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DEMO Design Activity in Europe: Progress and Updates

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This paper describes the progress of the DEMO Design Activities in Europe and particularly work done to address critical design integration issues that affect the machine configuration and performance, the plant concept layout and the selection of system design and technologies. Work continues to be primarily focused on the design integration of a pulsed baseline DEMO reactor concept, but a number of alternative configurations (e.g., a double-null divertor and a snowflake divertor as well as a flexi-pulsed-steady-state operation, etc.) are under preliminary study especially to evaluate their DEMO reactor relevance. Some initial considerations are given on the strategy to implement a structured design and technology down-selection, that progressively reviews and narrows options to arrive at the DEMO plant concept that addresses major system integration risks and offers the best probability to satisfy all stakeholder mission requirements. Finally, some recent technical achievements are highlighted.

Keywords - DEMO, Fusion reactors, Systems engineering, Systems codes, Design integration, Breeding blanket, Divertor

1. Introduction

As part of the Roadmap to Fusion Electricity Horizon 2020 [[1]], Europe initiated in 2014 a comprehensive design study of a DEMOnstration Fusion Reactor Prototype (DEMO) with the aim of generating around the middle of the century, several hundred MWs of net electricity and operating with a closed tritium fuel-cycle [[2]]. This is currently viewed as the remaining 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.

In accordance with the strategy and ambition of the Roadmap, key features of the European DEMO design and R&D approach include: (i) a strong philosophy of 'systems thinking' and emphasis on developing and evaluating system designs in the context of the wider integrated plant design; (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) where possible, modest extrapolations from the ITER physics and technology basis to minimize development risks; (iv) 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 coolants, power exhaust solution and configuration, power conversion systems, etc.).

The Roadmap emphasizes ITER as the crucial machine on which the validation of the DEMO physics and part of the technology basis depends. There is therefore a high degree of schedule dependency between ITER and DEMO, although 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 to ITER exploitation, but relies on a progressive flow of input from ITER for design and physics basis validation prior to authorization of DEMO construction.

Recent external events (i.e. the revision of the ITER schedule that now sets 2025 as the date of first-plasma and the start of D-T operations approximately a decade later) have motivated a review of the DEMO development strategy and schedule to ensure that the DEMO design activities maintain coherence with ITER exploitation and that there is adequate information from ITER and other supporting devices, to substantiate the DEMO design and physics basis at critical decision points whilst also respecting other external constraints and drivers on the schedule. This re-examination has also provided an opportunity to absorb lessons learnt from ITER in terms of project management, design maturity and the importance of a systems engineering approach to clearly establish system requirements, and manage systems integration during the Conceptual Design Phase. As such, the revised DEMO development strategy places strong emphasis on

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development of requirements, examination of systems integration aspects, traceable concept down-selection and assessment of design and project maturity through the implementation of a formal Gate Review Process.

This paper provides an overview of the development strategy, and also highlights the progress in the DEMO design and R&D activities since [2] in the Power Plant Physics and Technology (PPPT) Department of the EUROfusion Consortium by geographically distributed project teams involving many EU laboratories, universities, and industries in Europe. Sect. 2 provides an overview of the roadmap revision to adjust to the ITER delay, Sect. 3 describes the design approach. Sect. 4 describes the plant design process, reference configuration and the design choices under consideration. Finally, in Sect. 5 some of the main technical achievements are highlighted.

2. Setting the DEMO ambition

2.1 DEMO in the EU roadmap

DEMO in Europe is considered to be the nearest-term reactor design to follow ITER and capable of producing electricity, operating with a closed fuel-cycle and to be a facilitating machine between ITER and a commercial reactor. The main mission requirements of DEMO in Europe are summarized in Table 1.

Table 1 European DEMO goals [1]

− Conversion of heat into electricity (~500 MWe)
− Achieve tritium self-sufficiency (TBR>1)
− Reasonable availability/ Several full power years
− Minimize activation waste, no long-term storage
− DEMO as a component test facility and pathfinder to a First-of-a-Kind (FoaK) Commercial Power Plant.

It is a device which lies between ITER and FoaK fusion power plant. In terms of where in relation to a power plant it should be positioned, the Roadmap schedule sets the ambition to realize the DEMO objectives by the middle of this century - which has a strong bearing on this positioning. With this in mind, the overarching principles of the DEMO development strategy 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 licencability 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.

2.2 Basis for revisiting the DEMO design schedule

In the time elapsed since initiation of the activities under the EUROfusion consortium in 2014 [2], there have been a number of external and internal developments that challenge some of the assumptions underpinning the original DEMO schedule. This includes the revised ITER schedule, as well as a greater appreciation of the 'integration challenge' required to resolve all the DEMO systems in a robust plant architecture. These developments have necessitated a review of the DEMO development strategy and schedule to ensure that the DEMO design activities will maintain coherence with ITER exploitation, and that system integration issues, physics uncertainties and technology options are properly investigated and assessed, prior to initiation of the Engineering Design Phase and launching of major procurement activities.

The revised schedule aims to maintain the target of electricity production around the middle of this century (see [3]) seeking the most pragmatic compromise between maintaining an ambitious schedule on one hand and reducing technical and project risks to an acceptable level on the other. However, it is clear that technology and physics basis of DEMO must be validated by successful operation of ITER prior to authorization of construction (and hence commitment to the majority of the capital costs). The most critical and final major validation input, is the demonstration of D-T burning plasma scenarios with Q=10 in ITER that are scheduled to start around 2035, including the results of the TBM programme (short pulse in 2037 and long pulse in 2039) [4]. The revised DEMO schedule would therefore allow the DEMO design basis to be validated prior to launching the major procurement and construction activities. Other factors that have been taken into account in the revised schedule include; the limited availability of tritium supply [5] and taking advantage of the opportunity to extract maximum benefit from the experience of realizing ITER in terms of the design, licensing, and development of the industrial supply chain and construction experience.

Efforts are also being undertaken, to assess how to develop a credible strategy to position DEMO in order not to preclude the important - albeit ambitious - goal to enable (at least during the later phase of operation) an acceptable extrapolation from DEMO to a FoaK Fusion Power Plant. In particular, the feasibility of a 'flexi' DEMO that operates initially in inductively driven pulsed regime, with the possibility to be upgraded to a long-pulse or steady-state machine with a greater reliance on auxiliary current drive, targeting a higher capacity factor and therefore greater FoaK Fusion Power Plant relevance is being investigated [6]. This is discussed further in Sect. 4. However, it should be noted that the validity of this approach requires the confirmation of physics & technology development paths that can be accommodated in a single device.

