Fusion Electricity
A roadmap to the realisation of fusion energy
Annexes
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Annex 1. Mission 1 - Plasma regimes of operation of a fusion power plant

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1.1 Short description.
In a fusion power plant, plasma regimes need to ensure a high fusion gain $Q$ ($Q=P_{\text{fus}}/P_{\text{aux}}$). The Lawson criterion sets a limit on the $n\tau_E$ product (with $n$ the density and $\tau_E$ the energy confinement time) to achieve a given fusion gain. The energy confinement time is determined by plasma turbulence that has been progressively controlled by developing of more advanced regimes of operation. Our present understanding of transport processes in plasmas shows that the $n\tau_E$ product increases with the power plant dimension.

A number of regimes of operation have been developed in tokamaks over the years (see box), which are at varying degrees of readiness with respect to application in the fusion reactor. The Type I ELMy H-mode is the most developed regime and forms the basis for standard operation in ITER. However, a few points still require a demonstration.

Tokamak plasma regimes

L-mode: In its simplest mode of operation, the energy and particle confinement in a tokamak is limited by turbulence across the entire minor radius (rather than binary collisions leading to so-called neoclassical transport). The corresponding regime of operation has, relatively speaking, low global confinement and is called the L-mode.

H-mode: In magnetic configurations with an X-point geometry limiting the region of confined plasma and above a certain input power, the turbulence dominating the edge thermal and particle transport is spontaneously suppressed, leading to an increase in the global confinement of approximately a factor of two. This high confinement mode of operation is called the H-mode. Indeed the energy and particle content of the H-mode edge is so good that the plasma parameters typically increase until a stability limit is reached. Several variations on the H-mode regime exist, distinguished by the stability mechanism that is employed to achieve steady-state or quasi steady-state (stationary averaged over limit cycles) plasma conditions.

- **Type I ELMy H-mode**: The most widely studied steady H-mode is limited by discrete edge localised modes (ELMs). This is the H-mode regime with the highest confinement and is the basis for predicting performance in ITER. The main open issue is that the pulsed power load on first wall components due to unmitigated Type I ELMs is too large for ITER and a fusion power plant.

- **Type II ELMy H-mode**: At high plasma density and with strongly shaped magnetic configurations, the small, high frequency ELMs gradually replace Type I ELMs, eventually completely replacing large ELMs. To date, this regime has only been achieved at high density and thus at high edge collisionality. The combination of high density and low collisionality required for a reactor can only be achieved at large machine size. The present strategy is to base ITER on regimes, such as the Type I ELMy H-mode, which can be achieved at both high density and low collisionality (separately) in present machines. On the other hand, experiments in...
ITER with regimes such as the Type II H-mode will provide a definitive test of their reactor-relevance.

- **Type III ELMy H-mode**: The simplest method of ELM mitigation in present machines is gas puffing to increase the resistivity of the plasma edge to the point that the resistive stability limit is below that of the ideal limit. In this circumstance small Type III ELMs appear with much lower pulsed power loads on the plasma-facing components and global confinement intermediate between that of the L-mode and the Type I ELMy H-mode.

- **QH mode**: At low plasma collisionality and with external control of the torque introduced to the plasma, it is possible to generate a harmonic oscillation in the edge of the plasma that provides sufficient energy and particle transport achieve stationary conditions below the ideal edge stability limit and thus without ELMs. This is referred to as a quiescent H-mode or QH mode. Tests of this regime in ITER with high density in combination with low collisionality and with the lower available external momentum sources will determine the feasibility of applying this regime in a fusion power plant.

**AT modes**: To achieve truly steady-state operation in a tokamak the plasma current must be driven without use of the central solenoid for inductive current drive. Limits on recycling power in a fusion power plant mean that a significant fraction of the non-inductive current drive must be self-generated. The self-generated current scales with the normalised plasma pressure gradient, whose radial profile must be to a large extent aligned with the required total current profile for the regime. Calculation and proof-of-principle experiments show that manipulation of the plasma current profile can be used to improve the core plasma confinement and thus the pressure gradient there. There are several variants of plasma current profiles presently under investigation with the family of regimes collectively known as advanced modes of tokamak operation or AT modes. While it is possible to produce such regimes with either an L-mode or an H-mode edge, it is likely that the confinement advantage of the H-mode will mean that AT modes will be variations of H-modes, with the pros and cons of the various edge stability mechanisms discussed above.

**Hybrid modes**: As stated above, it has been possible to improve confinement by manipulating the plasma current profile. While this has not yet been developed into a fully non-inductive, advanced mode of operation, the technique has been used in regimes that should allow in ITER a combination of higher performance and longer pulse length. As these regimes are intermediate, in the sense of the fraction of self-generated current, between conventional H-modes and advanced modes of operation, they are referred to as hybrid modes of operation.

Specifically, plasma regimes of operation in a fusion reactor need to integrate:

1. Burning-plasma conditions in which fusion generated alpha particles are the dominant heating mechanism;
2. High density for achieving high fusion power and reactor-relevant values of the neutron wall load (~1-2MW/m²);
3. Large radiated power from a mantle surrounding the hot plasma core (to reduce the power conducted to the divertor and achieve acceptable divertor heat load);
4. Avoidance/mitigation of off-normal events (ELM, disruptions);
5. Diagnostics and actuators compatible with the reactor environment.
Requirements 1-5 are common to all magnetic fusion concepts. Plasma regimes that can achieve these conditions for long pulses (~8h in a Fusion Power Plant (FPP), relying in part on plasma current driven inductively by the tokamak’s central solenoid) have been already demonstrated in tokamaks and could be the basis for a conservative DEMO reactor. Fully steady-state plasma regimes can also be produced in tokamaks but they require to integrate also:

6. High values of the plasma beta and a large fraction of self generated bootstrap current;

As these non-inductive regimes of operation require more advanced control of the plasma and operation closer to stability limits, Requirement 5 for reactor-compatibility of diagnostics and actuators will be more restrictive.

The stellarator configuration is not prone to disruptions and is intrinsically steady-state but it has to pass a development path that makes it not suitable for DEMO on the time scale considered here. Its development path, described under Mission 8, aims at validating the stellarator as an option to the tokamak by the time of the market penetration of fusion.

1.2 Critical aspects for reactor application.

**Burning plasma.** In present experiments, the level of plasma heating and its radial profile can be to a large extent controlled externally. Alpha particle heating, on the other hand, is determined by the plasma density and temperature. Thus, a reactor will be a self-organized system with reduced control of and by plasma heating. In addition, fusion-generated alpha particles can drive collective instabilities that can limit plasma performance.

**High-density operation.** In tokamaks, the edge plasma density cannot be increased above an empirical limit (the Greenwald density limit). The investigation of the origin of the limit and of possible ways to overcome it is an active field of research. One option being tested is the use of plasma density peaking to simultaneously respect the Greenwald density limit for the edge plasma whilst increasing the core density so as to maximise the fusion yield. If reactor relevant regimes with \( n > n_0 \) cannot be achieved, high densities could be achieved by operating at higher plasma current and correspondingly higher toroidal magnetic fields (requiring a further development of super conducting magnet technology). For the purposes of the conservative DEMO, no credit for trans-Greenwald densities is taken, with research into accessing higher density as a high priority given the large potential gains in a FPP.

**Large radiated power.** The regimes of operation in present machines typically achieve high confinement for values of the power radiated from the region inside the separatrix (core radiation) below 30% of the input power. In a reactor the radiated fraction from the core will be larger than 70%. Within the scope of Mission 1, the critical issue is how to develop regimes with a sufficiently large fraction of heating power radiated from the core without reducing the energy confinement time and avoiding thermally unstable regimes.

**Avoidance/mitigation of off-normal events.** ITER can tolerate a few disruptions and the assumed disruption rate is in line with the present JET experience. However, a reactor should be almost disruption free. Thus, tokamak plasma scenarios must include specific provisions for disruption avoidance and their early detection, control and mitigation. In addition to disruptions, unmitigated Edge Localized Modes (ELMs)
will produce unacceptably high thermal loads on the plasma facing components and must be suppressed or reduced to very low amplitude.

**Diagnostics and actuators reactor compatible.** In a FPP, diagnostics have to operate in conditions that will limit the applicability of present systems: limited view, radiation hardness, simple design for reliability. For the actuators, a selection of the H&CD systems will be necessary on the basis of their capability of fulfilling DEMO needs (source, launcher, CD efficiency).

If steady state operation are considered the following points should be also addressed.

**High-β AT regimes.** A steady-state electricity output can in principle be achieved with pulsed plasma operation provided adequate heat storage is available. Nevertheless, steady state plasma regimes may have the benefit of reduced reactor size, lower fatigue of core components and lower cost of electricity. Steady-state operation in tokamaks requires accessing regimes at high beta that are characterized by a large fraction of the plasma current generated through the bootstrap mechanism. Since the plasma is close to stability boundaries, these regimes are more prone to disruptions and require a number of sophisticated diagnostic, control and actuator mechanisms. The limitation to reactor-compatible diagnostics and actuators discussed above is thus likely to be more restrictive for advanced modes of operation.

**Compatibility between high CD fraction and divertor operation.** Even in AT regimes, part of the plasma current must be driven by external current drive methods. The efficiency of plasma current drive increases with plasma temperature and decreases with density and thus is inversely proportional to the square of the plasma density at constant pressure. Divertor operation in a reactor, however, requires high edge plasma density values. Thus, a compromise must be found and the window in plasma density for AT regimes in a FPP operation will be restricted.

**1.3 Level of readiness now and after ITER.**

The achievement of Q=10/300s target on ITER relies on the Type-I ELM My H-mode regime at moderately high plasma density (85% of the Greenwald limit), energy confinement time in line with the ITER98y2 scaling, a moderate values of beta ($\beta_N \leq 2$) and about 40% of the heating power radiated from the region inside the magnetic separatrix (if partially detached divertor operation are assumed). This regime is well qualified. Indeed, ITER demonstration discharges have been successfully performed in several tokamaks. However, the regime’s use on ITER is subject to the suppression/mitigation of Edge Localized Modes, for which solutions (pellet pacing, resonant magnetic perturbations, etc.) have been already tested in medium size devices (a test in JET-size machine is still awaiting) and the confirmation that alpha particle collective modes do not limit plasma performance in this regime, as indicated by some theoretical studies. Verification of the effectiveness of alpha heating is a prime goal of ITER.

ITER also aims at demonstrating a fully steady–state scenario at Q=5. In order to meet this goal, the steady state scenario should achieve a confinement about 50% above the ITER98y2 scaling. Such a regime has not yet been demonstrated in present tokamaks although both theory and experiment suggest that it might attainable with appropriate tailoring of the plasma current profile. An intermediate step, a ‘hybrid’ of the conventional ELM My H-mode and a fully non-inductive advanced mode of operation has been demonstrated in several machines, providing a confinement
improvement of ~20%. This hybrid regime could be prototypical to those of a pulsed tokamak reactor provided large radiated power is simultaneously achieved.

Development of fully steady-state regimes requires a similar effort to that of the 1980s and 90s leading to the qualification of the ELMy H-mode (the H-mode was discovered in 1982 and the definitive confinement scaling published in 1998). The EU strategy is based on an intensive campaign of regime identification and optimisation on medium-sized tokamaks (both in the EU and outside), possibly supplement by initial size-scaling experiments in JET, and then a large machine test in JT-60SA in the 2020s. On the same time scale as was required for the conventional regime, it is hoped to provide a qualified regime for testing in Phase 2 of ITER operation (2030s). Given the time scales and the risks associated with designing to a plasma regime that does not yet exist, a conservative DEMO, with construction starting in 2030, should be based on the conventional or hybrid regime. Should DEMO be delayed for some reason, the Phase 2 test in ITER would provide definitive input as to whether an advanced regime of operation can be incorporated in DEMO.

In summary, ITER will extend the present qualification of the plasma regime of operation to burning plasma conditions. ITER can demonstrate whether full avoidance/control/mitigation of disruptions is possible at reactor scale. A fusion reactor is presently foreseen to be a relatively limited extrapolation in linear dimension from ITER and so with a similar physics. Thus ITER should be able to complete Mission 1. In this regard, it should be noted that ITER, in order to fully address the divertor issues discussed in Mission 2, will have to go beyond the achievement of the headline missions of Q=10 (inductive) and Q=5 (steady-state). In particular, it will be necessary to investigate the compatibility between high radiation and high confinement up to the maximum possible radiated power fraction and taking advantage of the proposed upgrades to reach the maximum possible level of input power.

1.4 Main risks and risk mitigation strategies.

From the considerations presented above it is clear that the achievement of Mission 1 is strictly linked to the success of ITER. The ITER Organization (IO) has made a detailed analysis of the risks to ITER Phase 1 operation and has identified the main R&D needs to mitigate those risks. The top ITER risks form the basis for the first nine entries in the Mission 1 risk register. These risks have then been supplemented by risks that are expected to be addressed in ITER – essentially regime integration at the reactor scale and burning plasma physics – and by a few risks specific to the step from ITER to DEMO.

The risk register has been used to develop a series of high-level work packages for risk mitigation. Generic machine and subsystem requirements are given for each package, e.g. ITER/DEMO magnetic geometry, metallic plasma-facing components, machines of varying size, integration into ITER plasma regimes. These requirements can be compared to list of tokamak capabilities given in Table 1. Risk mitigation of the conventional modes of operation foreseen for ITER Phase 1 and the conservative DEMO it expected to be provided primarily by JET in this decade and then ITER in the 2020s. For advanced modes of operation, the development path is via medium-sized tokamaks in this decade, JT-60SA in the 2020s and ITER Phase 2. Besides JET, in Europe most of the relevant capabilities are available in ASDEX Upgrade, which is
expected to play an important role during Horizon 2020. In addition, international collaborations on machines such as DIII-D, KSTAR and EAST would provide valuable additional information to EU understanding.

To accomplish all of the objectives of the work packages, some machine upgrades are required. The ITER heating & current drive systems will have to be upgraded in order to provide the plasma current profile control required for testing advanced modes in Phase 2. Preparing the options for this upgrade will require some investment in the present decade. In addition, while the use of tungsten armour for all plasma-facing components has been successfully tested in ASDEX Upgrade and is the only presently known solution for DEMO, it is felt that the step from a medium-sized tokamak to DEMO is too large and that a test of an all-tungsten wall in a larger machine is required. Ideally, this test would take place in ITER and the IO is presently evaluating the feasibility of changing the armour material in ITER. If that proves not to be possible, a tungsten test might be done in JT-60SA, provided agreement can be found with our Japanese colleagues. As a third option, a full tungsten wall could be tested in JET in the early 2020s. This would have the advantage of providing an earlier test but would be with inertially-cooled components and short plasma pulses.

While the work packages are milestone-based, it is recognised that theory & modelling must play a key role in the development of the plasma regimes for a FPP and it is intended that a significant fraction of resources be allocated to this type of work. Project aimed at better physics understanding might address topics such as the L-H transition, electron transport, the plasma response to externally applied non-axisymmetric fields and the physics of ELMs.

1.5 Options for Implementing the Work Packages and/or Gaps.

Virtually all of the work packages require operation in a diverted tokamak with as close as possible to the ITER/DEMO/FPP. The main machine parameters and capabilities for several such tokamaks, in the EU and outside, are listed in Table 1.1. Metal Plasma Facing Components (PFCs) are an essential component of Work Packages 1.1, 1.7-9, 1.14, 1.18 and 1.21 (presently AUG, C-Mod and JET, in the future, EAST, KSTAR and ITER). In addition, the regime development in Work Packages 1.3 and 1.20 can only be considered complete if proven in conjunction with a metal wall. Size scaling is an important element of Work Packages 1.1, 1.3-7, 1.9, 1.13, 1.14, 1.18 and 1.20. In addition, large machines are specifically required for (elements of) Work Packages 1.4, 1.10, 1.12, 1.17 and 1.21. For these reasons, ITER, JET (up to 2020) and JT-60SA (post-Horizon 2020) have unique roles to play. Overall, integration of the various work packages into self-consistent scenarios is crucial and under-represented in the attempt to divide the crucial issues into separate items. The present EU strategy, as set out in the Facility Review, must be to focus on a few, well-equipped machines including appropriate upgrades to address this need to integrate.

For inductive modes of operation, the main remaining issue is proving the compatibility of the regime with the first wall requirements (fuel retention, transient power loads such as disruptions and ELMs and, especially for DEMO, high steady-state power loads). The JET programme consisting of a focused push to high performance in inductive regimes of operation in 2013-15 followed by a demonstration in DT in 2017 (Work Package 1.3) should provide the necessary results.
for Work Packages 1.6 & 1.7 but will not allow enough experimental time to adequately address all the other crucial issues that require input from a large machine. Agreement with international partners to jointly upgrade and exploit JET, in particular with a new ELM coil system (Work Package 1.4) or an ECRH system (Work Package 1.15, 1.16 and 1.20), would require operation until ~2020 and would provide operational experience that would aid efficient exploitation of ITER. As outlined in the Facility Review, work at JET must continue to be supported by results from medium-sized tokamaks (Work Packages 1.1, 1.3-7, 1.9, 1.13, 1.14, 1.18 and 1.20 contain elements of size scaling). The metal wall in AUG means that it has presently a unique role in Work Packages 1.1, 1.7, 1.9, 1.14 and 1.18. It is also the obvious choice for contributing to research on high-density operation (Work Package 1.13). MAST has already made a strong contribution to the study of ELM avoidance/mitigation (Work Package 1.4) using their in-vessel coil system and this is expected to continue.

The EU strategy for developing advanced, non-inductive regimes of operation (Work Package 1.20) is based on the joint exploitation with Japan of JT-60SA, leading to a test in ITER Phase 2. At the moment, even the existence of these regimes, in conjunction with reasonable edge plasma parameters and thus wall loading, is not proven. This might be compared to the search for a steady-state H-mode regime, beginning with the discovery of the regime in ASDEX in 1982 and culminating in the famous 1998 multi-machine confinement scaling. The present situation for non-inductive modes of operation is similar to that of the late 1980s for the H-mode, suggesting that we have another decade ahead of us before a consolidated regime is achieved. On this logic, preparation for and efficient use of JT-60SA requires an aggressive programme during Horizon 2020 on existing tokamaks. These tokamaks must have for strong off-axis heating capabilities and, ultimately a close-fitting conducting wall to test operation above the no-wall beta limit. In Europe, the only present machine that is a candidate for this role is AUG, depending on the implementation of planned upgrades (see Table 1.1). TCV is also expected to play a role in this regard if the proposed upgrades to the heating and current drive systems can be implemented. On the other hand, tests of scenario existence and operating range can be usefully done on machines with carbon-based PFCs so collaborative programmes on DIII-D, EAST and KSTAR would also be of great value. Indeed, parallel development on several machines, as was the case for the conventional H-mode, will provide the most reliable regime in the shortest amount of time.

Tests of regime compatibility with FPP diagnostic and actuator boundary conditions (Work Package 1.19) should be an on-going part of the EU experimental programme. Such tests on machines that are already developing integrated operating regimes and already have well developed control and diagnostic systems (in the EU, AUG and JET) would not require large additional resources. The EU should proceed on AUG and JET inside the resources described above in order to inform and validate choices for further diagnostic development.

Overall, the level of risk mitigation provided by the above work packages is considered sufficient and for Mission 1 there is no need to build a new EU facility, with the recommendation to make use rather of the present relevant tokamaks in the medium term and to build strong international collaborations to facilitate, in
particular, the development of non-inductive modes of operation in parallel to ITER Phase 1.
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<td>Others...</td>
<td>Metal coated wall &amp; divertor optional in a later phase.</td>
<td>Snowflake divertor Configurational flexibility</td>
<td>Super-X divertor, off-axis NBCD</td>
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<td>No</td>
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Black: presently available
Blue: Agreed (year of expected availability)
Red: Proposed (proposed year of availability)

Authors: R. Neu (EFDA CSU), G. Federici (EFDA CSU), C. Linsmeier (IPP), C. Lowry (EFDA CSU), M. Rieth (KIT)

2.1 Short description.

The main challenge for the realization of a fusion power plant is the heat exhaust problem. The power that crosses the magnetic separatrix is diverted along the magnetic field line to a remote region (the divertor) where it is exhausted on actively cooled divertor targets. The heat flows in a narrow radial layer of width $\lambda_q$ (~ few mm at the midplane assumed in ITER) called scrape-off layer (SOL). ITER must exhaust ~150MW of heating power (fusion alpha particle power and auxiliary heating). If 40% of the heating power is radiated inside the magnetic separatrix 90MW will flow in the SOL (of which 60MW (i.e. 2/3) towards the divertor outer target). In order to achieve an acceptable heat load on the divertor targets, the latter are inclined at a shallow angle with respect to the magnetic field line and located in a region near the separatrix X-point with significant magnetic flux expansion. In this way the wetted area of the divertor targets in ITER can be increased up to ~2m$^2$. Thus, if all the heat entering the SOL ultimately ends on the divertor target (attached divertor regime), the power load would be ~30MW/m$^2$. However, such a value is above the present technological capability of ~10-20MW/m$^2$ for steady state power load based on water-cooled copper alloys. To further reduce the heat load, part of the power flowing in the SOL has to be radiated in the divertor, leading to the so-called partially detached regime. This requires plasma temperatures in the proximity of the divertor plate below 10eV [2.1, 2.2].

<table>
<thead>
<tr>
<th>Technological limit to the divertor targets</th>
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<td>The divertor target consists of a plasma-facing part (armour) that has to withstand the interaction with the plasma power and particle loads and is subject to erosion, the engineering constraints of the heat sink (i.e. the coolant confining structure) and the coolant.</td>
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<td>Significant progress has been made during the last two decades on the development of technologies for divertor high-heat-flux components cooled either with water or with helium. In the former case, prototypes fabricated either with carbon or with tungsten armours and with a Cu-alloy heat sink, have been successfully tested under cyclic loads up to 20MW/m$^2$ for application in ITER. In the latter case, solutions have been found that can withstand 10MW/m$^2$ for a large number of cycles. It should be recognized that these values are close to the ultimate technological limits set by the intrinsic limitations of the thermo-mechanical properties of the limited number of materials suited for this application in the fusion environment. Taking into account that these properties will degrade under neutron irradiation already at the level of a few displacements per atom (dpa), and considering additional design margins that need to be included for a reliable target design (e.g., to accommodate for transients, for tile misalignments etc.), the power handling limits above must be reduced to about 10MW/m$^2$ in the case of water-cooled components and to even lower values in the case of He-cooled components.</td>
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</table>
Low temperature, detached divertor conditions also reduce the erosion of the divertor armour. The main erosion mechanism in the divertor is physical sputtering by plasma and impurity ions. These impinging particles transfer energy to the atoms of the armour materials. If the transferred energy is large enough the target atoms can overcome the surface binding energy and leave the surface. The plasma temperature in front of the target defines the energy of the impinging particles by their Maxwell distribution and by the additional acceleration the charged particles experience in the so called plasma sheath in front of the surface. (For plasma temperatures below 5 eV, physical sputtering approaches zero for tungsten as an armour material and a typical impurity composition in fusion plasmas).

Being challenging for ITER, the problem of the heat exhaust becomes a serious issue in a fusion power plant. There the fusion power will be 5 and 7 times larger than that in ITER in devices with linear dimensions about 50% larger than ITER.

We mention here another function of the divertor, namely the particle control that is to provide adequate pumping capability to exhaust the neutralised gas, most notably the He ash, as well as to retain eroded impurities such that they will not enter the main plasma. Any of the divertor solutions proposed to be studied must satisfy these requirements together with those of heat exhaust.

Although much less prone to high power and particle loads, also the plasma facing components in the main chamber wall will receive power from radiation and particles and will undergo erosion. For ITER, Be melting and excessive erosion can hamper operation whereas for DEMO the choice of the plasma facing material as well as the cooling technology depends critically on the particle spectrum and the total absorbed power. Therefore all solution envisaged for the power exhaust in the divertor must also treat the main chamber issues in a consistent way.

Solutions for the heat exhaust in DEMO/FPP are presently being explored along three lines:
- Baseline divertor solution - a combination of radiative cooling and detachment;
- Innovative magnetic divertor configurations to achieve higher flux expansion spreading the heat over a larger area or to achieve longer divertor connection lengths and larger divertor radiated power;
- Advanced plasma-facing components (e.g. liquid metals) that could exhaust higher heat loads.

2.2 Critical aspects for reactor application.

Baseline solution.
Detached divertor conditions have been obtained in several tokamaks. They require (see box) significant temperature gradients along field lines to be established in such a way that the plasma temperature close to the target decreases below 10eV allowing volume recombination of the hydrogen isotopes. The easiest way to achieve this is by decreasing the power $P_{\text{SOL}}$ crossing the magnetic separatrix, e.g. by radiating a large amount of power from the plasma core. However, H-mode operations require a minimum amount of power ($P_{\text{th}}$) to be conducted through the separatrix. There is a
considerable uncertainty in the value of $P_{\text{thr}}$ in ITER and DEMO. If $P_{\text{thr}}$ is too high, detached H-Mode operation may not be possible. This point will be ultimately clarified by ITER.

Furthermore, detached conditions have to be quickly regained also in case of slow transients. For example, in case of a change in the power flowing into the SOL, the transition to attached conditions, if not counteracted in a few seconds, would lead to a destruction of the divertor target in ITER und similarly in DEMO.

The control of detachment requires robust sensors, algorithms and actuators. Although present day machines implemented already techniques serving their requirements, solutions for ITER and DEMO are yet to be developed.

**Divertor detachment**

Detached divertor conditions are characterized by a strong pressure gradient along magnetic field lines in front of the target and a reduced ion flux to the target. This is established by volume recombination of the plasma which becomes significant below a plasma temperature below 3 eV which requires that the SOL plasma is sufficiently collisional that significant temperature gradients along field lines can be established. High collisionality can be achieved by: reducing the power flowing in the SOL; increasing the SOL density at the midplane $n_a$; and producing magnetic configurations with the largest possible connection length between the midplane and the divertor target. As a consequence of the resulting low temperatures in the divertor, recombination of the fuel ions sets in, reducing the ion flux to the divertor target and thereby further the power load. In present day tokamaks, deterioration of the confinement due to influx of neutrals from the detached region into the core plasma is sometimes observed. However, numerical modelling shows that the ITER SOL plasma is opaque to neutrals and no impact on the core confinement is expected.

