How far is a Fusion Power Reactor from an Experimental Reactor?

R. Toschi[(1)], P. Barabaschi[(2)], D. Campbell[(1)], F. Elio[(2)], D. Maisonnier[(1)], D. Ward[(3)]

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1. Introduction

To support a request of very substantial resources to build and operate an experimental reactor such as ITER, it is necessary to show that such a device is well positioned on the route towards a reactor and not too far from the reactor in parameter space. In other words we should show that ITER objectives and design features are well chosen so that, following ITER, the development path to a reactor is not too long and not too risky. That would be the case if, assuming a successful completion of the ITER programme, we would be able to move directly and safely to a "first of a kind" reactor. The main conditions to be satisfied to reach this goal are:

- the design features and operation modes of ITER are "reactor relevant". If so we can take a conservative approach in designing a reactor namely we can maintain strong technical continuity between ITER and a reactor and therefore we ensure a solid technical basis to the reactor design. Where this continuity is not possible e.g. structural materials for high neutron fluence not yet available, we should make sure that an adequate R&D programme parallel to ITER is pursued.
- the extrapolation of key performance parameters from ITER to a reactor are moderate and supported by theoretical / experimental evidence.

To assess the reactor relevance of ITER, rather than a comparison between ITER and one of the existing reactor design, we chose to investigate which enhancements would be possible in the ITER design to move it towards a "first of a kind" reactor. This "constrained" approach has the great merit to be based on an existing engineering design, on extensive R&D and on well founded design criteria.

2. Recommendations to make a fusion power reactor an attractive source of electrical energy

To inject some perspective into this analysis on the approach to a reactor, we shall first summarise some recommendations made by the EU utilities and industry on the requirement that a fusion power reactor should satisfy to become an attractive source of electrical energy. The EU utilities / industry concentrated their attention on safety, waste disposal, operation and criteria for an economic assessment.

(1) EFDA-CSU, Garching (G)
(2) ITER JCT, Garching (G)
(3) EUR-UKAEA Association, Culham (GB)

Safety and waste disposal

- there should be no need for an emergency evacuation plan, under any accident driven by in-plant energies or due to the conceivable impact of ex-plant energies;
- no active systems should be required to achieve a safe shut-down state;
- no structure should approach its melting temperature under any accidental conditions;
- "defence in depth" and "prevention and mitigation of accidents" and in general ALARA principles should be applied as widely as possible;
- waste transport should be confined to centres close to the production and "integrated" sites, including both power plant and waste treatment/storage, should be considered;
- the fraction of waste which does not qualify for "clearance" or for recycling should be minimised and after an intermediate storage of less than 100 years, should go to a repository characterised by a maximum intrusion dose less than 1 mSv.

Operation

- operation should be steady state with power of about 1 Gwe for base load and have a friendly man-machine interface
- the lifetime should be about 40 years with the possibility of further extension up to 60 years for parts which are not replaceable
- maintenance procedures and reliability should be compatible with an availability of 75 -80 %. Only a few short unplanned shut-downs should occur in a year.

Economics

- since public acceptance is becoming more important than economics, economic comparison should be made with energy sources with comparable acceptability, e.g. fossil fuel, but including the economic impact of "externalities"
- licensing requirements and regulatory constraints should be strongly reduced as compared to fission plants
- the construction time should be minimised, possibly no more than 5 years
- standardisation should be pursued as much as possible

3. Enhancement of ITER plasma performance towards a reactor

The key parameters which determine the fusion performance are the plasma energy confinement, plasma density, and thermal beta. In addition, to maintain an acceptable Q (>20) in steady-state (i.e. non-inductive) operation, the plasma pressure must provide a sufficiently high bootstrap current contribution, so that the current drive requirement from the auxiliary heating systems is not excessive. Finally, the α-particle and auxiliary heating power must be exhausted while limiting the power flux to the divertor target, so as to keep thermal stresses within engineering constraints and to limit erosion rates. The plasma regime considered in this study is the same as in ITER, the ELMy H-mode with monotonic or shallow shear, for which the most extensive database exists. Thus the reference rules adopted for the calculation of fusion performance are as detailed in the ITER Physics Basis. The analysis does not rely on the exploitation of "advanced scenarios" in which the central plasma shear is reversed, the bootstrap current is well aligned with the total current and the energy confinement is enhanced by an internal transport barrier. In the following, we shall discuss the rationale for the choices made in this study, in particular, in relation to plasma energy confinement, plasma density, plasma beta, current drive efficiency, and particle and power exhaust.