In addition, it is presently foreseen that DEMO should play the role of a "Component Test Facility" and, while using a so-called 'driver blanket concept' (i.e., the near-full coverage blanket concept to be installed by day-1 to achieve tritium self-sufficiency and to extract the thermal power deposited mainly by the neutrons and convert this in electricity), it must be used to test and validate in properly designed and supported ports or segments more

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advanced breeding blanket concept(s) having the potential to be deployed in a FoaK reactor. Such flexibility and capabilities 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.

The fact that the DEMO schedule remains reliant on critical input from ITER in order to validate the physics and design basis before proceeding to construction, leads to the outcome that it is necessary to delay the Engineering Design Phase and Decision to Construct. Accordingly it has been recommended that the revised schedule aims to make best use of the time remaining prior to full ITER exploitation to conduct a Conceptual Design Phase. Therefore, the revised development plan consists of the following three phases (see Fig. 2): (i) a Pre-Concept Design Phase to explore a number of DEMO plant concepts and develop system requirements up to 2020 (ii) a Conceptual Design Phase to mature and validate the baseline concept up to 2027; and (iii) an Engineering Design Phase beginning roughly around 2030[1] to develop the detailed design and prepare for the launch of major procurement activities.

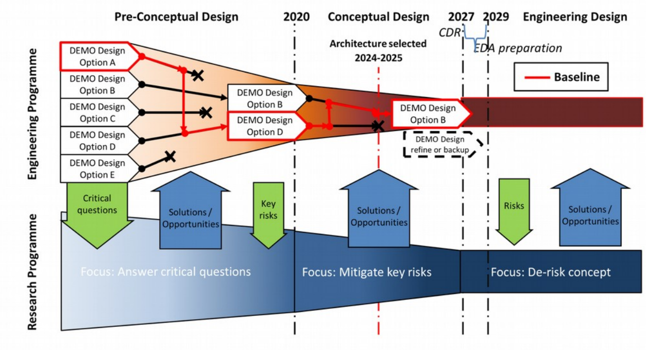

Figure 1– Design Strategy and Phases

Between each of the major phases, it is proposed that a phase gate review shall be carried out. A gate review is a formal review of all aspects of the project, including the evaluation of technical feasibility/risks associated with the design, but also aspects concerning cost, schedule, safety and any other aspects of importance to project stakeholders. The purpose of the gate reviews is for the project stakeholders to assess and determine whether continued investment in the project is warranted, considering the balance of risk/reward, and to assess the investment necessary to execute the subsequent phase of the project.

The Gate Review must therefore be carried out by suitability qualified and experienced individuals, and the gate exit criteria must be stringent and appropriate to justify the next phase of investment. There needs to be an understood level of maturity and hence stability in the design and technologies assessed at each Gate Review, and hence the Gate Exit Criteria must be developed in good time, and be realistically achievable.

Once a phase gate has been passed, the activities of the project must be reoriented to focus on the core scope of the next phase – the project should not be permitted to revisit or make major modifications to the design that were not planned for that phase and should have been resolved in the preceding phase. The proposed schedule of the gate reviews currently comprises a pre-CDR review in 2020 and a CDR in 2027. The activity to define the gate exit criteria is currently ongoing.

2.3 Overview of Pre-Concept Design Phase

During the Pre-Concept Design Phase, the focus is on establishing the system requirements through a top-down systems engineering process, identifying the main technical risk/feasibility issues, and assessing the potential of a number of DEMO candidate system architectures and design concepts to meet the requirements. The supporting R&D programme aims to respond to the critical questions and feasibility issue raised in the initial investigation of the options under investigation. As can be observed from Figure 1, during this phase, studies will continue to be focused on the baseline concept in order to ensure a thorough examination of integration issues and a level of coherence across the PPPT Work Packages (WPs) (see Sect. V). Fortunately, many of the integration issues are common across alternative plant concepts and so the work carried out will be of value regardless of the concept finally selected. A similar approach is applied at the sub-system level, where WPs are leading parallel investigations of candidate sub-system design concepts and technology options. In both cases, as the design work progresses, a process of concept feasibility assessment and down-selection will be implemented to progressively converge on the concept(s) that offers the highest probability of satisfying all requirements for minimal risk.

The Pre-Concept Design Phase shall therefore culminate in the selection of a plant concept with the highest likelihood of success by the end of 2020, and potentially one back-up alternative concept for risk mitigation and exploitation of potential opportunities (e.g. enabling technology dependent). The main activities that characterize the pre-Conceptual Design Phase are as follows:

a) Identification and agreement of the pre-CDR gate exit criteria.
b) Definition and substantiation of stakeholder and plant requirements, and cascading of functions and requirements to sub-systems;
c) Definition of the DEMO operational concept, including definition of operational phases, and main function required in each operational phase.
d) Identification and study of main technical risks and feasibility issues, in particular, those associated with design integration, technology development and safety (regulator acceptability).
e) Development of the baseline plant & building layout and demonstration of maintenance feasibility.

[1] Note that there is also a period between the completion of the CDR to allow for consolidation of the concept design before commencing the Engineering Design Phase

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f) Preliminary safety assessments, including assessment of alternative design a technology options.
g) Definition of the physics scenarios to be used for the concept design and identify and address physics basis development needs;
h) Completion of sensitivity analyses to understand the impact of uncertainties on physics assumptions (e.g, systems code, physics simulations, and engineering assessments);
i) Study of alternative plant concepts and assessment of their attractiveness against the reference baseline concept. Emphasis should be on engineering and operational challenges, safety, power conversion aspects of the power plant;
j) Screening and selection of possible sub-system design and technology options;
k) Identification of the main R&D requirements, including those that must be addressed with comprehensive programmes during the Concept Design Phase;
l) Building of relationships with industry and embedding industry experience in the design to ensure licensing, manufacturing and operational aspects are considered;
m) Preliminary cost estimates.

2.4 Overview of Conceptual Design Phase

The objective of the Conceptual Design Phase is to bring the baseline concept to a complete integrated system design so that detailed assessments of technical feasibility, safety, licensing issues and life-cycle costs can be undertaken, and preparations can be made for major procurement activities. It is paramount that system requirements and interfaces are validated to the extent that they can be frozen without a large risk of significant changes being required during Engineering Design Phase and procurement activities. The importance of validating requirements early in the programme, to avoid significant and costly changes later is illustrated in Figure 2 [7].

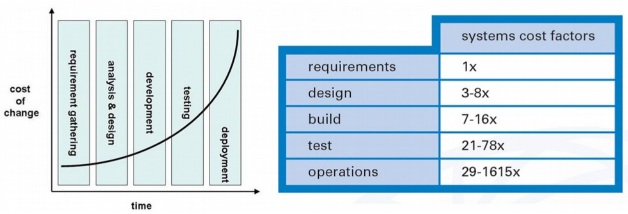

Figure 2 Costs of change to a system increase significantly past the pre-conceptual design phase (requirements capture and preliminary design studies).