The specific technology to be used for the plasma facing components is also a critical aspect. The heat sink material must at the same time have high thermal conductivity and high melting temperature, requirements that lead to the choice of copper alloys as heat sink material armoured with tungsten as plasma facing material. Precipitation hardened copper alloys such as CuCrZr will be used as heat sink for the divertor in ITER but their performance is expected to degrade under neutron irradiation already at 2-3 dpa, depending on the operating temperature which is typically the neutron damage per full power year expected for the copper heat sink material in a DEMO divertor (the expected neutron damage in the divertor of ITER is < 2 dpa for the whole ITER lifetime). Dispersion strengthened copper alloys such as Glidcop (or advanced Cu-based composite materials, see Mission 3) could be used for a water-cooled DEMO divertor since they have increased neutron resistance and could provide adequate structural integrity lifetimes comparable or longer than the unavoidable armour erosion lifetime, which will force periodic replacements of the divertor in a reactor. Additional areas of technology with promising prospects for further developments are tungsten composites (e.g. fiber-reinforced tungsten), which may overcome the problem of intrinsic brittleness of tungsten and the additional embrittlement caused by neutron irradiation, improved performance He-divertor designs, etc. The issues surrounding the materials development for Divertor materials will be covered in Mission 3.
Innovative geometries.
Two solutions are presently under investigation: the “snowflake” configuration [2.3, 2.4] and the “super-X” configuration [2.5, 2.6]. These solutions are at a proof of principle level of investigation in small and medium size tokamaks (see below). Critical aspects are the complexity of the magnetic configuration and the necessity to avoid in-vessel coils in DEMO/FPP. Preliminary design activities in the EFDA PPPT programme are ongoing to understand whether these solutions can be realistically integrated in a DEMO/FPP design, including the constraints arising from neutron shielding and remote maintenance.

Super-X Divertor:
In a super-X divertor the poloidal field coils are used to extract the boundary plasma into a separate divertor chamber. Moving the strike-point to larger R increases the geometric size of the plasma wetted area, reducing the target heat flux density. In addition a quasi-null in the divertor poloidal magnetic field is generated, which increases the parallel connection length – that is, a field line trajectory in this region is predominately toroidal so that the distance between the midplane and the target is substantially increased allowing further power loss along the field lines.

[Image]

Fig. 2.1. Planned super-X configuration in the MAST Upgrade Tokamak

Snow-Flake Divertor:
A snow flake diverted configuration is characterized by a second order null, i.e. not only the poloidal magnetic field vanishes (ordinary X-point) but also its first derivatives and the separatrix divides the poloidal plane into six sectors (giving the name to this configuration). The second-order null modifies the magnetic topology near the plasma boundary by expanding the flux 2-3 times more than in the ordinary X-point configuration. At the same time connection length in that region increases, reducing the local heat load to the divertor plates in a similar way as described above.
Liquid metals

Liquid metal-based solutions (Li, Ga, Sn) could provide a heat load capability of up to several tens of MW/m². In addition, they relieve the problem of damage to the surface. The problems here are twofold

- the compatibility with plasma operation due to the evaporation of the liquid metal surface if the operating temperature is too high and the issues related to MHD effects for plasma transients and/or for sufficiently high flow velocity.
- the fact that solution relying on evaporation probably cannot be used in a continuously operating device.

The difficulty of handling liquid metal loops in a tokamak’s vacuum environment means that any such system concept should be established as feasible with very high priority.

Liquid metals

The use of liquid metals as plasma facing materials inherently bears the advantage that they can act as plasma facing surface and as cooling medium simultaneously. Additionally, there is no neutron induced damage as in solid materials. In order not to dilute/contaminate the fusion plasma in a non-acceptable fashion, the sputtering yield must be sufficiently low, as for a solid surface. However, impurity influx into the main plasma induced by the generally higher vapour pressure compared to solids, adds to the physical sputtering. It has to be kept as low as possible by controlling its temperature. Depending on the technical solution, the cooling can be provided by the flowing liquid and/or by secondary conventional cooling. MHD forces which are induced by currents in the moving liquid itself or by plasma transients can distort the liquid surface in such a way that it no longer can provide its protective function.
2.3 Level of readiness now and after ITER.

**Baseline strategy**

Partial divertor detachment has been achieved but its behaviour cannot be described by the existing numerical codes in a predictive fashion. Operation close to detachment and close to the H-L threshold as foreseen in ITER requires still considerable efforts investigating the confinement behaviour and control issues. Ultimately, ITER should demonstrate by 2030 the applicability of its baseline strategy to the power loads relevant for a reactor. In case of success, and with further development of divertor modelling towards predictive capability, a full demonstration can then be made in DEMO.

The technology solutions for the ITER divertor targets have been qualified in accordance with the ITER divertor procurement strategy by using dedicated high-heat-flux on test facilities (e.g., electron- and/or ion-beams) and will be demonstrated in an operational environment on ITER. An early DEMO probably will use copper-based water-cooled components (that will have to be tested on high heat flux test facilities) while in parallel progressing with the R&D on W alloys (see Mission 3). If the R&D on W-alloys is successful, elements could possibly be tested on DEMO.

**Innovative geometries and liquid metals**

Some innovative geometries and liquid metals have been tested so far in short pulse, low power experiments and are at an early stage of development.

- The snowflake configuration has been produced in TCV and NSTX, showing a reduction of the heat load on the divertor plate. However, a proper test would require a modification of the divertor PFCs to have optimized divertor
geometry. Also, to show that the snowflake divertor is compatible with the particle control requirements, the divertor volume would have to be closed to show the efficient buildup of neutral pressure compared to the main chamber as well as the retention of eroded divertor armour material. The super-X configuration is going to be tested in MAST starting in 2015.

- Liquid metals have been tested in CDX-U, LTX, NSTX, T-10, T-11, HT-7, ISTTOK, FTU, TJ-II and EAST. These tests have mainly explored liquid lithium in a broad range of solutions (fast/slow flows, static pools, jet injection and capillary porous system). Experimental demonstrations of high power handling have been done so far by evaporating the Li and a closed Li circuit has not yet been demonstrated. These experiments have contributed to developing the necessary know-how on the effects of liquid metals on plasma conditions.

In order to assess the feasibility of both innovative geometries and liquid metals, further tests at high heat loads, longer pulses are needed before testing them in a device with higher plasma current (see below the gap analysis). In addition, the technical feasibility and integration of these solutions under DEMO conditions must be proven.

### 2.4 Main risks and risk mitigation strategies.

The main risks on the baseline strategy are:
- Too high H-L threshold power (P_{thr});
- Power decay length (\lambda_q) smaller than expected;
- High radiation fraction not achievable without too high dilution/contamination of plasma (Mission 1 risk)
- Lack of control of detachment.

If the stationary heat loads on ITER turn out to be too large, a possible solution would be an increase of the DEMO dimensions above what is presently foreseen. However, other limits (including cost constraints) may enter that prevent such a solution.

Risk mitigation strategies on detachment control and high radiative scenarios will involve R&D activities on existing devices as well as progress in the understanding of the L-H / H-L transition and detachment physics.

A detailed description of the risks is given in the risk register.

### 2.5 Options for Implementing the Work Packages and/or Gaps.

Most of the investigations in support of ITER and the baseline strategy of Mission 2 require operation in a diverted tokamak (see Table 1.1 in Mission 1 for main machine parameters and capabilities of such tokamaks, in the EU and outside) with as close as possible to the ITER/DEMO/FPP geometry and metal PFCs (C-Mod, AUG, JET, ITER, WP 2.1.1-5, 2.III.1-3). Some more basic questions on the compatibility of H-Mode operation with high radiative power exhaust or investigations on alternative modes of operation (e.g. no/small ELM regimes) can be performed in a broader suite of devices (Tokamaks: TCV, C-Mod, EAST, KSTAR, AUG, DIII-D, JET, JT-60SA,
ITER; WP 2.I.4-5 and 2.III.2-3 – linear devices: MAGNUM, Pilot PSI, PISCES, JULE PSI; WP 2.III.2-3). All of the above stated investigations have to be closely accompanied by adequate modelling to assess their potential for extrapolation because the baseline strategy can in principle only be fully implemented in ITER and its success will not be demonstrated before the Q ~ 10 milestone is achieved. Thus, in the absence of alternative solutions tested by 2030 at sufficiently large size and/or implementable in ITER, a failure of the baseline strategy would lead to a delay in the realization of fusion of ~20 years. For this reason, proof-of-principle experiments are being performed in order to assess the capabilities of alternative divertor geometries (Super-X: MAST-U, Snow-Flake: NSTX, TCV, WP 2.IV.1-2) and liquid metals (LDX, FTU, NSTX, KTM, WP 2.IV.3). Given the early stage of development, it is essential that these concepts will need not only to pass the proof-of-principle test but also the assessment of their technical and integration feasibility on DEMO, perhaps by adjusting the overall DEMO system design to the concept, before being explored any further.

Nevertheless, the extrapolation from proof-of-principle devices to ITER/DEMO based on divertor/edge modelling alone is considered too large. Thus, a vigorous programme has to be put in place in the next few years to secure the achievement of Mission 2. Depending on the details of the most promising chosen concept, a dedicated test on JET/JT-60SA (if at all possible, but at reduced heat load) or on a Divertor Tokamak Test (DTT) facility, entirely devoted to the divertor problem, will be necessary.

The need of a DTT facility has been for some time advocated within the fusion community [2.7]. The facility should be mainly aimed at the demonstration of innovative geometries and liquid metals at a scale that can be extrapolated directly to ITER and DEMO, but could also support the baseline strategy in case that it turns out to be successful because of its envisaged capability to provide a large P/R. However, the test of plasma facing components should be executed - whenever possible - on beam facilities that avoid the complication of tokamak operation.

Technical requirements of a DTT could be depending on the assessment of the different concepts:
- Large $P_{\text{SOL}}/R$
- Potential for large transient thermal loads
- Flexible magnetic configuration to investigate different geometry of the divertor
- Possibility of using liquid metals
- Actively cooled divertor components to allow a fast approach of equilibrium conditions

Furthermore it could benefit from:
- High $T_{\text{wall}}$ for retention studies (see Mission 1)
- Large particle fluence to investigate evolution of armour under combined power and particle load (see Mission 3)

The exact parameters of such a device cannot be fixed at this stage and are subject to a proper review, which is part of the work packages.
The evaluation of the scope and top-level requirements together with the definition of the technical boundary conditions for such a facility should be completed by the end of 2014. The analysis should consider the pros and cons of a single facility vs. a set of parallel, more targeted, experiments. Such a facility could be either a new device or an upgrade of an existing device, taking full advantage of the infrastructures already available in various laboratories. The schedule should be realistic to fit with the start of operation at the beginning of the next decade. In view of the large interest that this line of research is getting in US, RF, Kazakhstan and China, opportunities for international collaborations should be sought.

Materials and components tests should be preferably performed in dedicated test devices. The requirements will be covered in Mission 3, which is also charged with determining details of the necessary facilities. It is important to note that the testing of the PFC and high heat flux materials will involve the testing of systems of materials designed into engineering mock-ups, and hence the testing programme will have a strong interaction with the development of divertor concepts.

2.6 Description of the major Work Packages.
The Work Packages can be divided into four groups. The first group aims at demonstrating suitable schemes for controlled power exhaust in ITER and DEMO for the baseline divertor strategy. In order to extrapolate the results of these investigations, the second group of work packages concentrates on predictive tools for divertor/SOL and PWI modelling. The third group deals with specific issues of the erosion and damage of PFCs as dust production and melt layer behaviour. Finally the last area of investigation is devoted to the investigation of alternative power exhaust solutions for DEMO exploiting innovative divertor configurations and solutions with liquid PFCs.

References
[2.7] See e.g. R.J. Goldston Fusion Energy Conference 2010

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Note: This Annex summarises the main findings of the EFDA Materials Assessment Group (MAG), and independent group set up by the EFDA Power Plant Physics and Technology (PPP&T) Implementing Agreement Board at the request of the CCE-Fu. The full report of the MAG was delivered to the CCEFu and is given in document CCEFu57_7.1. The format of this Annex is, as a result, somewhat different, but the Risk Logs and Development Programmes are fully equivalent to those of the other Annexes, and are found in Annex 14 and 15.

3.1 Short description

The materials which require a degree of nuclear hardness for a Fusion Power Plant can be divided into categories by the missions they themselves fulfil. These are:

- **Structural Materials**, which provide the structures (walls, base-plates etc) for the in-vessel components, principally the Breeding Blanket (see Mission 4 and also below), the Divertor cassette structure (see Mission 2) and the Vacuum Vessel;
- **Plasma Facing materials**, which provide the ‘First Wall’ armour to take the plasma impingement on the Breeding Blanket and the Divertor;
- **High-heat flux materials**, which allow efficient conduction of heat fluxes to the in-vessel coolant channels, especially in the Divertor where the heat fluxes in a reactor will be significant (several 10s of MW.m⁻²) [note that in the Divertor region the armour has to have high-heat flux capability also];
- **Functional materials**, which are contained in the Breeding Blanket and take on the roles (See Mission 4) of neutron-multiplication and tritium breeding.

The Assessment Group considered the Material Development for the first three missions, as these are common to many Blanket and Divertor concepts. The **Functional Materials**, whilst no-less important, are intimately embedded in Blanket concepts and are therefore considered in Mission 4.

Material Damage in a Fusion Reactor

The internal components of a Fusion Reactor will be subjected to an intense bombardment of high energy neutrons (up to loadings of 2MW.m⁻²) from the plasma deuterium-tritium (D-T) fusion reaction. The neutron spectrum has a peak at the energy released in the D-T reaction (14.1 MeV):

\[ ^2D + ^3T \rightarrow ^4He (3.5MeV) + ^1n (14.1MeV) \]

Several critical phenomena are known to occur when neutrons are incident on material surfaces. The knowledge of the severity of the interactions is established in
irradiations with fission neutrons, where radiation “hardening”, radiation embrittlement, radiation swelling and radiation-induced thermal creep and high temperature helium embrittlement of grain boundaries have been measured [3.1]. These are primarily the result of displacements of atoms within the lattice following inelastic neutron-atom collisions, with the resultant formation of self-interstitial atoms and vacancies in the lattice that diffuse and aggregate to create a variety of extended defect complexes.

Metals and alloys with a ‘body-centred-cubic (bcc)’ crystal lattice structure, including iron and ferritic steels, and tungsten, show better resistance to prolonged irradiation, in terms of much lower swelling and lower embrittlement, than metals with face-centred-cubic (fcc) lattices. For example, the only class of steels that shows low swelling in the high-dose ~ 200 dpa limit, at temperatures close to 420°C, are ferritic (bcc) steels [3.2]. A comparison of the swelling under fission irradiation for ferritic steels against austenitic (fcc) steels is shown in Figure 3.1, and this comparison is usually taken as an unacceptable effect in terms of use of austenitic (eg. 316L stainless) steels in a high radiation environment. The effect of neutron collisional lattice damage is measured in ‘displacements per atom (dpa)’ and the onset of radiation hardening and accompanying change in ductility is shown for a typical ferritic-martensitic steel in Figures 3.2(a) and 3.2(b)[3.3]

Fig. 3.1: Comparative swelling after fission neutron irradiation for different steels.

![Fig. 3.1: Comparative swelling after fission neutron irradiation for different steels.](image)

Fig. 3.2: (a)(left) Ductile-Brittle-Transition-Temperature (DBTT) change for Reduced Activation Ferritic Martensitic steel (EUROFER) post- fission neutron irradiation at T ~ 300 -330°C (b)(right) Radiation hardening of EUROFER under fission irradiation at similar temperature. [Data from reference [3.3]]

The irradiation damage which causes the embrittlement and hardening is known to be ‘annealed’ if irradiation occurs at higher temperatures (above 300-350°C). This is due to the thermally-induced dissolution of small defect aggregates as the temperature (and hence energy) is increased. Post-irradiation tests, summarized in figures 3.2, on
steels after fission irradiation, show large hardening and an ~200°C increase in the
ductile to brittle transition temperature (DBTT) occurs for irradiation temperatures of
~ 300-350°C after ~20 dpa. This raises serious concerns about the viability of these
steels for fusion structures operating at or below 300-350°C without
mitigation measures, particularly since potential additional embrittlement from the
effects of neutrons with fusion-relevant energy levels, discussed in paragraphs below,
have not been included in the results.

Almost everything that is known about material degradation in intense neutron fluxes
comes from studies with fission (essentially moderate energy) neutrons. There are
however good reasons to suppose that the effects of the high-energy fusion neutrons
will be worse for a given neutron flux. One factor is that the inelastic lattice damage
collisions become at least an order of magnitude more probable at high neutron
energies (see Figure 3.3(a)). But the main factor is that primary knock-on atoms from
fusion neutron collisions have correspondingly more energy which directly leads to
more displacements. Moreover above a few MeV incident energy the neutrons
interact with the lattice atoms to produce transmutation products with accompanying
helium and hydrogen production (example in Figure 3.3(b)) in reactions of the type
A(n,α) B, and A(n, p)B. The helium atoms in particular are capable of coalescing into
microscopic helium bubbles which lead to further embrittlement of the material.

In addition to the neutron damage, the High heat flux and Plasma Facing metals
undergo bombardment by the ions and neutrals leaving the plasma. This is
particularly virulent at the divertor strike zones in a Tokamak, where the heat flux can
reach values over 20 MW.m⁻² and the plasma particle flux can reach values over 10²⁴
m⁻².s⁻¹. In these cases there will be a high level of sputtering and erosion of the
material surface. Robustness against this damage (which will eventually erode the
surface of the divertor and necessitate a replacement) is the key parameter in the
selection of divertor materials and is discussed below.

The objectives of Mission 3 may be stated as:

- To develop full scientific and engineering characterisation, and theoretical
  understanding of:
  - Neutron resistant structural materials able to withstand high levels
    of 14MeV neutron flux and maintain their structural and thermo-
mechanical properties, including in a welded condition, in a sufficiently wide window of operation and for a sufficiently long lifetime exposure (fluence) for DEMO reactor and Fusion Power Plant;

- **High heat flux and Plasma Facing Materials** able to withstand the combined effects of 14 MeV neutron flux and high intensity plasma ion/neutral bombardment and maintain their mechanical and thermomechanical properties and erosion resistance in a sufficiently wide window of operation and for a sufficiently high combined fluence for DEMO reactor and Fusion Power Plants;

- To produce and qualify suitable structural and high-heat flux materials that also exhibit reduced activation so as to avoid permanent waste repositories;
- To develop industrial production and fabrication methods for the Structural and High heat flux and Plasma facing materials, which could lead to cost savings;
- To aim, in the long run, for materials capable of extra high temperature to help achieve high thermodynamic efficiency for a Fusion Power Plant, and if possible for a DEMO machine.

### 3.2 Critical aspects for reactor application.

#### 3.2.1 Operating Ranges and engineering robustness of Materials

The internal components of a fusion reactor will experience a variety of operating conditions: temperature, temperature gradients, the local radiation dose, dose rate, and neutron energy spectrum. These parameters will have a significant impact on the materials selections and development. To define the materials requirements a *systems engineering approach* must be adopted viewing each of the major sub-assemblies which require structural, HHF and PF materials as *engineering materials systems*. Temperature distributions in the divertor and breeding blankets are going to be highly non-uniform, thus each component of these structures, must have its temperature lying within the boundaries of the temperature operating window for a given material, defined by the requirement that the material retains its structural integrity under neutron irradiation.

A temperature operating window for a material is normally defined in terms of its steady-state response to irradiation, or more generally in terms of stability of its microstructure over the timescale of operation of the material in an engineering application.¹

Important considerations in identifying and characterising the temperature operating window are:

- The identification of a time interval over which a material can operate outside its conservatively-defined temperature operating window, which has implications for the reactor design and modes of operation;
- The compatibility of (interaction between) the different materials used in a structure, as an essential requirement associated with the selection of candidate materials;

¹ In reality for a specific concept the situation is more complex and stress (primary and secondary) and strain, their cyclic ranges, number of cycles all define a highly complex multi-dimensional design space. Thus to take zero approximation, simply defined by upper and lower temperatures, has to be done with a safety margin at each end of the ‘window’.
- The effect of cyclic variation of the dose and temperature gradients on materials during reactor operation.

Thus, fusion power-generating systems impose conditions on the choice of a materials testing system, which should be able to address the question of qualification of materials in a broad range of time-dependent temperature and irradiation conditions.

The specific requirements that must be demonstrated by a material suitable for fusion reactor in-vessel components are:
- Resistance to neutron irradiation with acceptable dimensional stability with respect to void swelling, thermal creep, irradiation creep, stress relaxation, and growth;
- Acceptable strength, ductility, and toughness in the exposed environment;
- Acceptable resistance to creep rupture, fatigue cracking, creep-fatigue interactions in the exposed environment;
- Acceptable mechanical and physical properties against helium embrittlement;
- Acceptable chemical compatibility and corrosion resistance (including stress corrosion cracking and irradiation-assisted stress corrosion cracking) in the presence of coolants and process fluids and corrosion resistance in contact with fusion specific breeder materials (Li ceramics or Be);
- Acceptable integrity and low tritium retention of the subsurface of the most exposed areas of the armour within the divertor and first wall.

3.2.2 Manufacturing, Construction and Design issues Materials

The industrial development of the candidate materials for a Fusion Reactor is in its infancy. Thus, in addition to the above, manufacturing and construction issues also need to be addressed. The lessons learned from the Nuclear Fission research programme will be invaluable in this exercise.

The wide range of requirements arising need considerable development times and hence requires priority placed on ensuring that the industrial scale development is carried out in close co-operation with materials suppliers and fabricators. R&D is required to optimise, and in some cases develop, the forming and fabrication processes (heat treatments, forming of semi-finished products, powder metallurgy developments, etc.) and to assure the performance of any resultant joints and welds. The importance of the requirements for welded joints, including testing and qualification and additional requirements or optimisation that ensure reliable repair and replacement, where this is required, needs to be considered. This is a key area where Fusion developments will benefit from lessons learnt in the development of Fission reactors. Another lesson from fission is the crucial ability to design in monitoring of the safety performance of all the significant structures and components with adequately developed non-destructive monitoring and measuring techniques (robotic or otherwise) where access is restricted. It will be necessary to ensure acceptance by regulation and/or funding authorities through the development of Codes and Standards procedures for each of the materials selected for use. Here the lessons learnt from Fission (including the necessities for a Safety Case justification) will also be valuable. The experience of the licensing and code development for ITER will be a key input, discussed in section 3.7.3 below. In addition to these lessons learnt, some of the materials may require specific new methods to be developed. In all cases a set of defined "design allowable data" (with and without irradiation) for use
by the designer will be needed. Existing code design approaches that include irradiation (e.g., dependent on n-fluence) will need an extension to cover fusion operating conditions, i.e. to include He/H accumulation effects as additional variables.

### 3.3 Framework for Materials Development

The Fusion Roadmap requires a construction decision on DEMO in the early 2030s, with an accompanying tight timescale for Material development. A 2GW (thermal) DEMO concept has ~ 12 dpa/fpy neutron damage for first wall steel. For conservatism, ~ 15 dpa/fpy is taken as the benchmark for the first wall steel. The DEMO phase 1 has ‘starter components’ (Blanket and Divertor), and will have features in common with a ‘Component Test Facility’. The Materials Development programme objectives for the ‘starter’ blanket will be to withstand a 20 dpa limit for the first wall Blanket and ~ 10 dpa for the Divertor armour. The DEMO should have a second phase with second set of components, for final reactor qualification. This will involve operation of the Blanket first wall up to at least 50 dpa (steel equivalent). It will be necessary, for timely execution of the whole programme, to use the same coolant Balance of Plant for the Blankets and Divertors for the two DEMO phases.

The Materials Development, therefore, adopts the philosophy of identifying the **Baseline Materials** for the Blanket Structural steels, the First Wall and Divertor PF armour and the Divertor HHF substructure of DEMO.

For each Baseline Material a **Risk Log** is developed for the risks in using that material in DEMO (and hence being unable to qualify it for a Fusion Reactor). The Risk Log has Mitigations and a time-schedule for Mitigation (the latter set by consistency with the Roadmap). Some mitigations involve design- or operational- work-arounds, but some could be achieved by development of replacement materials with superior performance. The Mission 3 development therefore identifies such **Risk Mitigation or Back-up materials**. The Risk Mitigation (RM) materials themselves are in different current states of Technology Readiness Level (TRL). For those which have had some development, and are in the TRL3-5 range, Issues can be identified which need development to mitigate. However, rather than have ‘nested’ Risk Logs, it is more appropriate to visualise **Issues Tables** (see Annex 14) which must be addressed to develop these RM materials. The Risk Mitigation materials should be developed alongside the baseline until a review point.

The MAG recommends that at an appropriate time in the development of the DEMO design, presumably around the concept design review stage, a review of the Risk Mitigation material and corresponding Baseline materials is made and a choice to change the Baseline or continue with Baseline development be made. If the Risk Mitigation materials are not chosen for the baseline at this stage, they could still continue background development as ‘Advanced Materials’.

### 3.4 Baseline Materials for Fusion Power Reactors

**3.4.1 Breeding Blanket Structural Materials**

Regarding **Breeding Blanket Structural Materials**, the baseline material choice is the Reduced Activation Ferritic Martensitic (RAFM) Steel **EUROFER**. This was a
successful development of the EU Fusion programme and has a good overall balance of mechanical properties required (strength ductility, fracture toughness, creep resistance, fatigue resistance), and there is broad industrial experience in fabrication. It has sufficient corrosion resistance to liquid LiPb – at least under the flow conditions for a liquid metal breeder (as opposed to coolant flow) and for interface temperature <~475°C, and sufficient compatibility with He-gas cooling. Its compatibility with leading EU blanket concepts is therefore good, and it has been chosen as the structural material for the EU ITER Test Blanket Module (TBM) programme, described in Mission 4. As a Ferritic-Martensitic (FM) steel it has the relatively good n-irradiation stability of a bcc material (low swelling).

