a double null divertor on the TBR, etc. (see for example [[18],[19]]). Ref. [[20]] also discusses the results of a sensitivity analysis carried out to determine the impact on the performance (electrical output) and pulse duration as a result of varying a number of key physics parameters by ± 10% around the nominal value. The strongest sensitivities are found for the plasma elongation, confinement and density. Certainly the uncertainty on each parameter is not the same and in [[21]] a proposal for the probability distribution of system code input parameters is presented.

Because of space limitation, considerations here are limited to only a few representative aspects.

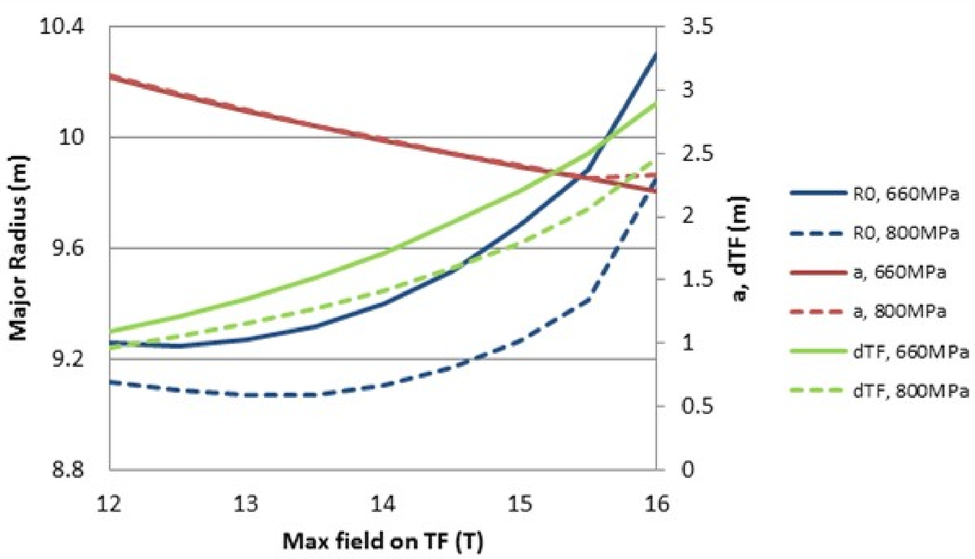
3.3.1 Peak field in the TF coils and impact on machine size:

In a superconducting tokamak magnet, the superconductor itself takes up relatively little volume in the winding pack and it has been found that the effect of the peak field in the TF coil winding pack on the superconductor critical current density is a secondary size driver: increasing this limit has relatively little effect on overall machine size [3]. As discussed above, higher field windings generate higher forces in the mechanical structures in and around the plasma and TF coils, including the TF coils themselves. The stresses in the inboard leg of the TF coil casing (the "nose") quickly reach the maximum allowable stress for a given geometry, providing an effective limit to the practically-achievable field. The solutions to this require either higher-strength structural materials than currently available, or an increase in machine size to provide space in the radial build for enlarged magnets. The stress varies as $\sim B^2/A$, where B is the toroidal field and A is the aspect ratio, and for a 12.5T peak field in DEMO we obtain an inboard leg width of around 1.3 m. Assuming an HTS conductor which could perform at 20T instead, the resulting forces would require a doubling in the radial thickness of the inner TF limb (as the toroidal width cannot be increased). This modelling also requires a small increase required in the winding pack size due to quench protection assumptions dealing with a higher stored magnetic energy, although the quench behaviour of HTS is different from LTS and this is currently not well captured in systems codes. This increase in TF thickness means that the potential higher field arising from the use of HTS alone does not result in a significantly smaller machine (and given other constraints on radial build, generally only if either aspect ratio increases or tritium breeding capability is removed from the inner wall of the tokamak). However, increasing the allowable stress in the structure of the coil (e.g. through the development of a higher performance cryogenic steel) can have a substantial effect on the overall machine size, particularly when compounded by the use of HTS.

Figure 3 shows the major radius R_0, minor radius a, and TF coil thickness dTF required in the systems code PROCESS by different maximum fields at the TF coil, for two different stress limits in the TF structural material (660 MPa and 800 MPa). Aspect ratio was left as a free parameter and the major radius was minimized by keeping the net electrical power, pulse length, etc. constant. Initially the machine size falls somewhat as the field rises – through an ability to achieve similar plasma output at higher aspect ratio, reducing the minor radius. (This increasing aspect ratio is monotonic throughout the modelled space.) Soon, however, the rapidly-increasing TF coil thickness begins to dominate the change in radial build and the device size increases overall, even as the plasma continues to shrink. Below that are cross-sections of TF coils showing casing and winding pack (WP) for coils in PROCESS at 12.5 T, 660 MPa and 16 T, 800 MPa assuming the use of HTS. The accompanying lines are 1 m long for scale. In this model, much of the WP in the latter case is occupied by structural material (conduit casing) as well.

This is not to say that the development of HTS coils is not a useless endeavor; there are many aspects of HTS conductors which are very promising. To place the focus, however, solely or principally on achieving higher field coils is misguided and neglects the necessity of overall design integration. Higher current density winding packs (reducing the width of the WP slightly), higher temperature operation (reducing cryoplant loads), higher margins to quench and therefore improved reliability, lower cost, and increased flux swing availability from the central solenoid are all reasonable and worthwhile R&D goals.

Alternative high risk R&D areas such as demountable coils (potentially simplifying maintenance and/or improving investment protection prospects), or segmented manufacture (solving mass production and transport issues) are also potentially attractive from a reactor perspective but rely on techniques that are not yet well developed even on the lab scale, and would require extensive R&D to raise them to a suitable technology readiness level. Basing DEMO design on such potential developments would greatly increase the project risks.



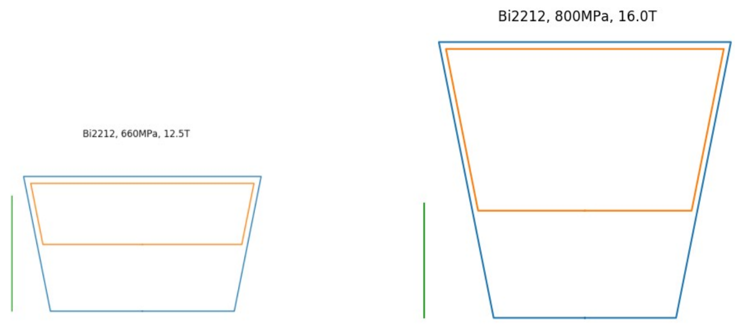


Figure. 3: Impact of the peak field in the TF conductor on radial build and machine sizing, from the systems code PROCESS (see text).

3.3.2 Wall plug efficiency of H&CD systems:

Albeit preliminarily, it is found that for the baseline pulsed DEMO 1, the auxiliary heating power requirements during the flat-top are rather modest (e.g., ~30 MW continuously for MHD control, plus some tens of MW for burn control when needed). On the contrary, for scenarios at high non-inductive current fraction like flexi-DEMO, auxiliary power for current drive is required at much higher levels than DEMO 1, also because the same control functions as for DEMO 1 have to be provided as well . Thus, for flexi-DEMO, achieving a high value of wall plug efficiency is crucial to ensure an acceptable electricity output. This circumstance can clearly be observed in Fig 4. For baseline DEMO, a reduction of 25% of the wall-plug efficiency (from 0.4 to 0.3) leads to a loss of about ~40 MW in the electrical power output, whereas in Flexi-DEMO (which already has a lower electricity output for the same fusion power than baseline DEMO because of the larger H&CD needs) the same reduction in η_{WP} leads to a decrease of 90 MW in the net electric power.

The curves in the figure have been built assuming $P_{el,0}$ = 500 MW for $P_{aux,0}$ = 50 MW at a wall-plug efficiency η_{WP0} of 0.4 (these values originate from the reference PROCESS run for DEMO 1). The other curves are built as a function of P_{aux} and η_{WP} with the formula

$$P_{el} = P_{el,0}+P_{aux,0}/\eta_{WP0} - P_{aux}/\eta_{WP}$$

thus assuming a constant fusion power and a constant power absorption from non-H&CD plant components.