To build the basis for demonstrating safety objectives can be met, the systems and components that are considered safety related or important for investment protection will be designed, and the plant licensing strategy will be established. To enable this, remaining decisions on sub-system design & technology options, and the reference physics scenarios must also be settled. Some of the key decisions that are expected to be made in this period include; selection of divertor configuration and first wall protection strategy; breeding blanket concept and coolant selection; remote maintenance (RM) strategy for in-vessel and ex-vessel components; H&CD mix selection and plasma operating scenario selection.

R&D work during this phase is expected to aim predominantly at the validation and maturation of critical technology elements, to establish confidence that the technology assumptions that underpin the DEMO baseline design are feasible. Large scale qualification and licensing demonstrations of systems and components are mainly foreseen during the Engineering Design Phase. Nevertheless, system level solutions upon which the plant concept is dependent should be validated during the Conceptual Design Phase, to mitigate the risks of significant overhaul once Engineering Design or procurement activities have been launched. In particular, the Remote Maintenance (RM) strategy is pivotal in the definition of much of the physical layout of the in-vessel components, vacuum vessel, magnets and the plant layout and buildings design. As there are strong implications on plant design and major front-end loaded procurements, it is important that the proposed RM strategy is confirmed through test-rig and trial demonstrations during the Conceptual Design Phase.

Finally, by the end of the phase, the concept design must be mature enough to develop credible development cost and schedule estimates for the subsequent Engineering Design Phase. The Concept Design Phase will culminate in a CDR Gate in 2027 where the reference design configuration is frozen in preparation of launching the Engineering Design Phase.

3. Design approach

3.1 Recap of main technology & design integration risks

ITER is a key facility in the EU strategy and the DEMO design and R&D is expected to benefit largely from the experience gained in the design, construction and operation of ITER. However, there are several differences between ITER and DEMO [8] and there are a number of technology, physics and licensing issues that remain to be addressed beyond ITER by DEMO. Some of the most prominent DEMO design technology decisions that are foreseen are summarized in Table 2.

Table 2 Main design integration and engineering development decisions.

- Breeding blanket concept and coolant selection
- Power conversion systems selection (i.e, direct or indirect)
- DEMO divertor configuration selection and first wall protection strategy.
- Remote maintenance schemes to achieve higher availability and interdependencies between nuclear plant layout and RM infrastructure. i.e, layout, access, shielding, removal, storage and handling of PFCs.
- Development and integration of novel tritium technology recycling and processing to demonstrate an technically feasible, efficient and economically viable tritium plant.
- Heating and Current Drive Systems selection mix

They play an important role in defining the tokamak size and the machine configuration as well as the plant layout. They also concern or interface with one or

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more key nuclear systems (e.g., primary heat transfer systems, tritium recovery and purification systems, heat exchangers, energy storage and power conversion systems) and therefore have a strong impact on safety, maintenance requirements and licensing.

These represent decisions that cannot be taken in isolation, but require the investigation of the variants in question within the context of the wider plant architecture. Investigating these fully, and managing the multiple associated configurations represents one of the major challenges and efforts during the Pre-Conceptual Design Phase. Moreover, discoveries in the plasma physics programme have highlighted large uncertainties in the DEMO plasma scenario assumptions. This has stimulated the need to increase confidence in key areas using a combination of theory and experimental activities to steer the formulation of the physics scenarios towards a credible point in the design space.

3.2 Systems engineering approach to support systems integration

In additional to the management and assessment of multiple configurations, the origins of many of the challenges in terms of systems integration can be attributed to the following characteristics inherent to the DEMO design; (i) the large number of sub-system interdependencies giving rise to a high degree of complexity in the overall system; (ii) the holistic, emergent behaviour of a tokamak; (iii) the large uncertainties in terms of physics and technology performance on which much of the design assumptions depend; (iv) the high level of integrity required of a design that must be subjected to nuclear licensing scrutiny; (v) the need to demonstrate maintainability, and high reliance on remote maintenance.

It has been identified that the implementation of a systems engineering approach led by a strong Lead Systems Integrator (LSI) is essential for the managing the development and integration of complex systems with a high degree of risk and novelty [7]. The systems engineering approach encompasses considering the spatial and physical integration between systems and components. In this regard, it is seen as a priority to develop a baseline configuration of the physical plant layout, to better understand the spatial/physical integration aspects from an early stage, to identify integration issues and improve coherency between system requirements. 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.

3.3 The Role of Industry

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 in PPPT industry tasks to date, and interactions with Gen IV projects, the Fusion Industry Innovation Forum (FIIF) and the DEMO Stakeholder Group (SHG), 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) 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 under the PPPT department. Some technical highlights from these tasks are introduced in Sect. 5.

4. Design choices under consideration

4.1 Current design baseline and definition of machine parameters

Work continues to be primarily focused on the design integration of a pulsed baseline DEMO plant concept, which serves as a "proxy" for more detailed design integration work (often critical in tokamak design), to understand integration risks and resolve design interface issues. Nevertheless, a number of alternative configurations (e.g., a flexi-pulsed-steady-state operation, a Double-Null (DN) or Snow-Flake (SF) divertor configuration, etc.) are being studied, albeit still preliminarily, (e.g., see Sect. 4.4 and Ref. [9]) to evaluate their DEMO relevance.

The process to define an appropriate set of plant parameters and technical features starts with the definition of the high-level requirements that the DEMO plant must achieve (e.g., net electricity output, tritium self-sufficiency, operation mode, etc.) and involves trade-offs

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between the attractiveness and technical risk associated with the various design options.