3.4.2 HHF and PFC Materials

Regarding Divertor and First Wall Materials, Tungsten is the material of choice for state-of-the-art technology in the plasma-facing components, mainly because of the high threshold energy for sputtering by hydrogen isotopes (around 100-200 eV) and the low retention of tritium in the material. The Roadmap DEMO concept will have an unshielded Divertor Strike zone power loading which will be very dependent on the parametrisation of the plasma power scrape-off length (see Mission 2 discussion), and it is likely that heat loads of the order of 10-20MW.m⁻² will have to be handled. The neutronics simulations show that the neutron damage level in the Divertor PFC tungsten will be much lower than that in the Blanket Steel (<=~ 3-5 dpa per full power year), whilst for structural tungsten the requirement may be as low as ~ 1.5 dpa per fpy.

The Structural HHF materials used in the high-heat flux components of the Divertor depend on the cooling concept. Two options exist: a high-temperature He-cooled design (using W-based materials as heat sinks and structural material) and a low-temperature water-cooled design, relying on Cu-alloys as heat sinks. The decision between concepts has to be based not only on materials parameters but on an overall systems engineering approach. It is noted however that only water-cooled concepts (for instance those on ITER) come near the ~20 MW.m⁻² power handling. Thus if the DEMO coolant has to be fixed at the first phase, and cannot be changed, this practically dictates that the DEMO Divertor has to be a water-cooled concept, with Cu-alloy HHF substructure.

3.5 Risks and Risk Mitigation for Baseline Materials

3.5.1 Breeding Blanket Structural Materials

There are several risks attached to the use of EUROFER in the starter blanket of DEMO and hence in a Fusion Power reactor. The Risk Log for EUROFER, including the risk mitigations, is attached in Annex 14 Table 3.1. The most serious risk, with very high impact on any DEMO design, relates to low temperature embrittlement. The exact low temperature limit, currently derived from fission irradiations, is uncertain because of the added effect of the helium embrittlement from 14 MeV neutrons, but robust operation of the blanket should only be guaranteed if the first wall structure is irradiated above 350°C and with less than 30 dpa/500 appm He. Characterisation of this has a high priority and will require 14 MeV neutron irradiation and a co-ordinated
extensive pre-cursor programme with isotopically-tailored melts with fission irradiation\(^2\) and high-energy ion implantation experiments

Low-temperature embrittlement, coupled with EUROFER’s decline in strength above 550°C gives a difficult, relatively narrow, temperature operating window for a Breeding Blanket and make it difficult to envisage a high-temperature coolant loop (with high thermal efficiency) with a EUROFER blanket as its ‘front end’. This makes it a difficult choice for an Advanced Fusion Power Plant, but its use with DEMO is possible with mitigation operational work-arounds in place.

The lower embrittlement temperature for EUROFER makes it problematic to use in a water-cooled blanket such as the Water Cooled Lithium Lead concept, where 290-320°C operating temperature is required. There is some indication that some melts of EUROFER have had superior embrittlement properties around 300°C. As the Roadmap for DEMO includes revisiting the feasibility of a WCLL breeding option (see Mission 4 Risk Mitigation), a development programme to push EUROFER to a lower embrittlement temperature is needed. Also, as with all 8-9% Cr FM steels, corrosion under irradiation would be an issue if water were to be used in a Breeding Blanket, so a development programme is needed here.

The issues with EUROFER, and the central importance of the Blanket Structure, make it necessary to pursue a ‘risk mitigation’ programme for the structural steels, either as complete replacements for EUROFER, or to complement the use of EUROFER at zones of high irradiation, and in high temperature applications. Two lines of approach are identified:

- Adapting developments outside fusion for “Generation IV Ferritic-Martensitic steels” with improved high temperature creep strength (up to ~ 650°C) achieved using Thermo-mechanical treatment (TMT) to improve the microstructure and density of radiation defect recombination centres;
- Pursuing the development in the Fusion programme of Oxide Dispersion Strengthened (ODS) alloys, in development for a decade but still at an early stage in terms of industrialisation, again these have a similar high temperature creep strength.

These alternatives have reached reasonable levels of development (TRL3-5), and drawback issues can be identified. The Issues Table for these steels is attached as Annex 14 Table 3.2 The TMT-modified FM steels have the drawbacks of:

- Only limited development of reduced activation variants;
- Lack of fission irradiation data and data on helium transmutation embrittlement.

They are also expected to have low-temperature embrittlement problems, but their high density of nanoscale precipitates and microstructures are predicted to give superior performance to EUROFER on low temperature and helium embrittlement, and, as ‘classical’ steels their industrial fabrication and welding development should be relatively straightforward.

The ODS programmes intend to complement RAFM steels with a structural material aiming at both higher temperature and improved neutron-irradiation resistance, and

\(^2\) Using, eg. Steels doped with Boron-10 trace, so that the fission-energy reaction \(n + ^{10}\text{B} \rightarrow ^{4}\text{He} + ^{7}\text{Li}\) take place, producing helium in the lattice (whereas He-implantations overwhelmingly produce surface effects).
with dispersed oxides intended to provide precipitate sites to ‘fix’ the helium gas bubbles generated by high-energy fusion neutron interaction and thus prevent movement to grain boundaries and enhanced embrittlement. The ODS steels currently have the following drawbacks:
- The early batches produced typically have low fracture toughness at room temperature, thus implying further basic development, with a prospect that impurity control could overcome this;
- Fabrication of components will be difficult as the current materials have anisotropic mechanical properties unless a proper thermo-mechanical treatment is followed;
- The quality of experimentally produced heats is highly variable.

Moreover the steels are only available in small (kg size), laboratory made quantities, and the process for manufacture, with much powder metallurgy and intensive thermo-mechanical treatment has to be industrialised. Other serious issues for ODS, relatively worse than the high temperature Generation IV FM steels are the lack of welding techniques available, and the poor knowledge of the non-irradiated engineering parameter database in the case of 14Cr steels. The ODS steels also have low-temperature radiation embrittlement problems, but there are experimental indications that ODS steels exhibit less radiation-induced hardening than conventional RAFM steels, as oxide dispersion is acting as effective point defects sinks. Therefore less severe low-temperature radiation embrittlement is expected.

The central issue of Blanket Structure is so important, that both lines of Risk Mitigation should be pursued as part of Mission 3.

### 3.5.2 HHF and PFC Materials

Many risks are attached to the use of tungsten armour or structural material. The Risk Log for tungsten is attached as Annex 14 Table 3.3. The highest Risks relating to tungsten as a PFC material are:
- Uncertainty in the occurrence of macroscopic enhanced erosion effects in neutron irradiated tungsten materials;
- Uncertainty in the H isotope retention behaviour following neutron irradiation.
  For both these effects there is an urgent need for plasma stream experiments on neutron-irradiated tungsten.

Also in the category of very-high impact risk is the intrinsic brittleness of tungsten and the lack of radiation embrittlement data. Moreover, if tungsten were to be used structurally, as opposed to for PFC armour, even the un-irradiated materials database has some serious shortcomings (eg. thermal fatigue data). For tungsten used as a Blanket PFC covering, there is the need to develop self-passivating tungsten alloys to guard against oxidation/deflagration in an up-to-air accident scenario.

Copper-alloys are considered as the material for the HHF heat sinks in the water-cooled Divertor design. Their most serious issues relate to the rapid loss of ductility under irradiation at temperatures < 180°C (operating temperature should be kept above 200°C). This may necessitate composite material development. Design use of copper alloys (without better composites) may have to be restricted to substructure/coolant systems in the immediate vicinity of the Divertor strike zones, where neutronics simulations show that the dpa damage is reduced by a factor ~ 3.
compared to the outer edges of the Divertor (figures ~ 3 dpa/fpy are predicted). The Risk Log for copper alloys is attached in Annex 14 Table 3.4.

At present, no developed alternative to tungsten is apparent for the Divertor. The discussion on Mission 2 notes the possibilities of using Liquid Lithium (based on some data from present generation tokamaks), but also notes that the development of a complete Lithium circulation system (not just the use of Latent heat properties of melting lithium) is required, and this puts the solution more in the line of a coolant development.

The clear ‘show-stopper’ nature of the Divertor development, and the known risks with Tungsten and copper alloys make it essential to develop a Risk Mitigation programme for HHF and PFC Armour. The Materials Development programme will initially concentrate on these promising candidates:

- Fibre and foil reinforced composites of Copper and Tungsten;
- Copper-tungsten laminates;
- Tungsten-copper functionally-graded materials.

There is also the need of development of ceramic barriers coatings for tungsten and copper to prevent the permeation of tritium from the PFC/HHF structure through to the coolant (needed in the case of water, helium or liquid metal cooled concepts). The Risk Mitigation and barrier materials have already been fabricated under laboratory conditions, but are at a very early stage of development (TRL 1-2), therefore there are no developed Issues Tables.

3.6 Materials development programmes

The Materials Development programmes for both Baseline and Risk Mitigation materials are shown in the Gantt chart (Annex 16).

3.6.1 Breeding Blanket Structural Materials

As a Blanket Steel material, EUROFER is being developed for the testing of Test Blanket Modules (TBM) in ITER. This is discussed in Mission 4, and is an essential step to reduce technical risks and uncertainties associated with the reference EU concepts for Breeding Blankets, the Helium-Cooled Lithium Lead (HCLL) and Helium-Cooled Pebble Bed (HCPB). The development programme is shown in the Gantt chart. This development is comprehensive, but it will not develop EUROFER as a suitable material for a Water Cooled Blanket, such as the WCLL, which should be investigated as Risk Mitigation for Mission 4 (see Annex 4). Thus a development along these lines is shown in the Gantt chart.3

The proposed development programmes for the Risk Mitigation Generation IV RAFM steels and ODS Steels is shown in the Gantt chart. The programme would aim for a ‘down-selection’ to one Risk Mitigation steel around 2017, and hence the non-selected parallel path at that point would then be de-emphasised and possibly stopped if progress had not reached a demonstrable level of TRL ~4. If the development was

3 There is some evidence (see e.g. Ref.[3.3]) that there is a EUROFER melt with better embrittlement properties at 300C. The Japanese programme is also developing RAFM steels along these lines.
still looking promising, for instance for very high temperature operation, it could be pursued as part of the programme to achieve acceptable cost of electricity (Mission 7).

The choice between Blanket concepts would be made at the end of Horizon 2020, and at this point the Baseline Structural Steel would be confirmed (see Mission 4).

### 3.6.2 Blanket Structural Materials status after ITER DT phase

As previously noted, the testing of Test Blanket Modules (TBM) in ITER is viewed as an essential step to reduce technical risks and uncertainties associated with the demonstration of power extraction and tritium breeding technologies essential for a reactor. This data is required for validation of scientific understanding and predictive capabilities; demonstration of the principles of tritium self-sufficiency in practical systems; development of the technology necessary to install breeding capabilities in next-step machines; and providing the first integrated experimental results on reliability, safety, environmental impact, and efficiency of fusion energy extraction systems.

The fabrication of the TBM will require the industrial qualification of a number of materials technologies (i.e., diffusion welding for the box fabrication) as well as the development and qualification of functional materials. All these activities are included in the TBM R&D programme conducted by F4E. At the end of the ITER DT phase of the TBM programme, in 2030, the development of the TBM in general and the Baseline EUROFER in particular (or its replacement) is estimated to be at TRL 6.

### 3.6.3 HHF and PFC Materials

The Development Programmes for the Tungsten, Copper Alloys and Risk Mitigation composites are shown in the Gantt chart. Also shown are two related development programmes:
- for a self-passive first wall tungsten alloy cladding should be pursued with high priority as an aid to regulatory licensing of DEMO, to avoid oxidation products release in the event of a Loss of Vacuum Accident;
- Tritium permeation barrier development (see Mission 4).

For the Risk Mitigation materials a ‘down-selection’ in the lines being pursued is foreseen in around 2017, with a choice of baseline necessary at the end of Horizon 2020. The Baseline material assembly would then be tested in the Divertor Test Tokamak under relevant plasma and heat flux conditions (see Mission 2). As part of the Development programme, it is recommended that the Fusion community should investigate the possibility of joining a larger ‘umbrella’ research proposal to investigate these materials under extreme conditions, perhaps in the format of the previous EXTREMAT programme.

### 3.7 Materials Research Facilities

Notwithstanding the previous recommendations about the use of isotopically-tailored fission irradiation data to establish helium embrittlement effects, it is clear that data from irradiation under a ‘Fusion neutron spectrum’ is essential as a precursor to final engineering decision on DEMO materials. The proposed path to construction of
a source capable of irradiating materials with a neutron spectrum approximating to that of a Fusion Reactor (for shorthand, denoted in this report as a ‘14 MeV neutron source’ is discussed below in section 3.7.3. Such a source will not be available for many years (actually after the end of Horizon 2020), so we first discuss what can be done with Fission irradiation sources.

3.7.1 Fission Irradiation Sources
As discussed above, the irradiation of materials by a Fission neutron spectrum can give the information needed about the effects of neutron-induced lattice displacement damage (so-called ‘dpa effects’). The Development Plans for steels and HHF/PFC materials all feature irradiation in Fission reactors followed by testing, and this is an essential step in all the materials’ development. For instance as part of the down-selection process for a Risk Mitigation steel, there are irradiations of small melts to 10 dpa by 2018. Large melts of the selected Risk Mitigation steel (which may become the baseline continue through the end of Horizon 2020, and out to 2024. For steels, these irradiations will consist of studies with isotopically-tailored melts (to give information on Helium-embrittlement effects in the steel lattice), and Helium-ion beam implantation in the steel melts. The latter give some information, but have to be treated with caution, as they are limited to the surface (few hundreds of microns depth) of the metal.

The facilities currently available for Fission are listed in Table 3.1 attached. Adequate facilities exist if the EU is able to utilise reactor payload time in other ITER Parties’ devices. The facilities of Table 3.1 will be supplemented soon by the Jules Horowitz reactor (France), capable of 10-12 dpa/full-power year. Although this device is intended “for the needs of the EU Fission Industry”, its use for fusion in the later years of Horizon 2020 is essential. A similar remark can be made about the PALLAS successor to the Dutch HFR at Petten. The possibility of the use of Accelerator Driven Systems (such as MYRRHA) for fusion application will have to be assessed in due time.

3.7.2 High Heat Flux and Plasma Stream Devices for non-irradiated testing
A programme of testing for HHF/PFC Materials (see Gantt chart) would involve first tests of divertor/first wall concepts/prototypes in un-irradiated state in one of the several high heat flux facilities in Europe (FE 200, JUDITH 1,2,3, GLADIS, HELOKA, KATHELO). Specific questions concerning the combination of plasma and power-loads (steady state and transients) can be investigated in high flux linear devices (Pilot PSI, Magnum). Some specific tests in plasma gun facilities (e.g. 2-MK200 in Russia) / QSPAs (Russia, Ukraine) for investigation of the material behaviour under intense heat loads during short (~few ms) transients could be necessary.

3.7.3 Plasma Stream facility for Irradiated Material testing
As the highest Risks for tungsten as a PFC material are related to the uncertainty in the plasma erosion effect and tritium retention of neutron irradiated tungsten materials (effects of neutron-induced defects), a test of the PFC materials (Baseline and Risk Mitigation) by a plasma stream after irradiation has a very high priority early in Horizon 2020. There is currently no plasma facility in Europe, or indeed worldwide, capable of handling irradiated materials. The EU Fusion programme should complete,
early in Horizon 2020, the development and commissioning of a plasma-stream experimental facility capable of handle activated samples (e.g., JULE-PSI at Juelich).

3.7.4 14 MeV Neutron source
Irradiation under a ‘Fusion neutron spectrum’ is essential as a precursor to final engineering decision on DEMO materials, in order to get the realistic data on Helium – embrittlement following high-energy neutron-induced transmutation. The extent of this data, provided it is available in a timely fashion, can fall short of the full ‘qualification’ of materials. This is enabled by the choice of the, essentially-unirradiated Vacuum Vessel as the primary boundary, and the engineered ‘defence in depth’ provisions (see Mission 5).

The milestone for provision of this data to match the DEMO construction decision in the early 2030s would be to achieve at least 30dpa damage in the Baseline steel samples by 2026. It is to be noted that the Codes and Standards exercise for these materials has to be started in Horizon 2020, and is a gradual process, ‘completed’ by the 14 MeV analysed input. The Codes and Standards milestone for DEMO should be 2028.

To fulfil the 14-MeV materials irradiation requirements for a Fusion Power Plant require a dedicated source. The most appropriate source is based on a so-called ‘light-ion stripping’ accelerator neutron source. The candidates are based on deuteron beams in the 25-40 MeV range incident on a light ion (Lithium or carbon) target. The neutrons produced in such a source, have a peak around 14 MeV, and their spectrum closely mimics that of a Fusion Power reactor. There has been a long-standing project to implement such a source, currently being taken forward under the EU-Japanese Broader Approach programme. This is the International Fusion Materials Irradiation Facility (IFMIF), based on twin 40 MeV, 125mA deuteron beams on a liquid Lithium target [3.4, 3.5].

As Fusion materials development will continue in the long-term (post-2030) the long-term aim for development and qualification is fulfilled by IFMIF, due to the high dose rate (20 dpa/full-power-year) and sufficient irradiation volume (500cc) available. The schedule of IFMIF [3.5] is too late, however, for a decision on Fusion Materials for DEMO in 2028. Thus, for Risk Mitigation, a minimum of 14 MeV neutron data should be provided to the DEMO programme by 2026 and this requires an accelerator 14 MeV neutron source capable of irradiating a sufficient mission volume of critical materials to a level of 30dpa to establish critical properties. An approximate view on this mission volume is in the region of 70cc, but the Fusion community will adjudicate on this volume early in the 2013 programme.

An early start on a more basic 14MeV accelerator is envisioned. This has been christened ‘pilot-IFMIF’. It is intended that this basic accelerator will grow out of the Broader Approach ‘IFMIF EVEDA’ programme, and also that it could be developed into the full-IFMIF at an appropriate stage (i.e. during the DEMO construction in the early-2030s).

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4 EVEDA – Engineering Validation and Engineering Design Activities.
The EU Fusion programme will set up an urgent examination of alternatives for pilot-IFMIF in the 2013 activities.

Typical of the alternatives is the ‘Early Neutron Source’ (ENS) proposal for a reduced (26MeV, 125 mA) version of IFMIF on a smaller liquid Lithium target. A potential schedule for this device (assuming funding is available in the last years of Horizon 2020) is shown in the Gantt chart. This is capable of 12 dpa/fpy damage fluence for steels and could provide the required 30 dpa damage by 2026.

References

Ref: Report of the Consultants" Meeting on Role of Research Reactors in Material Research for Nuclear Fusion Technology, IAEA Vienna
<table>
<thead>
<tr>
<th>Facility Name, country</th>
<th>Fast neutron &gt; 0.1 MEV $[10^{18} / \text{m}^2 \text{s}]$</th>
<th>[dpa/fpy] in Fe</th>
<th>Volume $[\text{cm}^3]$</th>
<th>Temp $[^\circ\text{C}]$</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>BR-2, B</td>
<td>1.5-3 &lt;1.0</td>
<td>&lt;3 &lt;1</td>
<td>90 250</td>
<td>50-1000 50-1000</td>
<td>Different rigs and capsules. Most appr. ~300°C RAFM steels. Some in-situ (slow tensile, fatigue)</td>
</tr>
<tr>
<td>HFR, NL</td>
<td>2.5</td>
<td>&lt;7 &lt;1</td>
<td>81500</td>
<td>80-1100</td>
<td>In-situ rigs</td>
</tr>
<tr>
<td>HFIR, US</td>
<td>1.1 5.3</td>
<td>&lt;18 &lt;7</td>
<td>100 700</td>
<td>300-1500</td>
<td>Accelerated testing in small volume</td>
</tr>
<tr>
<td>BOR-60, Russia</td>
<td>~30 total</td>
<td>~20</td>
<td>350</td>
<td>300[3.6]-700</td>
<td>Only passive instrumentation (and not for complete campaigns - rigs have T-control for one period - then moved to “equivalent” positions Claim 300°C, but this is actually the inlet-T and 335/340 is more realistic.</td>
</tr>
<tr>
<td>BN-600, Russia [3.6]</td>
<td>~ 65 total</td>
<td>&gt;20</td>
<td>350</td>
<td>375-700</td>
<td>Only passive instrumentation</td>
</tr>
<tr>
<td>JOYO, Jp</td>
<td>&gt; 50 total</td>
<td>~30</td>
<td>?</td>
<td>300-700</td>
<td>The only fast breeder with in-situ test (swelling, creep), reasonable T-control (+-4K!).</td>
</tr>
<tr>
<td>FBTR-Kalpakkam, In</td>
<td>1-10 in 15 campaigns</td>
<td>No info on total volume</td>
<td>Pictures indicate capsules of ~100.</td>
<td></td>
<td>Based on some publication and on presentations. So far the reactor has been mainly used for fission testing (fuel elements) Non-instrumented capsules for steel and nickel testing under fabrication.</td>
</tr>
</tbody>
</table>

Table 3.1 List of potential Test Reactors for fusion material testing

Authors: G. Federici (EFDA CSU)
Acknowledged contributions from: C. Bachmann (EFDA CSU), J. Harman (EFDA CSU), C. Morlock (EFDA CSU), Y. Poitevin (F4E), I. Ricapito (F4E), A. Li-Puma (CEA), P. Sardain (CEA), C. Day (KIT), P. Norajitra (KIT), L. Boccaccini (KIT).

4.1 Short description

Fusion power is captured and produced in an integrated first-wall/blanket system that surrounds the plasma. This system must operate at a high temperature to efficiently convert fusion power into electricity. Furthermore, tritium fuel has to be bred by capturing fusion neutrons in lithium-bearing materials\(^5\). Additional systems associated with the power extraction and tritium breeding blanket include additional shielding of various components (e.g., superconducting magnets), heat transport loops, coolant chemistry control, heat exchangers, and systems to recover and process bred tritium from the blanket and tritium in the plasma exhaust. Figure 4.1 (from [4.1]) shows the fuel cycle and power extraction streams and functions.

![Fig. 4.1](image-url)  
**Fig. 4.1** Primary fusion fuel handling systems and interfaces: (red) tritium breeding and neutron energy capture in the blanket, (green) extraction of fusion energy and bred tritium, (blue) purification and recycle plasma exhaust (Source [4.1])

The main functions of the blanket/fuel cycle system can be summarised as follows:

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\(^5\) The fuel for a deuterium tritium (DT) fusion reactor consists of a naturally occurring isotope of hydrogen (deuterium) and another isotope that must be bred using neutrons produced by the plasma in lithium-bearing blankets.
• **Blanket tritium breeding and heat production**: utilize and manage fusion neutrons by breeding tritium, converting neutron energy to heat. This region is exposed to high neutron fluence, especially in the first ~20 cm closest to the plasma.

• **Blanket tritium and heat extraction**: generate high-grade heat suitable for conversion to electricity through a heat exchanger and turbine cycle (see Mission 6 BoP). Extract tritium from the breeding blanket and send it to the purification and recycle loop.

• **Tritium recovery and recycle**: recover unburned tritium from the reactor exhaust and tritium extracted from the breeding blanket. This region must handle large tritium inventories and must prevent tritium release to the environment. Accountancy of tritium in the fuel cycle for both safety and proliferation, given that kilogram quantities of tritium are circulating daily through the system, is another aspect of this challenge.

• **Shielding**: the in-vessel blanket systems shield vacuum vessel, magnets and other equipment outside the reactor from nuclear radiation.

At the moment civilian tritium mostly comes from CANDU fission reactors. Today, there are 20 operating Canadian (from Ontario Power Generation) and 4 Korean CANDU reactors. Tritium is recovered from these reactors at the Tritium Recovery Facility (Darlington) and the WTRF (Wolsong). Presently, 20-25 kg of tritium are available. It is expected that the ITER experiment will make use of 20 kg of tritium. For DEMO it is expected that an initial inventory of the order of 5-10 kg will be needed\(^6\). Assuming that: (i) the Canadian tritium recovery rate is ~1.5 kg/yr and Korea adds about 20% to this; (ii) the tritium recovery rate will remain at this level until 2025; and (iii) that thereafter it will decrease rapidly (reactors reach end-of-life), it is foreseen that about 15 kg should be available in 2040 (without including ITER operations) \([4.2]\). This should leave enough tritium for the initial phase of DEMO operation but tritium self-sufficiency and minimisation of inventory build-up will need to be demonstrated as first priority objectives.

### Main Breeding Blanket Design Concepts \([4.3]\)

A variety of breeding blanket design concepts has been considered, ranging from more conservative concepts to higher-risk higher-payoff concepts for future reactors. The major candidate breeding materials consist of lithium ceramic breeders and liquid metal breeders. The degree of conservatism in the concept is often linked with the choice of structural material since more advanced concepts generally require operation at high temperature to provide for high cycle efficiency and power production performance and, thus, a greater degree of extrapolation in structural material properties and technology. The choice of structural material in turn influences the choice of breeding material and coolant based on accommodation of key issues such as material compatibility, material properties degradation and temperature limits.

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\(^6\) Tritium start-up inventories depend strongly on tritium burnup fraction \((\dot{f}_b)\), tritium fueling efficiency \((\eta_f)\), and tritium processing time \((t_p)\)
The main elements of the blanket are the breeder, the neutron multiplier, the coolant and the structural material. The most common classification can be made on the basis of the physical nature of the breeder (see below).

The performance and attractiveness of a blanket concept are dependent on a number of parameters, (e.g., power production for a given plant size, safety, availability, design and fabrication, tritium, economics) and are also coupled with the design and requirements of other reactor systems. For example, any blanket must play a significant role in shielding the magnets and breeding performance competes with shielding performance in space restricted regions such as the mid-section of the inboard region. Blankets with higher specific shielding capability and with a higher local breeding ratio would provide a substantial advantage in this case. The blanket must also be compatible with other systems. For example, it is often desirable to have the same coolant for both blanket and divertor, which places additional demands on the coolant heat transfer performance.