For this reason, it is also important to investigate which H&CD configuration ensures the better performance in terms of coupling with the plasma. To drive the ~6 MA which are necessary in the actual flexi-DEMO scenario, in fact, an efficiency larger than 60 kA/MW is required to maintain the injected CD power below 100 MW

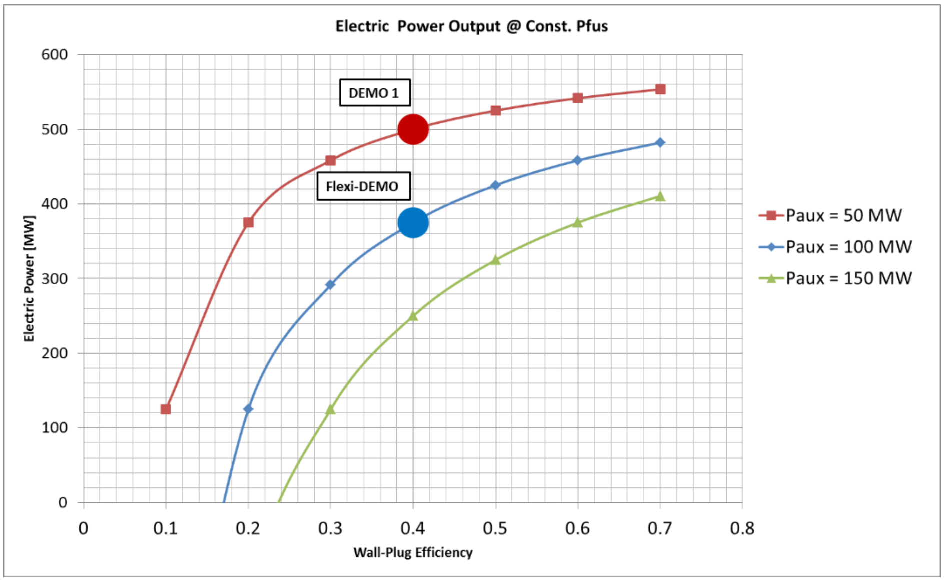


Figure 4: Impact of the wall plug-efficiency (the ratio of power coupled to the plasma to the electrical power required to run the system) of the current-drive systems on the DEMO electricity output. The curves are built at constant fusion power and constant non-H&CD power absorption, assuming a 500 MW power production for 50 MW of auxiliary power at η_{WP}=0.4

3.3.3 Plasma elongation:

In using a systems code to explore the parameters which play the strongest roles in determining plasma performance (and hence impact most on device size), the plasma shaping is found to play a strong role due to allowing higher plasma current at fixed edge safety factor q_{95} (Fig. 3). This is due to having a number of fixed assumptions (e.g. operating Greenwald density fraction, q_{95}), which combine through the IPB98(y,2) energy confinement scaling to give $W{\sim}\kappa^{2.5}$, and thus $P_{fus}{\sim}\kappa^{5}$. Increasing the plasma triangularity, δ, also aids this effect. However, these are not free parameters. As well as complex physics effects not easily captured in a systems code such as impacts on the pedestal, ELMs, and MHD activity, δ is limited by the shaping ability of the PF coilset which must respect current density limits and tokamak access for remote handling, and elongation is limited by the plasma vertical stability. Vertical stability could be improved by reducing the aspect ratio A but this increases the plasma minor radius and decreases the achievable field in the plasma, generally driving the device size up. (Elongation dependency on A is captured in the dataset in Fig. 3 above.) Otherwise, the inclusion of toroidally-conducting inserts into the blanket in key locations could help, at the possible cost of TBR and a significant increase in engineering complexity. As ever, the final choice of plasma parameter values must respect what is reasonably achievable within whole-machine engineering constraints.

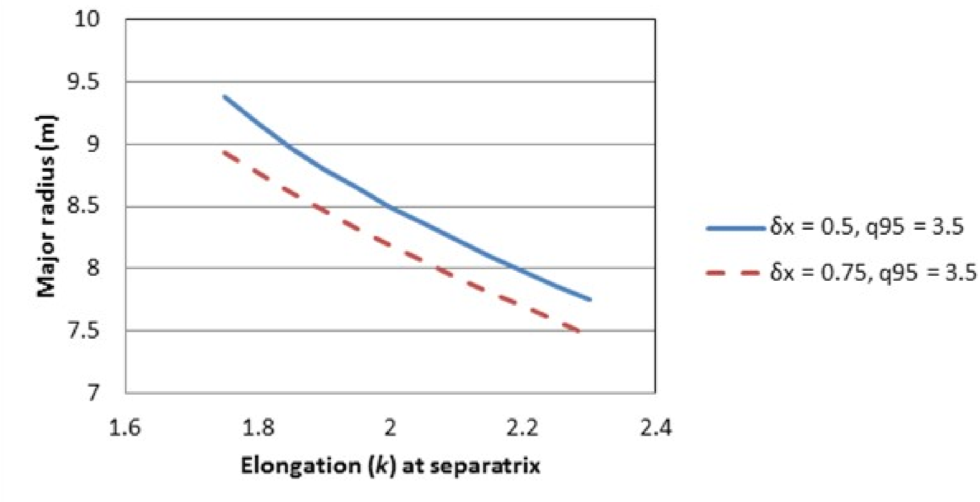


Fig. 5: Impact of the elongation on DEMO machine sizing

5. Plant Design and Integration Studies

4.1 Systems engineering approach to support systems integration

The focus during the pre-concept design phase is on a design integration approach, based on systems engineering, which is recognized to be essential from an early stage to identify and address the engineering and operational challenges, and prioritise the required technology and physics R&D. This approach is not limited to only considering requirements definition and propagation traceability but also encompasses considering the spatial and physical integration between systems and components. In this regard, the development of a baseline configuration of the physical plant layout, is seen as a priority to better understand the spatial/physical integration aspects from an early stage, to identify integration issues and improve coherency between system requirements. This is not intended to represent fixed and exclusive design choice but rather a "proxy" of possible design options to be used to identify technical issues, both in engineering and physics that need to be resolved in DEMO. Experience with ITER indicates that it is important to initiate this activity early, so that major integration issues can be identified and resolved before critical aspects of the design are frozen, or major procurement activities are launched. This philosophy of developing systems designs in a holistic, integrated fashion is a fundamental principle of the systems engineering approach. The baseline systems architecture and plant layout is continually evolving, being updated as new information comes to light, but it represents the current 'best' option and acts as a central reference point to all contributors.

4.2 Key Design Integration Issues

Certain design, physics or technology choices are so integral to the plant architecture, that they have significant implications on a large number of systems that must be integrated into the plant. If such choices are made in isolation, they could have adverse effects on the design on the plant as a whole, adding risks and complexity to the design and increasing the difficulties for the integration of one or more systems and ultimately costs. Therefore, a thorough examination of system integration aspects is essential to ensure that the integrated view of the plant is maintained from the very beginning and all factors affected by the numerous design choices to be made are identified, evaluated, and properly weighted. Implementation of this approach provides an opportunity for overall design convergence, reduction of integration risk and minimization of life-cycle costs at an early stage of the design. The risk of postponing integration, assuming that it restricts innovation and inhibits an attractive DEMO plant, is that designers remain oblivious of integration issues and developing design solutions that cannot be integrated in practice.

A limited number of Key Design Integration Issues (KDIIs) that have a strong impact on tokamak and plant design architecture, safety, maintainability and licensing have been selected for study during the pre-concept design phase (see Table 4 and Ref. [[22]]). A number of design options for each of these KDIIs are being studied and the Gate Review G1 planned in 2020 (see Sect. 2.2) will evaluate and down select, if possible, the most attractive design options.

Table 4 Example of DEMO key design integration issues (KDIIs) being studied in the pre-concept design phase.

Key Design Integration Issues	**Design Options**
1) *Wall protection to withstand plasma transients*	a) plasma conforming wall w/o limiters; b) guard limiters; c) double-null divertor
2) *Integrated design of breeding blanket and ancillary systems related to the use of helium or water as a coolants for the blanket and impact on the overall plant design*	a) water cooled breeding blanket and auxiliaries; b) He-cooled breeding blanket and auxiliaries
3) *Design integration risks arising from advanced magnetic divertor configurations*	a) single-null; b) double-null; c) Super-X; d) snowflake divertors
4) *Breeding blanket vertical segment-based architecture*	a) full segmented blanket; b) poloidally-segmented blanket
5) *Power Conversion System Options*	a) indirect; b) direct (w/o or a small Energy Storage System)
6) *Integrated design of tokamak building concepts incl. ex-vessel maintenance*	Different building options including licensing and remote maintainability access constrains.
7) *Pumping concepts based on tritium direct recirculation*	a) direct recycle; b) indirect recycle.
8) *Development of a reliable plasma-operating scenario including supporting systems (e.g., Heating and Current Drive (HCD) and plasma diagnostics/control systems.*	alternative (to ITER) plasma scenario

They have been selected because: (1) the equivalent technical solutions adopted for ITER are not DEMO relevant (e.g. different materials, design requirements, coolant types, operating conditions, plasma conditions, etc.), or (2) no relevant design/operation information is expected from ITER due to the different missions between the two devices. Example of items of the former category are the protection of the first wall from plasma transients by using easily replaceable guard limiters, advanced divertor configurations with potentially higher wetted areas, different blanket maintenance schemes, pumps based on the direct internal recycling, a robust plasma scenario with much higher radiative power fractions than ITER and robust solutions to minimise/supress ELMs and disruptions, etc. Examples of items of the latter category are, for example, the breeding blanket to breed and extract high grade heat, the design of the balance of plant including power conversion system to convert heat into electricity.

What follows is brief description of the KDIIs under study.