Energy Confinement

The ITER recommended scaling for the H-mode power threshold and that for H-mode energy confinement have been used in this study. For modelling of steady-state scenarios in ITER, a confinement enhancement factor of H_H=1.2 is used, which is modest compared to the enhancement which can be obtained under appropriate conditions in current experiments. Moreover, it is well within the range of scatter relative to the scaling which is observed in the existing H-mode confinement database. This enhancement factor is also applied in this study, though for values of plasma density in the vicinity of the Greenwald value.

Plasma Density

To increase Q it is essential, in a given device, to increase the fusion power density, i.e. beta, while maintaining the sufficient input power to provide the required component of non-inductive current drive. To make efficient use of the plasma beta, operation at high density is favoured. In fact the approximation $P_{fus} \sim n^2T^2$ is valid in the region of 12keV, but at lower densities and higher temperatures, fusion power varies more weakly than T^2, and the loss of fusion power is not compensated by the increase in current drive efficiency which derives from the higher temperature. These consideration effectively constrain a viable plasma scenario to operate in the vicinity of the Greenwald density. Some advantage can be gained in fusion power production for a given line average density if the density profile is slightly peaked. However, a further constraint on plasma density arises from the need to limit the power flux to the divertor target, which, as shown by experiments and modelling, implies a high separatrix density. In ITER, steady-state operation at Q=5 assumed values of n/n_{GW} of ~0.7 or 1-1.4 (ITER '98). In this study the line average density has been adjusted to match the Greenwald value, but a modest peaking has been allowed ($n(0)/n_{ped} \sim 1.6$). Maintenance of H-mode quality confinement at densities in the vicinity of the Greenwald value is recognized as a major issue for the tokamak programme. However, progress has been made in recent years, in particular by exploiting the technique of inboard pellet launch developed in ASDEX Upgrade, where H-mode confinement has been maintained at line average densities in excess of the Greenwald value. The effectiveness of this technique has been confirmed in DIII-D and JET. Moreover, in the latter experiment, inboard pellet launch allowed density profile peaking factors in excess of the value specified here to be obtained, although at reduced confinement. Profile peaking at line average densities around the Greenwald value combined with H-mode confinement has also been observed in long pulse gas-fuelled plasmas in ASDEX Upgrade and DIII-D. These experiments indicate that H-mode operation at plasma densities in the vicinity of the Greenwald value is a realistic prospect and that techniques are available which provide access to peaked density profiles. In particular, the achievement of significant density profile peaking in plasmas at the JET scale with modest pellet velocities (several $100ms^{-1}$) is a promising result which is worthy of further development.

Plasma Beta

Two principle processes can be expected to limit plasma beta in the scenario considered here. In plasmas with monotonic q-profiles, or weak central shear, neoclassical tearing modes (NTMs) have been observed to limit β_N in many devices. Experiments in ASDEX Upgrade and COMPASS have shown that NTMs can be stabilized by current drive and this offers a promising approach to the control of these modes at the reactor scale. Ideal mhd is expected to limit the β_N to a value of β_N~3 (the value assumed in ITER). In cases where the ideal mhd kink is stabilized by the presence of a resistive wall, the resistive wall mode (RWM) can persist, as observed for example in DIII-D. Theory indicates that stabilization of the RWM using an active feedback control system based on external coils is feasible, though experiments are at an early stage. If this technique is successful, it should allow access to the value assumed here of β_N=3.8 (including 0.6 due to fast particles). Note that there is some uncertainty in the role of fast particles in beta limiting processes, with some analyses indicating that a stabilizing role could be expected. It should also be noted that theoretical analyses indicate that with appropriate control of the plasma shape and profiles, considerably higher values of β_N should be accessible.