4.2 Critical aspects for reactor application.

4.2.1 Blanket tritium breeding and power extraction

The major material and design issues generally associated with the different breeding blanket concepts can be summarized as follows [4.3]:

a) Solid breeders blankets

This class of concepts includes a combination of a ceramic breeder, a beryllium-based multiplier (e.g., Be or Be-alloys). Near term concepts use a ferritic/martensitic steel as structural material. More advanced concepts using higher temperature structural materials to increase the coolant temperature (in particular in combination with He coolant) have been proposed [4.4, 4.5, 4.6]. The ceramic breeder and Be multiplier can be in the form of sintered blocks or pebble beds. Currently, the main candidate breeder materials are (in the order of decreasing lithium density): lithium orthosilicate (Li$_4$SiO$_4$) and lithium metatitanate (Li$_2$TiO$_3$). Two different coolants have been considered: water [4.7] and helium [4.8].

In general solid breeders (with Be multiplier) have the advantages:

- To require smaller radial thickness and Li$_6$ enrichment (40% in case of Li$_4$SiO$_4$) to achieve a sufficient tritium breeding ratio (TBR~1.14). This is due to the favourable neutron multiplication characteristic of Be (see [4.9]).
- To rely on a simple and efficient mechanism of tritium extraction from the ceramic pebbles with a low pressure He purge flow;
- To better control of the parasitic permeation of tritium to the coolant that is a safety issue and possibility to avoid of permeation barriers.
- To reach higher energy multiplication in the breeder zone (e.g., 1.33 [4.9]) that would help increase the total plant efficiency.

The critical aspects are listed below

- Chemical compatibility between the Be multiplier and water/air if water is used as a coolant or in case of air/water ingress in an accident scenario. Hydrogen production due to the Be-water reaction is a key safety issue.
• Tritium production, release and trapping characteristics of the breeding material and Be multiplier. Tritium permeation to the coolant is also an important issue.

• Thermo-mechanical interactions between the pebbles and the structure including neutron irradiation effects. Thermal and mechanical property degradation will affect temperature control and thermo-mechanical performance. This sets limits on the allowable power density due to the relatively low thermal conductivity of the ceramic breeder and on the blanket lifetime due to irradiation damages in the ceramic breeder and beryllium.

• Li burn-up in the ceramic.

• Cost of fabrication and necessary re-processing of the ceramic breeder and beryllium multiplier. For tritium breeding reasons, the lithium contained in this material must be enriched to 30–60% Li-6 (above the natural level of 7.5%).

• Compatibility of Be with structural material.

• Availability of Be material to be used in future reactors (hundreds of tons)

In light of the above, currently the most promising ceramic blanket concept in Europe is the helium cooled pebble bed (HCPB) [4.4], which is one of concept that Europe is planning to test as part of the ITER blanket test programme. A water-cooled concept [4.7] is instead proposed by Japan.

b) Liquid breeders

The eutectic lead-lithium alloy LiPb is one of the most attractive breeder/multiplier materials due to its good tritium breeding capability, its relatively large thermal conductivity, and its relative immunity to irradiation damage. Nevertheless there are issues of Li burn-up, of transmutation and of activation (direct or due to corrosion products) that require methods for chemical control/purification. It can lead to tritium self-sufficiency without employing additional neutron multipliers and allows for tritium extraction outside the vacuum vessel. LiPb has also the advantage of being almost inert in air and of having only a relatively mild and controlled reaction with water. In addition, LiPb can also be used as a coolant in advanced concept (see below). The following main types of blanket exist:

Self-cooled Blanket: This concept relies on a liquid metal that acts as a coolant and as a breeder and to avoid unacceptably high magneto-hydrodynamic (MHD) pressure drops, one needs to electrically insulate the flowing breeder from the conducting walls.

He-cooled LiPb blanket (HCLL): This concept relies on a liquid metal LiPb that acts as a breeder and He as a coolant. This blanket will be tested in ITER in the form of test blanket modules (TBMs) [4.10].

Water-cooled LiPb blanket (WCLL): This concept relies on a liquid metal LiPb that acts as a breeder and water as a coolant. The main issues are the control of the LiPb water interaction in case of an accidental guillotine rupture of a cooling tube, the control/ minimisation of the tritium permeation from LiPb to water and the risk of embrittlement of the selected steel as structural material resulting from operation at temperature lower than 350°C. The first two issues can be alleviated with appropriate counter-measures such as dimensioning the LiPb container to the
water-pressure, using double-wall tubes as coolant pipes (increasing the blanket reliability and availability at the same time), and applying tritium permeation barriers on the cooling tubes. Some of these issues were partially addressed in an R&D program performed in the EU several years ago [4.11, 4.12].

**Dual-coolant LiPb blanket concept (DCLL)** – This concept relies on a LiPb breeder/coolant that is flowing sufficiently fast to remove both the bred tritium and the majority of heat from the reactor. A second helium coolant is used to cool the structures especially the plasma exposed front part (i.e, the first wall). In this case, the MHD pressure drops in flowing LiPb are minimised by using SiC inserts that do not have structural functions. Degradation of thermal conductivity of SiC-composites by neutron irradiation should not be a problem since this material serves here only as a thermal insulator. However, helium transmutation in SiC is very large and resulting effects must be better understood. The main issue is the limitation due to the maximum allowable first wall temperature and from the compatibility of the structural material with LiPb, limiting the allowable interface temperature to about 550°C. Use of oxide-dispersion-strengthened (ODS) steels with their higher strength-based temperature limit would increase operation capabilities but welding requirements would make the fabrication more difficult. [4.13-15]. These issues are covered in Mission 3

Main common issues of the liquid metal blanket concepts mentioned above are:

- Corrosion of the pipes and blanket structures by circulating LiPb,
- Efficient extraction and purification of tritium from LiPb at high temperature
- Control of tritium leakage and minimisation of permeation to coolants
- Polonium and other transmutation products control in irradiated LiPb.
- Cost of Li\textsubscript{6} enrichment. LiPb rely up to 90% of Li\textsubscript{6} enrichment to minimise the radial thickness of the breeder zone.

**4.2.2 Fuel cycle: tritium control and handling**

Tritium is among the most mobile of elements and can readily permeate through metallic structures, especially those at elevated temperatures with large surface areas, such as those in the heat transport system. On the other hand, the expected legal tritium release limits for fusion are extremely low. Thus, accurate accountancy of tritium in a fusion power plant is a challenge, especially the tritium tracking between inner and outer fuel cycle.

Controlling tritium permeation through adequate permeation barriers (by additional coating layers and/or chemical conditioning of the surfaces) is essential to minimize accumulation of tritium in certain areas of a fusion power system. Such barriers may also mitigate corrosion or provide electrical insulation. However, long lifetimes of these barriers under environments combining irradiation, high temperatures, thermal cycling, and corrosion must be demonstrated.

**4.3 Level of readiness now and after ITER.**

The HCLL and HCPB blankets are the European reference concepts for breeding blanket in DEMO and, as a consequence, they will be tested in ITER in form of TBMs [4.16]
4.3.1 Blanket tritium breeding and power extraction

Testing of Test Blanket Modules (TBM) in ITER is viewed as an essential step to reduce technical risks and uncertainties associated with the demonstration of power extraction and tritium breeding technologies essential for a reactor\(^7\). This data is required for validation of scientific understanding and predictive capabilities; demonstration of the principles of tritium self-sufficiency in practical systems; development of the technology necessary to install breeding capabilities in next-step machines; and providing the first integrated experimental results on reliability, safety, environmental impact, and efficiency of fusion energy extraction systems. Most of the open issues should be addressed as part of the TBM programme, but risks and gaps will remain after ITER (see below).

The fabrication of the TBM will require the industrial qualification of a number of technologies (i.e., diffusion welding for the box fabrication) as well as the development and qualification of functional materials. All these activities are included in the TBM R&D programme conducted by F4E.

4.3.2 Other aspects of the Fuel Cycle

In addition to the TBM Programme, operation of ITER itself will yield invaluable information on large-scale DT processing and tritium accountancy in a working fusion environment. Data will be collected on fuel cycle systems in the following areas: (i) operation at higher flow rates and shorter recycle times; (ii) regulatory compliance; (iii) large tritium inventory; (iv) processing technology demonstration; (v) RAMI; (vi) tritium safety data; and (vi) integrated operations in fusion environment.

Nevertheless, the current design of the tritium extraction/purification systems for HCLL/HCPB TBMs is driven by reliability/availability requirements and by the need to assure operation flexibility in order to support the TBM experimental program. On the contrary, the capability of these systems to achieve demanding performance objectives, mainly in terms of maximization of tritium extraction efficiency (around 80%) and minimization of tritium inventory, is not considered as an aspect of primary importance.

Consequently, when referring to DEMO, not only the design but also the technology selection criteria for the tritium extraction/purification systems must be fully revised in the light of the DEMO fuel cycle characteristics and requirements. In particular, processing rates will need to be increased by more than an order of magnitude. The time to produce on-spec products will need to be reduced by about an order of magnitude. Duty cycle requirements will require not only increasing the reliability of systems, but also converting them from manually operated experimental systems, to

\(^7\) The Test Blanket Module (TBM) programme on ITER has the following goals for each of the concepts tested:\(^2\): (i) demonstrate T breeding performance and verify on-line T recovery and control systems; (ii) demonstrate high-grade heat extraction suitable for electricity generation; (iii) validate and calibrate the design tools and the database used in the blanket design process including neutronics, electromagnetic, heat transfer and hydraulics; (iv) demonstrate the integral performance of blanket systems under different loading conditions; (v) observe possible irradiation effects on the performance of the blanket modules; (vi) confirm maintenance approach and equipment; (vii) obtain some very limited information on the reliability of test blanket modules and blanket systems.
automated production systems. Meeting these future needs will require both development of new technologies and extensions and refinements of existing technologies. It should also be noted that the ITER tritium systems will largely be a production system with little opportunity for experimentation outside what is needed for operations.

The technology readiness level (TRL) for the various areas is summarised in Table 4.1 below. In the application of TRL’s the relevance of the environment and importance of scale suggest that large extrapolations will remain even after successful operation of ITER. Supportive R&D activities must be planned to achieve the necessary fuel system maturity (see below).
<table>
<thead>
<tr>
<th>Technology Area</th>
<th>Readiness now</th>
<th>Readiness after ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Blanket tritium breeding and power extraction</strong></td>
<td>TRL 2-3</td>
<td>- Experimental results available on tritium breeding in capsules/specimens irradiated in fission reactors.</td>
</tr>
<tr>
<td></td>
<td>TRL 6</td>
<td>- TBM programme will provide all the know-how for the fabrication of the blanket box, welding, etc. and the functional material qualification for the HCLL and HCPB concepts</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- TBM will provide the first integrated experimental results on reliability, safety, fusion energy extraction systems.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Efficient removal and processing of tritium not fully addressed by ITER</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Materials operating conditions similar to DEMO blankets, but very low n-fluence</td>
</tr>
<tr>
<td><strong>Blanket breeding control</strong></td>
<td>TRL 3-4</td>
<td>- Good understanding and basic nuclear cross-section data is available. Sophisticated 3D numerical models of neutron transport have been developed, and some subscale integral benchmarking data exist.</td>
</tr>
<tr>
<td></td>
<td>TRL 6</td>
<td>- ITER will provide a great opportunity to further refine the models and provide direct data for tritium breeding.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Design optimisation may be required for blankets that provide marginal tritium breeding (such as some ceramic breeder blankets). The technologies for active control still require formulation, prototyping and demonstration</td>
</tr>
<tr>
<td><strong>Tritium recovery</strong></td>
<td>TRL 3</td>
<td>- For most breeding blanket concepts, techniques for extracting tritium have been identified, and for some concepts, proofs of principle chemistry tests have been carried out (See for example for liquid breeders Ref. [4.17] and for ceramic breeders [4.18] and references therein).</td>
</tr>
<tr>
<td></td>
<td>TRL 4-5</td>
<td>- Only limited scale tests and partial validation expected in ITER. However, because of different scale and requirements, the adopted technologies, especially for tritium extraction from blanket and accurate accountancy might not be DEMO relevant.</td>
</tr>
<tr>
<td><strong>Fuel Cycle and tritium accountancy</strong></td>
<td>TRL 4</td>
<td>- Experience available from Tritium Labs in EU (e.g., the Tritium Laboratory of Karlsruhe) and in the US (e.g., Tritium Systems Test Assembly (TSTA) that began operations in 80’s and contained all of the systems required to process DT fusion fuel.</td>
</tr>
<tr>
<td></td>
<td>TRL 6</td>
<td>- The level of control, reliability, and throughput will be higher in a power plant. Overall, tritium processing has advanced to TRL level 4, and should approach level 6 following successful operation of ITER and, in parallel, conducting R&amp;D on accurate dynamic tritium accountancy methods.</td>
</tr>
</tbody>
</table>
4.4 Main risks and risk mitigation strategies.

The risk of proceeding to a device requiring tritium breeding, without the testing experience and knowledge gained from ITER TBM Programme is judged to be unacceptably high. The ITER TBM programme should address most of the issues for the blanket concepts to be tested that for Europe are He-cooled pebble bed (with ceramic breeder and Be multiplier), and the He-cooled LiPb [4.16, 4.19]. Nevertheless, there remain some technical risks and uncertainties. Some of the major risks are discussed in the Annexed Risk Register.

In particular, the choice of the coolant for a demonstration fusion power plant is still open and this is expected to have a substantial impact on the design, operation, maintenance, reliability and safety. It is generally agreed that water should be considered as the divertor coolant for a DEMO starting construction in the early 2030s as the divertor surface heat flux conditions prove to be beyond helium power handling capabilities (specific R&D in this regard is proposed in Mission 2). However, the choice between helium and pressurised water as the breeding blanket coolants is still open. Technical issues influencing the choice include (i) thermal power conversion efficiency; (ii) pumping power requirements; (iii) power handling requirements of the first-wall; (iv) inner blanket thickness (n-shielding and streaming / void space with helium); (v) achievable tritium breeding ratio; (vi) breeder tritium extraction; (vii) tritium permeation and primary coolant tritium purification and control; (viii) chemical reactivity, coolant leakage; (ix) design integration and feasibility of BoP. A great deal of attention must be devoted in Horizon 2020 to consistently analyse and resolve the technical issues of a breeding blanket and balance of plant with helium and with water as primary coolant in order to make an informed selection by 2020.

If, as a result of the analysis done during Horizon 2020, a different breeding blanket module needs to be tested in the ITER TBM programme, this will have to be done during ITER phase II with a delay of the DEMO blanket development programme. Possible risk mitigation may arise from some sharing of information on the TBM programme among the ITER parties and from participation in the operation of a nuclear component test facility if built by an ITER party. For example, China is designing a Chinese Fusion Engineering Test Reactor and this facility should start tritium operation around 2030. Options for potential participation in the exploitation of such a facility, for example, by testing alternative blanket concepts should be seriously pursued.

4.5. Description of the major Work Packages.

The TBM programme remains undoubtedly an essential step of the DEMO blanket development programme and should address most of the open issues. Nevertheless, a sustained complementary programme of technology R&D is required to reduce the risks to DEMO blanket development that cannot be fully explored in ITER, and/or to develop adequate knowledge to evaluate alternatives to the mainline concepts (see below).

R&D and design activities are mainly required in the following three areas:
a) R&D/ Design on HCPB/HCLL blanket concepts complementary to the TBM Programme
b) R&D/ Design on a water-cooled DEMO back-up blanket concept (WCLL)
c) R&D/ Design on an advanced blanket concept (DCLL).

Details on the work packages are provided in the Annex. Existing facilities that could be used for specific breeding blanket/ first wall and coolant studies are briefly described below together with areas of possible gaps.

Several liquid metal loops are available to investigate
i) *materials compatibility characterisation* e.g., PICOLO (FZK) as specific LiPb corrosion loop, LIFUS III (ENEA) oriented to ascertain the corrosion / erosion rates in IFMIF representative conditions, Latvian loop for EUROFER corrosion experiments in LiPb flow. LIFUS-5 (ENEA) and RELA III (ENEA) are instead facilities to study effects of large and small water leak in LiPb, respectively. VIVALDI (ENEA) and CORELLI II (ENEA) are facilities to investigate the efficiency of coating materials in the presence of LiPb alloy, also under thermal-mechanical fatigue. MELODIE (CEA) is a facility to perform hydrodynamic corrosion test of welded joints in LiPb under WCLL representative conditions.

ii) *Technology*: e.g., TRIEX (ENEA) aimed at studying T extraction from LiPb.

iii) *development of components and thermal-hydraulics*: e.g., MEKKA (KIT) to investigate liquid metal flows in the presence of a strong magnetic field, DIADEMO (CEA) dedicated to the thermo-mechanical qualification of HCLL TBM mock-ups; EBBTF (ENEA) aimed at characterising behaviour of solid and liquid metals breeding blankets in representative geometrical configurations.

Specific He-loops oriented to specific TBM applications are: HeFUS 3 (ENEA) dedicated to the thermo-mechanical qualification of HCPB TBM mock-ups (scale 1:4); HELOKA (FZK) dedicated to testing full-scale TBM and advanced divertor modules.

Suitable high-heat flux tests stands are: FE 200 high heat flux test facility (AREVA NP and CEA), GLADIS IPP, HELOKA & KATHELO KIT (in 2013), JUDITH 1, 2, 3 FZJ (the only test facilities where radio-active samples/mock-ups could be handled).

Suitable fission reactors (for test on functional materials (see Mission 3 for tests on structural and PFC materials) are: HFR, Petten, HFIR, Oak Ridge; Jules Horwitz Reactor, Cadarache (to be setup); Bor-60, Russia (fixed temperature only); BR-2, Mol (low doses only), BRR (KFKI Budapest), LVR-15 (NRI Rez, Czech Rep.), OSIRIS (CEA France).

Areas of potential gaps

- **Blanket test facility (thermomechanical, thermo-fluid tests)**
  In order to study the effect of a more integrated environment, both in terms of the loading conditions and blanket functions and materials, a multi-effect blanket thermomechanical, thermo-fluid MHD test facility is envisioned where blanket/first wall components can be tested under combined loading conditions for long periods of time. A similar facility has been recently proposed in the US [4.1]. The key aspects of such tests will be to: (i) acquire precise measurements of
thermomechanical and thermofluid performance of mock-ups for comparison to and validation of simulation capabilities, and (ii) gain failure modes, frequencies and effects data for representative blanket systems with prototypic materials, temperatures, and under simulated fusion loading conditions. Nuclear heating simulation could be in this case achieved via embedded heaters, surface heaters, or induction heaters. Measurements of mock-up and loop temperature, strain, and coolant flow would be used for quantitative comparison against simulations. Compatible sensors and their integration into test modules will be a key requirement for performing such measurements. In addition, longer term processes such as corrosion, transport and deposition would also be quantifiably measured with concentration measurements and witness plate samples. Hydrogen transport and permeation can also be investigated in an integrated fashion.

- Fusion Neutron Source (FNS) (see Mission 3)

References


[4.19] ITER Research Plan, 12 Nov 2009 / 2.2 / 2FB8AC.
Annex 5. Mission 5 - Implementation of fusion safety aspects

Authors: G. Federici (EFDA CSU), W. Gulden, N. Taylor (ITER Organization)

Acknowledged contributions from: D. Jones (AREVA TA)

5.1 Short description

Fusion power offers substantial safety and environmental advantages (S&E) compared with other baseload electricity generation technologies. Some of the advantages are an inherently safe process to operate the fusion plasma, low mobilizable radioactive inventory stored in the plant, relatively low radioactivity fuel and no greenhouse gas emissions. The Power Plant Conceptual Study [5.1] has shown that there is no need for evacuation in the case of an accidental event in a fusion power plant. Worker safety is also incorporated into fusion designs, so that site workers are protected against radiation exposure that must be maintained as low as reasonably achievable. The radioactivity induced by the neutrons in the materials surrounding the plasma chambers can be reduced through careful choice of materials and the induced radioactivity lifetimes can be short enough to avoid an accumulating radioactive waste burden. A main target of research is to ensure that in a future power plant all materials should be recyclable to minimize the environmental burdens of radioactive waste, chemically toxic waste or mixed waste for future generations.

Safety and environmental features of a fusion power plant will be mainly determined by in-vessel materials, by the design of components to extract the power and to produce and confine tritium, and by the choice of maintenance procedures. Studies of the safety and environmental impacts of fusion, both of ITER and its precursors (e.g., NET) and of future power plant concepts have shown that a good performance can be achieved [5.1, 5.2, 5.3]. Significant progress has been achieved during the last 10-15 years, especially in Europe, mainly due to the ITER Project.

Fusion Power Safety Objectives

- To protect workers, the public and the environment from harm;
- To ensure in normal operation that exposure to hazards within the facility and due to release of hazardous material from the facility is controlled, kept below prescribed limits and minimized to be as low as reasonably achievable;
- To ensure that the likelihood of accidents is minimized and that their consequences are bounded;
- To ensure that the consequences of more frequent incidents, if any, are minor;
- To apply a safety approach that limits the hazards from accidents such that in any event there is no need for public evacuation on technical grounds;
- To minimize radioactive waste hazards and volumes and ensure that they are as low as reasonably achievable.

In pursuing these objectives, use will be made of established safety principles, such as Defence in Depth, maintaining doses as low as reasonably achievable (ALARA), and maximum use of passive features to implement safety functions.
5.2 Critical aspects for reactor application.  
As shown by the ITER experience [5.3, 5.4, 5.5], progressive integrated safety design analysis will be needed during all phases of the DEMO design. Because of the interlinks between safety requirements and design choices (e.g., materials, coolants and operating conditions), safety analyses will play an important role in the selection of the most promising concepts and must be constantly updated to match the progressive evolution of the design.

The role of safety analysis in DEMO should include:

- Defining the safety approach, application of defence in depth by emphasizing passive safety.
- Establishing safety criteria, starting with offsite release criteria and working down to individual systems. This step was required in ITER to effectively implement safety objectives.
- Proposing integrated safety design approaches. For example, removing safety constraints from in-vessel components to the extent possible and evolving coordinated and comprehensive strategies for radiological confinement.
- Identifying the safety functions and the consequent safety requirements for the design.
- Assisting system designers with safety assessments.
- Assessing the design of systems with respect to their impact on other systems, so that safety experts focused on potential events could evaluate where one system affects another and system interconnections.
- Identifying potential safety issues early in the design, and performing safety research and development activities.
- Demonstrating the safety margins in the design.

Although there are differences between ITER and DEMO (see table 6.1 Mission 6), a lot of what has been learned on safety from ITER experience can be extrapolated to DEMO. The most important areas of fusion safety that are expected to bear a strong impact in the design of DEMO are:

Source Terms and Safety Analysis. Public acceptance and safety of fusion energy is strongly dependent upon the ability to reliably control and confine the radioactive inventories (e.g., mainly tritium and dust that represent the so-called fusion source terms) and on the energy sources available to “mobilize” them. In terms of dust, the key uncertainties are the mechanisms and the amount of dust generated in the machine, its location and the potential for explosive dust mixtures in the presence of hydrogen and air in certain accident sequences. In terms of tritium, for high temperature breeding blankets, the key tritium issues include tritium accountancy, tritium control and permeation. R&D is needed (e.g., tritium permeation barriers) to help better define and hopefully resolve the issue prior to DEMO. In particular, the problem of tritium control requires careful considerations. ITER represents a large step forward in the handling of DEMO-scale tritium flow rates, but ITER will not fully address removal and processing of tritium from candidate breeder blanket
systems nor will its tritium systems be available as test facilities to develop improvements still needed in processing time and system availability.

Understanding and quantifying the fusion source terms will be required for design and later for licensing activities. A series of system-level computational tools needed to analyse the response of a fusion system to an off-normal event or accident are available and have been used, ultimately, for the design process of ITER. They are expected to be improved and fully validated during ITER operation; however it is likely that in some areas (e.g., tritium transport, chemical reactions, dust and hydrogen explosions, magnet arcing) additional data could be required to validate key models prior to a DEMO, particularly where the materials in use differ from those in ITER.

Functional failure modes and effects analyses (FFMEA) have been normally used to identify accident sequences. They consider single failures and their effects. Probabilistic Risk Analysis (PRA) is concerned with combinations of failures. Both techniques seek to identify failures and bad events before they occur so that defensive measures can be taken without harm to people. However, for a full PRA a well-developed design is required and this can only come later during the engineering design.

**Integrated Safety in Design and Licensing.** The licensing process of DEMO may require that design and fabrication of all components with safety functions meet strict nuclear-based design codes and a rigorous quality control and assurance process. However, there exists a lack of knowledge base and a lack of extensive operating experience for fusion components, as compared to fission reactors.

A key safety decision made early in the ITER project was to shift the safety burden away from plasma physics, plasma control, diagnostics, the divertor, first wall/blanket, and magnets to vessels, heat transport systems, and the tritium plant. By definition, ITER’s physics and plasma facing components would be experimental, and an ITER objective was to maximize the flexibility of testing and experimentation during operation. This approach reduced the safety risk by removing all safety needs from the components and systems whose behaviours are least known.

As a result, no plasma-facing components served as part of the radiological control boundary, and wide allowances were preserved in case of their failure. The ITER safety design did, however, require limiting the in-vessel inventory of dust and tritium, thereby placing a greater safety burden on the first confinement boundary.

One of the important design integration issues was defining the first confinement boundary. Choice of the vacuum vessel and its extensions was a logical alternative that took advantage of pre-existing fusion design approaches of high quality and high reliability vacuum vessels. The vacuum vessel can meet the low failure rate criteria with robust construction and double walls that confined tritium, neutron activated materials, and chemically toxic materials. Unfortunately, there are very many vacuum vessel penetrations that are necessary to operate the tokamak. The integration challenge was determining the boundary perimeter for these penetrations. The designers required many vacuum vessel interfaces to the vacuum pumping system, the radiofrequency plasma heating systems, the fuelling system, the diagnostics and their ports, penetrations for cooling system piping, and the maintenance access ports with
port plugs. All of these systems extended the vacuum vessel strong barrier boundary and are part of the first confinement barrier, so their robustness has to be demonstrated as part of the licensing process.