KDII/1 - The power handling capability of first wall in DEMO (that represent the non-breeding part of the breeding blanket) is rather limited (~1-1.5 MW/m^2 for a water cooled concept and ~1 MW/m^2 for helium) due to the requirements of using radiation hardened materials like EUROFER steel for the structures and the cooling pipes of the breeding blanket, and high temperature high pressure coolant for efficient energy conversion. While these limits are deemed to be achievable in nominal conditions in the present DEMO blanket designs, the issue remain the occurrence of plasma transients. Because of the requirement to achieve effective breeding and tritium self-sufficiency, the first-wall in DEMO must be relatively thin (few mms) in order not to adversely deplete the neutron flux entering the breeding regions. If adequate provisions are not included in the design, the occurrence of few of these events may severely damage the first wall or even lead to a breach of the cooling pipes and the consequent loss of coolant. Preliminary results of the work commenced to address this problem are describe elsewhere [23].

KDII/2 - It is generally agreed that water should be considered as the divertor coolant for DEMO design as the divertor surface heat flux conditions prove to be beyond present helium power handling capabilities [24]. However, the choice of the coolant for the breeding blanket is still open and two options are presently considered: water and helium. This has recently motivated a critical re-evaluation of the technical choices for the DEMO breeding blanket and the TBM concepts to be tested by Europe in ITER [25] (see also Sect. 5.1). The integration aspects related to the choice of the breeding blanket coolant affects the overall design layout of the DEMO plant, and bears a strong impact on design integration, maintenance, safety because of his interfaces with all key nuclear systems. Technical issues influencing the choice include: (i) thermal power conversion efficiency; (ii) pumping power requirements; (iii) required power handling capabilities of the blanket first-wall; (iv) n-irradiation structural material mechanical properties; (v) n-shielding requirements (e.g., reduce the blanket thickness that is critical at the inboard side); (vi) achievable tritium breeding ratio; (vii) breeder tritium extraction; (viii) tritium permeation and tritium inventory control and purification; (ix) chemical reactivity, coolant leakages and chronic release; (x) design integration and feasibility of BoP; and (xi) design of safety system like the Vacuum vessel Pressure suppression System (VVPSS) that shall contain and confine the primary coolant in case of in-vessel Loss of Coolant Accident (LOCA) keeping the Vacuum Vessel (VV) pressure below the limit presently set to 2 bar (as in ITER). Studies are progressing to investigate all the aspects related to the selection of the coolant for the blanket and the BoP. Preliminary results of this work is describe elsewhere [26].

KDII/3 - The choice of the divertor configuration is a crucial design aspects, and there are still uncertainties as to whether the design concept adopted by ITER can be used in DEMO, or if alternative solutions are required. Experimental data on highly-radiating (90%+) plasmas, as required by ITER, and models of their energy confinement are scarce. Attractive alternative divertor configurations including double null (DN), snowflake (SF) and super-X (SX), might offer the possibility of distributing the divertor load on larger wetted areas which result from either increased number of strike points or flux expansion, or of stably increasing the level of SoL/divertor radiation to decrease the power density on the plate. The physics performance of these advanced divertor configurations is being investigated [27], but there are serious concerns on the implications arising from the engineering requirements for example of integrating additional coils, provide additional neutron-shielding and more complex remote maintenance provisions. Specific work is underway to assess the impact of incorporating these alternative configurations into DEMO whilst respecting requirements on remote handling access, forces on coils, plasma control and performance, etc. Initial results of this work [28,29] indicate that the greatest challenges relating to each configuration are: (a) for DN, finding an appropriate segmentation and efficient remote maintenance scheme; (b) for SF, understanding of the physics allowing high stable X-point radiation, and also control of the positions of the divertor limbs under plasma movement (Fig. 5); (c) for SX, finding a coil layout which achieves the target plasma equilibrium while respecting coil force limits and remote maintenance access. This choice is additionally complicated by the fact that the SX concept only ameliorates the heat load at the outer target, meaning that the benefits are very limited except using a DN SX configuration, which would also substantially increase the magnetised volume of the machine.

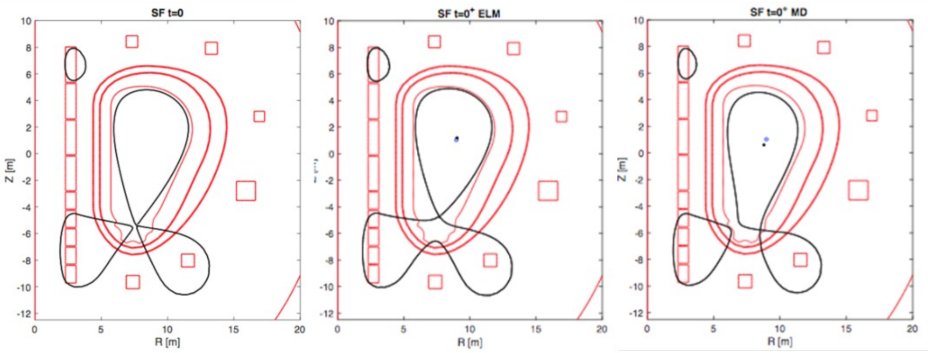


Figure 5: Simulations of SF limb positions in normal equilibrium (left), following an ELM (middle), and following a vertical displacement (right). In each of the two disturbed cases, the limb positions and relative power-sharing are substantially altered and risk placing high power densities on unprotected in-vessel components. [28]

KDII/4 - The remote maintenance DEMO design process is developing an integrated and consistent strategic approach to meet the high level plant requirements and allow remote or manual operation throughout the active areas of the plant. Due to the performance trade-off between the operational performance of in-vessel components and the remote handling suitability, the interrelationships and possible interacting challenges of extracting breeder blankets are being actively investigated in the pre-concept design stage. The most significant risks remain those related to the control of the large, relatively flexible, in-vessel components and the challenging in-bore welding and inspection of service pipes (see Sect. 5.3). Results of work conducted to date to investigate blanket handling solutions has shown that there are significant risks associated with the handling of full blankets segments due to: the limited space in the port; the control issues associated with the stiffness of the mover and payload; seismic constraints; and the consequence of a dropped load. This has led to the consideration of alternative blanket segmentation concepts and this has become one of the key design integration issues to be solved in the concept design phase. Along with the wider integration programme, alternative architectures that can accommodate limiters and a double null divertor configuration to establish an effective configuration that meets the DEMO plant requirements are being considered [30].

KDII/5 - Considerations related to the characteristics of the BoP play an important role in the design and the licencing of the DEMO plant. Emphasis at this early design phase has been on a few important aspects of BoP, particularly the PHTS [31,32] and the relevant Power Conversion System (PCS) [33] because of their technical complexity and strong impact on design integration, maintenance, safety [34]. The pulsed operation foreseen for DEMO is particularly demanding for the Plant Electrical System (PES), the BoP and the PCS. In a conventional (fission) Nuclear Power Plant (NPP) the main components of the BoP/PCS, are designed for steady state operation and not for very frequent transients. A large Intermediate Energy Storage System (ESS) is currently considered, which add complexity and cost to the plant. Alternative design options featuring a more direct coupling of the PHTS to PCS, requiring only about 10% of nominal flow for the steam turbine and thus much smaller storage of molten salt are being studied. Main features of this new concept are: about 10% of nominal flow is required by steam turbine; much smaller storage of molten salt (see Sect 4.3.2).

KDII/6 – Work is underway with the support of a nuclear architect-engineer company to investigate various plant building layouts and assess the feasibility of a number of technically foreseeable solutions (see Sect. 4.3.1). Attention on concepts of buildings that enable on adequate space provision for ex-vessel maintenance (e.g. adequate space around the components) and for reducing the radiation exposure to the personnel (improve shielding/make use as much as possible of remote maintenance) and, at the same time, meet stringent safety and licensing criteria as compartmentation of areas with redundant safety systems, minimisation of radioactive inventories or and, enthalpies, fire loads, segregation of circuits to minimise liquid operating releases to environment, etc.

KDII/7 – A novel fuel cycle architecture, based on the concept of Direct Internal Recycling (DIR) [35] is being investigated to minimise the tritium inventory. The extrapolation of the concept used by ITER will not be feasible for DEMO because of the much larger throughput. Thus, proofing the feasibility of this concept is a high priority. This involves the so-called KALPUREX process [36] that replaces the discontinuous pumping used in past fuel cycle architectures with continuous pumps. These are based on mercury (mercury vapour diffusion and mercury liquid ring) to be fully tritium-compatible, and a large scale demonstration unit for this technology is under preparation [37]. But KALPUREX is also adding a completely new functionality to the fuel cycle, namely the separation close to the torus which is fulfilled by a metal foil pump. Although superpermeation, the basic physics principle behind the behaviour of a metal foil pump, is well known, it has never been implemented in an engineering design of a technical component such as a pump. A first proof-of-principle was achieved [38], but a much more robust R&D programme is under implementation to ensure the performance and feasibility of this concept and before dropping the backup solution that consists of a three-stage cryopump with distributed pumping that is also being investigated.