Steady-state operation

This requires that sufficient auxiliary heating power is available to drive ~50% of the plasma current and that the radial distribution of the driven current complements the bootstrap current profile so that the total current profile satisfies any global requirements and is robust against mhd instabilities. On theoretical grounds it is expected that the current drive efficiency, γ_{CD}, of auxiliary heating systems, should increase with electron temperature, a prediction which has been confirmed in numerous experiments. Therefore, the high temperatures which are anticipated in fusion reactors imply that for the major auxiliary heating systems the current drive efficiency should be substantially greater than in existing experiments. The current drive efficiency for the range of plasma parameters considered in this study can be approximated by, $\gamma_{CD} = 0.35\frac{T_e\ (\mathrm{keV})}{10}$, and this has been exploited in the exploration of scenarios using the ITER PRETOR code. In the representative case considered here, for which no explicit assumptions have been made in relation to the H&CD systems, this yields a current drive efficiency of 0.63 and 43% of the current is driven externally.

Particle exhaust

Suitable divertor plasma conditions must be maintained so that helium ash can be exhausted at a rate equivalent to that at which α-particles are produced to avoid poisoning the plasma and quenching the burn. Evidence from existing experiments and from modelling of ITER indicates that with an ELMy H-mode edge, helium can indeed be exhausted at the required rate and that the exhaust rate is determined by divertor conditions rather than core particle confinement. The value of τ_{He}^*/τ_E=5 assumed here (as in ITER), which yields $n_{He}(0)/n_e(0)$~7%, is based on the experimental and modelling database assembled for ITER. This indicates that the total fuel throughput available is a key parameter in determining the helium exhaust rate and that this is likely to set the requirement for fuel throughput and reprocessing in a reactor.

Power exhaust

Dissipation of a substantial fraction of the α-particle and auxiliary power before it reaches the divertor target is the key plasma-wall interaction issue for a power plant. For the plasma parameters in question, the sum of core bremsstrahlung, synchrotron emission and impurity line radiation from intrinsic impurities is approximately equal to the proposed current drive power (~80MW). This implies, that the loss power due to conduction and convection across the separatrix is approximately equal to the α-power. On the basis of arguments relating to surface erosion rates and tritium retention, tungsten appears as the most suitable plasma facing material for the divertor target for steady-state reactor operation. This implies, however, that a maximum peak power flux of ~15 MWm^{-2} would be permissible and the divertor plasma temperature should be reduced below ~10eV to ensure that the erosion rate is acceptable. Modelling of the factors influencing the lifetime of the ITER divertor target has shown that melting due to high energy pulse from ELMs and disruptions would be limiting factors for tungsten PFCs. Although current predictions for the amplitude of type I ELMs in ITER indicate that target ablation could be problematic, development of regimes with more benign ELMs continues within the tokamak programme.. For example, scenarios in which high plasma shaping allows access to type II ELMs while maintaining high confinement are promising, but require further exploration to confirm their viability under reactor conditions. Considerable advances are also being made in the avoidance and mitigation of disruptions. Automated systems for recognition of disruption precursor conditions are now being implemented and, together with the rapid developments in control software and computing power, offer the prospect that potential disruption conditions can be routinely avoided. Progress in this area could therefore make disruptions a rare and 'off-normal' event. In addition, the routine application of disruption mitigation techniques, such as 'killer pellets', should allow the potential for damaging plasma-wall interactions at disruptions to be substantially reduced.

Divertor scenarios developed for the ITER '98 illustrate a promising approach to control of the power flux in a reactor. Based on operating regimes in existing experiments, such as the CDH-mode in ASDEX Upgrade, dissipation of exhaust power in ITER '98 utilized the concept of 'impurity seeding', in which a noble gas such as neon or argon is injected into the divertor plasma at low concentrations (<1%). This results in a substantial increase in line radiation in the edge region of the bulk plasma and in the divertor. Two dimensional numerical modelling of the ITER '98 divertor showed that, at high densities and by exploiting radiation from intrinsic and seeded impurities, a 'partially detached' divertor regime could be developed which allowed the power conducted and convected to the target to be reduced to about one third. Access to partially detached divertor conditions are an important aspect of the analysis, since this mode of operation preferentially reduces the target heat flux and electron temperature close to the separatrix as a result not only of impurity radiation, but also of loss processes such as volumetric recombination and ion-neutral friction processes. For the case considered here (~400MW of α-power), the ITER '98 scenario would limit the peak heat flux to the divertor target to below 15MWm^{-2} and the plasma temperature at the plate to < 10 eV.