Hence, safety integration became increasingly important with each of the systems that provide a part of the confinement barrier by penetrating the vacuum vessel. The limiting amounts for accidental release of these materials are determined by the adopted safety approach, such as the implementation of the no-evacuation criteria. The effectiveness of release mitigation strategies can only be developed, evaluated, and refined through safety integration in design.

**Materials Life Cycle Management** - Fusion offers salient safety advantages relative to other sources of energy, but generates due to the need of replacement of plasma facing components a sizable amount of radioactive materials that may not be acceptable in existing low-level waste repositories due to the large quantities and the level of tritium contamination. Materials life-cycle management and environmental needs are broad and certainly influenced by design configurations, materials selection, and operational performance. Proper handling of the activated materials is important to the future of fusion energy.

The capacity and the compatibility with tritium contamination of operational commercial repositories in Europe is unclear and must be determined because it could have significant implications on material selection and waste management in DEMO. To guarantee the environmental potential of fusion, geological disposal should be avoided. Focus should be placed on more attractive scenarios, such as recycling and reuse of activated materials within the nuclear industry and clearance or free-release to the commercial market if materials contain traces of radioactivity. Mixed tritiated and radioactive wastes need special attention and may require further development of techniques for detritiation of bulk materials.

An R&D program is proposed below to address some of the main issues identified.
5.3 Level of readiness now and after ITER.

<table>
<thead>
<tr>
<th>Technology</th>
<th>Readiness now</th>
<th>Readiness after ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Quantification of the source term</td>
<td>poor, and need to be over-conservative in estimates</td>
<td>improved in some areas (e.g. dust, in-vessel T retention), but relevance may depend on DEMO materials choices</td>
</tr>
<tr>
<td>- Quantification of the energy source</td>
<td>moderate, but effects of large plasma events needs improvement</td>
<td>improved, particularly with respect to plasma events</td>
</tr>
<tr>
<td>- Design Rules for Component Qualification</td>
<td>poor</td>
<td>improved for some small components (e.g. windows) but for structural materials there is little relevance. TBMs are too low fluence to be useful.</td>
</tr>
<tr>
<td>- Waste Management</td>
<td>Poor, need development of recycling and disposal</td>
<td>Some useful experience in component handling, detritiation, can be expected, but it’s only a small part of what is needed</td>
</tr>
</tbody>
</table>

5.4 Main risks and risk mitigation strategies.  
(see Annex 14 for more details).

<table>
<thead>
<tr>
<th>Risks</th>
<th>Risk Mitigation</th>
</tr>
</thead>
<tbody>
<tr>
<td>DEMO design that does not sufficiently make use of fusion specific safety features.</td>
<td>Start activity by definition of Safety Design Requirements for DEMO. Define the safety approach and criteria.</td>
</tr>
<tr>
<td>Limits on tritium releases in gaseous and liquid effluents difficult to meet</td>
<td>Development and demonstration of large-scale high-reliability and high-efficiency detritiation technologies (both water detritiation and air detritiation)</td>
</tr>
<tr>
<td>Large quantities of tritiated waste and of radioactive waste requiring geological disposal.</td>
<td>Develop recycling options. Develop and demonstrate large-scale, efficient detritiation systems from solid waste, identify isotope concentration levels acceptable in materials to avoid/minimise radioactive wastes requiring geological disposal.</td>
</tr>
</tbody>
</table>

5.5. Description of the Major Work Packages.  
Safety-related activities in Horizon 2020 are described in an appendix with emphasis on the most important issues during the phase of concept definition and conceptual

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Components for use in DEMO must be qualified to validate the design and demonstrate safety roles of key components are satisfied. Guidelines on the safety classification to be used for DEMO in-vessel components, taking credit to the largest extent possible from ITER experience (see considerations above) is needed. Available alternatives must be also clarified and implications on the licensing process arising from these assumptions must be determined. Simple acceptance of fission based codes is likely to lead to unacceptable restrictions on operations or unnecessary expense to meet safety margins that are not needed for fusion systems.
design activities. In addition, safety-related R&D that is specific to a component design concept, e.g., chemical reactivity of breeders and multipliers with waters, or common to more than one concept, e.g., development of T-control techniques, are described elsewhere (see Mission 4).

A) Integrated Safety Analysis.
Work in this area is primarily meant to understand and quantify the radioactive and toxic source terms in DEMO and the energies which could mobilize them, as a function of design assumptions (which are expected to evolve) and will include code development/improvement and analysis work. Experimental verification by properly designed experiments is also an important component of this R&D. Similarly to ITER, establishing safety criteria and integrated safety design approaches for DEMO will be a prerequisite to all safety assessments and will be needed to identify potential safety issues early in the design and corresponding safety R&D activities. This will be done by producing a Safety Requirements Document (SRD) for DEMO by 2016.

If, depending on the design choices to be made, explosion risks\(^9\) in DEMO remain important, there is a need to improve and validate hydrogen combustion modelling in Computation fluido-dynamic CFD codes and to develop and experimentally validate dust explosion models. There will be also the need to develop a better strategy for fire prevention, detection and extinguishing because the potential for fire in all parts of the nuclear plant may present a greater risk than the accident initiators previously studied.

As far as occupational radiation exposure is concerned, DEMO may be quite different from ITER. Respecting a collective dose target (for ITER, 500 person.mSv/year) may be very challenging. Although very difficult to fully define at the conceptual design stage, minimizing hands-on maintenance operations is important and should be part of the RH strategy (as part Mission 6).

Finally, as a result of concerns heightened after Fukushima, analysis of “beyond design basis” events has become more important. In this context, it should be expected that the following activities should be carried out: 1) Evaluate impact of external hazards in the safety case of DEMO, e.g., by considering combinations of events (e.g., plane crash + fire, earthquake + flooding), or postulate even lower likelihood events; 2) Assess impact arising from total loss of power (including emergency power); 3) Analyze and reduce risks arising from a hydrogen explosion (see above).

B. Integrated Safety in Design and Licensing
Complex and extensive verification and validation tools and supporting safety performance databases will be required to fulfil licensing and regulatory requirements. Detailed regulatory requirements and associated regulatory guidance applicable to fusion power reactors currently do not exist. Identification and detailed accounting of fusion specific requirements and guidelines is necessary prior to assessment of licensing specifications for establishing a licensing framework for DEMO.

\(^9\) Potential for explosion risks comes from three areas: (i) “Conventional” explosion risks; (ii) Hydrogen explosion risks in fuel cycle; and (iii) In-vessel H and H/dust explosion risks.
From the very beginning, an integrated safety design approach should be followed for DEMO. For example, removing safety constraints from in-vessel components to the extent possible and evolving coordinated and comprehensive radiological confinement boundaries. In addition there will be the need to propose and develop a licensing framework based on safety goal policies to ensure that conceptual design and operation scenarios are consistent with safety performance goals. The necessary inputs would include, at a minimum the following reports: 1. Safety Requirements Document (SRD) (final by 2016), 2. Safety analysis report including comprehensive identification of hazards, identification of safety functions and the corresponding safety credit to be given to systems, structures and components (by 2015); 3. Transient and accident analysis, including demonstration of ultimate safety margin in arbitrary beyond-design basis scenarios; etc. code and standard identifications, hazard analysis. (2018-2020)

C. Radioactive waste management strategy for fusion. Beyond the need to avoid producing high-level waste, there is a need to establish a complete waste management strategy that examines all the types of waste anticipated for DEMO, given a more restricted regulatory environment for disposal of radioactive material expected in the future.

The strategy should include waste reduction, recycling and material clearance procedures, remote handling and treatment of tritium-containing materials, separation of materials from complex components, detritiation and tritium capturing and handling, and refabrication of recyclable and clearable materials. Examples of specific research needs include:

- Studies to determine if clearance and/or recycling are really feasible and if yes, develop viable industrial processes. In particular, one should investigate fusion-specific clearance limits that could be issued by legal authorities and the required infrastructure and requirements to make simple or complex recycling a viable option for fusion and existing recycling and clearance infrastructure and market. This study should also characterize and assess the quantity of waste generated by the recycling process.
- Develop and demonstrate large-scale, efficient detritiation systems capable of removing and handling the majority of tritium from activated materials. Existing repositories may not accept tritium-contaminated active waste.

References
Annex 6. Mission 6 - Integrated DEMO Design and system development

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Acknowledged contributions from: J. Harman (EFDA CSU), C. Morlock (EFDA CSU), B. Meszaros (EFDA CSU), T. Bonicelli (F4E), P. Bruzzone (CRPP), D. Ward (CCFE), E. Surrey (CCFE), L. Porte (CRPP), A. Loving (CCFE), C. Day (KIT), P. Sardain (CEA), D. Buckthorpe (AMEC), E. Bogusch (Areva)

6.1 Short description

DEMO will be the single step between ITER and a first-of-a-kind commercial reactor. It has various missions but a universal definition does not exist. Also, it is rather difficult, to define a set of top-level requirements for devices like DEMO without specifying beforehand a large set of hypotheses, both in physics and in technology. However, a definition of the top-level requirements for DEMO compatible with several studies undertaken [6.1, 6.2] and more recently the Ad hoc Group on DEMO Activities, CCE-FU 49/6.7 of March 2010 [6.3] is to:

- Demonstrate a workable solution for all physics and technology questions associated with capturing the energy released by burning plasma and converting it to a useful power flow in a safe, reliable, and sustainable manner through the successful integration of many systems and physical processes.
- Demonstrate significant net (~ several hundreds of MW) electricity production with self-sufficient fuel supply.
- Achieve realistic availability targets.

Currently, no pre-conceptual design exists for DEMO. Work carried out in the past in Europe on fusion reactor design focussed on the Power Plant Conceptual Studies [6.4] with emphasis on the assessment of the safety, environmental, societal and economic features of fusion power, and less on feasibility studies10.

As part of the recently established EFDA Power Plant Physics and Technology (PPPT) Department, key physics and technology prerequisites for a number of options for DEMO are being analysed, with the purpose of identifying the most urgent technical issues that need to be solved to address the remaining knowledge gaps. It should be noted that the problem of the power exhaust is one of the main outstanding issues and is going to be a major driver of machine size and mode of operation (Mission 2).

In light of the above, an integrated design effort is viewed as essential to reduce the uncertainties and the technical risks associated with the various design options and to shed light on key design trade-offs and constraints within an integrated system for

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10 The Power Plant Conceptual Study (PPCS) – completed in April 2005, has been a study of conceptual designs of five commercial fusion power plants and the main emphasis was on system integration. It focused on five power plant models which are illustrative of a wider spectrum of possibilities. They are all based on the tokamak concept and they have approximately the same net electrical power output, 1500 MWe. These span a range from relatively near-term, based on limited technology and plasma physics extrapolations, to an advanced conception.
DEMO. For example, maximizing power cycle thermal efficiency requires operation at high temperatures, with upper limits set by coolant and structural compatibility, thermal stress, material behaviour and plasma thermal loading considerations. Identifying serious limiting factors and design issues can guide research toward high-leverage and high-payoff issues while minimizing program risk. Such detailed advanced design studies also allow the assessment of RAMS (Reliability, Availability, Maintainability and Safety) issues for fusion that are strongly linked to design assumptions.

Integrated system analysis and conceptual design is expected to play a major role in supporting program evaluation and decision points and in guiding fusion R&D towards clear and measurable deliverables and practical products. This will include: definition of system requirements and analysis of trade-offs; analysis of potential design options and identification of high leverage issues and potential showstoppers; evaluation of systems/components performance and key limiting factors; assessment of feasibility issues; concept verification and basic (small-scale) technology R&D demonstration; conducting design integration studies and resolution of main interface problems. In particular, it is essential to develop improved integrated designs that satisfy availability and safety requirements by applying lessons learned from ITER, other experiments and remote handling facilities and to make design choices that promote reliability and maintainability.

6.2 Critical aspects for reactor application

In addition to the problems of power exhaust (Mission 2); structural materials, and heat sink/ armour materials (Mission 3); breeding blankets (Mission 4); and safety (Mission 5); there are a number of important systems, whose performance and reliability will strongly affect the feasibility and performance of DEMO. These include: Superconducting Magnet systems; H&CD systems; Remote Maintenance systems; Vacuum and Pumping systems; Diagnostics and Control Systems; and the Balance of Plant (BoP). The main technical risks and risk mitigation strategies are indicated below together with technical readiness levels (TRL) both now and after ITER.

Since the mission requirements of a near-term DEMO put more emphasis on solutions with high technical readiness levels and realistic performance and component reliability, rather than on high-efficiency, the discussion here and the R&D priorities defined in the work packages are defined to achieve these goals. Nevertheless these goals remain very ambitious. More advanced technological solutions that can only be developed on a longer timescale are instead included in Mission 7. The main differences between ITER and DEMO are summarised in Table 6.1

Table 6.1 Main Differences between ITER and DEMO

<table>
<thead>
<tr>
<th>ITER</th>
<th>DEMO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimental device with physics and technology missions.</td>
<td>Nearer to a commercial power plant, but with some development missions.</td>
</tr>
<tr>
<td>400 s pulses (some longer at lower power). Long dwell time.</td>
<td>Long pulses (a few hours), quasi-steady state.</td>
</tr>
<tr>
<td>Experimental campaigns. Outages for maintenance, component replacements.</td>
<td>Design as a system must maximize availability.</td>
</tr>
</tbody>
</table>
Large number of diagnostics. Only those diagnostics required for operation.

Multiple heating & current drive systems. Optimized set of heating & current drive systems.

Large design margins, necessitated by uncertainties and lack of fully appropriate design codes. With ITER (and other) experience, design should have smaller uncertainties.

Cooling system optimized for minimum stresses and sized for modest heat rejection. Cooling system optimized for electricity generation efficiency (e.g. much higher temperature.)

Test blanket modules introduce range of diverse concepts. Single blanket concept.

Unique one-off design optimized for experimental goals within cost constraints. Move towards design choices suitable for series production.

No tritium breeding requirement (except very small quantity in TBM). Tritium breeding needed to demonstrate self-sufficiency.

Conventional 316 stainless steel structure. Nuclear hardened, novel reduced activation materials as structure for breeding blanket.

Very modest lifetime neutron fluence, low dpa and He production. High fluence, significant materials damage.

Licensed as Nuclear Facility, but like a laboratory, not a reactor. Licensing as reactor more likely.

Licensing as experimental facility allows some credit for experimental nature (e.g. no dependence of safety on plasma behaviour). Stricter approach may be necessary to avoid large design margins.

“Progressive start-up” permits staged approach to licensing. “Progressive start-up” should also be possible (e.g., Utilize a "starter" blanket configuration using moderate-performance materials (which don't affect regulatory approval) and then switch to blankets with a more advanced-performance material after a few MW-yr/m² (see Mission 3)).

During conceptual design (including “EDA”), licensing in any ITER party had to be possible. Fewer constraints.

**Magnets**

For fusion applications, high current cables are needed and CICC (Cable In Conduit Conductor) have been the principal vehicle. CICC cables are complex structures, and there are still areas of their performance that are poorly understood, especially with regard to fatigue effects. The complex strain state of strands within the CICC are not quantitatively understood yet as is clear from the degradation effects seen in the ITER test cables after multiple loading (both magnetic field loading and system warm-up and cool-down) cycles [6.5, 6.6].

Whilst it has been demonstrated that Nb₃Sn conductors can produce a magnetic field strength of up to 18T, the limiting factor for field strength is the ability of the coil material to withstand the stress from internal forces. The force in the coil increases with the major radius (R₀) and the field strength squared. From ITER to DEMO, the increase in force and therefore cross sectional area of, e.g., the TF coils will be approximately 68%. This has implications for both system cost and design integration.

In general, the issues for Low Temperature Superconductor (LTS) conductor and magnet technology for DEMO are 1) improved performance; b) increased lifetime and reliability; and c) cost reduction.
H&CD systems

H&CD systems are among the technologically more advanced systems under construction for ITER, and DEMO will benefit largely from this development. The discussion here is limited to Neutral Beam Injection (NBI) Systems and Electron Cyclotron Heating and Current Drive (ECH). Additional systems (e.g. Ion Cyclotron Radiofrequency (ICH) and Lower Hybrid Waves (LH)) are available but their final use in DEMO will be determined based on successful installation and operation in ITER.

ICH is a lower cost solution, but still has open questions related to coupling in a variety of plasma scenarios and edge densities. In addition, current drive using ICH is not well developed and is concentrated near the plasma axis whilst off-axis current drive is usually required for confinement improvement (central current drive may be sufficient if the goal is only to increase pulse duration). Hence, it would mainly be a candidate for a pulsed DEMO, not so much for a steady state DEMO in which CD requirements are very important.

LH is the most efficient CD source and is a good candidate for saving flux during $I_p$ ramp-up, for lengthening the pulse duration and for accessing and sustaining advanced scenarios. However, it has coupling and accessibility problems in DEMO and hence it must be further studied if its use is possible in DEMO at all. In particular, the compatibility of the launcher with close proximity to a fusion plasma is an important issue. The technology of the sources and of the coupling structures requires further development but without tests on ITER it will be difficult to assess this system.

For a steady-state DEMO, NBI offers a high CD efficiency. In addition, it can be flexibly used at different values of $B$ and $I_p$ during plasma commissioning development. It cannot be used for MHD control and hence cannot be used as an exclusive H&CD system on both DEMOs. Technical development risks exist with beam energies of 1 MeV but will be addressed as part of the ITER R&D Programme. It is unlikely that energies above 1MeV will be proposed for DEMO since increase to 1.5MeV would increase the CD efficiency by about 10%, which could also be achieved by lowering the plasma density and increasing the electron temperature. NBI has a relatively low ‘wall-plug efficiency’, and it is desirable to carry out an R&D programme to improve this via neutraliser development and energy recovery systems. These do not need tokamak facilities for development, as test beds will suffice.

The R&D that will be made for ITER will allow the reuse of technology for DEMO. The reliability of the source can be assessed only after the test in the Neutral Beam Test Facility (NBTF) and in ITER; modularity is mandatory for a DEMO and could improve reliability of the overall system.

ECH is technically mature for shaping the current density profile in a unique way and for controlling MHD instabilities (e.g. NTMs, sawtooth instabilities, etc.). However, its off-axis CD efficiency is lower than that predicted for NBI and hence it is at present not clear if it can serve as an exclusive CD method in a steady state DEMO. The technical readiness level (TRL) of ECH is similar to NBI. The main development needed beyond ITER involves the choice of the frequency, which is probably higher than in ITER, and the step tune-ability, which would allow a launcher without movable parts. Remote steering as an alternative/ risk mitigation strategy for step-
tunability should also be investigated. Coupled to step-tunability is the issue of the feasibility and performance of broadband windows. The increase of power per unit can be important to increase the margin of operation but should be always weighed against the need for system modularity for high reliability. Overall ‘wall-plug’ efficiency is already high (≥50%) and its improvement is more relevant for long-term application in FPP (see Mission 7).

**Diagnostics and Integrated Control Systems**

DEMO diagnostics and associated control systems are constrained by the extreme environmental conditions, mostly due to the high neutron flux and fluence, and the stringent requirements on reliability, availability and maintainability. At the same time, plasma operation must be even more robust than in ITER since e.g. the pulse length is increased and disruptions have to be fully avoided (see Mission 1).

The design of ITER diagnostics has progressed a number of issues, but it is expected that some of the diagnostics used in ITER will not be available for DEMO.

**Remote Maintenance System**

The development of the remote maintenance system for DEMO is driven by the need to maximise the overall plant availability, the strongest downward driver of cost of electricity in a fusion power plant and, therefore, to minimise the plant down time for the foreseen maintenance operations. Delivering a reactor-relevant maintenance concept will drive the design toward fewer numbers of replaceable maintenance modules (e.g. multi-module blanket segments; divertor cassettes, etc) and eliminating complex in-vessel operations that are commonly used on experimental fusion devices. In-vessel environmental conditions such as radiation, activation and decay heating will be far more aggressive than those to be experienced in ITER. This will restrict the type of maintenance operations that can be performed in-vessel and also the type of equipment that can be deployed, possibly ruling out optical camera systems entirely.

Design integration and remote maintenance compatibility of components must be an essential consideration throughout the design phase of maintained equipment. For example, attachment of in-vessel components to the vacuum vessel or shield is a particularly critical aspect with respect to RH compatibility and thermal compensation.

The reliability of the plant maintenance system is essential to ensure that down time is minimised. This again drives toward the use of proven technology and simplified operations as well as the deployment of dependable rescue systems.

This work is particularly important within the DEMO work programme because the ITER maintenance scheme for in-vessel components, in particular blanket systems, is not power plant-relevant.

**Vacuum and Pumping**

Due to the requirement for DEMO to have a self-sufficient tritium fuel cycle and also operate for extended pulses (i.e. 2-4 hrs), the critical aspects that need to be addressed by the torus Vacuum and Pumping area are: (1) effective tritium separation (in regions close to the divertor) to minimise throughput requirements of the tritium plant; and (2) development of a pumping train of continuously operating vacuum pumps.
Key drivers in the development of the new technology must be the reduction of the tritium inventory build-up and the minimisation of the processing times (both targets to minimise start-up inventory) by giving the pumps the additional function to act as a short-cut to the tritium plant.

The third aspect with special relevance to DEMO is to ensure an economically attractive high duty factor. This means to reduce the dwell times between the shots to an absolute minimum. For ITER, to maintain a certain dwell time was not a requirement. For the DEMO dwell period, it is mandatory to achieve the necessary start-up vacuum pressure in the order of maximum 20-30 min (this is the order of the CS coil re-charging time). This is a new and challenging requirement for the DEMO torus pumping system.

All vacuum pumping systems have to show a high level of system integration, as their performance strongly depends on the distance to the volume they have to pump and the configuration of the pumping duct.

**Balance of Plant**

The overall thermal efficiency of the balance of plant (BoP) will be limited by design decisions made concerning the structural material of the in-vessel components (e.g. EUROFER) and the choice of coolant. The potentially viable options for both primary and secondary coolants are for the primary loops: helium, pressurised water and lithium-lead (blanket); pressurized water and helium (divertor); and for the secondary loop: helium, pressurized water, molten-salt.

The choice of a water-cooled divertor (see Mission 2) would remove the possibility of higher grade heat being introduced into the primary cooling loop in order to improve the overall thermal efficiency (as would be the case for a high-temperature helium-cooled divertor). However, the heat output from the divertor could still be usefully transferred to the balance of plant.

A pulsed DEMO machine will have implications for the BoP in terms of thermal transients and fatigue due to cyclic loadings. In order to prove the principle of connection of a pulsed energy system to the grid, it may be necessary to introduce a Thermal Energy Store to buffer the thermal transients and reduce the cyclic loading effect however which would further increase cost and complexity of the BoP.

Although heat exchangers have been built in the past for helium and CO2 gas, currently helium-based BoP technology is considered rather immature compared with water-based systems and no “off-the-shelf” helium to steam heat exchangers appear to be available for the likely conditions of DEMO. The choice of helium as a coolant also introduces various mechanical and materials issues that will have to be resolved through a development programme (e.g. erosion; high sticktion on moving parts). Furthermore, the Generation IV programme in Europe seems less likely to develop helium-based BoP in favour of Sodium and Liquid Metals.

Nuclear safety considerations are also paramount and it is expected that DEMO will have to employ an indirect cycle for the BoP architecture. Even so, the control of contamination and tritium transport will have to be carefully addressed and the choice of a water coolant will introduce gamma doses from $^{16}\text{N}$ decay and therefore additional shielding requirements.

Maintenance considerations for the BoP are also very important and therefore the overall BoP layout in terms of design integration; typical planned maintenance /
inspection schedules (i.e. turbines, heat exchangers); and robust design and failure analysis will all be critical aspects in the development programme.

6.3 Level of readiness now and after ITER

ITER construction and operation will be a huge accomplishment and represent a significant step forward in many engineering areas associated with fusion. ITER will contribute to the knowledge base needed for a DEMO reactor in many areas. However, there are several areas where gaps exist.

Table 6.2 Technological readiness of the various systems.

<table>
<thead>
<tr>
<th>Technology Area</th>
<th>Readiness now</th>
<th>Readiness after ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Superconducting magnets</td>
<td>TRL 4 - ITER technology developed in the 90’s. - Demonstrated in model coils and small fusion experiments in EU and Asia</td>
<td>TRL 7 - Still some issues with the superconductor but TRL will be very high after ITER. - Some optimisation will be necessary</td>
</tr>
<tr>
<td>H&amp;CD systems</td>
<td></td>
<td></td>
</tr>
<tr>
<td>NB (1 MeV)</td>
<td>TRL 3 - 4 Increasing TRL 5 - 6 after successful completion of MITICA</td>
<td>TRL 7 – 8 - After successful operation in ITER</td>
</tr>
<tr>
<td>ECH (1 MW. 170 GHz) – steerable mirrors</td>
<td>TRL 4 - 5 - After ITER development programme</td>
<td>TRL 6 - 7 - After commissioning and successful test in ITER (but ITER presently does not plan step-tuneable gyrotrons)</td>
</tr>
<tr>
<td>Diagnostics (see also Mission 1)</td>
<td>TRL 3-4 - ITER technology</td>
<td>TRL 3 - 4 - some of the diagnostics used in ITER will not be available for DEMO.</td>
</tr>
<tr>
<td>Remote Maintenance</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Divertor maintenance</td>
<td>TRL 4 - 5 - After successful completion of the DTP programme</td>
<td>TRL 6 - 7 - After demonstration in ITER (note: Radiation levels may preclude the use of visual feedback in DEMO)</td>
</tr>
<tr>
<td>Blanket maintenance</td>
<td>TRL 1 – 2 - DEMO blanket maintenance scheme completely different to ITER</td>
<td>TRL 4 - After mock-up trials conducted in 2017 (as part of DEMO programme)</td>
</tr>
<tr>
<td>Ex-vessel</td>
<td>TRL 3 - 4</td>
<td>TRL 6 – 7</td>
</tr>
<tr>
<td>BoP maintenance</td>
<td>TRL 7 - Relevant technology transferable from fission plants</td>
<td>- Not applicable - No BoP in ITER</td>
</tr>
<tr>
<td>Vacuum and Pumping</td>
<td>TRL 6 - Prototype demonstration in a relevant environment</td>
<td>TRL 9 - System proven through successful mission operations</td>
</tr>
<tr>
<td>Cryopump w/o separation function</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cryopump with separation function</td>
<td>TRL 4 - Component validation in laboratory environment</td>
<td>Not applicable (ITER uses different pumps)</td>
</tr>
</tbody>
</table>
6.4 Main risks and risk mitigation strategies

See risk register in Annex.