KDII/8 – The design of any fusion device is strong affected by the assumed plasma operating scenario. The development of the plasma scenario for DEMO is based on two high level criteria, namely ensuring a sufficiently high fusion power outcome to maintain a satisfactorily high net electrical output, and being able to guarantee the integrity of the plasma-facing-components (PFCs) for a sufficiently long operation time. Such requirements have to be met in an integrated approach to the machine design. This means that a suitable plasma scenario for DEMO, and in general for every next generation electricity producing device, has to be developed by taking into account from the very beginning all the constraints originating from the engineering and technological aspects of the design. In particular, the role of diagnostics, H&CD and fuelling and pumping have been identified as crucial. Thus, the activities carried out in the PCD phase are focussing not only towards a deeper understanding of the physics governing the DEMO plasmas, but also on a stricter interaction with the design and technological development process, in order to rely from the earliest phases on solutions which are compatible with an integrated and consistent approach to the machine design [39]. This effort has led early on to the identification of large risks related to the problems of divertor detachment control and to the effective reliability of ELMs mitigation schemes (to be demonstrated in ITER), because a small number of such events could cause serious damage to the divertor targets (see also Sect. 3.2). In DEMO-1, a single Type-1 ELM event will be sufficient to melt the divertor target tungsten surface, and 50-100 unmitigated ELMs would result in the erosion of the entire target thickness [17]. As natural ELMs are foreseen with a frequency of ~1Hz, ELM mitigation measures do not provide a credible solution for DEMO, and thus, it is very likely that an ELMy H-mode cannot be used as plasma operating regime in DEMO. This is perhaps true also for ITER. Therefore in ELM free regimes, even if at somewhat reduced H-factor and pedestal density might need to be considered for DEMO. R&D in present devices must be focussed towards building knowledge on such regimes (e.g. I-Mode or QH-Mode), namely to find operating boundaries, confinement and L-H transition scalings.

4.3 Plant Design

4.3.1 Plant Equipment Buildings

Preliminary DEMO plant layout configurations have been developed in collaboration with FRAMATOME GmbH (formerly AREVA) for the two options of using either water or helium to remove the heat from the breeding blanket. They are useful to identify the major buildings and structures needed to contain the plant equipment (see [3,34] and references therein) and their main characteristics in terms of dimensions. Fig. 6 shows an elevation view of the plant and a comparison with the European Pressurized Reactor (EPR) design. It includes several systems (i.e. breeding blanket PHTS, secondary loops, NBI, ECH, Magnet Feeders, Toroidal Magnet Quench Protection System, Cryogenic distribution and Vacuum Vessel Pressure Suppression System (VVPSS) for mitigating in-vessel Loss of Coolant Accidents (LOCA), etc.). This layout serves to help identify system integration issues, and to develop a technically feasible, operable, and a maintainable and safe plant design. It enables the identification of areas in which there are significant technical uncertainties, and to provide a clear basis for safety and cost analysis and further improvements. Other buildings such as the control building and the turbine building are similar to those in other nuclear plants, and their arrangements can be adapted readily to this plant. The conceptual design is deemed to be a feasible and consistent with current technology and industry practice. However, investigation into the impact of plant maintenance and the potential limitations coming from the licensing regulation, which were only given preliminary consideration in this study, must be continued in the future.

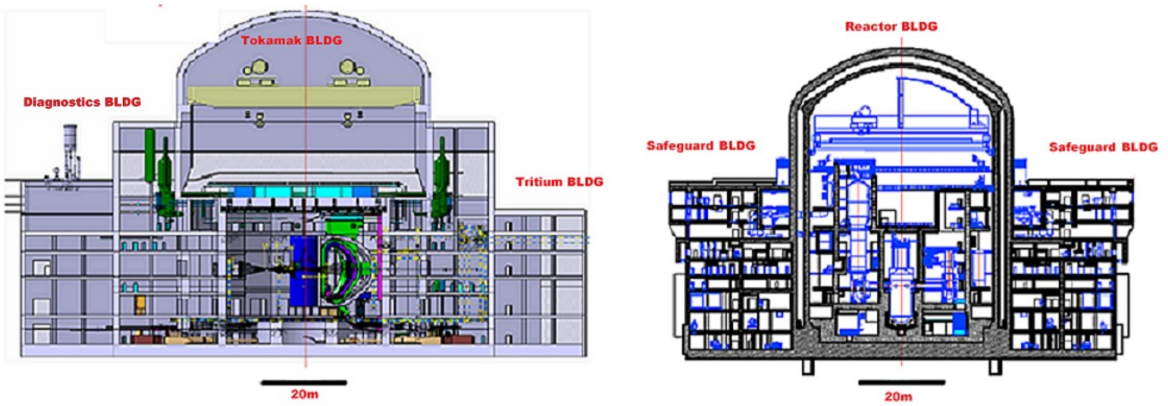


Figure 6 DEMO Tokamak Building Complex (compared with EPR)

4.3.2 Balance of Plant (BoP)

The primary coolant in DEMO must remove the heat by neutrons from the plasma and deposited volumetrically in the surrounding in-vessel structures (80% of the total fusion power). The remaining part (~20%) of the fusion power (fusion alpha particles) with the addition of the auxiliary heating power (~100MW) constitutes the so called "power exhaust", and is deposited as surface heat by plasma-facing-components (PFCs), i.e., first wall and divertors. Taking into account exothermal heat produced by nuclear reaction (about 1.2-1.3 energy multiplication factor depending on materials adopted), in a reactor of about 2 GW of fusion power, the blanket system has to remove about 1500 MW of nuclear power. Conversion of this energy at adequate thermodynamic efficiencies requires that the coolants are at high temperature and pressure. This has a strong influence on reactor engineering. Preliminary considerations related to the design of the Plant Electrical Systems (PES) are reported elsewhere [42].

The requirements of the DEMO BoP are very demanding in comparison with the similar systems of a fission power plant (NPP). Different cooling fluids, different temperatures and pressures and pulsed operation represent significant challenges to the design of the heat transfer and conversion system as well as the very large and, in part, pulsed electrical power requested by the different electrical loads necessary for the fusion reactor (several times bigger than the electrical power requested in a nuclear or conventional power plant) [[40]]. Any effort to reduce the complexity of the DEMO BoP, through simplification and a rationalization of the design and operation of the main reactor systems are expected to have beneficial returns on the design of BoP systems, the safety the operation of the plant and ultimately of the costs.

Work is ongoing with the strong support of relevant industry, to investigate the design of both options of helium and water as coolants for the breeding blanket to advance the design of PHTSs, Intermediate Heat Transfer System (IHTS) and PCS and to assess the readiness of the technologies postulated for a plant that operate with an Energy Storage System (ESS) [[41],[42],[43],[44],[45]]. Fig. 7 shows the preliminary layout and Table 5 summarises the main characteristics of main BoP equipment parameters for the case of a Helium Cooled Pebble Bed (HCPB) and a Water-Cooled Lithium Lead (WCLL) concept, respectively. Such work is useful to: (i) assess dimensions of main components (e.g. HEX, circulators/ pumps, pipes, collectors); (ii) identify technical feasibility issues; (iii) understand commercial availability and R&D needs; and (iv) establish layout requirements and evaluate integration implications with other systems. An attractive alternative design option is being investigated providing a more direct coupling of the PHTS to the PCS with a much smaller ESS. In this case, only about 10% of nominal flow would be used by the steam turbine during the dwell period and a much smaller storage of molten salt (HITEC) would be required.