4. Enhancement of ITER engineering performance towards a reactor

We assume that ITER will achieve the performance objectives stated and will provide an operating experience sufficient to define sets of optimum, reliable and reactor relevant operating parameters. As a consequence certain provisions implemented in ITER and typical of an experimental device, such as design margins against off-normal transients (e.g. frequent disruptions) and wide flexibility in choosing operating regimes and configurations, should be considerably reduced in a reactor. Furthermore, the design of a reactor would benefit from the absence of significant fatigue loads in the main mechanical structures. On the other hand, compared to ITER, a reactor will require higher fusion power density, with correspondingly higher wall loading, higher coolant temperature in the in-vessel components for energy conversion efficiency, Tritium-breeding blanket, longer life time and shorter replacement time for the in-vessel components to achieve higher availability. Therefore, in the process to adapt ITER engineering to a "first of a kind" reactor some specifications will be more severe, but others can be relaxed, opening the possibility of alternative design solutions. Where applicable the same design criteria, allowables etc. as in ITER have been used in this process.

Magnets

The ITER magnet engineering and technology appear fully compatible with the reactor requirements. In a reactor, compared to ITER, the reduced demand for flexibility in the magnetic configuration would allow the implementation of more compact solutions e.g. a layer wound central solenoid and bucking of TF coil configuration (as in ITER '98). Furthermore, the central solenoid inductive flux can be reduced, there are no significant fatigue loads and a more efficient TF coil square conductor (instead of shear plates) could be considered. All this would give the possibility, within the present ITER radial in-board build of reinforcing the TF coil structure, so as to increase the magnetic field on axis to 6.4 T (and correspondingly on the conductor to 13.7 T), and of increasing the shield by about 15 cm. This shielding would be sufficient to maintain the same level of nuclear heat deposition on the conductor in the reactor as in ITER in spite of the higher wall loading, but it would allow about two full power year of operation before reaching the allowed radiation dose on the TF coil insulator. A shielding suitable for 30 years (i.e. 20 fpy) operation would probably lead to an increase of the ITER major radius by about 5 % from 6.2 to 6.5 m).

First wall and breeding blanket

The concept here adopted for the breeding blanket the is one extensively studied in the EU by the FZK-Laboratory, namely the He-cooled pebble bed ceramic blanket. A blanket of this type, although with water cooling, has been designed to be installed in ITER in case of an extended operation phase. In the reactor case, the high temperature of the blanket He coolant (> 500°C) would require some design adaptations of the ITER attachment concept. The modules in the reactor case would be larger than in ITER to reduce their number and therefore the required time for the full blanket replacement. The breeding blanket modules on the inboard would be about 40 cm thick and they would offer an adequate shielding to the permanent mechanical structure behind, including the vacuum vessel. The higher operating temperature of the blanket with respect to ITER and the larger size of the modules require an improvement of the cooling manifolds and of the mechanical attachment, since the vacuum vessel operating temperature remains at 100°C,with the same thermal shield and the same first nuclear confinement of ITER. The coolant can be routed to the modules by manifolds mounted inside the vessel and restrained on the wall behind the blanket modules. To withstand the large difference between their working temperature (~500°C) and the vessel, the manifolds will be built in high strength Ti alloy, rather than steel, because the lower coefficient of thermal expansion and Young's module produce a reasonable thermal stress (below 300 MPa). The connector between the blanket modules and the coolant manifolds is coaxial and provided with flexible branches as in ITER, but the single seal between the coolant and the vacuum will be simpler because a He leak would not disturb the plasma as much as water. Thus a metal gasket can be probably used to improve the remote handing. The larger blanket modules and the temperature difference will respect to the vessel requires flexible supports, which can be derived from those of ITER by increasing the spoke length and thickness while they are fixed by the same Inconel 718 bolt. This upgraded version of the flexible support is already envisaged in ITER for the breeding blanket test modules which will be mounted in the equatorial ports.