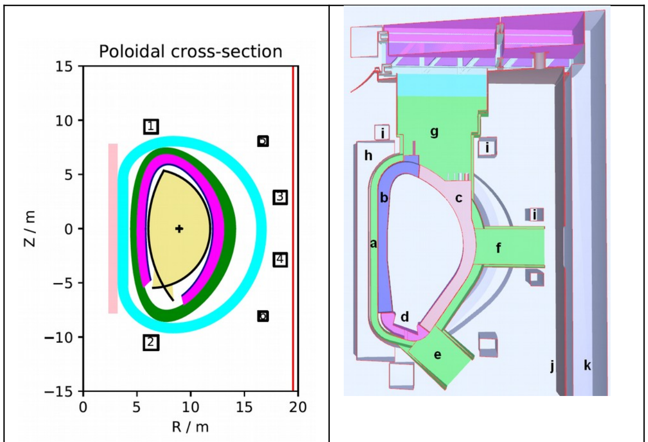
System codes such as PROCESS [10] and SYCOMORE [11] representing the full plant by capturing the interactions between (usually simplified) models of all the important plant subsystems are currently being used to underpin EU DEMO design studies and to find meaningful design points. The physics and technology basis entering European system code studies for DEMO is described elsewhere [12]. Although various optimization criteria can be used in a systems code to arrive at a single optimum design point, the three overarching criteria that are used in this early design phase to set the minimum tokamak size are (1) the divertor protection, (2) the access to the H-mode, and (3) the maximum field in the conductor of the TF coils. The divertor power handling has been found from the very beginning to be an important size-driver in DEMO [13]. The major radius, *R* is driven amongst others by parameters related to H-mode operation, divertor protection and the peak magnetic field in the TF coils conductor. Figure 3 in Ref. [14] shows the dependence of the major radius on f_{LH} and P_{sep}/R and peak field at the TF conductor from PROCESS calculations. The figure clearly shows that DEMO size is effectively driven by the divertor power handling capability, P_{sep}/R, (currently assumed to be ~17 MW/m). As we need $P_{sep}> f_{LH}P_{LH}$ to safely operate in H-mode (i.e., $f_{LH} f_{LH} \approx$ ~ 1.1-1.3) a reduction of machine size at constant performance can be realized, if (a) the divertor power handling capability can be increased far beyond values used in ITER (i.e. P_{sep}/R >17 MW/m) or (b) The required power crossing the separatrix to access high confinement must be decreased significantly, and (c) the peak allowable magnetic field at the TF coil conductor, the engineering current density in the TF coil winding pack, and higher mechanical strength cryogenic steels (achieving e.g. σ= 800 MPa) are developed and qualified.

Table 3 shows the major characteristics of the current DEMO baseline design point. Table 4 describes the main design assumptions in the current baseline design, whilst the design choices that are still open and are foreseen to be resolved at a later stage, e.g., during the concept design phase were presented previously in Table 2. A cross section of DEMO configuration is shown in Fig. 3.

Table 3 Key DEMO parameters (EU DEMO 2017)

Characteristics	**Value**
- Aspect ratio	*3.1*
- Major/minor radius (m)	*8.9/2.9*
- Plasma current (MA)	*19.0*
- Elongation/triangularity (95%)	*1.65/0.33*
- Toroidal field, axis/coil-peak (T)	*4.9/11.1*
- Auxiliary heating power - flat top (MW)	*50*
Performance (inductive)	
- Fusion power (MW)	*1998*
- Neutron wall loading (MW/m^2)	*1.04*
- Burn time (s)	*3600*
- Dwell time (s)	*<600*
- Volt-sec capability / Volt-sec for bum (Vs)	*695/359*
- Loop voltage (V)	*0.042*
- $\beta_{N,tot}$	*2.9%*
- Av electron temperature (keV)	*12.8*
- Av. Electr. density / Greenwald density limit (10^{20} m^{-3})	*0.79 / 0.73*
- Z_{eff}	*2.2*
- Plasma stored energy (GJ)	*1.25*
- Divertor challenge quantifier P_{sep} B / (q A R) (MW T/ m)	*9.2*

It should be noted that some of the underlying physics and technology assumptions (e.g., physics: confinement, plasma pressure, fraction of non-inductive plasma current; technology: maximum magnetic field in the superconducting coils, allowable surface heat loads in the divertor, neutron load limits on the structural materials, maximum recirculating power, thermodynamic efficiency, etc) are subject to uncertainties that bear a strong impact on the dimensioning of the tokamak system and the design and technology of the components surrounding the plasma. Therefore, studies are underway for the systematic treatment of uncertainties with system codes in order to establish their impact on the DEMO design, to establish quantify design impact and converge towards a robust design point [15].



(i) (ii)

Fig. 3: (i) Elevation view of the tokamak as generated by PROCESS; (ii) Tokamak radial-build: a) vacuum-vessel; breeding blanket (inboard); c) breeding blanket (outboard); d) divertor; e) lover port; f (equatorial port; g) upper port; h) toroidal field coils; i) poloidal field coils; j) cryostat; k) bioshield

Table 4 Current EU DEMO design assumptions

Main design assumptions
– P_{fus}~2000 MW~ 500 MW_e
– Pulses > 2 h
– Single-null water cooled divertor; PFC armour: W
– Low Temperature Super Conducting magnets Nb_3Sn (grading)
– B_{max} conductor ~12 T
– EUROFER for IVCs, AISI ITER-grade 316 for VV
– In-vessel RH: vertical (blanket)/ horizontal (divertor)
– DEMO plant lifetime (design) 10 fpy
– Neutron wall loading (average) ~ 1 MW/m2
– Thermal conversion efficiency > 30%
– Tritium fuel cycle: self sufficient
– Blanket lifetime • Starter blanket(a): 20 dpa • Second blanket : 50 dpa
– Reactor availability: A scenario is assumed in which the availability of a DEMO plant during its initial years of operation (stage I starter blanket) is relatively low (max 20%) and increases (in stage II) to about 30%.

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4.2 Supporting analysis studies

A number of studies that substantiate the design features and technical characteristic of the baseline and that have strong implications on plant parameter selection and architectural plant layout have been carried out. Emphasis in this early design phase is given on studies that address design integration risks and engineering/operational challenges that affect the overall layout of the DEMO plant and its performance, or that bear a strong impact on, maintenance, safety, or environmental impact. The rationale for the selected configuration of the main tokamak components and the main design integration issues are described in [16].

Some of the studies that have strong implications on machine parameter selection that underpin the main design features, or are ongoing studies to address the open design choices are summarised below:

- Sensitivity studies to determine impact of uncertainties of key physics engineering assumptions that affect plasma performance (see Ref. [14]).
- Trade-off studies for key design parameters to understand the impact on plasma performance, integration, maintenance, etc. Most notably, for the plasma aspect ratio [7] and the reduction of the thickness of the outboard breeding blanket [the number of TF coils [16].
- Preliminary analysis to determine the implications of the assumed plant availability and RM shutdown durations during various phases of operations on the plant tritium fuel cycle performance requirements to meet the high level plant requirements: the minimisation of the tritium start-up inventory, provision of a tritium stockpile buffer in the event of unforeseen shutdowns, and provision of tritium for a future fusion power plant [17].
- Safety analysis have been started to evaluate the response of the systems to abnormal events, and to guide the design to minimize potential accident consequences. Initially, Functional Failure Modes and Effects Analyses were completed for all key systems, identifying a set of Reference Event scenarios [18]. These are now the subject of computer modelling using established safety codes [19]. Inventories of tritium and activation products, potential source terms for these postulated accidents, have been re-evaluated [20], supported by neutronics and activation analyses [21] and assessments of sputtering [22] and activated corrosion products [23]. Experimental studies are also being performed to validate some of the codes and models in use, where existing data is inadequate [24,25,26,27]. Other safety and environmental issues being addressed include the minimization of routine tritium releases during normal operation, by comprehensively identifying the potential sources, and seeking to minimize these and restrict their pathways for release. A provisional study of the potentially largest contributors to occupational radiation exposure is also in progress, with the aim of influencing design choices to minimize potential doses. All these topics, together with others, are chosen to address a full range of safety issues [28], and to ensure that safety is fully taken into consideration in the conceptual DEMO design.
- Preliminary assessments of radioactive waste have been performed, focused on the influence of design options on the quantity and classification of waste [29]. R&D has been launched on techniques for detritiation of solid waste, and on the feasibility of recycling, together with industrial partners.
- Extensive neutronic analysis to confirm the ability of the adopted design solutions to achieve adequate TBR, shielding and activation levels (e.g., see for example Ref. [16, 29, 30]).
- Preliminary studies to integrate auxiliary system such as H&CD (EC, NBI), fuelling and diagnostics systems. Aspects being analyzed include: the opening in the breeding blankets and the impact on the breeding blanket segment design, remote maintainability, neutronics impact on the systems themselves and on other systems (e.g. shielding of the TF coils), safety [31].
- Global thermal-hydraulic analyses of the DEMO plant including the blanket and PHTS and provide a fast design tool to optimize the thermal-hydraulic performances and to support accidental analyses and the dimensioning of the associated systems (like the VVPSS) [32].
- Assessment of the plasma vertical stability and impact of thermal transients. It was found that the instability growth rate strongly depends on the assumed plasma elongation (k_{95}=1.65) and the distance of the plasma from passive structures, such as the vacuum vessel, which is the nearest toroidally continuous passively conducting structure to the plasma. The 3D effect of non-toroidally continuous breeding blanket modules and ports, is also taken into account. The analysis of the vertical stability control indicates that for the current configuration the passive stabilization satisfy controllability (m_s=0.3, [33,34]). The active stabilization was evaluated on optimised equilibria, with reduced distance between the plasma centroid and the magnetic axis, which improved the decoupling between plasma perturbations and vertical movement. The maximum power needed to control vertically the plasma, evaluated on 5cm VDE and plasma perturbations (ELMs, minor disruptions), results in a maximum electric power of 140MW (reactive power) during flatop (ITER flattop requirement = 200MW). This analysis needs to be re-run with 16TF coils geometry, (i.e. with PF coils farther from the plasma).
- Rapid vacuum pump-down of the plasma chamber to 1 mPa prior to the initiation of the next pulse and use of ECH assisted start-up to minimize the dwell-time to less than 10 min could have a beneficial impact on the minimization of the adverse effects of pulsing on the heat exchanger and turbine. Initial results by modelling are encouraging and the model used for the EC assisted

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breakdown is now being verified experimentally on the impurity conditions level relevant for DEMO.

- As the heat loads in a fusion device are poorly characterized [35], and their impact to the design of the in-vessel components is very important, a study has been carried out to investigate the impact of off-normal thermal transient loads. The results of this study are described elsewhere [36], but the main conclusion is that dedicated protections are going to be required in some areas (i.e., top of the machine). [16]. Alternative configurations, including the use of a second divertor on the top tokamak are also being considered [37]. In the case of a DN no additional PFC protection may be needed on the top, due to better active VS and radial vertical plasma stability control, and predictability of the possible plasma-wall contact areas. However, alternatives present disadvantages and integration challenges (e.g, TBR, maintenance) that need to be further investigated.

4.3 Preliminary physical plant layout definition

A first DEMO plant layout study has been performed in collaboration with AREVA GmbH to identify the major buildings and structures needed to contain the plant equipment (see [38,39]). The preliminary layout serves to help identify system integration issues, and to develop a technically feasible, operable, and maintainable 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. A first definition of the DEMO Tokamak & plant building layout has been outlined identifying important systems to be located there and their main characteristics in terms of dimensions. 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 of plant maintenance, which was only given preliminary consideration in this study, must be continued in the future. The problem of radioactive contamination of the plant and equipment must be given serious consideration in this regard, and in this vein, further design work is planned.

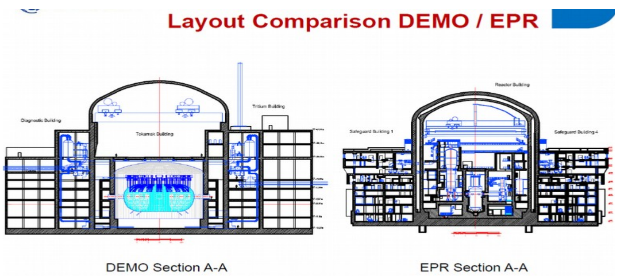

Fig. 4: DEMO Tokamak Building Complex (compared with EPR)

4.4 Alternative design tracks

The choice of the divertor configuration and first wall design are crucial design and operation aspects, and there are still uncertainties as to whether some of the design choices and technical solutions adopted by ITER can be used, or alternative solutions are required. The conventional solution to this problem, using an ITER-like divertor, is to assume that excess power can be radiated away from the main plasma and in the scrape-off layer (SoL); however, experimental data on highly-radiating plasmas and models of their energy confinement are scarce. Attractive alternative divertor configurations include the Double null (DN), the snowflake (SF) and super-X (SX). They 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 of the level of SoL/divertor radiation.

Previous work reported in [40] has investigated the performance and requirements of such divertor solutions by expanding around the existing DEMO baseline design, keeping the same (or similar) gross machine geometry and target plasma parameters such as shaping and current. However the outcomes of this work must now be re-integrated into a more complete design which accounts for the impact of e.g. the additional PF coils required for the altered equilibrium and changes in achievable plasma shape on overall plant performance. For example, the additional coils required for achieving an SF configuration, while maintaining acceptable remote handling (RH) access, depresses the available flux-swing from the central solenoid, requiring re-optimisation of the radial build if a two hour target pulse is to be achieved. Figure 5 shows, with very rough calculations, the dependence of the tokamak major radius for different P_{sep}/R values for an SF divertor, taking into account achievable shaping and flux swing effects. The current baseline value is also plotted. It can be seen that $P_{sep}/R >$ 27 MW/m — equivalent to radiating around 200 MW in the Scrape-of-layer and X-point, twice the conventional value — is required before the benefits of implementing an SF divertor become apparent. In addition, the increased X-point radiation implies a radiation load of up to 1.2 MW/m^2 on the first wall close to the divertor which must also be considered. Depending on the design of the divertor coolant system, there is also potentially an impact on RM time (which is dominated by pipe cutting and welding) and hence machine availability. The current work aims to provide a rapid integrated engineering assessment of the impacts of incorporating these solutions into a DEMO design and provide targets for the performance such configurations must achieve to be considered as viable and competitive for DEMO and future fusion power plants.