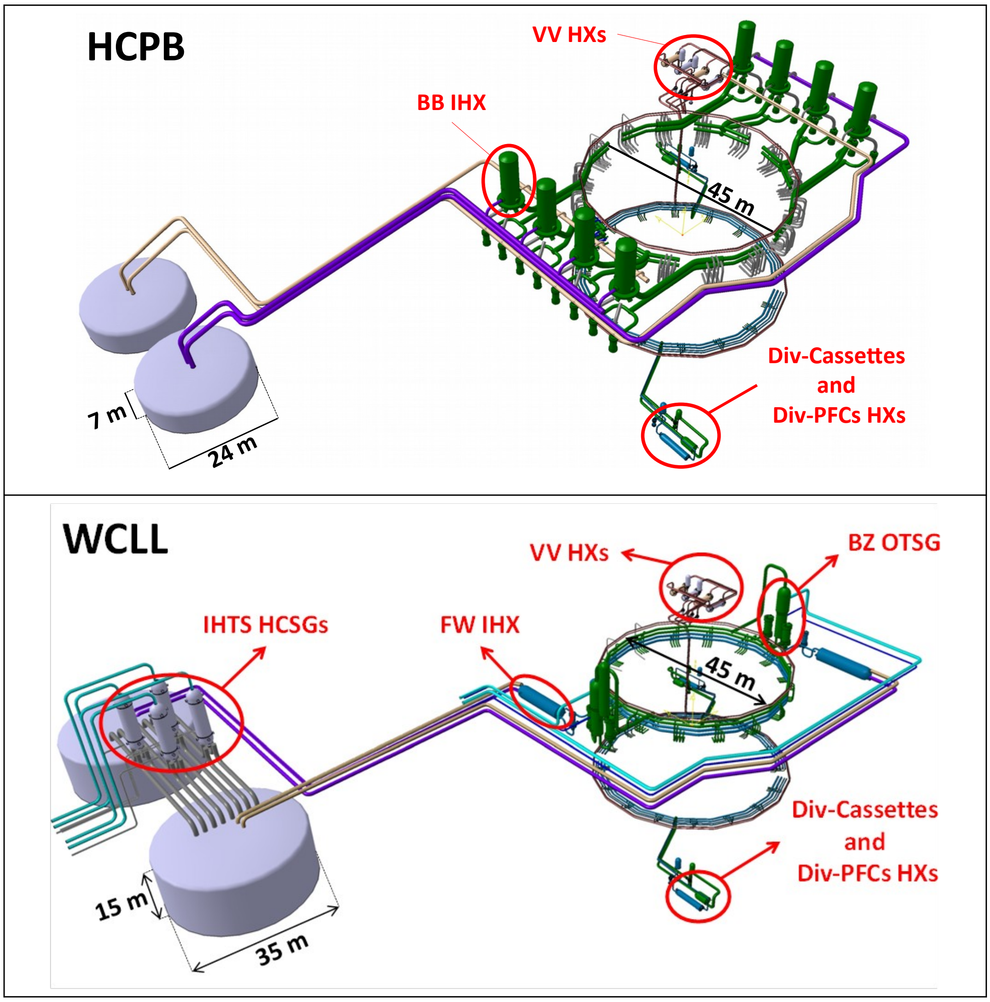


Figure 7 Layout of PHTS and IHTS for HCPB and WCLL

Figure 7 Layout of PHTS and IHTS for HCPB and WCLL BB: Breeding Blanket, Div: Divertor, VV: Vacuum Vessel, PFCs: Plasma Facing Components, IHTS: Intermediate Heat Transfer System, BZ: Breeding Zone, FW: First Wall, IHX: Intermediate Heat eXchanger, OTSG: Once-Through Steam Generator, HCSG: Helical Coil Steam Generator.

Table 5 Representative characteristics of main BoP equipment

BoP main systems/ equipment	**HCPB**	**WCLL**	**Fission EPR** (for comparison)
# of separated primary coolant systems:	14	8	1
- *BB*	*8*	*2 (4 loops)*	*N.A.*
- *Div*	*4*	*4*	*N.A.*
- *VV*	*2*	*2*	*N.A.*
- *RCS*	*N.A.*	*N.A.*	*1 (4 loops)*
# of primary HX/SGs	14	10	4
- *BB*	*8*	*4*	*N.A.*
- *Div*	*4*	*4*	*N.A.*
- *VV*	*2*	*2*	*N.A.*
- *RCS*	*N.A.*	*N.A.*	*4*
# of pressurisers	6	8	1

IHTS	1	1	N.A.
IHTS HCSGs	T.B.D	4	N.A.
MS tanks	2	2	N.A
PCS steam cycle	Dual Superheated Rankine cycle	Superheated Rankine Cycle (B&W PWR like)	Saturated Rankine Cycle
Overall piping length [km]			
PHTSs:	6.7	5.5	0.1
- *BB* - *Div* - *VV* - *RCS*	*2.9* *2.3* *1.5* *N.A.*	*1.7* *2.3* *1.5* *N.A.*	*N.A.* *N.A.* *N.A.* *0.1*
IHTS	0.8	1.2	N.A.
Coolant inventories [m^3]			
PHTSs:	2423	1173	460
- *BB* - *Div* - *VV* - *RCS*	*1680* *173* *570* *N.A.*	*430* *173* *570* *N.A.*	*N.A.* *N.A.* *N.A.* *460*
IHTS tanks	6000	22000	N.A.

Acronyms. *BB: Breeding Blanket; Div: Divertor; VV: Vacuum Vessel; PHTS: Primary Heat Transfer System; IHTS: Intermediate Heat Transfer System; RCS: Reactor Coolant System; HX: Heat eXchanger; SG: Steam Generator; MS: Molten Salt; PCS: Power Conversion System.*

6. Design Readiness and Maturation Strategy for Critical Technologies for DEMO

5.1 Breeding blanket

Achieving tritium self-sufficiency will be an unescapable requirement for any next-step fusion nuclear facility beyond ITER. Just as an example, a DEMO with a fusion power of about 2 GW will consume circa 111 kg of tritium per full power year (fpy), and this clearly underscore the indispensable requirement for the breeding blanket to produce and enable extraction of the bred tritium to achieve tritium self-sufficiency. It should also be kept in mind that ITER operation will largely use up currently known civilian stocks of tritium, from CANDU-type fission reactors, and that tritium supply considerations are very important to define the implementation timeline of a DEMO device, which must breed tritium from the very beginning and use a significant amount of tritium (5-15 kg) for start-up operation [[46],[47]]. This points to the urgent need to monitor the future availability of tritium and to understand the impact on limited resources on the timeline of DEMO. However, there is very little that the fusion community can do to exert an effect on the supply side, as tritium is a by-product of the operation of these reactors and not the primary economic incentive. Defense stockpiles of tritium are unlikely ever to be shared, and commercial CANDU operators will not alter their plans just to sell more tritium for the start-up of the first fusion power plants. In the short-term it is recommended to monitor the production of tritium in HWRs and estimate the available supply commercially. If, at some point in the future, it looks as though the demand for DEMO will exceed the supply from CANDUs, then action would have to be taken. It is likely that production of significant amounts of tritium from a dedicated source would be very expensive and take a long time. The "tritium window" as it was once defined by Paul Rutherford [[48]] is not open indefinitely. Based on current estimates, we believe it would be open until around 2050, after which it closes quite rapidly, unless the future of the CANDU reactor program turns out much more favorably than could presently be expected. The most advantageous way to fit fusion development into the tritium window would be to timely construct DEMO after ITER on the presently current timetable in Europe. Any program strategy that substantially delays substantially the DEMO step places fusion at risk, by allowing the unique and effectively irreplaceable tritium resource to decay to levels, which may be insufficient to complete fusion's technological development.

In spite of its criticality for fusion development, no fusion blanket has ever been built or tested. Hence, its crucial integrated functions and reliability in DEMO and future power plant are by no means assured. ITER presents a first and unique opportunity to test the response of breeding blanket materials and representative component mock-ups, specifically called Test Blanket Modules (TBMs) at relevant operating conditions, in an actual fusion environment, albeit at very low neutron fluences (see for example [46,[49]] and references therein).

Recently, work on the DEMO pre-conceptual design in Europe has also clearly shown that some of the technical features of the breeding blanket (e.g., the type of coolant, the type of breeder, the type of neutron multiplier) impact not only the design of the breeding blanket itself but also the design of the interfacing systems and, as a consequence, the overall design layout of the nuclear plant, and bear a strong impact on design integration, maintenance, safety because of his interfaces with all key nuclear systems. This has led to change of the design and R&D strategy for the DEMO breeding blanket and the ITER Test Blanket Modules (TBM) [25]. Focus is now on the two most promising and mature blanket concepts for DEMO. i.e., the HCPB and the WCLL, with limited R&D activity on the other concepts (e.g., Dual Coolant Lithium Led (DCLL)) – see Fig. 8 [[50],[51]]. Accordingly, one of the two helium-cooled test blanket module in ITER, the one using lithium lead as a breeder/ multiplier (HCLL) has been replaced with a WCLL. This will enable testing both high temperature/high pressure coolants (helium and water) and breeder/neutron multiplier materials combinations (PbLi and ceramics/ Be), which is perceived to be the best strategy to minimize the technical risks and gaps.

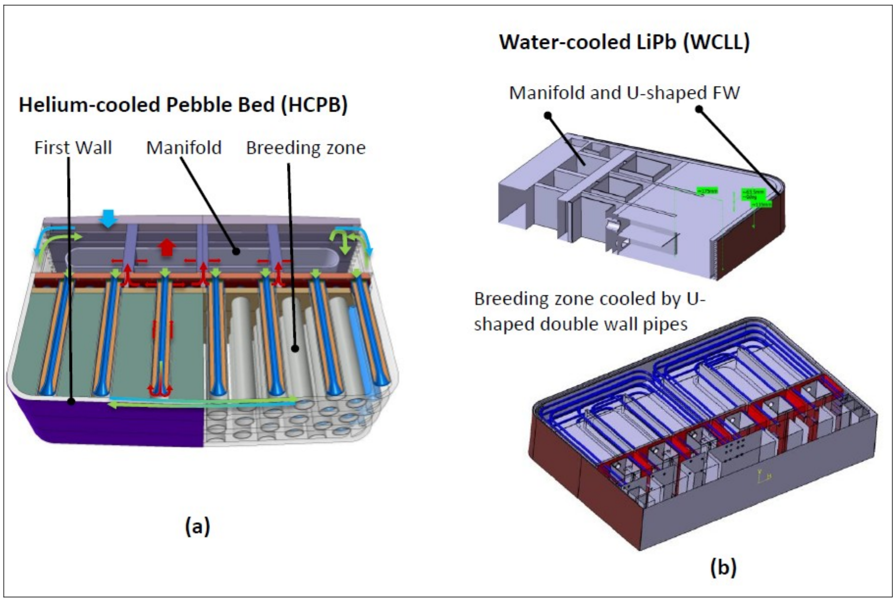


Figure 8 Schematic representations of (a) HCPB and (b) WCLL breeding blanket concepts.