6.5. Description of the major Work Packages

The specific activities outlined here form the basis of an R&D program which relies on, as much as possible, ITER experience and on using existing infrastructure and testing facilities. Emphasis is on performance and reliability more than on efficiency

**DEMO Plant Level System Integration**

The integration of the various systems identified above within the overall DEMO plant is an essential activity within Mission 6. It is necessary to assess the influence of key design drivers such as primary coolant choice; H&CD configuration; maintenance targets etc. on the achievement of the overall plant mission requirements. Furthermore, it is clear that the impact that design drivers have on individual systems may not become full apparent unless view from the plant level.

**Superconducting Magnet System**

The integration of the Superconducting Magnet system within the DEMO plant is an essential task within the DEMO CDA phase. This will involve establishing requirements, functions and interfaces with many other systems to ensure that aspects such as plasma stability; overall sizing of coil components; access to vacuum vessel ports; etc; are considered from the outset. The baseline technology for DEMO is proposed as Low Temperature Superconducting (LTS) Niobium Tin (Nb$_3$Sn). This technology is the current state-of-the-art in fusion superconducting magnet systems and its application at near-DEMO conditions will be demonstrated by ITER within the DEMO CDA phase. Nb$_3$Sn technology has a mature industrial-scale manufacturing base that is capable of constructing conductors to meet the needs of DEMO sized coils. During the CDA phase the selection of a Nb$_3$Sn conductor that fulfils the critical DEMO requirements is the first priority. It is foreseen that prototype conductors can be built and tested by mid-2015. An iteration of the prototype may be

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11 Advantages of indirect cycles: 1) Relaxed radioactivity containment requirements; 2) isolation of PCU from the primary circuits (minimize risk of missile accident); 3) decrease O&M due to non-contaminated PCU; and 4) continuation of the Class I boundary area \( \rightarrow \) more design flexibility with respect to material choices.
necessary, taking another two years. Depending on the conductor choice, the related coil winding issues will be reviewed. Other aspects such as electrical insulation, structural material (mostly the one of the "conduit") and joint design will be addressed.

**Heating & Current Drive Systems**

**Neutral Beam**

It is expected that the Neutral Beam system for DEMO will employ, as much as possible, the technology that is deployed and tested both at the Neutral Beam Test Facility (NBTF) “MITICA” in Padova, Italy and then within ITER itself. However, a number of technical risks have been identified that should be addressed during the DEMO CDA phase.

The overall layout of the DEMO Neutral Beamlines and the integration of the Neutral Beam system into DEMO clearly have major influence on risks associated with performance; maintainability; reliability; redundancy; fault tolerance; safety; etc. For example, the question of one large ion source per beamline versus a number of smaller ion sources per beamline should be addressed.

R&D into ion source technologies is viewed as an important activity to mitigate risks associated with caesium consumption (i.e. complex RH operation to replace Cs results in lower availability of NB system) and also loading issues due to back-streaming ions.

A number of risks contribute to the overall efficiency of the neutral beam system (i.e. stripping losses; beam divergence; neutralisation efficiency) and various work packages will attempt to mitigate these through accelerator design and energy recovery systems.

**Electron Cyclotron**

The main development needed beyond ITER involves the choice of the frequency and the step tune-ability function. The increase of power per unit can be important to increase the margin of operation but should be always weighted against the need of modularity for high reliability. Present facilities (KIT until ITER gyrotron development is over, then CRPP) do not seem to need substantial improvements beyond that already underway.

**Diagnostics and Integrated Control Systems**

A number of activities should be planned in Horizon 2020. They include (see also Mission 1):

- Building on the ITER analysis, assess principal restrictions to diagnostics for machine protection, basic control and advanced control in DEMO.
- Screening of diagnostic techniques and methodologies and assess long lead diagnostics relevant R&D.
- Assess and develop novel approaches to feedback control of the plasma with ‘sparse’ data systems and robust actuators.
- Following screening study, develop further hardened versions of key ‘essential minimum diagnostic set’.
- Following feedback control developments, develop novel diagnostic systems and data analysis tools.
- Assess the issues related with the integration of the diagnostics on DEMO.
Remote Maintenance
The integration of the Remote Maintenance system within the DEMO plant is an essential task within the DEMO CDA phase. This will involve establishing requirements, functions and interfaces with many other systems to ensure that plant availability and maintainability are considered from the outset.

In-vessel radiation doses will be far higher in DEMO when compared to ITER and in turn this may restrict the type of maintenance equipment and operations that can be performed. A thorough understanding of the DEMO radiation map will be developed and used to help mitigate associated risks.

As the blanket maintenance requirements are far removed from ITER, a completely novel concept must be developed and substantiated. A strong integration with the in-vessel components design team is foreseen.

Furthermore, it is foreseen that the substantiation of design concepts for aspects such as in-vessel attachments; remote maintenance transporters; servo manipulators; will require the use of extensive mock-up and test facilities.

Vacuum and Pumping
The work packages for Vacuum and Pumping are split into three main areas: (1) Development of cyropumps with a separation function; (2) Development of continuously working pumps and (3) development of a fuel cycle simulator.

Balance of Plant (BoP)
The work packages for the Balance of Plant are made up of the following areas: (1) System Integration; (2) Modelling, analysis and design of BoP architectures; (3) Application of existing BoP technologies and development needs (4) Energy Storage Systems.

Within (1), the unique system integration issues associated with attaching electricity generation equipment to a fusion reactor shall be investigated in-depth. Risks associated with efficiency; performance; pulsed operation; maintainability; reliability; redundancy; fault tolerance; cost; safety; etc shall be tackled at the system-level.

In (2), a number of possible architectures for both water-cooled and helium-cooled BoP shall be designed, modelled, analysed and evaluated using appropriate tools and the involvement of industrial experts. Closely integrated will be (3) to engage closely with industry in order to ascertain what existing and proposed technologies will be available for application to the DEMO plant. Finally (4) addresses the risk associated with pulsed operation through investigation of energy storage systems within the BoP.

References


Annex 7. Mission 7 - Competitive cost of electricity
(Long Term Attractiveness Fusion Power)

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Acknowledged contributions from: D. Stork (CCFE), J. Harman (EFDA), E. Surrey (CCFE), D. Ward (CCFE), L. Porte (CRPP), P. Bruzzone (CRPP)

7.1 Short description

Nuclear fusion will not be ready for commercial power generation in the first half of this century; it remains nevertheless an attractive energy solution and arguably, perhaps along with solar power and breeder fission reactors, the only truly sustainable option for large-scale base load supply in the long-term. If the research and development in fusion energy delivers the advances predicted, then it could achieve this aim in the second half of this century.

In order for fusion power to be a major provider of future energy supply, the development programme must establish its technical feasibility and its economic feasibility, as well as guaranteeing safety and environmental performance. The primary goal is to demonstrate technical feasibility, whilst maintaining safety and environmental performance, but at the same time economic performance must be maintained as a guiding principle. It is not appropriate to set a specific cost target that fusion must reach, as this varies substantially between countries, even today, and will be highly dependent on future developments which are largely unknown such as the future cost of fuels and other commodities, costs associated with pollution constraints and, particularly uncertain at present, the penalties and costs associated with carbon emissions and abatement. An example of this uncertainty is the approximate doubling of the price of capital intensive energy technologies in the period 2003-2008, along with a price of oil which increased by a factor of 6 over that period. Determining a single price point for the future energy supply system is clearly enormously uncertain and cannot be used as a strong driver of the development programme.

In the period before the commodity price explosion (2003-2008), it was concluded broadly that the expected internal costs of fusion electricity would be competitive with typical renewables (without storage costs) and about 50% greater than coal (without emission abatement costs) or fission. A study by Maisonnier et al. [7.1] using a mathematical model with the latest data on costs and varying the free parameters of the design so as to minimise the cost of electricity, concluded that the expected costs to supply electricity from fusion, if the programme was technically successful, would vary between €0.03-0.09/kWh which would make fusion power competitive with other sources of energy. An earlier paper by Ward [7.2], stated that a mature fusion technology could supply electricity in the range €0.03-0.07/kWh. As indicated above, these studies are very dependent on the assumptions made about both technical outcomes and the economic background such as commodity prices, but serve as a guide to the sort of cost range expected.

In a coherent development programme, there must be a balance between R&D on
delivering the core programme, addressing risk of failure of that programme, and on concept improvement/cost reduction. The balance of expenditure in these 3 areas is, of course, a matter of judgement and will depend on the perception of future energy markets; for instance will the growth in energy consumption in the developing world be easily satisfied with existing and planned energy technologies or will it present large stresses which induce volatility and high prices which will hold back development? This mission addresses the concept improvement/cost reduction theme – developing more advanced technologies than are strictly necessary for technical success, in order to explore cost reduction strategies.

The attractiveness of a fusion system (economics, safety and environmental features) is mainly determined by materials and design of systems associated with harnessing fusion power. However at present these systems are at a low technical readiness level (TRL) with high uncertainty as to the performance of envisioned solutions and material systems. An integrated program of advanced R&D needs to focus on developing alternative approaches that could allow for incremental improvements, or radical approaches that could substantially affect other systems and reduce cost.

Example of radical changes are for example development of H&CD systems with very high efficiency, the use of High Temperature Superconductor (HTS) materials and magnet systems to avoid the use of helium, the development of advanced technology divertor solutions, that would become attractive if we find reliable way to minimise the divertor surface heat flux, the development of high temperature structural materials, such as ODS, etc.

A program which fulfils the research gaps and research activities described here can potentially revolutionize the design of magnetic fusion devices but only on the long term.

### Elements for an Attractive Fusion Electric Power Source

- Cost advantage over other available energy options
- Eased licensing process
- No evacuation plan needed
- No high-level waste produced
- Reliable, available, and stable electrical power production
- No local or global atmospheric impact
- Closed on-site fuel cycle
- High fuel availability
- Plant capability of load-following
- Availability in a range of unit sizes

### 7.2 Critical aspects for reactor application.

Fusion technologies that have the potential to strongly impact the attractiveness of fusion are:

- **H&CD system with high (60%) wall plug efficiency.** This includes primarily the development of technologies for energy recovery and better neutralization efficiency (in the case of NBI). Some of these are already included as part of the R&D for DEMO (Mission 6). In addition, the photo neutraliser could offer the
potential for high neutralisation efficiency of the negative ion precursor beams anticipated for DEMO for NBI systems. The deployment of high power, high efficiency ECH systems is also considered under this area of development.

- **High temperature Superconductors (HTS) Magnets.** HTS magnets offer the opportunity for higher magnetic fields at higher operating temperatures and margins. This in turn would lead to a higher overall efficiency from the fusion power plant due to higher energy density and lower cryogenic power requirements respectively. In addition, if demountable TF magnets could be developed, they would offer a potentially transformative technological innovation with simpler maintenance methods (i.e. larger access ports) and improved availability. The application of REBCO (Rare Earth Barium Copper Oxide) for the conductor seems the most likely candidate at the moment but feasibility of a cable needs to be demonstrated (see 3-year programme elaborated in 3PT). A prototype conductor would be more expensive than for LTSC. Most of the development in this area is expected to come through investments outside fusion and synergies with the fusion programme should be sought. Nevertheless, there is also the need to maintain know-how in the laboratories.

- **Advanced Divertor Heat Removal Technologies.** Development of He-cooled divertor concept should be continued to further improve performance and durability of concepts developed so far. In addition, alternative heat flux enhancement techniques should be examined in more detail at DEMO relevant parameters. These include improvements to jet impingement via cascade or surface roughness modifications as well as novel coolant and material selections. These alternative heat transfer removal techniques may provide higher heat flux removal capability and/or be more robust under conditions which will degrade the performance of concepts currently under development. On the other hand, while there are certainly gains to be made by improving convective heat transfer through these enhancement techniques, they produce diminishing returns on performance as the conductive element of the heat transfer through the system from plasma to coolant becomes dominating.

- **Advanced Breeding Blanket Concepts (DCLL) (see Mission 4).** The DCLL uses flowing PbLi as both breeder and coolant for the breeding zones, while utilizing high pressure helium to cool all structures including those surrounding the breeding zone. Flow channel inserts made of a SiC-composite in all liquid metal ducts serve as electrical and thermal insulator, enabling a liquid metal exit temperature about 200K higher than the maximum temperature of the steel structure.

- Activities included in Mission 4 aim at the development of a reduced performance DCLL design that uses EUROFER as a structural material (i.e., max temperature 500-550°C) [7.3] also leading to lower power conversion efficiency. Options included here consider instead more aggressive designs that push the LiPb outlet temperature to the compatibility limit of LiPb and SiC, perhaps even exceeding the creep strength temperature limit of present generation fusion steel structures.

- **Advanced structural high temperature materials** A significant assortment of materials and alloys for high temperature applications are in use in the petrochemical, metals processing, and aerospace industries, but a very limited
number of these materials have been tested or qualified for use in nuclear reactor-related systems.

Issues and recommendation for further developments in this area are part of Mission 3.

7.3 Level of readiness now and after ITER.

By definition the technology readiness level of the options presented in Mission 7 is quite low (TRL 1-2) and will be unaffected by ITER.

7.4 Main risks and risk mitigation strategies.

See Annex - Risk register

7.5. Description of the major Work Packages.

Much work remains to be done establishing the technology and engineering foundations of these advanced technology options and some of the most important activities are described here.

Photo-neutraliser

The photo-neutraliser offers the potential for high neutralisation efficiency of the negative ion precursor beams anticipated for DEMO. Although this technology has been under discussion for several decades [7.4], there has been no attempt to build a prototype, mainly due to the high laser power required and the perceived disadvantage of complex equipment [7.5]. More recent studies for fusion power plants have shown the advantages of deploying the photo-neutraliser to boost the wall plug efficiency of the NBI system and reduce the re-circulating power requirement [7.6]. There have been studies since (see the review in [7.7] and [7.8, 7.9]) that can be basically reduced to two designs of cavity – Fabry-Perot (resonant) and direct drive. The advantage of the Fabry-Perot is the reduced laser power (~tens of kW) needed as a result of the cavity resonance. However, maintaining this condition is challenging and in recent studies [7.10] the required cavity length is estimated to be 20-30m. The direct drive cavity by contrast can be the same dimensions as a conventional gas neutraliser but requires in the order of 800kW laser power, depending on the degree of neutralisation required. The geometry of the photo-neutraliser, imposed by power considerations, will most likely require development of tall, narrow ion sources or the stacking of multiple small injectors to produce narrow beams. This geometry is also compatible with energy recovery but the impact on system integration has not been assessed. The combination of the two technologies allows more flexibility as the energy recovery acts a risk reduction for the photo-neutraliser; the use of energy recovery reduces the sensitivity of wall plug to photo-neutralisation.

The development roadmap presented here considers both options, together with energy recovery, as there is an optimisation process to consider involving reduced photo-neutralisation efficiency coupled with energy recovery to deliver the optimum combination of wall plug efficiency and technological feasibility.
The key points can be summarised as follows:

- Both Fabry-Perot and direct drive cavities require further technical work to assess ultimate feasibility
- Integration into the NBI system / overall plant needs to be investigated and optimum performance identified
- cw lasers ($\lambda=1064\text{nm}$) of sufficient intensity (100W to 10kW depending on gain) need to be developed
- Suitable materials for the cavity and mirror production methods need to be identified
- The effect of the beam and associated plasma, neutron irradiation and operating conditions on the cavity structure need to be assessed
- Use of energy recovery can reduce the required neutralisation whilst improving the wall plug efficiency
- Beam geometry needs to be developed to match the photo-neutraliser cavity
- Energy recovery from large, high energy beams has not been demonstrated
- SIPHORE (SIngle gap PHOto-neutralizer energy REcovery injector), based on the photo-detachment of negative ions and energy recovery of unneutralised ions will address high finesse cavity issues using experience gained in the VIRGO project\(^\text{12}\).
- Ion source suitable for photo-neutraliser cavity.

**ECH high power multi stage depressed collectors**

A development programme is proposed to show the feasibility of depressed collectors for very high average power ($\geq 1$ MW) gyrotron oscillators for fusion applications. Initial calculations indicate that collection efficiencies exceeding 75% can be realized. Use of such a collector on a gyrotron with a base efficiency of 30% would increase the efficiency to over 60%. Such a dramatic increase in efficiency would result in substantial savings in the realization and operation of high power electron cyclotron heating systems.

Initially, a study of the feasibility of the use of collectors on megawatt gyrotrons with frequencies from 100 GHz to 240 GHz should be made, followed by a detailed design of a multi-stage depressed collector that should then be fabricated and tested in a second phase.

**HTS Magnets**

Development, analysis, fabrication and testing of small scale prototypes (2-3) of HTS cables for currents of the order of 5-10 kA. These tests should be conducted on existing facilities, at KIT, accepting the limitations of the facility FPI (He bath rather than forced circulation, approximate temperature control, short length zone at high B, shorter than the torsion pitch of the cables). The objective is to verify various concepts of cable, all bases of tapes of coated conductor (REBCO). The programme should also include experiments ad-hoc to verify the properties and the specific problems of each conductor concept to be developed. Milestone 2015: evaluate results (analysis and tests) and take account possible further industrial developments, to be expected outside of fusion for the field of HTS (improved tapes, or filaments).

\(^{12}\) F. Acernese et al., Status of Virgo, Class. Quantum Grav. 25 (2008) 114045
The fabrication of 1-2 full-size conductor prototypes is linked to the results of previous work and should be postponed towards the end of 2019-2020. Iteration on the sub-size may be required instead. The test of the full size should be conducted at conditions of interest in SULTAN with relative modest investments.

A large number of Associations involved in this development effort in view of the modest investment to be made in roadmap 2020 is seen as a potential risk to reduce efficiency of work.

**Advanced Divertor Heat Removal Technologies**

Alternative heat flux enhancement techniques exist or are being developed outside fusion that should be examined in more detail at DEMO relevant parameters. These include improvements to jet impingement via cascade or surface roughness modifications as well as novel coolant and material selections. These alternative heat transfer removal techniques may provide higher heat flux removal capability and/or be more robust under conditions which will degrade the performance of concepts currently under development. On the other hand, while there are certainly gains to be made by improving convective heat transfer through these enhancement techniques, they produce diminishing returns on performance as the conductive element of the heat transfer through the system from plasma to coolant becomes dominating.

A number of innovative divertor solutions, both in terms of materials and magnetic configurations, are going to be investigated as part of Mission 2. Screening of candidate solutions is planned as part of Mission 2 together with a preliminary evaluation of their attractiveness and reactor relevancy. Activities specified here pertain only to advanced He-cooled divertor solutions.

**Advanced Breeding Blanket Concepts (DCLL) (see also Mission 4)**

The specifications of the most urgent activities for the less aggressive DCLL design option is included as part of Mission 4 with emphasis of initial R&D and design activities on recognised potential showstoppers (design and performance of SiC flow channel insert (FCI), MHD experiments and simulations of flow control and heat transfer, and tritium behavior in lead lithium alloys. Once these issues are fully addressed design integration and blanket mock-up fabrication and testing on large-scale experiments can progress. Because a DEMO using a DCLL breeding blanket may require a refractory metal alloy heat exchanger, in order to keep the tritium inventory in the heat exchangers and the permeation from the PbLi cooling loops into the confinement building at manageable levels, very aggressive PbLi tritium extraction techniques (efficiency > 80%) and tritium diffusion barriers will have to be developed as included in Mission 4.

More advanced concepts of DCLL that rely on much higher temperature structural material should be explored as part of Mission 7. Eventually, even more aggressive blanket solutions, such as SCLL with exit temperatures up to 1100°C and power conversion efficiencies potentially up to 60% could, could be considered. This concept relies on a liquid metal that acts as a coolant and as a breeder and to avoid unacceptably high magneto-hydrodynamic (MHD) pressure drops, one needs to electrically insulate the flowing breeder from the conducting walls.
High temperature structural materials

The economics of fusion will depend to a large extent on efficient conversion to electricity of all of the emissions from the plasma. Recirculating power within the plant, such as that required for current drive, places additional demands on conversion efficiency. One of the most important requirements to achieve effective utilization of the helium cooling option appears to be the development of high temperature (> 600-650°C) structural materials that are compatible with impurities in helium. There are not many suitable materials candidates and in all cases they do not have an established industrial basis for reliable operation.

The R&D programme for the development of high temperature structural materials is included as part of Mission 3.

References
Annex 8. Mission 8 - Stellarator development

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Acknowledged contributions from: P. Castejon (CIEMAT), A. Dinklage (IPP), T. Klinger (IPP), F. Schauer (IPP), R. Wolf (IPP)

8.1 Short description

The Stellarator magnetic confinement configuration was proposed in 1950 by Lyman Spitzer and a number of devices were built in the 1950s and 1960s. The non-optimised magnetic configuration of earlier devices resulted in poor energy confinement and much better results were achieved using Tokamaks in the 1970s and 1980s. This led to the Stellarator line falling out of favour. Progress in understanding and computational capabilities allowed better Stellarator designs and a renewed interest in the 1990s, when a number of new devices were built. The key issue is the optimisation of the magnetic configuration in order to minimise particle transport, and the design of such optimised Stellarators was possible only in the 1980s.

Stellarator Magnetic confinement configurations

In a Stellarator the required helical twisting of the magnetic field lines needed for particle confinement is achieved solely with external coils. This results in a complex 3D geometry of the plasma and the magnetic field coils. A Stellarator configuration can be achieved with a continuous helical coil or a number of discrete coils producing a similar field (Torsatron Stellarator), a helical coil to confine the plasma, together with a pair of poloidal field coils to provide a vertical field (Heliotron Stellarator), a twist of the magnetic axis around a toroidal coil (Heliac Stellarator) and utilizing an optimised modular coil set designed to simultaneously achieve good energetic particle confinement and stability properties (Helias Stellarator).

The Helias Stellarator configuration is thought to be the most promising Stellarator concept for a power plant, with a modular engineering design and optimised magnetic field properties. The Wendelstein 7-X device is based on a five field-period Helias configuration.

Stellarators have intrinsic advantages relative to the Tokamak, mainly the inherent steady state capability and the absence of current driven instabilities, both connected to the fact that they do not need an internal plasma current to maintain the magnetic geometry. Present day stellarator experiments show higher density operation, well above the Greenwald density limit of Tokamaks, without hard MHD stability or equilibrium beta limits and so far, optimised Stellarator configurations operational boundaries are characterized by a degradation of the confinement, reducing the need of feedback control near operational limits. If confirmed on larger devices and understood from first principles, these will be big advantages over the tokamak in terms of reactor prospects. The freedom in designing the magnetic field of Stellarators allows the optimisation of neoclassical transport; fast particle losses; MHD stability (due to the inherent negative magnetic shear) and can accommodate different divertor
concepts. To improve the equilibrium boundaries, it is possible to design configurations with limited Shafranov shift (there are even examples of non-optimised Stellarator with negligible Shafranov shift, like TJ-II [8.1]). New computing capabilities open further opportunities for the further optimisation of these concepts [8.2, 8.3].

On the other hand, there is a larger performance uncertainty, when extrapolating to larger devices such as Stellarator Fusion Power Plant, due to a less mature physics basis when compared to Tokamaks. There is no universal confinement scaling for Stellarators due to the unknown balance between neoclassical and turbulent transport, a much larger variety in magnetic geometries and a smaller experimental data base in a more restricted range of operation parameters. Important sources of uncertainty in extrapolating the confinement of Stellarators to larger devices is the very unfavourable temperature dependence ($T^{7/2}$) of particle transport in the collisionless limit and the absence of validated physics-based turbulent transport models.

In addition, the 3D geometry means that the power Exhaust / Divertor solutions are complex, and these have so far been tested only to a limited extent. There is an unfavourable tendency for impurity accumulation which requires suitable regimes of operation. The 3D geometry of the Stellarator also poses a number of engineering challenges for the design and construction of the Super Conducnting Magnets, plasma facing components, blanket and shield. More specifically, the engineering challenges are related to the space availability for the internal components such as the blanket / shield required for tritium breeding, power extraction and protection of radiation damage sensitive superconductive magnets, the integration and performance issues of the divertor, coil spacing, bend radius, assembly and remote handling requirements.

8.2 Critical aspects for reactor application

Energy and Particle Confinement

Particle confinement is a central issue for Stellarator optimisation, including the confinement of high energy particles and related alpha-heating and the avoidance of impurity accumulation. The energy and particle confinement of larger optimised Stellarators are expected to be comparable to Tokamaks [8.4], approximately twice the ISS04 [8.5] International Stellarator Confinement Scaling compiled using the available data sets from the world’s larger non-optimized or partially-optimised Stellarators built prior to 2004. However, due to the unknown contribution of turbulent transport when moving closer to reactor conditions at reduced neoclassical transport, the confirmation of the energy and particle confinement of optimised Stellarators is one of the main objectives of the W7-X experiment.
Particle and Heat Exhaust

The problem of Particle and Heat Exhaust in Stellarators is also a key issue for Stellarators as it is for Tokamaks. Power Exhaust / Divertor solutions include concepts such as the Island Divertor foreseen for WENDELSTEIN 7-X (W7-X), the Helical Divertor in LHD or a flux expansion Divertor proposed for NCSX. The power handling capability of the different Divertor solutions varies, but it is reasonable to assume in general that to keep the thermal load below the technical limits of 5–10 MW/m² in a Stellarator Power Plant, up to 90% of the alpha particle power must be radiated; most likely through the achievement of plasma detachment conditions, as is the case in Tokamak DEMO conceptual designs. It is possible to design high aspect ratio (non-compact) Stellarators that maximize the wetted surface of the divertor tiles, hence decreasing the power flux. The high-density operation contributes to a higher Bremsstrahlung radiation and plasma detachment operation was demonstrated in W7-AS⁸ experiments. Nevertheless, adequate Stellarator steady state operation regimes which combines good energy confinement with acceptable heat loads, density control and impurity concentration with relevant high-Z wall materials still need to be developed.