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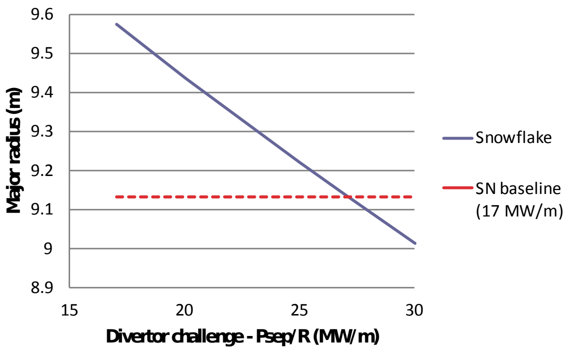

Figure 5: Plot showing major radius as a function of divertor physics performance (quantified as P_{sep}/R) for a snowflake divertor device targeting 500MW net electrical output and a 2 hour pulse length, taking magnet design, plasma performance, and additional space requirements into account. The dotted red line is the size of the ITER-like baseline design.

Considerations are also given to design a machine designed to initially operate in a short pulse mode (e.g., 1 hr), with conservative physics assumptions, but that could move to steady-state operation with foreseeable improvements in physics and current drive, based on latter-stage ITER scenarios. This option is referred to as flexi-DEMO [6]. The scope and extent of feasible engineering upgrades for a fully nuclear device, and improvements in physics are being assessed, in order to gain an understanding of what range of operating scenarios could be covered within a single machine.

Finally, a potential attractive option for improving plasma performance and reducing the tokamak size, and hence capital cost, is increasing the toroidal field. This could be accomplished through the use of high temperature superconductors (HTS), which remain superconducting at higher fields than conventional superconductors. The higher field allows higher plasma current at a given safety factor, and thus higher density, better energy confinement, and higher fusion power density. However achieving a high field places increased stresses on the structural elements of the TF coils and there is a coupling between the H-mode power threshold, which has a *B* dependence, and the power handling capability of the divertor which penalises high fields [11], HTS magnet models are being incorporated into European systems codes to investigate whether the potential benefits can be realised in the light of other assumptions about technology, and conversely which other technologies and materials need to improve — or reduce uncertainty in their performance — in order to take advantage of higher fields.

5. Main Achievements/ Accomplishments

The main recent achievements in the DEMO systems design and R&D activities are briefly summarized here. Work is conducted in the EUROfusion Consortium by geographically distributed project teams involving many EU laboratories, universities, and industries in Europe under the coordination of a LSI. Because of the limitation in space, we briefly report in this paper only on activities that address main integration risks or affect crucial system interfaces. However, references to work published in these proceedings or reported elsewhere are provided.

Breeding Blanket (WPBB) – Undisputedly, the breeding blanket is one of the most important and novel parts of DEMO and because of the numerous remaining uncertainties and feasibility concerns, a selection of the DEMO breeding blanket now is premature and a sustained programme of technology R&D is being implemented by EUROfusion – in the work package Breeding Blanket (WPBB). The design work conducted to date has clearly shown that some of the technical features of the blanket, e.g., the coolant, the liquid breeder, the tritium extraction pervasively affect the overall design layout of the DEMO plant, and bear a strong impact on design integration, maintenance, safety because of their interfaces with key nuclear systems (e.g., primary heat transfer systems, tritium recovery and purification systems, and power conversion systems). As it is foreseen to have DEMO, operating around 2050-2060, the time constraints allows only limited technological extrapolation that can be considered for the DEMO Breeding Blanket. The risk is to provide a system that cannot fulfil the goal to qualify a blanket to be used in a commercial reactor. As DEMO is operating as component test facility for the blanket, the main long term goals can be pursued using test blankets (they are called "advanced blanket") while the requirements of the basic (called driver blanket) can be reduced and made compatible with the available/mature technology, also if these technologies are not optimized for the long term goals. At the moment, four design options with different level of design and technology readiness are still considered as potential driver blankets within WPBB, utilizing helium, water, and LiPb as coolants and a solid or LiPb as tritium breeder/neutron multiplier [[41],[42],[43]]. The strategy is to arrive to the DEMO driver blanket down selection around the year 2024 [48] by taking into account design and R&D input obtained not only in the area of blanket, but safety, materials, BoP, remote maintenance etc. This will enable a DEMO plant concept design review by 2027, to be followed by an engineering activity to be concluded to allow a start of construction around 2040. For both cases with helium and water as coolant a preliminary design layout and performance analysis of the primary heat transfer system (PHTS), and power conversion system have been performed taking into account the coolant pipes layout and the required mass flow rates. This enables the estimation of the coolant inventory and the associated enthalpy, which together with the PHTS system segregation and layout are essential data for progressing with accidental safety analyses and for the design of key systems like the vacuum vessel pressure suppression system (VVPSS), which is an important safety-class component. Moreover recent considerable R&D progress in the following areas should be mentioned: production and characterization of tritium permeation and corrosion barriers and of suitable tritium extraction technologies, manufacturing and welding of EUROFER sub-

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components, characterization of double-wall pipes for water cooled concepts, fabrication and characterization of functional gradient tungsten layers for the first-wall, fabrication of advanced functional material for solid breeder blankets.

All the phases of the ITER TBM programme (design, construction, safety and licensing, procurement, integration and testing) play and will play a very important role in confirming/validating the choice of the breeding blankets to be installed in DEMO and of some technologies of their ancillary systems (e.g, PbLi circuit, cooling systems, coolant tritium purification systems and tritium extraction systems). However, to enable a consistent DEMO construction decision in time, the TBM programme would need to test the best combination of design options regarding coolants, breeding materials and technologies that could effectively minimise the technical risks for DEMO. However, the selection of the TBM to be tested in ITER was made by Europe in 2008, in absence of a comprehensive DEMO design study, and assuming that parallel advanced development in areas of Balance of Plant of Nuclear Systems and structural materials were to be expected from Fission Industry and in particular from the development of He-cooled advance fission systems (i.e., Gen. IV). A technical assessment is currently in progress in Europe to review this choice in view of latest findings and consider the possibility and the technical and financial implications to replace one of the two He-cooled with a water-cooled (WCLL) TBM in order to be able to test in ITER both coolants (Helium and water) and breeder material (PbLi and ceramic/Be): this is perceived to be the best strategy to minimise remaining technical risks and gaps to arrive to a consolidated design for the driver breeding blanket for DEMO. A decision is foreseen by the end of the year.

Balance of Plant (WPBOP) - Work is ongoing to assess the design, design integration and technological problems posed by the PHTSs for the breeding blanket (two options are currently considered He: 300-500°C, 80 bar; water: 292°C-328°C, 150bar), divertor and protection limiters (Water: 130°C-150°C) and the vacuum vessel (Water: 190°C-200°C) and to examine feasible solutions. [44,45,46,47] Such work provides a perspective on the design challenges related to both technologies and can act as a guide for further R&D. In particular, it 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. In addition, the pulsed nature of the currently considered reactor operation imposes unique design problems on the energy conversion system. In DEMO, energy is generated in the reactor for 120 min (burn time); the reactor is then shut down for 10 min (down time) for recharge.