To further minimize the risks, DEMO is being design to act as a Component Test Facility for the breeding blanket. This means that while operating with a near-full coverage breeding blanket, called "driver", which must be installed by day-1 to achieve tritium self-sufficiency and extract the thermal power and convert this in electricity, DEMO will test and validate in a number of ports, more advanced breeding blanket concept(s) that have the potential to be deployed in a future FoaK fusion power plant. The idea to test advanced blanket concepts in a reactor operating with a conservative breeding blanket design is not new. Early considerations were already given to this in the 80's (see for example [[52],[53]]. Such flexibility and capabilities, however, have to be properly investigated early in the conceptual design phase and formalized as high level requirements, since they have major implications on the plant architecture, and systems requirements. This implies that adequate equipment external to the DEMO basic device (test loops) must also be installed at the beginning, or provision made for its later installation. The design features of the test elements should be compatible, reliable and safe enough not to jeopardize the operation of the DEMO Plant. The detailed design of the test elements will be done during the conceptual design phase.

The selection of the "driver" breeding blanket and most promising advanced blankets is now impossible because of the existing uncertainties. A sustained programme of technology R&D is under implementation (see Fig. 9) to reduce the risks and a decision of the DEMO driver breeding blanket is now planned by the first half of the next decade (see Sect. 2.2) by taking into account design and R&D input obtained not only in the area of breeding blanket and TBM, but safety, materials, BoP and remote maintenance, etc. [26]. This will enable a DEMO plant concept to be coherently designed for a design review by 2027 (see Sect. 2). The design, R&D and testing of TBMs in ITER is viewed as an essential step to reduce the remaining technical risks and uncertainties associated with the demonstration of power extraction and tritium breeding technologies essential for a DEMO fusion power plant. This is required for: (i) developing and validating the scientific understanding and predictive capabilities; (ii) demonstrating the principles of tritium self-sufficiency in practical systems; (iii) developing and qualifying the breeding technologies to be used in next-step machines (i.e., DEMO); (iv) providing the first integrated experimental results on safety, environmental impact, and efficiency of tritium extraction systems;

and (v) providing initial components and operational reliability data for different ancillary systems (e.g, PbLi circuit, cooling systems, coolant purification systems and tritium extraction systems). The lesson to be learnt by the design and R&D of the ITER TBMs (both breeding boxes and ancillary systems) is viewed to be particularly valuable to aid the development and the down selection of the DEMO breeding blanket concept and will be discussed later in this paper. The completion of the TBM R&D phase II program is mandatory for the verification of the choice of the "driver" blanket, with validation being completed before starting DEMO construction.

However, large gaps would exist even with a successful TBM programme. It is nevertheless clear that risks and gaps will remain after ITER and, therefore, a sound and complementary R&D Program for DEMO to address long time performance at higher neutron fluence and high reliability is needed. In particular, vigorous materials irradiation in the limited number of In particular, vigorous materials irradiation in the limited number of existing fission research Material Test Reactors (MTRs) and ultimately in a DEMO-Oriented Neutron Source like IFMIF-DONES [[54]] is urgently required together with the construction of a limited number of dedicated non-nuclear blanket test facilities (or upgrade of the existing ones) for testing integrated multi-effect blanket behaviour.

In addition, to minimise the risks, associated with the lack of n-irradiation data for the candidate structural material (EUROFER) and consolidated nuclear fusion design criteria, it is foreseen that DEMO will operate replacing the set of blanket once. It will utilise a first set of blankets (called 'starter") with a 20 dpa damage limit in the first-wall steel (EUROFER) and conservative design margins and then switch to a second set of blankets with a 50 dpa damage limit with an optimized design, and if available, improved structural materials that need to be qualified in advance. Because it is unfeasible to change the BoP, the same coolant must be used while switching from the first set to the second set of blankets. Selecting 20 dpa as a limit, is due to the fact that irradiation of structural material of interest at this dose values can be simulated with sufficient accuracy in existing Material Test Reactors (MTRs), because the level of the He production (to be expected up to this fluence in a 14 MeV fusion spectrum) is still relative modest (~300-500 appm, to significantly affect material properties. Fusion irradiation data to be provided in a DEMO oriented fusion neutron Source (DONeS) [54] foreseen to become operative by the end of the decade will be important to validate data collected in MTRs and extend irradiation data at higher fluences, relevant for the second set of breeding blankets.

This type of progressive licencing 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 [[55]]. Licensing approval for operation up to moderate damage and activation 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 mechanical design (larger design margin) to cope with uncertainties in the reactor loads and performances.

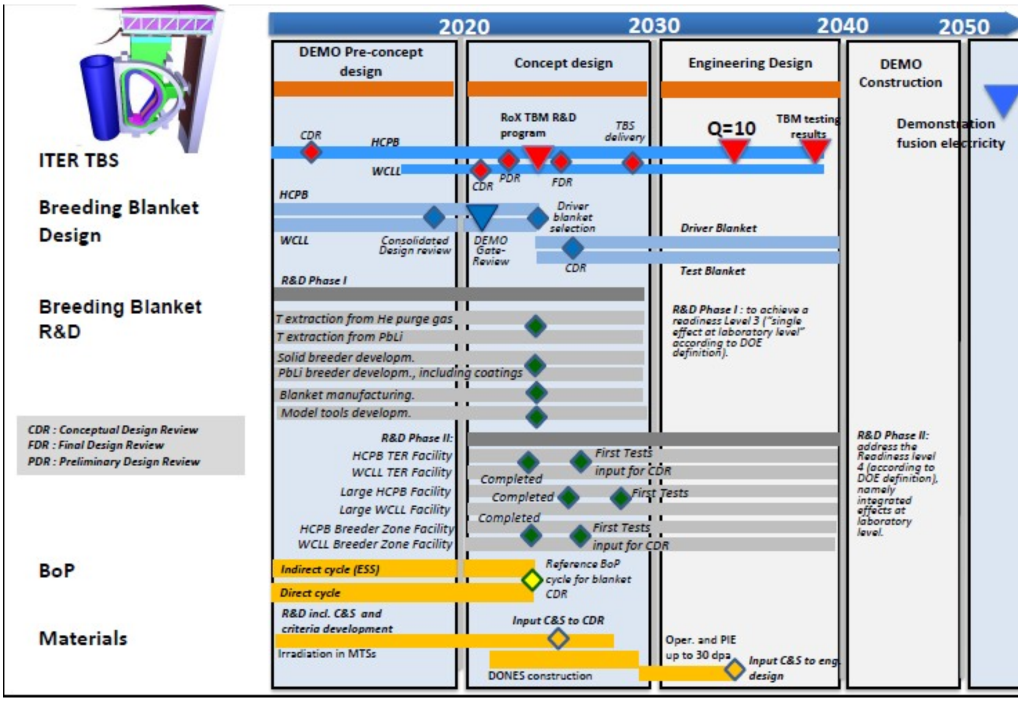


Figure 9 New design and R&D strategy proposed to re-align the ITER TBM and DEMO breeding blanket

5.2 Superconducting magnets

The DEMO magnet system includes presently 16 Toroidal Field (TF) coils, which provide the toroidal field needed for stable plasma operation, 6 Poloidal Field (PF) coils, which define and control the shape of the plasma configuration and stabilize the vertical position of the plasma and 5 modules for the CS magnet, which inductively establishes and maintains the plasma current.

For the TF coils, four winding pack (WP) options are proposed: one solution reproduces the ITER concept with radial plates, whereas the other three designs [56] explore different winding approaches (pancakes vs. layers) without radial plates, and manufacturing techniques (react & wind vs. wind & react Nb_3Sn), with the aim of improving the performance of conductors and propose cost effective solutions for the magnet system.

Out of these three alternative conductor designs, one option (TF WP#1) [57, 58] is based on the react & wind (R&W) method for Nb_3Sn magnets, in which the rectangular conductor is wound after the heat treatment, which is carried out without the stainless steel (SS) conduit and electrical insulation. The main advantage of this approach is the reduction of the effective strain acting on the superconducting cable, with an improvement of the transport capability of the superconducting strands. The drawback is that, after heat treatment, Nb_3Sn becomes fragile and the winding procedure of the coil becomes more problematic. The WP is wound in single layers (SL); this approach allows the grading of both superconductor (SC) and SS cross-sections in the different layers, with a relevant saving on costs. The results of the first experimental campaigns [57,58] helped to improve the layout of the conductor. Tests carried out on the last design [59] have demonstrated that there is not degradation of the current sharing temperature (Tcs) of the conductor after 1000 electro-magnetic (e-m) cycles and 4 thermal cycles. The AC losses of the cable proved to be low.