Burning Plasma

ITER will provide important information on the development of Burning Plasmas and the related technologies, but there are a number of Stellarator specific issues that cannot be addressed in Tokamaks. Compared to Tokamaks, Stellarator reactors will operate at higher densities and lower temperatures. Consequently, the slowing down time of fast particles is significantly smaller in Stellarators and the fast particle beta is diminished accordingly. But the intrinsic 3D geometry significantly changes the Alfvén spectrum and related wave-particle interaction and the redistribution mechanisms with respect to those in a Tokamak, therefore, there is a large uncertainty on the behaviour of the alpha collective instabilities and related fast particle anomalous transport and losses.

Stellarators Devices

Wendelstein 7-X is a Helias Stellarator currently being built in Greifswald, Germany by the Max-Planck-Institut für Plasmaphysik (IPP), which will be completed by 2015.

Helically Symmetric eXperiment (HSX) is the first Stellarator to use a quasi-symmetric magnetic field, confirming the prediction that quasisymmetry would reduce transport.

The Large Helical Device (LHD) is a fusion research device in Toki, Gifu, Japan and is the largest superconducting Stellarator in the world, employing a Heliotron magnetic field configuration originally developed in Japan.

The National Compact Stellarator Experiment (NCSX) is a Compact Stellarator of which construction started at Princeton Plasma Physics Laboratory but was not completed.

TJ-II is a flexible Heliac installed at Spain's National Fusion Laboratory.
Engineering of the Super Conducting Magnets, Blanket and related Maintenance

The 3D geometry of the Stellarator poses a number of engineering challenges for the design and construction of the Super Conducting Magnets. However, the stored magnetic energy is considerably smaller than in a Tokamak, there are neither disruptions nor the need for a central solenoid (transformer) or poloidal coils. Nevertheless, for an upgraded HELIAS reactor with higher field and Nb3Sn superconductor, the technical challenges concerning the magnet system and cryostat can be compared to those of ITER [8.6].

Concerning the plasma facing components and blanket & shield, the same concepts as in a Tokamak can be applied, but have to be adapted to the specific plasma vessel geometry.

The integration and the remote maintenance of the internal components (and their coolant rooting pipes) required for the power exhaust (divertor), neutron shielding, power extraction and tritium breeding (blanket) are expected to be more challenging than in Tokamaks because of the complex geometry, making it necessary to have a larger variety of individually shaped blanket and divertor elements. There is, therefore, the need to work on the development of the required remote maintenance strategies for Stellarator configurations. For example, for a HELIAS reactor, the in-vessel component maintenance will have to be performed either through ports with a cross section of $\approx 6 \times 2$ m$^2$, direct vertical access could be possible via at least one port per module, which is comparable to corresponding Tokamak ports. In-vessel rail systems need to follow helical paths. Alternatively or in addition, a separation of the torus for quick exchange of large parts has been proposed [8.7].

As in a Tokamak, the engineering complexity must be balanced against the high-availability requirements and the components life time needed for a Stellarator Fusion Power Plant in a nuclear environment. Stellarator reactors share with Tokamaks all the issues related with material neutron damage, remote maintenance and tritium fuel cycle.

8.3 Level of readiness

Stellarator experiments are at least one generation behind Tokamaks since an optimization of the 3D magnetic field structure is required to mitigate the classical Stellarator’s show-stoppers [8.7], related to insufficient particle (including fast particles) and energy confinement.

The better confinement properties of optimised Stellarators have been observed in small devices such as W7-AS [8.10] and HSX [8.11] and also in the largest existent device, LHD, where the beneficial effect of an optimised magnetic field was demonstrated by an inward shift of the configuration [8.12], but these will have to be confirmed in the W7-X.

The qualification of the most feasible Divertor concept is part of the ongoing Stellarator program. The island Divertor concept has been investigated on W7-AS with up to 90% radiative power fraction in high-density H-modes, but at high collisionality. The helical Divertor is being assessed in LHD and flux-expansion divertors are yet to be investigated.
Stable operation at reactor relevant (thermal) beta values has been reached in W7-AS and LHD, but at low values of the magnetic field.

8.4 Main risks and risk mitigation strategies

Energy and Particle Confinement
With regard to the reactor physics basis, the validation of the energy and particle confinement of optimised Stellarators is one of the main objectives of the W7-X experiment. The neoclassic part of the physics underlying these optimisation goals is considered to be well understood and it is quite likely that the achievement of most of these goals concerning neoclassical behaviour will already be verified during the early years of W7-X operation. However, due to the power limitation in the early phase of operation, it is uncertain whether sufficiently high beta values will be reached in this phase of operation and the role of turbulent transport fully assessed.

Particle and Heat Exhaust
Even after all optimisation goals of the W7-X experiment have been demonstrated, advanced Stellarators of this type will only become attractive reactor candidates if they can maintain high-performance (high-β, low-ρ*, low-ν*) steady-state plasmas (discharges of several minutes. This is required to surpass all physical and technical time constants of relevance) with viable Divertor performance (high radiated-power fraction with at least partial detachment) without loss of density control and without impurity accumulation. As a superconducting device with 10 MW continuous (over 30 minutes) of ECRH and a high-heat-flux Divertor capable of handling power loads of 10 MW/m² (following initial operation with an inertially cooled divertor), the W7-X experiment will possess the technical capabilities necessary to achieve such integrated reactor-relevant scenarios. Therefore, since long-pulse operation in Stellarators is largely unknown, the investigation of steady-state plasmas will be the main objective of the second phase of the W7-X experimental programme.

Burning Plasma Physics
The experience from ITER on Burning Plasmas physics will be valuable in many areas but there are also topics for which the Tokamak results will be of limited use to Stellarators. W7-X cannot simultaneously access integrated scenarios with values of ν*, β, and ρ* equal to those of a reactor. Hence, Stellarator specific aspects of the physics of burning plasmas (α-particle heating and exhaust, α-particle effects on stability) are not assessable in W7-X.

Therefore, a device will be required to develop the required Burning Plasmas scenarios, in particular, regarding the Stellarator specific α-particle dynamics and fast particle collective effects. This next step HELIAS Burning Plasma device will establish the physics basis for a Stellarator reactor [8.13].
Engineering of the Super Conducting Magnets, Blanket and related Maintenance
A next step HELIAS Burning Plasma device is also required to demonstrate a field configuration compatible with the optimisation required for adequate plasma confinement, together with the engineering constraints posed by reactor compatible nuclear shielding and remote maintenance. This Burning Plasma Stellarator experiment could have a limited activation < 5 dpa, but together with the results from the Tokamak DEMO in terms of low activation materials and blanket developments, provides the required knowledge for the construction of a Stellarator Fusion Power Plant. It remains to be studied what the actual smallest size of a machine fulfilling the requirement of dominant α-heating will be, but below, we give an idea what it could look like.

Conceptual studies for burning plasma HELIAS experiments
(Wobig et al in Nuclear Fusion 43 (2003) 889.)

HELIAS ignition experiment is an upgraded version of the Wendelstein 7-X experiment. The magnetic configuration is a four-period HELIAS configuration, major radius 18 m, plasma radius 2.0 m and B = 4.5 T, which allows one to use NbTi superconductors at 4.5 K. The design criteria of the ignition experiment HSR 4/18i are the following: The experiment should demonstrate a safe and reliable route to ignition; self-sustained burn without external heating; steady-state operation during several hundred seconds; reliability of the technical components and tritium breeding in a test blanket. The plasma parameters of the ignition experiment are: peak density 2–3 x 10^{20}m^{-3}, peak temperature 11–15 keV, average beta 3.6% and fusion power 1500–1700MW. The highest mechanical stresses in the steel casing are below 750MPa, inside the winding pack the non-linear behaviour of the imbedding has been taken into account leading to a maximum compression stress of 70MPa in the windings.

8.5 Gaps
- The values of the dimensionless parameters (high-β, low-ρ*, low-ν*) expected for W7-X will typically differ from those of a reactor plasma by factors between 1.5 and 3 from 1D neo-classic modelling considerations. However, these simulations predict much better confinement than the International Scaling Laws. Therefore, a direct extrapolation of W7-X to a Stellarator DEMO / Burning plasma size device would, not be without risks, although the degree of risk would certainly be reduced by significant progress in the theoretical models used to explain (and subsequently predict) W7-X and ITER results.

- It is not clear if the island Divertor will show performance adequate to extrapolate to a burning plasma Stellarator experiment. If this should not be the case, investigation of Divertor solutions different from the Island Divertor requires new experiments, since these cannot be accessed in W7-X.

- Optimised magnetic field configurations required for alpha particle confinement requires new experiments since this cannot be achieved in W7-X, preferable a burning plasma experiment.
• Stellarator specific power plant technologies such as the Super Conduccing Magnets, plasma facing components, blanket, shield and related maintenance strategy will have to be developed and an integrated and tested in a next step device.

The decision on an intermediate step Burning Plasma device and whether the remaining physical issues might also be adequately addressed in a smaller device cannot be assessed until W7-X is fully exploited to ensure sufficient predictive capability for a next step device.

8.6 Opportunities for Stellarator contributions to research for ITER

The Stellarator programme should exploit synergies with Tokamaks wherever meaningful. Due to their inherent 3D structure, Stellarators will be a crucial element for testing the predictive capability of plasma models. In addition, several 3D effects important for Tokamaks will benefit from understanding gained on Stellarators, but also from the tools being developed to adequately describe Stellarator physics. An important area is 3D effects on divertor and edge physics, where tokamak research has recently entered with the application of helical fields for ELM control.

Another important aspect is the possibility to run very long pulses (up to 30 min after the completion in 2019) in W7-X. This will allow PWI studies on these timescales, which might be needed to better understand the long-term dynamics of the wall. On the technology side, developing and testing the elements needed to run the machine in steady state will bring a lot of experience for high heat flux component development, but also for operating a machine under these conditions (e.g. coping with ECRH stray radiation in steady state or protecting the first wall by monitoring it in real time and reacting to local overheating).

8.7. Description of the major Work Packages.

8.7.1 W7-X exploitation

The main programmatic priority is the study of the energy and particle confinement of optimised Stellarators and the qualification of a Divertor under steady-state conditions via the scientific exploitation of the Wendelstein W7-X experiment.

The W7-X programme is divided into three parts: The plan for Operational Phase 1 (2015 – 2017) is part of the present project plan of W7-X [8.14].

<table>
<thead>
<tr>
<th>W7-X Operational Phase 1</th>
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<tr>
<td>W7-X will start operations in 2015 with 8MW of heating power and an inertially cooled test-divertor unit (TDU). Only the vacuum vessel wall and a few wall elements are water-cooled. The heating system consists of 8MW electron-cyclotron resonance (ECR) heating and 8MW neutral beam injection (NBI) heating. With such a configuration, discharges of 5-10s pulse length with up to a total of 8MW heating (either ECR or NBI or combined) can be achieved.</td>
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Operational Phase 2 (>2019) is based on the medium term plan for the completion of W7-X which aims at establishing the full steady state capability with actively cooled PFCs.

Operational Phase 3 (>2023) covers remaining items which, based on present-day knowledge, need to be investigated for the demonstration of the reactor capability of the Stellarator concept.

8.7.2 Theory Development and Modelling/ Stellarator optimization

The complexity and diversity of the Stellarator configurations means that it is not feasible to investigate experimentally all possible options, therefore, the search for the best configuration relies on theory and modelling to a great extent. An optimization with respect to the physics together with the engineering constraints must capitalise on the advances in understanding and computational capabilities. The engineering constraints will become more important as the devices approach reactor conditions. Therefore, in parallel to the exploitation of W7-X, a significant effort must be devoted to model development and further Stellarator Optimisation. For example, further optimisation of the physics properties could also include turbulent transport ingredients, in addition to the neo-classic optimisation and possibility give less weight to other aspects now deemed less crucial such as MHD stability. In particular, relaxing some of the stability optimisation criteria could be performed together with the optimisation of the coil complexity, space for blanket / shield and maintainability.

References

[8.2] A. Gómez-Iglesias et al., Computing and Informatics 31, 31 (2012),
Annex 9. Education and training needs in support of the EU fusion roadmap.

Authors: Niek Lopes Cardozo (FOM), Francesco Romanelli (EFDA CSU), Christian Schoenfelder (AREVA), Sandor Zoletnik (KFKI)

9.1 The challenge: from a research program to an industrial development program

The realization of the roadmap calls for fundamental changes in the composition of the workforce in fusion, and therefore in the Human Resource (HR) strategy. In the coming decades the development of fusion goes from a science-driven, lab-based exercise to an industry-driven and technology-driven program. Moreover, fusion makes the transition from essentially non-nuclear technology to nuclear technology. ITER is the first step in this triple transition, which is also driven by the development of DEMO.

Whereas this transition in the workforce will be gradual, the changes in the training programme and in particular in the education programmes that feed workforce, must anticipate the required changes in the workforce by 5 to 10 years. Therefore, a forward-looking action plan on Education and Training has to be implemented already now.

9.2 The fusion Education and Training in Europe is well-organised

The fusion community in Europe is well set up to implement the necessary HR-strategy. There are three actions or organizations that together cover everything concerning education and training at the European level. These are:

- For education, at Master and PhD-level: Fusenet (the European Fusion Education Network), the umbrella organization under which all fusion education, from Master (and earlier) to PhD, is coordinated. Members of Fusenet are universities with a fusion curriculum, fusion research labs as well as industries that are involved in fusion. Fusenet has developed joint criteria for the European fusion master and doctorate, coordinates the summer schools, acts as matchmaker for internships between students and industry, and organizes joint educational events and tools. The possibilities in fusion education are made accessible to students via the Fusenet website www.fusenet.eu. Fusenet was initiated with an FP7 grant and is now an independent legal entity.

- For training: EFDA (European Fusion Development Agreement) is the organization under Euratom in which all European fusion laboratories collaborate to carry out the joint European fusion research program. EFDA runs the Goal Oriented Training (GOT) programme, which selects candidates and training proposals and financially (partially) supports a 3-year training programme. GOT delivers ~40 trained engineers per year in targeted fusion technology areas. EFDA also runs the Fusion Fellowship programme, which promotes excellence among young researchers by competitively making available ~10 two-year post-doctoral grants per year.

- The Fusion Industry and Innovation Forum (FIIF), formed by industrial organizations that are involved in the development of fusion energy, helps to integrate industry in the development of DEMO and supports the training and
education activities. As an increasing number of fusion-related jobs will be in industry, the FIIF is essential in the formulation of the required competences and needs, which can then be provided by Fusenet and EFDA.

EFDA, Fusenet and FIIF are well harmonized; there is regular contact at board level as well as informal contact at working level.

Fusion for Energy (F4E), as procurement agency, presently does not have a direct role in education and training, but it does provide input to the organisations above concerning the priorities in the training programmes.

9.3 What needs to be done in Education and Training.

The research field presently counts about 2300 professionals (FTE), with a total work force of 4000. The expectation is that with proper concentration and direction of the work forces, the number of research staff will remain approximately constant in the transition to the ITER and then DEMO-era. But during this period, a shift in the competence portfolio from physics to (nuclear) technology, materials and engineering must be effectuated, which calls for an effective management of the education of the new fusion researchers at the master and PhD-level. This can be managed effectively between Fusenet, EFDA and FIIF.

In industry, ITER represents new work for an estimated total 5000 FTE (professional and technical) in a broad variety of disciplines. Moving towards DEMO, this number would have to roughly double, to 10000 FTE.

In industry, we distinguish the non-fusion specific nuclear technology, and the fusion-specific (nuclear) technology. In the non-fusion specific nuclear technology, the competences needed in fusion are very similar to those in fission, and the demands of the fusion field can be accommodated within a small increase of the manpower in the much larger fission field. The fusion-specific nuclear technology, on the other hand, concerning e.g. system engineering, plant control, materials, tritium generation and fuel cycle, calls for a proactive human resource management strategy.

A workforce of some 2500 professionals in research and an equivalent but growing number in industry, with an average professional life of 25 years, requires about 200 new professionals be injected into the system every year. Statistics show that of the students who finish a PhD in fusion, about one third make a career in fusion, others mostly going into industry, finance or non-fusion related academic positions. Similarly, at the MSc-level only about 30% of students with a fusion specialisation – which is now available in several countries, under Fusenet coordination – will continue into a PhD. This is considered to be a healthy fraction, which provides the fusion community the choice of the best candidates, whereas also the student, after having done a fusion master, may decide to pursue a career in another direction.

Thus, a healthy system should aim at some 300 PhD students and an equivalent number of engineers (either PhD students or trainees) active in the fusion, with an appropriate spread over topics in fusion engineering and physics. The actual number may be lower depending on the need for fusion-PhDs in industry. The scope for specialized fusion masters well exceeds 500 per year. These numbers have to be taken with the consideration that the fusion system will also take up professionals with a non-fusion background, in particular if they bring a specialisation that is needed in the program, who would then have to be trained in the fusion-specific competences.

In summary, the upcoming transitions in the fusion programme call for a proactive HR-strategy. This calls for a thorough analysis, based on new questionnaires covering
both research and industry, and regular monitoring of the development of composition of the workforce.

Based on available information, we arrive at the following breakdown of the needs for Education and Training in the coming period.

9.4 Breakdown of the Education and Training needs: six classes

To maintain the research workforce at the required strength, while at the same time managing the gradual shift towards more engineering competences, EFDA and Fusenet are well set up to attract the required number of students and coordinate their education in such a way that the competence profile of the workforce adapted to the needs. The financial needs for this coordinated E&T programme for Horizon 2020 are

A. Support to Fusenet to coordinate education, support joint events and mobility of students, support joint materials etc. In order to achieve the required number of fusion MSc’s that can enter the system, fusion must be made very attractive to students. Mobility, joint events, good materials and support to interesting internships are important in this respect. Fusenet will also support specific actions to improve the integration of countries that do not (yet) have a strong national fusion program, in particular in the new Member States. To allow Fusenet to carry out these coordination and support actions, a budget of 500 k€/year is required.

B. Direct support to PhD projects in the fusion institutions or universities. The total number of PhD students in fusion is about 300 and may have to progressively grow to 600, depending on the need for PhD’s in industry, which has to be assessed. Also in this action special attention is paid to the integration of fusion projects in the new Member States, which calls for additional mobility support of PhD students to allow them to do part of their research in one of the established fusion research centers, and some support for their home thesis supervisors to help building up the local fusion environment. In the coming period, the support to PhD-projects needs to be supported with a budget of 9 M€/year.

C. A follow-up of the EFDA Goal Oriented Training programme: training of (mostly) engineers that enter the fusion research field for a total of 30 per year. Full cost: 13.5 M€/year.

D. Fellowships (two-year) awarded on an individual basis for a total of about 10 per year. Full cost 1.5M€/year.

Next to that, there is the need for two new types of training programmes that do not yet exist:

E. Condensed, in-company training of engineers who work in an industrial environment and are involved in fusion-related tasks, so that they are fully

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13 The dedicated action to stimulate and support participation in the fusion program by researchers and their PhD students in the new Member States, which involves support to MSc students (mostly to support mobility and allow them to participate in specialized fusion education) and to PhD-students (mostly to support mobility to allow them to carry out part of their research project in one of the established fusion research centers, and a modest support for their supervisors), is estimated to call for a budget of about 100 k€/year, and can be accommodated by Fusenet and EFDA.
trained for the regular engineering aspects of their task but need to get a background in fusion-specific engineering. This training would (partially) be paid by the company and should be tailored to the host’s specific needs. It is quite possible that such training could be offered commercially by specialised training centres, but this needs to be followed carefully.

F. Specific training of professionals and technicians who are already specialized in fusion, but need to acquire knowledge of technologies and standards that are associated with the transition of fusion to a fully nuclear technology. This could be in the field of licensing, regulation, (nuclear) safety, balance of plant etc. As noted above, all generic nuclear technology training and education is well covered by the fission field, to which fusion is a small perturbation. So EFDA, in collaboration with FIIF and Fusenet, will have to carefully assess the needs for new fusion-specific technology and organize the training based on this assessment. The funding for this training would have to come out of the EU fusion budget.

9.5 Facilities.
Education at the master level needs to be supported by hands-on experience of the students, which can be acquired with relatively simple laboratory experiments and internships in research groups and/or industry. In this phase it is important that the students can actually steer their experiments, which is generally not possible at the large devices.

Training, on the other hand, will in general call for the most realistic environments, including in particular the nuclear and safety aspects. This is typically found at the very largest devices in operation. JET is in the best position to be used as the training facility for ITER by the various ITER Parties. Appropriate provisions need to be in place to ensure that European students can benefit at best from this opportunity.
10.1 Introduction

The programme outlined in the eight Missions aims at a project-oriented approach focusing on the goal of achieving Fusion Energy on a direct path. However, in many areas of Fusion Physics, a first-principles understanding has not yet been achieved to the point where quantitative predictions can be made. While in the project-oriented approach, this is often dealt with in a more empirical, scaling-type approach, a vigorous underlying programme should be continued in the EURATOM member states to progress physics understanding. Such a programme should be supported purely on the basis of scientific excellence and should involve both theory and experiment. As an indicative number, it is proposed that about 10% of the resources be devoted to this programme.

Due to the inherent nonlinear nature of plasma physics, quantitative understanding often requires addressing problems on large computers, and computational physics is a discipline that is of great importance for fusion plasma physics. Hence, High Performance Computing (HPC) and related supporting activities, both in support of basic research and in support of the modelling effort under the project-oriented activities, should be adequately funded in the programme.

On the experimental side, we note that Basic Research is well suited also for devices that lack the elements to contribute to the Missions targeted at ITER and DEMO in an integrated manner. Rather, fundamental experiments, aimed e.g. at validation of a particular model, can be easier implemented in smaller, flexible devices, that may even be equipped with dedicated tools not applicable in larger machines.

10.2 Main Topics of Basic Research

In the proposed Roadmap, basic research is meant to address several areas in which fundamental understanding is required to reliably predict the integrated plasma behaviour in ITER and DEMO from first principles. Hence, it addresses ingredients necessary to reach a validated 'numerical tokamak' (and a 'numerical stellarator' as well). Such an instrument must in the end be at hand to allow a point design of a fusion reactor. Areas in which basic research will be important in addition to the programmatic approach given in the Missions are the prediction of the kinetic plasma profiles, i.e. understanding of turbulent transport as well as the interaction of fast particles with plasma stability, on the micro and the macro level. Better understanding of the basic underlying physics mechanisms will also help to devise active control means for the confinement of fusion plasmas.

10.2.1 Edge plasma physics

The biggest challenge to achieve predictive capability for the kinetic profiles lies in the understanding of the plasma edge region, in particular the physics of the H-mode and its pedestal. This is at present not at all understood from basic principles. Pedestal width and height for both density and temperature cannot be predicted today from any
ab initio theory. Likewise, the H-mode power threshold scaling is not understood and there are well-documented deviations from the scaling at the low and high density ends. Furthermore, ideal MHD edge stability is intimately linked to the pedestal parameters via Edge Localised Modes (ELMs). While linear stability analysis for ELM onset has made progress, nonlinear ELM modelling that would predict ELM affected zone and ELM losses is still in its infancy. Recently, also the effects of 3-d fields on the plasma, in particular the edge, have gained more importance and basic research should cover this aspect as well. Finally, dimensionless scalings fail to explain the Greenwald limit, indicating that we are missing an important part of the physics. Since proximity to the Greenwald limit is a 'reserve' to recover fusion power if other scalings turn out to be too optimistic, this is also an important area.

Hence, we propose the following topics in the area of edge plasma physics as foci of basic research aimed at supporting the Missions.

**L-H transition:** while a basic paradigm to explain the L-H transition in terms of turbulence suppression by sheared rotation has been established, no ab-initio code exists that simulates an L-H transition without forcing the boundary conditions (such as background radial electric field). Apart from the strong academic challenge that this fact presents, it also means that we know empirically that the linear density scaling of the L-H and H-L transition is only valid in an intermediate density range (and strongly deviates from linear outside this interval), but we have no predictive capability where this density window lies in future large devices.

**Properties of the H-mode pedestal:** The value of the edge pedestal temperature, which is crucial in determining the central value of the temperature due to profile stiffness, and also the density pedestal parameters are not at all understood from first principles. It is not even clear how to scale them, since atomic physics may play a role, similarity experiments are very tricky to interpret. The theoretical understanding is further complicated by the close link to the ELMs, which introduce nonlinear MHD that has to be modelled correctly to understand the impact on the pedestal. Here, understanding is also important in view of ELM mitigation (see also Mission 1).

**Power fluxes into the SOL:** Connected to the understanding of the edge is also the spatial and temporal structure of the power flux out of the plasma. This has to be modelled correctly in order to understand completely the complex link between plasma wall interaction and operational scenarios, as dealt with in Mission 2. While the development of integrated models for divertor performance is part of Mission 2, we note here that there are still fundamental issues to be studied in this area such as a correct description of parallel heat transport in a long mean-free-path regime. Also, present divertor modelling usually uses empirically determined perpendicular transport coefficients while finally, ab-initio SOL turbulence modelling should be included. This will be crucial to predict the power fall-off length, but also to understand the fluxes to the wall in the main chamber and to predict their erosion.

**Greenwald limit:** The Greenwald limit, although empirically very well documented in tokamaks and RFPs, is still not understood in terms of physics. A particular concern when extrapolating operational scenarios to ITER is that the Greenwald limit cannot be expressed in the dimensionless parameters $\rho, \beta, v^*$ alone, so that the ITER edge will be close to $n/n_{GW} \sim 1$, but at much lower collisionality than present day experiments, making an extrapolation of our present empirical knowledge of pedestal quantities very difficult. Understanding of the Greenwald limit is hence vital to obtain
real predictive capability. A credible model should also explain why stellarators appear not to exhibit a density limit of similar nature.