The first wall and the structural materials of the in-vessel components in general would be low activation ferritic - martensitic steel, e. g. the EUROFER with 7-10 % Cr WT (which replace Mo, Ni, Nb in previous steels) now being developed in the EU. This steel, in the ODS (Oxide Dispersed Strengthened) version, could work at a temperature up to 650˚C. The He coolant outlet temperature could be at least 530˚C, allowing a thermal-electric energy conversion efficiency of about 38%. The main issue concerning this steel when exposed to intense neutron irradiation is the increase of the DBTT above that at room temperature when irradiated at temperatures below 400˚C (at higher temperature irradiation up to 150 dpa in fast breeder did not indicate a serious degradation of mechanical properties). We assume a lifetime up to 120 dpa which, under an average wall loading of about 2 MW/m^2, would lead to an average lifetime of the breeding blanket of about 5 fpy. The breeder / multiplier, on the basis of present data, should have a lifetime compatible with the structural material lifetime. An analysis of the impact of the ferromagnetic property of this steel on the magnetic field configuration (e.g. ripple, poloidal fields penetration) confirms its compatibility with the tokamak configuration. The first wall could operate with peak heat loads up to 0.8 MW/m^2. The breeding ratio and the thermo-mechanical behaviour of such a blanket module will be tested in ITER. However, an important R&D activity which must be carried out in parallel with ITER concerns the behaviour of the materials under neutron irradiation to high fluence (~12MWa/m^2).

Divertor

The prediction of the operating conditions of the divertor targets is very uncertain but certainly very severe. Under these circumstances we chose first to define, on the basis of the extensive ITER R&D, the technologically admissible operating conditions and then we attempted to identify, using present modelling, a plasma scenario which would satisfy these conditions. Tests conducted on ITER divertor target mock-ups (monoblock and macro-brush tungsten armour 10 mm thick, on Cu-alloy heat sink cooled with water) showed a heat removal capability of up to 20 MW/m^2 under cycling conditions. The targets should have a lifetime of about 2.5 fpy to satisfy the availability requirement of the reactor. The most severe factor affecting the lifetime of the target is the erosion of the armour. A target such as that tested would probably meet the required lifetime under the condition of peak heat flux ≤ 15 MW/m^2, a few disruptions during its lifetime, no high power ELMs, and plasma temperature at the plate < 10 eV. According to present modelling, a plasma regime ("semi-detached") which could satisfy these conditions could be implemented provided that a large fraction (~70 %) of the SOL power is lost by radiation along the SOL and in the divertor area. To achieve this goal, impurities (e.g. Neon, Argon) will be injected, but attention should be paid to their impact on the erosion rate. Another limiting factor in the divertor lifetime may be represented by the neutron damage. The Cu-alloy heat sink in a mono-block structure working at a moderate temperature would tolerate, based on present data, up to 40 dpa, compatible with a lifetime of 2.5 fpy, but more data are needed concerning the behaviour of tungsten. The divertor thermal energy, although at low temperature, would be used for electrical energy production via preheating of the feed water from a steam turbine: this would result in a negligible efficiency loss compared to the optimum feed water preheating by steam bleeding.