Another conductor option (TF WP#2) also has a layer-wound, graded structure, but based on Nb3Sn double-layers (DL), and a wind & react (W&R) Cable-in-Conduit Conductor (CICC) concept [60]. Due to the rectangular shape of the conductor, the effective strain on the superconduting material is lower than in ITER circular TF conductors, allowing a more efficient use of the SC. Compared to WP#1, the cost saving is lower but still considerable with respect to an equivalent pancake-wound coil. The conductor has been experimentally qualified; it didn't present any degradation of the performances after e-m and thermal cycles, whereas the level of AC losses was rather high and shall be improved in the future design.

An additional conductor option (TF WP#3) is wound in double pancakes (DP) and the conductor is based on Nb_3Sn W&R fabrication process [61]. The technology adopted is inspired to the ITER one, but without radial plates. The TF conductor proposal is square CICC, with a central spiral inserted in a thick square SS jacket to compensate the absence of radial plates. The conductor is in preparation and will be tested in 2019.

Thermal-hydraulic and mechanical analyses carried out for all WPs [56] have provided encouraging results, with some critical aspects that will be solved in future designs.

For the CS coil two designs have been proposed: the first is based on a pancake wound W&R Nb3Sn conductor, like in ITER [61]. The second concept [62, 63] is based on a hybrid design with layer-wound sub-coils using Rare-Earth Barium Copper Oxide (REBCO), R&W Nb_3Sn, and NbTi conductors in the high, medium and low field sections, respectively. A sketch of a section of the coil and of the high temperature superconductor (HTS) conductor based on REBCO stacked tapes is shown in Fig. 10. Compared to the first option, the hybrid configuration allows to keep the same flux with reduced size [62] or increasing the flux (and therefore the duration of the flat-top phase) keeping the same size [63].

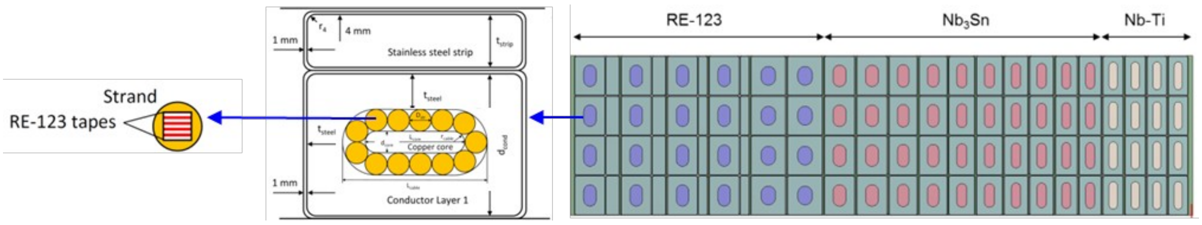


Fig. 10 Sketch of a section of the hybrid CS coil and of the HTS conductor based on REBCO stacked tapes.

In order to solve the open issues for HTS CICCs, the manufacturing of strands and cables made of REBCO tapes, followed by mechanical and e-m experimental investigations, are carried out [64,65]. In addition, an International collaboration with the Chinese team that is designing the magnet system of the Chinese Fusion Experimental test reactor (CFETR)), has been lunched to study the quench dynamics of HTS CICCs. The results collected from the experimental quench tests will be used to adapt and tune the thermal-hydraulic codes (developed for low temperature SC) to make them suitable for simulating the phenomenon in HTS conductors.

The overall strategy for achieving a consistent design during the concept design phase of the DEMO project is to identify the risks connected to each variant of the magnet design and implement actions to mitigate the risks. The objective is to collect all relevant information by 2024 in order to proceed with a down-selection of a reference option and a back-up variant for each magnet sub-system (see Sect. 2.2).

5.3 Remote maintenance

Maintenance presents many integration challenges across a wide range of plant. Some of these integration challenges are fundamental to the layout of the tokamak, the maintenance plan must therefore be resolved at an early stage to minimise the risk of costly redesign of DEMO [66]. Two examples of these integration challenges are the handling of the blankets and the service connections required for the in-vessel components. The conceptual design has been completed for a solution to the blanket handling requirements using a parallel kinematic mechanism and testing has been carried out on the proof-of-principle in-bore laser pipe cutting and welding tools and the pipe alignment system. Considerable effort and advances have also been made with the integration of the port maintenance equipment with the evolving component designs.

5.5.1 Blanket Handling

The design process has highlighted the technology risks associated with the handling of the large in-vessel components, through the narrow ports. The mitigation requires the development of new control system algorithms and simulation tools, designed to maximise the use of the sensors available to it [67]. To this end, the JET Telescopic Articulated Remote Mast (TARM) (see Fig. 11) has been refurbished and will be used to conduct initial tests of the novel control algorithms and structural simulation models with sensor integration and fusion [68]

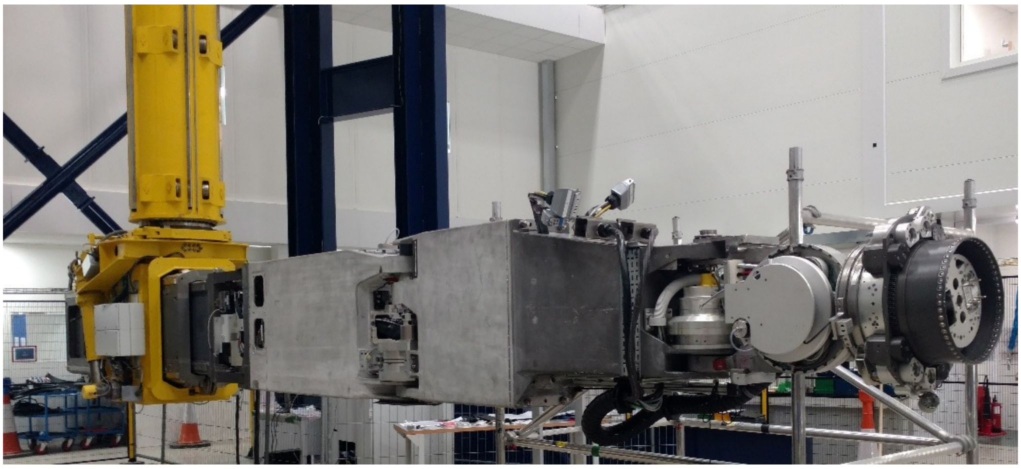


Fig. 11. Refurbished TARM manipulator mounted from new deployment frame

Given the high complexity and the need for very high levels of safety, reliability, accuracy and speed of DEMO remote handling operations, it is vital that the control systems are tested using realistic sensors, movers and payloads to demonstrate feasibility during the concept design phase. The testing will also provide, model validation and the demonstration of standard maintenance strategies, interfaces, tooling and recovery and rescue schemes. A scheme has been proposed for a maintenance test facility in which to perform these tests and thereby mitigate the maturation risks during the concept design phase and beyond. The proposal includes a layout and order of cost.

5.5.2 Service Joining

The laser pipe cutting and welding trials and pipe alignment tests (see Fig. 12) on the proof-of-principle designs have demonstrated the feasibility of the mechanical design of the pipe clamping, pipe alignment and laser spot position accuracy. They have also demonstrated that miniaturised laser optics can achieve the cutting and welding of the pipes, but that further development is required to improve the power handling.

Figure 12. Pipe alignment proof-of-principle pipe module side test assembly

To achieve a suitable a technology readiness level for the service joining system by the concept design review, welding and pipe alignment tests need to be performed in conjunction with the handling of the complete pipe modules, as part of integrated port design tests. This requires physical testing of the equipment in appropriately realistic test rigs in the maintenance facility.

The integrated testing of the systems in the upper and lower ports is required to demonstrate the feasibility of what is likely to be the most complex and space constrained areas of maintenance for DEMO.

7. Harnessing ITER Competence, Role of Industry and of International Collaborations

6.1 Harnessing ITER Competence

A key facet of the EU Fusion Roadmap, is to ensure that DEMO is positioned to capitalize on the industry competence gained from the design, construction and operation of ITER. It is argued that if DEMO is positioned too long after ITER, then there is a risk that this competence and highly skilled workforce would be lost. The updated DEMO schedule, has maintained the principle of the original Roadmap in this regard – taking into account the delays in ITER delivery – to ensure that DEMO development activities are phased to facilitate the transfer of skills & competence from ITER to DEMO. To further analyse the question of optimum phasing for transfer of industry competence between ITER and DEMO, it is important to differentiate between the exact nature of the industry skills required during different phases – some of these industry skills are highly specialised to fusion engineering – whereas others are more generic and are found in a number of industry sectors. For instance, there will undoubtedly be a big gap between the ITER buildings design and construction. However, this area of industry expertise is transferrable between fusion and fission new-build projects, and hence is not likely to be lost in the intervening period between ITER and DEMO building design & construction. On the other hand fusion specific industry expertise in areas such as Remote Maintenance, H&CD System, Diagnostics, Breeding Blanket and Low Temperature Superconducting Magnet design & fabrication etc. are indeed highly specialised and hence the phasing between ITER and DEMO for these elements needs closer analysis as is it at more risk of being lost if the intervening period between their respective utilisations is too great.