An Intermediate Heat Transfer System (IHTS)/ equipped with an Energy Storage System (ESS) using Molten Salt as heat transfer fluid is being investigated to mitigate the impact of plasma pulsing on the steam turbines, other Power Conversion System (PCS) equipment's and the electrical grid. However, this introduces complexity in the plant and work is ongoing, involving industry, to investigate the impact of a direct coupling of PHTS to PCS relying in small auxiliary power sources to ensure safe turbine operation in down time Preliminary conceptual designs for both HCPB PHTS and WCLL PHTS and relevant PCS have been developed. They are considered to be feasible and no major showstoppers have been identified. However, some aspects have to be further investigated before drawing any firm conclusions. Table 5 summarizes main issues which have been identified for the blanket coolants. For both PHTS He and water, an assessment of the cost will start as soon as the initial design is finalized, later this year.

Divertor (WPDIV) - In the last two years a significant progress has been achieved with regard to the technology development for plasma-facing components (PFCs) of vertical targets and the thermohydraulic design of cooling circuit for the entire divertor.

Table 5: Main issues of blanket coolants (the fusion power is 2037 MW and the power deposited in the HCPB and WCLL are is 2102 MW and 1843 MW, respectively.

	Issue	HCPB	WCLL	Notes/assessment/optimization
PHTS Circuits	Dimension	9 separated circuits	2 separated circuits (4 Loops)	HEX available. Assumed Circulators of 9 MW (under construction Circulators of 5 MW)
PHTS Pumping power	Huge	150 MW	15.8 MW	HCPB: More accurate assessment of Δp on going (if Δp is lower, the pumping power will be reduced)
PHTS Length of Pipes	Dimension/layout integration/Inspection & Test/Cost	9 km (DN_max 800)	1.8 km (DN_max 750)	Under discussion: if pipes f 1.3 m diameter is available for HCPB. It will allow reduction of length of pipes and space reservation for PHTS (pipe length ~ 3km)
N16, N17 in PHTS	Radiation doses in the area where the PHTS is localised	No issues	Significant radiation dose during plasma	Need shielding and accurate layout of PHTS versus sensible equipment (e.g. I&C)
In-VV LOCA	Need of an expansion volume to accommodate the pressure transients below the VV design pressure	Very huge (≈120000/55000 m3 EV dry/wet)	Reasonable (≈ 1600 m^3)	Analyses on going to see: 1) the possibility to reduce partially such volume in case of HCPB through a reduction of the He inventory involved in the reference accident scenario; 2) the possibility to use the Active Maintenance Facility area or some area of the tokamak building as an expansion volume
Ex-Vessel LOCA	Need of an adequate volume with safety characteristics to accommodate the accident	Huge	Very huge	Analysis on going (layout definition and relevant safety analyses) to see the adequacy of the few volumes available in the tokamak building. Isolation valves might reduce the inventory of water lost in the accident (under investigation).

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Energy Storage System, e.g. Molten Salt (if confirmed necessary) to compensate the pulsed operation drawback on turbine and electrical grid	Increase complexity of the plant	Reasonable Volume of the Molten Salt Tanks (~11200 m^3)	Very big volume of the Molten Salt Tanks (~ 55.000 m^3)	Analysis on going to soft down the issue • Reduction of dwell time (i.e. from 30m to 10m, the volume of the molten salt will reduce almost proportionally) • Elimination of the energy storage system (direct coupling). Under study (with support of Industry) • Increase of the power of the Auxiliary Boiler • Motorization of the Electrical Generator
Tokamak building	Dimension of the secondary confinement barrier	Quite big	Reasonable	Optimization on going for He (see above, e.g. the possibility to use bigger pipes diameter)
ACP and Tritium in PHTS	Dimension of the auxiliary systems to maintain concentrations < safety limits	Issue only for Tritium	Issue for tritium and ACP	Analyses on going. Reference ITER licensed solution for WCLL
Chronic releases	Very limited quantity permitted	Potential issue	Ok (NPP operating exp.)	Under assessment
Operating Nuclear Experience (of components of the PHTS and PCS)	For Licensing, the operating experience on systems and components is very important	Limited	Significant	To be collected, some R&D necessary for HCPB

Currently, 7 different design concepts are being developed for water-cooled and one for helium-cooled target [48]. Besides the conventional ITER-like tungsten monoblock model, advanced design concepts were devised with novel composite materials or nonconventional design solutions. Small scale mock-ups were successfully manufactured by means of tailored joining methods and qualified by dedicated nondestructive inspection. The mock-ups are being tested in a hydrogen beam irradiation facility to evaluate high-heat-flux fatigue performance using cold as well as hot (130°C) coolant water. The mock-ups of 4 target concepts withstood at least 100 load cycles at 20 MW/m² while the mock-ups of the remaining concepts are still in preparation or testing. The pipework of the PFC cooling circuit and the internal ribbed structure of cassette body were designed and optimized [49]. 3D CFD analysis verified that the cooling scheme assured required power exhaust capability with reasonable thermohydraulic performance and operation temperature range compatible with the structural materials [50]. Progress on the ongoing physics work including investigation of innovative divertors is described elsewhere ([51]).

Remote Maintenance (WPRM) – Highlights of the recent technical work which is progressing with the development of the remote maintenance (RM) system, including the engineering design of a solution to the blanket handling requirements using a Parallel Kinematic Mechanism [52], the release for manufacture of proof-of-principle laser pipe cutting and welding tools deployed in-bore and the integration of in-vessel and ex-vessel maintenance equipment concepts with the evolving component designs and plant layout are reported in [53].

The RM 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. The RM system requirements captured in the Dynamic Object Oriented Requirements System (DOORS) have been updated to reflect the developments in plant and RM system designs. The technical risk register was further developed, identifying the WPRM high level risks during the concept design phase. 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.

As the RM system has developed, three different transfer strategies have been proposed [54]. In the first strategy separate casks are transferred to the vertical port to deploy and remove remote handling systems and components. In the second, a vertical transport cask remains docked to the vertical port throughout the maintenance process and smaller casks are docked to the transfer cask to deploy and remove remote handling systems and components. Currently, the remote handling systems and components are transferred through a hot-cell above the reactor. The conclusion of this work is that, by eliminating casks from the maintenance strategy, there was a significant reduction in the time taken to complete a DEMO maintenance campaign, because it eliminates the time taken to dock and undock the casks which is often on the critical path for the maintenance operations. Development of the Active Maintenance Facility design to reflect the latest transfer strategy has also shown that the reduction in the number of transfer casks significantly reduces that storage space required in the facility.