Maintenance

Scheduled maintenance, i.e. periodic replacement of divertor and breeding blanket, will have the highest impact on the reactor availability. One way to minimise the time required to replace the blanket is to minimise the number of modules. To handle larger and heavier modules, it is necessary to improve the payload and dexterity capabilities of the ITER transporter (a vehicle on a monorail with a maximum payload of 4.5 T). Taking advantage of the fact that the blanket replacement will be preceded by the divertor removal, a simple but robust transporter (~12 T payload capability) has been conceived which runs on the two divertor rails. The number of modules could then be reduced from 426 in ITER to about 180. Given the significant variation of the wall loading poloidally, the blanket replacement could be staged (e.g. three stages, one every 2.5 fpy, on occasion of divertor replacement), optimising scheduled maintenance time as well as waste disposal. Maintenance time estimates, including about 20 % contingency, indicate that three replacement stages over 7.5 fpy, involving 360 modules and three full divertor removals, would require 36 months (~ 2/3 for the blanket, ~ 1/3 for the divertor) including statutory inspections. This estimate is in line with those of ITER '98 (replacement time for 730 modules up to 24 months) and account should be taken of the experience to be gained during ITER operation. For the unscheduled maintenance any estimate is very difficult because of the insufficient data base of the components/systems reliability: we have assumed here about 13 % of the operating time (i.e. about twice as much as of the today fission reactor). For unscheduled maintenance requiring the removal of a single module it seems appropriate to use an articulated boom, which would have the shortest deployment time. To meet the reactor requirement for unscheduled maintenance will probably be a high technical challenge for a fusion reactor: in this case, given the rather extensive scheduled maintenance, any measures relating to preventive maintenance of the overall plant should be implemented. In conclusion it seems reasonable to aim, after the experience in ITER, to a target availability of 65 % composed of (in parenthesis the target indicated by the utilities for a commercial reactor): operation 90 months (90), scheduled maintenance 36 months (18), unscheduled maintenance 12 months (8), total calendar time 138 months (116), overall availability 65 % (78).

A "first of a kind" ITER-type power reactor

Main parameters

Major radius (m)	6.2
Minor radius (m)	1.8
Elongation (95% flux)	1.7
Triangularity (95%)	0.33
Toroidal field on axis (T)	6.4
Plasma Current (MA)	15
Safety factor, q_{95}	3.8
Normalised β_N (including fast α's)	3.8
Bootstrap fraction, f_{bs}	0.53
Confinement coefficient, H_H	1.2
Plasma density, $\langle n \rangle$ (10^{20} m^{-3})	1.2
n_{line} / n_{GW}	1
n_0 / n_{ped}	1.6
Av. Electron temperature (keV)	18.7
Av. Neutron wall loading (MW/m^2)	2.2
Aux. heating power (MW)	80
Fusion power (MW)	2000
Q	25
Blanket energy gain	1.35
Total thermal power (MW)	2600
Thermal to electrical efficiency (%)	38
Gross electrical power (MW)	1000
Recirculating power (%)	~ 25
Operation time (fpy)	~ 2*
Availability (%)	65

* To reach 20 fpy (i.e. ~ 30 yrs. lifetime) the ITER major radius should be increased by ~ 5 % for extra shield (from 6.2 to 6.5 m).

5. Conclusions

A brief analysis has been conducted to assess the technical relationship between ITER and a "first of a kind" reactor. In outlining a possible reactor we have constrained ourselves to the minimum of extrapolations and / or design modifications necessary to make an ITER-type device meet reactor requirements, so as to maintain a strong technical continuity between ITER and a reactor. This conservative approach seems the most appropriate at this stage because we wish to stress the reactor relevance of the ITER programme. Furthermore a reactor design which draws most of its database from the actual construction and operation of the preceding device will deserve the highest confidence for success. In the reactor the plasma performance has been evaluated using the same scalings and modelling as in ITER and only two parameters (beta and density) have been enhanced by about 30 % compared to ITER. Similarly, the reactor engineering follows very closely the design criteria of ITER and takes advantage of the fact that, through the ITER programme, will be possible to establish well defined, reliable and optimised operating conditions. Therefore certain design margins, typical of an experimental device as ITER (e.g. allowing to experiment many different operating conditions and configurations, to stand off-normal events and fatigue, to test new components) may be reduced in a reactor. Last but not least the main safety issues of a reactor will be addressed in ITER.

Following this approach we have outlined a reactor which will retain the size and most of the design features of ITER although it will require a strong R&D programme to develop materials resistant to intense neutron irradiation. This ITER-type reactor would produce 1000 MWe in steady state with an availability of 65 %.

We can therefore give a positive answer to the initial questions by concluding that ITER programme, if carried out successfully, is indeed a solid and sufficient basis from which to move directly and safely to a "first of a kind" reactor.

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