From a phasing perspective, one can observe from Fig. 1 that since the detailed procurement & installation of these fusion specific elements has been delayed in the revised ITER schedule – then DEMO schedule postponement is necessary to maintain the logic of the phasing and skills transfer envisaged in the Roadmap. In particular the sequencing of the ITER $2^{nd}/3^{rd}/4^{th}$ assembly phases (where major items of plant will be installed) will conclude during the DEMO EDA phase – and hence there is good alignment and likelihood of direct transfer of industry competence in these areas from ITER to DEMO in specialist fusion engineering capability.

Currently, frequent technical exchanges with the ITER Organization (IO) have been organised to ensure the benefit of sharing the lesson learned and design experience especially in the following areas: (i) tokamak building design; (ii) plant layout; (iii) Safety and licensing; (iv) systems engineering; (v) neutron shielding concept; (vi) port plug port integration and remote maintenance; (vii) in-cryostat maintenance; (viii) thermal shield design; (ix) design of magnet feeders; (x) Vacuum Vessel cooling loops; and (xi) H&CD and Diagnostics integration.

6.2 Industry Involvement

Lessons learnt from comparable projects, have highlighted the importance of involving industry during the early phases of the design development – especially for complex nuclear infrastructures. For instance, Gen IV programmes have leveraged impressive industry support, and engaged with industry as a partner from the outset. Work conducted to date in DEMO have highlighted a number of areas where harnessing of industry competencies can have significant impact during the conceptual phases in areas such as; (i) support in establishing systems and project management processes to deliver the project; (ii) translation of experience in obtaining construction and operational licenses for nuclear infrastructures, as well as pre-qualification of components and systems; (iii) assessments of design and technology maturity and prospects for licensing; (iv) experience in industrial plant design and integration; (v) development of concepts for major components and systems that incorporate manufacturability considerations; (vi) and cost assessments.

Conversely, engaging industry in the DEMO design activities early, allows the possibility to build a familiarity within industry of the particular challenges associated with DEMO. Furthermore, it provides some continuity for industrial suppliers in the interim period following completion of ITER procurements – but prior to the launch of major DEMO procurements – to maintain some interest and engagement in fusion. It also provides some opportunity for industry to steer the design direction, and encourages industry to participate not only as a supplier, but also as an important stakeholder within the project. Aligned to the scope and strategy described above, a number of tasks have been undertaken with industry so far. These include: the development of a DEMO plant layout (Sect. 4.3.1), the design of the vacuum vessel, the cryoplant and cryodistribution systems etc. Delaying the undertaking of DEMO Engineering Design too far beyond the end of construction of ITER will risk dissipating and losing this experience and interest of Industry.

6.3 International Collaborations

The following International collaborative efforts on DEMO design and R&D are acknowledged:

6.3.1 Japan: Broader Approach - IFERC

Joint DEMO Design Activities (DDA) were established in 2011 to address the most critical DEMO design and material R&D issues and investigate feasible DEMO design concepts. The preparation activities for DEMO that arise from the Broader Approach (BA) Agreement between Europe and Japan are part of the statutory tasks of Fusion for Energy. However, due the priority given to the ITER construction activities, F4E has not engaged directly in this work, but has established since 2014 a collaboration with EUROfusion (formerly with EFDA) to conduct the domestic activities in the DEMO area as part of the work programme of the EUROfusion PPPT Department. These are expected to continue beyond 2020 as part of the so-called post-BA phase (2021-2025) [[69]].

6.3.2 China: DEMO/ CFETR technical exchange and areas of technology collaborations

China has a very ambitious plan to exploit fusion energy for electricity production as quickly as possible to offset a foreseeable large increase in energy demand. During the last decade China has made substantial progress in reducing some of the large gaps in a number of key fusion physics and technology areas. These gaps stem solely from the fact that, for historical reasons, China did not start as early as the Western Countries to work on fusion. Successful construction and operation of medium-size superconducting tokamaks such as EAST is a tangible sign of this progress [[70]]. Similarly, contributions to ITER procurements in many important areas, show the impressive capabilities of Chinese Fusion Laboratories and Industry. China is currently developing the conceptual design of a nuclear fusion facility called the Chinese Fusion Engineering Test Reactor (CFETR) [[71]]. Nevertheless, at the moment China neither possesses the fully required technical knowledge nor has access to key fusion reactor technologies e.g. blanket, divertors, remote handling, etc. to successfully embark in the construction of a post-ITER nuclear device (e.g., CFETR and/or DEMO). Joint Design Activities between CFETR and EU DEMO design teams, do not exist at the moment but there are plans to improve design exchange in this area. Bi-annual technical design exchange meetings are organised to (i) exchange as detailed as possible latest progress on CFETR and EU DEMO designs; (ii) clarify the rationale for top-level requirements, operational parameters for the Plant and technical options being considered for the components/ systems. Also discuss readiness levels of solutions being considered, including readiness before and after ITER; (iii) identify commonalities and differences in assumptions (physics and technologies); and finally (iv) present / discuss project implementation schedules. Also number of technology collaborations are being implemented in the area of breeding blanket, superconductive magnets and remote maintenance.

6.3.3 US: Upgrade and operation of the Magnetohydrodynamic PbLi Experiment (MaPLE)

A multiple-effect facility, Magnetohydrodynamic PbLi Experiment (MaPLE-U) [[72]], has been upgraded at UCLA with funds from the DOE Office of Science/Fusion Energy Sciences and in partnership with EUROfusion and six European Fusion Laboratories. The facility is a first-of-a-kind in the world and has been designed to investigate 3-D MHD thermofluid multiple-effects and material interactions for liquid metal breeder/coolant flow systems for fusion energy. The facility's construction and commissioning were completed in summer 2018 and the facility started operation in August 2018. The first series of experiments were very successful, and the results provide confirmation of the recent UCLA discovery based on advanced 3-D MHD modeling that multiple-effects such as heating and temperature gradients in addition to gravity and magnetic field result in instabilities and flow reversal in all types of liquid metal blankets. This contrasts with the assumption made by fusion researchers over the past 30 years that the flow is stable and laminar based on separate effect modeling and experiments. The new results on MaPLE-U indicate the need for an intensive program of experiments and modelling to provide an understanding and a new database with which liquid metal blankets can be prudently designed and operated. This research is a key part of the planned US/UCLA-EUROfusion collaboration, and is important for all liquid metal blankets, such as WCLL, DCLL and HCLL.

6.3.4 Fission Reactor Irradiation Experiment in the HFIR Reactor at Oak Ridge National Laboratory

High fluence irradiation experiments to close gaps in the EUROFER data base are underway in the HFIR Reactor at Oak Ridge National Laboratory based a Karlsruhe Institute of Technology (KIT)-ORNL contract. This is complementary to the two most powerful reactors in the EU BR2 (Mol) and HFR (Petten).The ORNL reactor offers a unique opportunity in terms of fluence (superior to the current EU reactors). Currently two out of nine campaigns launched in 2017 and 2018 are operated by ORNL.

8. Concluding Remarks

There are still differences of opinions around the world on how to bridge the gaps between ITER and a fusion power plant. However, there are outstanding issues common to any next major facility after ITER, whether a component test facility, a Pilot Plant, DEMO, or other. These include the need to develop foreseeable sound technical solutions for the problems of power exhaust, tritium breeding, cooling and extraction of high–grade heat from the breeding blanket, remote maintenance for the in-vessel components, robust magnet designs, qualified structural and PFC materials, nuclear safety, etc. The European strategy foresees a DEMO Power Plant to follow ITER to be built and operational around the middle of this century. The staged design approach that is being implemented to design DEMO in Europe is described in this paper. This is based on (i) developing and evaluating system designs in the context of the wider integrated plant design. A more systems oriented approach has brought clarity to a number of critical design issues and has provided a clear path for urgent R&D. (ii) Targeted technology R&D and system design studies that are driven by the requirements of the DEMO plant concept and respond to critical design feasibility and integration risks. (iii) Evaluation of multiple design options and parallel investigations for systems and/or technologies with high technical risk or novelty (e.g., the choice of breeding blanket technology and coolant, power exhaust solution and configuration, BOP and PCS, etc.). This has led to a new strategy for the DEMO breeding blanket and a change of the TBM concepts to be tested in ITER. (iv) Evaluation of the design and technology readiness of the foreseeable technical solutions, together with a technology maturation and down selection strategy to bound development risks by adopting structured and transparent Gate Reviews (pre-CDR Gate 2020).

It should be noted that this approach represents an important change in the EU fusion laboratory culture and that involvement of industry and exploitation of international collaborations on a number of critical areas is desirable. In particular, incorporating lessons learned from the ITER design and construction, building of relationships with industry and embedding industry experience in the design are needed to ensure early attention is given to industrial feasibility, costs, nuclear safety and licensing aspects.

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