The maintenance duration estimate was developed in conjunction with a RAMI assessment based on the sequence and durations from the duration estimate. It considered the availability of the Blanket Handling RM System equipment and procedures to indicate the impact of reliability and down time on the time to perform the blanket exchange. It concluded that the availability of the 2015 RMS design, implemented with high reliability equipment at high cost, is 94%, with most of the downtime caused by the pipe cutting and welding operation and the consequence of the long repair duration following drops, slipping and collisions of heavy loads. Two suitable RAMI IT codes were also identified; 'FlexSim' and 'Plant Simulation'.

Three concepts for the Divertor Cassette Mover have been developed and compared along with designs for the fixation and earth bonding tooling and work is underway to integrate the pipes, cassette mover and vacuum pumping systems in the divertor port. The conclusion of this work is that the current tokamak configuration does not allow practicable RM solutions which are efficient and recoverable. The work on the Adaptive Positional Control System has been extended with further research into state-of-the-art techniques for control of systems that deform significantly under load. A first test of the system will be conducted in 2018 using the JET ex-vessel RM

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deployment system; TARM (Telescopic Articulating Remote Mast) which has been transferred to the RACE building and work is underway to recommission the system with updated control hardware.

Tritium Fuelling and Vacuum (WPTFV) - The design of the DEMO Tritium Systems is developed considering the following guiding principles [55,56]: (i) full application of the Direct Internal Recycling concept leading to two continuous re-cycle loops in addition to an outer loop with classical isotope separation and tritium plant exhaust detritiation technologies; (ii) Tritium inventory minimisation, requiring the continual recirculation of gases without storage, avoiding hold-ups of tritium in each process stage, and immediate use of tritium released from tritium breeder blankets (without intermediate storage). To increase the burn-up fraction, an additional exhaust gas re-injection loop is under consideration. (iii) Environmental protection and dose minimization under normal operating and accident conditions.

As for the concept development and simulation of DEMO vacuum systems, the main activities include: (i) commissioning of the mechanical pump train at the JET Active Gas Handling System to be ready for operation for the initial tritium phase of the DT campaign 2018-2020. This pump unit will teach valuable and first-of-its-kind lessons on the combination of a vapour diffusion type pump with a liquid ring pump, both operated with a common mercury circuit [57]. (ii) Development of a first complete metal foil pump module; following the extensive characterization tests of the metal foil as such [58].

In the area of matter injection, efforts concentrate on two candidate pellet injection concepts from the magnetic high field side to benefit as much as possible from drift effects, either using a curved guide tube system with a centrifuge, or a free flight option from the top requiring the currently less mature double stage gas gun technology [59].

The advances and main achievements of the other Work Packages are described elsewhere. For the Diagnostics and Control Systems (WPDC) see [60,61]; for the H&CD systems (WPHCD) see [62,63,64,65] and references therein, for the DEMO Magnets (WPMAG) see [66,67,68], for the activities on Materials (WPMAT) see [69,70], for Safety and Environment (WPSAE) see Sect. [28], for Socio Economics Studies (WPSES) see [71,72], for ENS [73].

Specific contributions of Industry **-** There is a general perception that contributions from industry to fusion are only related to manufacturing activities/ hardware rather than design and safety engineering of complex nuclear plants. We strongly believe that greater progress could be made with a better use of the experience existing in industry and to this extent we have started a number of activities in 2016/17, including:

(a) *DEMO plant layout* - A general plant arrangement and the internal arrangement of the main systems have been produced by AREVA GmbH taking into account information and experience available from the design and construction of ITER and design principles used for fission reactor plants (see Sect. 4.1). The study considered mainly the tokamak system (i.e., the heat source) and its main auxiliary (cryosystems, power supplies, H&CD systems), and the Primary Heat Transfer System (PHTS) that removes the thermal energy generated in the blanket, divertor, and vacuum vessel from the reactor, converts the thermal energy generated into electrical energy, and rejects waste heat to the atmosphere.

(b). *Methodology for assessment and evaluation of DEMO design & technology options* – In order to prepare for the down-selection of design & technology options, it is important that a structured and traceable methodology for decision making is developed (or identified from existing approaches) and validated to be suitable for application to the DEMO programme. AREVA GmbH performed the task of developing a multi-criteria decision making methodology for DEMO, combining analytical data, and expert judgement to evaluate options against high-level figures of merit. The methodology emphasizes a thorough evaluation of system interactions and second-order effects in order so that decisions are not taken in isolation. An initial case study was undertaken, and it is expected that this method will be further developed to be applied to support decision making during down-selection process in later stages of the programme.

(c). *Design for robustness and manufacture of critical components/systems such as the vacuum vessel (VV).* An alternative full sector fabrication approach that had been proposed by MAN-Deggendorf as one of the options for the ITER VV was adapted and studied for the DEMO VV . The advantage of this approach is that it allows for the correction of fabrication tolerances throughout the manufacturing process. The study resulted in the adaptation of the VV design concept to the fabrication including the requirement for full volumetric inspection. The developed DEMO VV design requires only 2D-formed sheets, which can be formed via the common forming processes rolling and bending, no expensive 3D-formed sheets are required. The developed fabrication concept for the inboard wall reduces significantly the amount of welds. These and other special solutions contribute to a fabrication- and examination-friendly design, which will eventually reduce cost and manufacturing risk and provide the basis for achieving precise manufacturing tolerances.

6. Concluding remarks

The demonstration of production of electricity around the middle of this century in a DEMO that demonstrates a closed tritium fuel cycle represents the primary objective of the fusion development program in Europe. The approach followed in Europe to achieve this goal is outlined in this paper together with a preliminary description of the design solutions being considered and results of the R&D programme. This includes:

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- Modest extrapolations from the ITER physics and technology basis to minimize development risks.
- An integrated design approach to understand 1) the requirements and 2) the interactions of systems in context, and develop a coherent integrated DEMO concept design.
- 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 coolants, power exhaust solution and configuration, etc.)
- Design Phase Gate Reviews to effectively assess Design Maturity/System Design Readiness
- Design integration risks and engineering and operational challenges arising from power conversion aspects and feasibility/ reliability of the BoP together with the relevant impact on the interfacing systems, safety and remote maintenance.
- Targeted technology R&D and sub-system design studies driven by requirements of the DEMO system and respond to critical design feasibility and integration risks.

This approach represents a significant change in the EU fusion community culture. There has been an increase of the involvement of industry in the design and monitoring process from the early stage to ensure that early attention is given to industrial feasibility, costs, nuclear safety and licensing aspects, and to maintain and strengthen international collaborative efforts aiming at jointly tackling the challenges in a cost-effective fashion.

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