10.2.2 Core physics

First principles understanding for the core plasma region is more complete, where modelling is capable of predicting reasonably well the core temperatures for given boundary (i.e. pedestal top) temperatures. However, conditions for formation and confinement improvement due to Internal Transport Barriers (ITBs) are largely not understood and also the understanding of momentum transport is still weak. Concerning core MHD stability, the main limitations to fusion performance are Neoclassical Tearing Modes NTMs (for conventional scenarios) and Resistive Wall Modes RWMs (for advanced scenarios). In both areas, we still lack sufficient physics understanding to quantitatively characterise their occurrence and the schemes that can be applied for their control.

Hence, we propose the following topics in the area of core physics as foci of basic research aimed at supporting the Missions:

**Momentum transport:** This is the area of core transport with lowest predictive capability. Although rotation does not directly affect fusion performance, it enters into many areas indirectly, such as heat and particle transport by allowing barrier formation, or RWM stability by providing rotational stabilisation. In ITER, contrary to many present day devices, intrinsic rotation will probably dominate over external torque e.g. by NBI and hence, we need to have predictive capability of how large the expected intrinsic rotation will be, and the expected impact on confinement and stability.

**ITBs:** The different kinetic profiles can all show ITBs, with the mechanisms for their formation and the scaling of the associated confinement improvement not necessarily being identical. We therefore need progress in this area towards a predictive capability. We note here that improved core confinement not associated with ITBs, such as e.g. in the 'hybrid' regime, also still awaits a first principles based understanding.

**NTMs:** They constitute the practical $\beta$-limit to discharges with monotonic q-profile. While a $\rho_p$-scaling for their onset has been established empirically, the precise nature of the seeding physics, which determines the scaling of the onset conditions in detail, is not yet available. The same unknown physics at small island width determines the power requirements for NTM stabilisation by ECCD, which still has large uncertainties when extrapolated to ITER. No first principle calculation exists of NTM evolution for the full island width range.

**RWMs:** They pose the ultimate limit to $\beta$ in reversed shear scenarios. Recent experimental progress highlights the possibility of rotational stabilisation of RWMs at very low Mach number, which is considered to be related to kinetic effects (see also section 2.3). On the other hand, future reactor-grade devices will have low intrinsic rotation and limited means for external momentum input, so that predictive capability is urgently needed.

10.2.3 Interaction of fast particles with MHD stability

Since burning plasma physics is the main physics mission of ITER, it will be important to prepare the experiments conducted from ~2025 on such that maximum scientific gain can be made. The understanding gained on ITER is expected to allow
proceeding with confidence to DEMO. In the area of the interaction of fast particles with MHD stability, the basic mechanisms of stabilisation by orbit effects or destabilisation by the fast particle pressure have been identified, but the nonlinear evolution cannot be always predicted from first principles calculations. This, however, ultimately decides the saturation amplitude of fast particle affected MHD modes and hence the level of additional transport of fast particles in the presence of a large population of fusion $\alpha$-particles. Thus, Basic Research should also address this area in order to prepare the respective models for the final validation on ITER.

Hence, we propose the following topics in the area of the interaction of fast particles with MHD stability as foci of basic research aimed at supporting the Missions:

**Contribution of fast particles to global $\beta$-limits:** The contribution of fast particles to the $\beta$-limit, both global and to $\beta_{fast}$, is twofold. First, the fast particle pressure will contribute to the global MHD limit and this can, for an isotropic fast particle distribution (such as $\alpha$-heating), be viewed as setting the global MHD limit to $\beta_{\text{total}}$ instead of $\beta_{\text{thermal}}$. However, there is also the stabilising effect of a fast particle population on global MHD modes, and this can be quite strong also for the modes responsible for global $\beta$-limits, such as the RWM. The balance between stabilising and destabilising effects on these modes will finally determine the global $\beta$-limit, so that quantitative understanding is needed for both effects.

**Damping of fast particle driven modes:** while linear stability of fast particle driven Alfvénic modes is quite advanced, a quantitative understanding of the damping is lacking, partly because there are several competing effects that have to be taken into account. However, the balance of drive and damping will finally determine the saturation level and hence, the transport of fast particles by these modes. Hence, this must also be understood on a quantitative level, which again opens a wide range of research possibilities for both theory and experiment.

**Fast particle interaction with turbulence:** this area covers two aspects, firstly the transport of fast particles by ambient plasma turbulence. Although this should be low because of the gyro-averaging effect, recent experiments on NBCD indicate that there could be a finite effect and this should be understood better from first principles. Second, the inverse process, i.e. stabilisation or destabilisation of micro-instabilities by fast particles, is usually deemed unimportant in present day experiments due to the fact that fast particles have much larger orbits than the eddy size of turbulence, but in the ITER and DEMO parameter range, this will change and fast particles could also play a role on the scale of turbulence. It will be important to investigate these mechanisms to the extent possible already before ITER operation as to best prepare the analysis.

**10.2.4 Materials**

Materials degradation from radiation starts with collisions between high-energy neutrons and lattice atoms and then develops into radiation-induced microstructures that degrade mechanical properties, dimensional stability, thermo-physical properties, and resistance to environmental attack. This is true in both fission and fusion systems. Although there is a vast number of physical changes induced by irradiation, some phenomena are either already understood at a sufficient (albeit rudimentary) level for evaluating feasibility issues, or are not critical determining factors in establishing the feasibility of using materials in irradiation environments. Conversely, there are several fundamental radiation effects phenomena that critically need improved
understanding in order to assess candidate material options. These include defect production and evolution in alloys (as opposed to pure metals), coupled solute-defect diffusion processes in concentrated alloys, microstructural evolution of coupled point defect sinks (e.g., voids and dislocations), and the effects of point defect clusters, surface active transmutant impurities and helium on localized deformation and fracture toughness.

Theory and modeling can be used to develop understanding of known critical physical phenomena, and a set of computational tools that adequately describes radiation events and subsequent microstructural development and provides from this microstructure an accurate description of property changes in engineering materials would be a tremendously valuable tool for system designers. While developing these computational tools into a truly predictive capability is a difficult and time-consuming challenge, the knowledge gained from development of the individual model components will provide valuable insight to any on-going materials design effort. These models cannot be developed without a parallel set of validation experiments. The interplay between new experimental data and advanced theory or computational tools is critical to the development of accurate radiation response models.

10.3 The role of integrated modelling

While the individual elements mentioned above can often be tackled in separated efforts, integrated modelling towards a 'numerical tokamak' requires a large coordinated effort between model generation, implication, validation and integration. In particular, the following priorities can be identified:

- Building confidence in the predictive capability of the integrated modelling tools - to meet this goal an exhaustive programme in numerical code verification and experimental physics validation involving a broad range of expertises need to be supported over a range of devices.

- Providing broad access to the integrated modelling tools both as a production level facility and as a test bed for new developments - to meet this goal the supporting code integration framework and computational resources need to be established and made available. The access and use of a centrally maintained suite of modelling tools provide also the necessary basis for experimental validation over the full range of facilities (including spherical tokamaks and stellarators).

- Supporting the development of modular sets of codes and modules covering a range of physics areas at different physics fidelity targeting the needs of integrated modelling – the transition in terms of robustness and performance needed in going from research to production level codes need to be supported. In addition, the research needs stemming from coupled multi-scale physics as well as the challenge of bridging multiple numerical techniques within a single framework provides a research area of its own.
Annex 11. Resources and implementation scheme.

*Note:* This Annex is being revised following the evolution of the discussion on the budget for Horizon 2020 and on the new Consortium implementation scheme.
Annex 12. Industrial involvement in the realization of fusion as an energy source.

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12.1 Introduction
In the coming decades the development of fusion will move from a science-driven, lab-based exercise to an industry-driven and technology-driven program. This requires that industry progressively shifts its role from that of provider of high-tech components to that of driver of the fusion development. This will be a step wise process with industry, possibly in the form of consortia including research laboratories and universities, working closely with research partners.

A timely market penetration of fusion energy requires that DEMO be the only step between ITER and the commercial fusion power plant. The “Early DEMO” approach, advocated in the present roadmap, minimizes the time for the production of electricity from fusion but, in order to maintain DEMO as a single step, it has to promote the necessary innovation, both in industry and in research, to ensure that industry is able to take full responsibility for the commercial fusion power plant after successful DEMO operation. For this reason, DEMO cannot be defined and designed by research laboratories alone, but requires the full involvement of industry in all technological and systems aspects of the design.

Innovation in industry includes the development of enabling technologies and the selection of cost-effective technical solutions. This requires an early involvement of industry as a full partner in a number of key areas: utilities, in specific areas such as fuel cycle, and system engineering companies for the general layout, safety and integration, manufacturers for the components with the largest capital investments and for the development of advanced materials and technologies.

Innovation in research includes the investigation of advanced concepts in the critical areas (e.g., heat exhaust, tritium breeding, magnets, remote handling, etc.). Cutting-edge research must be promoted and research activities that have exhausted their innovation potential should be closed. Throughout the conceptual design phase there should be a systematic narrowing of alternative design solutions. However the exploitation of the full innovation potential of fusion research should be strongly pursued and strengthened through the implementation of the Technology Transfer project currently under EFDA. In this respect the fusion laboratories, universities and industry must collaborate to provide a win-win situation delivering the objectives of the Innovation Union.
Industry is present in all the Roadmap Missions, but in this Annex its strategic involvement in Mission 3 (material development), Mission 6 (DEMO design) and Mission 7 (low cost of electricity) is emphasized.

12.2 Present involvement of European industry in fusion.
European industry is presently involved with the ITER construction that represents a turnover of ~6 B€ over ~10 years and involves ~5000 full-time equivalent. In addition, the completion of the construction of W7-X and of the Broader Approach projects (JT-60SA and IFMIF EVEDA) are important examples of industrial participation in the fusion programme. A list of the main areas of involvement of European industry in ITER component supply is given below:
- Nuclear buildings
- Superconducting magnets
- Vacuum vessel
- High heat-flux components
- Shielding blanket
- Millimetre wave sources
- Diagnostics
- Cryoplant system
- Tritium plant

In the short term, participation in ITER construction is likely to produce know-how that can be transferable to other industrial applications. In the long term it will be necessary to implement an industrial approach to the transfer of knowledge from the construction and operation of ITER to Industry that will be required for the design of DEMO.

The DEMO design and R&D will, in addition, involve industrial competence in the area of the balance of plant.

Some of the competences in the fields listed above are non-fusion specific and will remain available after the end of the ITER construction independently of the investment in fusion. However, other competences are fusion specific and, to avoid a rapid loss of expertise at the end of the ITER construction, they will need to be supported to bridge the gap with the start of the DEMO EDA. For these fusion specific technologies it will also be necessary to provide training for industrial competences (See Annex 9).

12.3 Involvement of industry in the DEMO definition and design (Mission 6)
The precise definition of the DEMO goals needs to involve industry in order to ensure that all the areas that are critical for market penetration are properly addressed. Industry must also ensure that the design choices are soundly based on the existing industrial capacity in Europe. DEMO will be the first opportunity for the utilities and supply industry to assess the economic, social, and environmental potential of fusion.

The aspects that will likely determine whether industry is prepared to support the commercial deployment of fusion are:
- A well-defined licensing process and the minimization of waste management requirements, implying a good public acceptance of fusion;
- Capital investment in line with other technologies for base-load electricity generation;
- A prospect of acceptable costs through sufficiently high system reliability and overall plant availability.

These aspects will need to be specifically addressed with the relevant stakeholders. In particular, licensing and waste management will require the involvement of safety authorities and regulators.

The participation of industry could evolve as described below.

**DEMO definition and Conceptual Design Activity (CDA) (2012-2020).** During this phase EFDA (or the organization that will be in place after 2013) will progress the definition of the DEMO design goals and will execute the CDA in consultation with the Fusion Industry Innovation Forum and the participation of industrial experts. A review of the DEMO concept definition and of the CDA results should be made, involving the relevant stakeholders. This review should involve utilities and system engineering companies as for Gen IV programme to ensure that before launching engineering design activities, there is full acceptance of the proposal by these stakeholders. The implementation of this phase will require the use of individual contracts for industrial experts to work within the EFDA organisation in the specific areas where industrial knowledge and practices are required. The organisation or project team should specifically include a systems engineering group. In addition, in order to broaden the industrial landscape and prepare for strong consortia in the EDA phase, the skills and knowledge of both universities and industry should be tapped to provide innovative solutions to difficult problems.

**DEMO Engineering Design Activity (EDA) (2021-2030).** During this phase consortia of industries and laboratories should be formed for the necessary design and R&D activity. Realistic cost estimates will be developed and independently reviewed to allow the selection of the most appropriate design options and to ensure that the cost remain within the given budget.

Specific areas where industry involvement is considered critical are:
- Technical solution for the major DEMO components with the lowest cost and simplest manufacturing routes (e.g. magnet simplification); standardisation of parts.
- Safety Engineering & Licensing Support
- Reliability / Dependability Engineering
- Digital Engineering (i.e. CAD, CAE, VR)
- Design for Manufacture
- Design for Assembly & Maintenance (including remote handling)
- Balance of Plant design and integration
- High level of component reliability, maintainability, inspectability for DEMO
Definition, together with the research laboratories, of the priorities in the technology development
- Project Management, Scheduling and Risk processes
- Cost Analysis / Cost optimisation
- Manufacturing process development
- Human factors analysis
- Development of Codes and Standards.

In addition, some participation of industrial experts in the ITER commissioning, operation and exploitation (together with the IO members and the scientists and engineers from the fusion labs) is envisaged in order to gain sufficient expertise in the operation of nuclear fusion reactors and to gather plasma and fusion technology know-how to be used for DEMO design and engineering activities.

12.4 Involvement of industry in the material development (Mission 3)
Materials development must include strong emphasis on the industrialisation of the candidate materials, including issues of industrial-scale production and joining techniques, with a strong participation, as a full partner, of industries responsible for fabrication, qualification, testing and components manufacturing according to codes and standards. Major industrial organisations could provide advice from Gen IV programmes and facilitate the interaction with the broader supply chain.

An effort should be made on seeking synergies with other community-funded advanced materials programmes, for example, the EXTREMAT project. The experience of the Research Fund for Coal and Steel has shown that if grants with substantial Community share are available there is a substantial direct involvement of industries. Nuclear industry materials groups are also heavily involved in the SNTEP (Sustainable Nuclear Technology Platform).14

During Horizon 2020, partnerships between research laboratories and industries should be facilitated by EFDA through support to joint programmes. Resources for R&D activities could be made available also from other Community programmes. These joint programmes could naturally evolve during 2020-30 in consortia between industry and laboratories for the DEMO EDA phase.

12.5 Involvement of industry in the target of low cost of electricity (Mission 7)
Market penetration will require fusion energy to be competitive. Although it is difficult to reliably predict the cost of different energy sources in 2050, the ITER experience has even more than before underlined the need for the programme to strongly pursue lowering the cost of electricity from fusion through low capital cost.

This could be ensured by the DEMO project (acting as customer agency) providing only essential requirements to a competitive process among industrial consortia to deliver the complete system rather than providing detailed specifications for subsystems to a spectrum of separate industrial companies, as for ITER. It is not uncommon in the aerospace field to have in the early phase of a project two prime

14 Its main task force ESNII (European Sustainable Nuclear Industry Initiative) is supporting projects like ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration 600 MWe), MYRRHA (Multipurpose hYbrid Research Reactor for High tech Applications) and ALLEGRO (gas Fast Reactor) accounting for a ~10B€ budget.
contractors work in parallel and come with different approaches for the same satellite. The best approach or sometimes a combination of the two proposals is then chosen as the final design.

12.6 A policy for industry involvement in Horizon 2020
Industrial involvement needs a policy to develop and maintain industrial competence in fusion-specific areas after the completion of the ITER construction and in advance of the DEMO EDA. An early launch of the DEMO EDA in the 2020-30 decade would facilitate maintaining these competences. However, without specific provisions the know-how accumulated during the ITER construction phase, expected to end around 2017, competencies could rapidly disappear before the start of the DEMO EDA. The training of specific competences is analyzed in Annex 9. Here we consider the necessary actions for the competence existing in industry.

Participation in the DEMO CDA. This should be done through the delegation of key industrial personnel to the project team with appropriate support from their parent organisation and could be managed either directly by EFDA or by the Associates involved in the programme.

Participation in ITER assembly, commissioning and exploitation. This participation could be made with the same provisions foreseen for the participation of scientists and engineers and similar to those presently used for the exploitation of JET (e.g. long-term secondment in the Operation team).

Knowledge management system. Such a system should be set up to conserve both explicit (engineering and design data, integration procedures, operations, test procedures and test results, etc.) and implicit knowledge (knowledge of individuals) acquired during the development and operation of ITER for future generations of engineers and operators. The data-base of explicit knowledge will be set up by the IO. To preserve the implicit knowledge entails setting up a systematic, structured approach to transfer the know-how of retiring experts to their successors; a lessons-learned management system should be set up to capture lessons gained from mistakes made during system engineering, design, integration, test and operations, together with a best-practice conservation system which captures "tips and tricks" gained by experience.

Legal aspects. An area that requires specific attention is that of legal aspects of know-how management (Intellectual Property Rights, patents, non-discloser technologies). This should be discussed well in advance of the start of the DEMO EDA to facilitate the involvement of industry. The period between the end of the ITER construction (~2017) and the start of the DEMO EDA should be used to review the lesson learned from the ITER construction and the results of the DEMO CDA in order to define the best implementation tools for the ITER EDA.

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13.1 Introduction

Since the 1950s, fusion research has been carried out worldwide in the frame of international collaboration. The most important collaboration at present is the ITER agreement, with the objective of the construction and exploitation of ITER. The Broader Approach agreement between Europe and Japan, setup for the construction of the JT-60SA tokamak, engineering validation phase of IFMIF, DEMO studies and provision of computing resources for theory and modelling, is one of the most important examples of European participation in international agreements outside ITER. In addition, there are a number of bi-lateral and multilateral international agreements with the aim of advancing the development of fusion as part of an accompanying programme in support of and in complement to ITER. These agreements allow non-European participation in the European programmes, and vice-versa, they allow European participation in the US and Japanese programmes, and more recently on the emerging programmes of China, South Korea and India.

International collaborations offer a number of opportunities for the implementation of the roadmap, capitalising on the intellectual diversity of the fusion community, and from sharing of resources.

13.2 Exploitation of ITER

The rules for the ITER exploitation are foreseen in the ITER agreement. Europe should support the participation in ITER through mechanisms such as short- and long-term secondment of personnel to the IO and contractual arrangements with the European laboratories similar to the present JET Orders and Notifications.

The ITER exploitation will require specific collaborations on some of the foreseen enhancements. The most noticeable example is the use of Lower Hybrid waves for current drive in ITER, for which a test of a launcher in an ELMy device is still needed. The related R&D should be done in the context of international collaboration with other ITER parties.

The possibility of know-how sharing on the Test Blanket Module programme with other ITER parties should be also carefully considered in order to complement the information that will be obtained through the two TBM concepts that Europe will test, should the DEMO analysis show that other lines have to be investigated.
13.3 Exploitation of JET within an International Collaboration Framework

The unique capabilities of JET, namely the ITER-like wall (ILW), its tritium capability and its size, make JET a key device for preparing the first phase of ITER operation and exploitation.

The further internationalization of JET, possibly under the umbrella of the ITER IO, has been strongly recommended by the Wagner Panel and is considered a prerequisite to operate JET beyond the goal of complete exploitation of the ILW including a DT experiment, i.e. beyond 2017. The US and Russia have a long-standing tradition of participation in JET and have shown interest in participating in projects that will require JET operation beyond 2017. The US has led the feasibility study for a set of in-vessel resonant magnetic perturbation (RMP) coils and Russia has participated in the feasibility study of an ECRH system. At the moment, India is the ITER party that has shown a clear interest in being strongly involved with JET and a one-year project for the design and R&D for the RMP coils has been launched. This could be followed by the procurement of the entire system as in-kind contribution from one or more ITER parties.

In addition to the ITER parties, Brazil has a growing participation in the JET programme through a collaboration that involves CRPP and MIT. Discussions are ongoing on how to progress further this collaboration.

The internationalization of JET will have to bring a substantial contribution to the European programme, either by sharing the JET operation costs, through participation in-kind or in personnel, or by supporting other parts of the programme. On the other hand, training of staff at JET is recognised by the Wagner Panel as a potentially main contributor to the necessary process of preparing for efficient ITER operation. In particular, the Panel believes that: “A joint JET operation by all ITER parties in the coming years could be an important step in that direction, especially by providing a training ground for ITER scientist and engineers.” In light of the larger operating cost of ITER, the continued use of JET in this international environment has the potential for generating considerable savings to the EU programme.

13.4 Exploitation of JT-60SA

As the main ITER satellite device, the JT-60SA programme will focus on the preparation of the second phase of ITER operation aiming at long pulse discharges. The JT-60SA construction is presently made within the framework of the Broader Approach agreement, through the voluntary contribution of a few European countries. While originally, that implied a joint exploitation of the device for a few years, it is now agreed with Japan that Europe will be able to participate to the exploitation of the device, which is now scheduled to start operating in 2019. This joint exploitation period will continue for as long as the EU will support the operation budget with a share of the cost foreseen to be 25% of the total, in line with the present BA provisions. The level of participation is assumed to be 100 ppy/y with 10% on site work. As for ITER, this should be supported with provisions similar to those presently used for the JET exploitation. Enhancements targeted at achieving the JT-60SA goals should be implemented in 2020-30.
13.5 **Divertor development for DEMO**

As the divertor power handling requirements in DEMO are even more demanding than in ITER, a Divertor Tokamak Test facility (DTT) might be required to validate a DEMO divertor solution. An International Collaboration on a joint DTT facility offers a possibility of pooling joint resources in this area. In particular, the US and Japan have advocated the need for and shown interest in such a facility.

13.6 **Construction of a pilot-IFMIF project**

The present IFMIF EVEDA phase is supposed to end in 2017. By this time, 3 major prototypes will have been designed, manufactured, commissioned and operated in order to produce experimental backing of the IFMIF Design: 1) in Rokkasho (Japan): an Accelerator prototype (“LIPAc”): 1 line of 125 mA deuterons with 9 MeV (vs. the 40 MeV of IFMIF); (2) in Oarai (Japan): a Lithium Test Loop (“LTL”): Width of Li Target 100 mm (vs. the 260 mm of IFMIF); and in Karlsruhe: a High Flux Test Module prototype (“HELOKA”): 1:1 prototype + He cooling loop.

In the frame of the post-BA activities, F4E is presently exploring, in close interaction with JAEA, the possibility of extending the prototypes in a follow-up project towards an early neutron source (pilot-IFMIF Project) potentially assembled at the Rokkasho site. This device could provide early access to key and urgent investigations with strong impact on the design of an early-DEMO, e.g., determination of embrittlement effects in EUROFER (or alternative structural materials) at fusion relevant He/dpa condition, by measuring high quality data for the ductile-to-brittle transition temperature and fracture toughness (i.e. well defined irradiation temperature with no significant temperature excursions) around the expected threshold 30 to 50 dpa (requiring damage rates of typically 10-15 dpa/fpy).

13.7 **Use of research reactors for material irradiation studies**

Non-EU research fission reactors, e.g. nuclear research reactors in Dimitrovgrad in Ulyanovsk Oblast, Russia, have been already used for material qualification.

Considerable progress has been made in the past in elucidating the basic mechanisms of materials degradation under neutron irradiation by utilizing a variety of irradiation sources (e.g. fission reactors, ion beams, etc.) coupled with a robust theory and modelling effort.

Thermal and fast fission research reactors exist today and are available for irradiation services in the US, Europe and Asia; however, the fast reactors are located in Russia and Asia (e.g., China and Japan). For example, the BOR-60 Reactor is currently capable of producing 50 dpa in two years of operation. Significant advantages of this source are its immediate ability for high dose uniform irradiation over large samples with a well characterized spectrum. Technical limitations are the very low helium and hydrogen build-up, transmutation gas production rates similar to mixed spectrum reactors, and handling activated samples. Fusion relevant helium generation can be achieved in thermal and fast reactors by isotopic tailoring of specimens.
Options should be explored to effectively use relevant international fission reactors for a number of urgent irradiation experiments as opposed to pursuing only EU capabilities that are limited in this field to reduce the uncertainties in some areas, to investigate phenomena that bear important implications on the design of an early DEMO (e.g., determine the effect on the DBTT in EUROFER (or alternative materials) and degradation of properties of ceramic breeders and beryllium at high-dose, etc.) and to optimise use of the limited irradiation volume in IFMIF. Some ongoing scientific collaborations already exist (e.g., IEA technology Agreement on Fusion Materials), but more effective utilisation strategies and collaborations would be needed to clearly define and execute experimental campaigns to make the needed advance on time. Also an effort should be made to secure access during the second half of this decade to the new material testing reactor, the Jules Horowitz Reactor (JHR), presently under construction at CEA Cadarache (France). This research reactor will be mainly devoted to Gen IV testing materials testing, but the facility is also open to work with the fusion community if requested.

**13.8 Use of nuclear component test facilities in other countries**

There is a diverse programme emerging around the world in the area of component test facilities, in particular with the proposal of building a Chinese Fusion Engineering Testing Reactor (CFETR) facility in China, where some level of tritium breeding is foreseen. In the US, proposals such as the Fusion Nuclear Science Facility (FNSF) have been also put forward. It is important for Europe and other international parties to share know-how in this area by mutual participation in each other’s programmes.

These specific collaborations will benefit from other smaller scale joint projects on DEMO R&D, in particular making use of the infrastructure developed with Japan during the BA for this purpose.

**13.9 International collaboration on stellarator research**

As the European Stellarator Programme will focus on the Helias line, other alternative stellarator lines could be explored in the frame of international collaborations. The Heliotron line has yet to demonstrate an integrated scenario simultaneously achieving high-beta, high temperatures and sufficient confinement combined with a suitable divertor operation. Without pre-judging the success of ongoing attempts to provide this demonstration, studies on the Heliotron stellarator line should continue as part of international collaborations in order to extend the physics basis of stellarators and the exploration of the helical divertor concept, in particular in collaboration with Japan at LHD.

Compact stellarators offer the possibility of more compact stellarator reactor designs, comparable to advanced tokamak power plants. The US has shown a particular interest in exploring this line of research, offering opportunities of further international collaborations in the area of stellarator development.