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ITER: an integrated approach to ignited plasmas

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The programmatic objective of the International Thermonuclear Experimental Reactor (ITER) calls for it to demonstrate controlled ignition and extended burn of deuterium–tritium plasmas, to demonstrate essential fusion reactor technologies in an integrated system, and to perform integrated testing of high-heat flux and nuclear components.

The ITER design, as embodied in the Final Design Report, and its physics basis are presented and the projections of plasma performance are summarized together with the assessment of ITER's engineering feasibility and of the progress in validating technology R&D.

The future prospects for ITER and its potential central role as an integrated physics and engineering experiment in the overall development of controlled fusion as a source of useful energy are discussed.

Keywords: plasma confinement; ignition; divertor; superconducting magnet; remote handling; tritium

1. Introduction

Recent years have witnessed remarkable progress in the efforts to realize the potential of thermonuclear fusion as a practical source of useful energy here on Earth. The achievements of the present generation of large tokamak experiments—appropriately represented in this issue by colleagues from JET and TFTR—have carried us to the threshold of fusion reactor conditions (e.g. fusion power output 16 MW in D–T burn in JET (Gibson *et al.* 1998), and Q—the ratio of power produced from fusion reactions to that needed from auxiliary sources in order to maintain the required plasma temperature and density—approaching 1 in JET D–T experiments (Gibson *et al.* 1998) and equivalent to *ca.* 1.25 in JT-60U (Tobita *et al.* 1999)). While these leading machines have been pushing forward the technical boundaries, complementary research in a wide range of small to medium-sized devices has furthered our understanding of the physics phenomena and has informed the key scaling functions.

With this experience and with the benefit of the database thus generated, the world fusion development programme has now reached the point where it is ready and able to address the next logical challenge: to explore the domain of burning plasmas in which α-particles will be the main source of plasma heating and thus one of the major determinants of plasma behaviour. This challenge is beyond the capabilities of present machines; it requires a step change in size and major departures in other essential design features. The required change in scope to achieve ignition needs to incorporate and integrate key technological features of a future fusion power reactor. An experiment of this kind can provide a robust experimental foundation for the design of a first demonstration fusion power station.

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The International Thermonuclear Experimental Reactor (ITER) was conceived as a collaborative venture by the world's leading fusion programmes—the European Union, Japan, Russia and the USA—to take such a next step jointly in a framework of international collaboration (ITER EDA Agreement 1992). ITER's programmatic objective reflects the policy decision of the Parties to integrate the study of burning/ignited plasmas and of key technologies in a single machine as the most efficient approach to the goal of fusion as a practical energy source. The objective is

> to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. ITER would accomplish this by demonstrating controlled ignition and sustained burn of deuterium–tritium plasmas, with steady state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat-flux and nuclear components required to use fusion energy for practical purposes. (ITER EDA Agreement 1992)

This paper will seek to describe the ITER project in terms of its organization and objectives and to report and evaluate the progress made in design and supporting research and development. It is argued that ITER is ready to be realized and will be fully capable of fulfilling its key role as the next step in the development of fusion.

The paper does not address the issues of ITER's safety and environmental characteristics, which are extensively documented elsewhere (ITER 1999*a*), with the conclusion that ITER will meet its objective of demonstrating the safety and environmental potential of fusion power.

2. The ITER EDA Agreement: objectives and main parameters

The ITER project originated from summit discussions in the late 1980s involving President Gorbachev, President Mitterand and President Reagan. Thereafter, acting on an invitation from the IAEA, the four Parties undertook joint conceptual design activities (CDA) from 1988 to 1990. The success of this work, both technically and organizationally, provided the basis for jointly undertaking Engineering Design Activities from July 1992 under the terms of a six-year intergovernmental 'ITER EDA Agreement' under IAEA auspices.

Under the terms of the Agreement, the Parties jointly undertake work 'to produce a detailed, complete and fully integrated design of ITER and all technical data necessary for future decisions on the construction of ITER' (ITER EDA Agreement 1992). Overall responsibility for the project's direction rests with the ITER Council, at which the four Parties are represented. Technical activities are directed and coordinated by the ITER Director, who is assisted by an international 'Joint Central Team' located at three joint Work Sites in Europe (Garching), Japan (Naka) and USA (San Diego). 'Home Teams' based in each of the four Parties perform specific design tasks and carry out validating R&D. Within this framework progress of the design and related work has been reflected in a number of major milestone design reports throughout the six year initial duration of the EDA, culminating in a 'Final Design Report, Cost Review and Safety Analysis' (FDR) (ITER 1992; ITER EDA 1994), which was accepted by ITER Council in February 1998, and, following extensive domestic reviews in each of the Parties, was approved by the Council in June 1998.

Table 1. *ITER main parameters and dimensions*

total fusion power	1.5 GW
neutron wall loading	1 MW m^{-2}
plasma major radius	8.1 m
plasma minor radius	2.8 m
plasma current	21 MA
plasma inductive burn time	$\geqslant$ 1000 s
ellipticity (@95% flux surface)	$\sim$1.6
triangularity (@95% flux surface)	$\sim$0.24
toroidal field @8.1 m radius	5.7 T
maximum TF field @ TF coil	12.5 T
TF ripple at separatrix	< 1%
auxiliary heating power	100 MW

Detailed technical objectives and approaches to determine the best practicable way to achieve ITER's programmatic objective for ITER were established by the Parties at the start of the EDA phase. In terms of plasma performance they require ITER to have 'confinement capability to reach controlled ignition and to demonstrate controlled ignition and extended burn in inductive pulses with a flat-top duration of approximately 1000 s. ITER should also aim to demonstrate steady-state operation using non-inductive current drive in reactor-relevant plasmas' (ITER 1999*a*). For the purposes of engineering performance and testing, ITER should demonstrate the availability of technologies essential for a fusion reactor, test components for a reactor and test design concepts of tritium breeding blankets relevant to a reactor. To this end, 'the average neutron wall loading should be about 1 MW m^{-2} and the machine should be capable of at least 1 MWy m^{-2} to carry out longer time-integral and materials tests'. (ITER 1999*a*)

The main parameters and overall dimensions of ITER presented in the FDR are as summarized in table 1 (also see figure 1).

3. Summary of ITER design features

In line with its scientific and technological mission, the ITER magnet system is superconducting and comprises 20 toroidal field (TF) coils, a central solenoid (CS), nine external poloidal field (PF) coils, and three sets of correction coils. The centring force on toroidal magnets is reacted by the central solenoid. Coil cases for the toroidal field coils are used to support the poloidal coils. The TF and CS coils use as conductors Nb_3Sn and are manufactured using the so-called 'wind, react, transfer' process. The PF and correction coils have NbTi superconductor that can be wound *in situ*.

The vacuum vessel is a double-wall stainless-steel structure fabricated from 20 toroidally divided sectors, containing internal shield plates and ferromagnetic inserts to reduce TF ripple. There are access ports at the top, middle and bottom of each sector. The vessel is suspended from the toroidal field coil cases by 20 vacuum vessel vertical supports, and spaced from them by horizontal supports. In this way the vacuum vessel and the various magnet systems combine to form an integrated overall assembly, which simplifies the equilibration of electromagnetic loads. Gravity sup-

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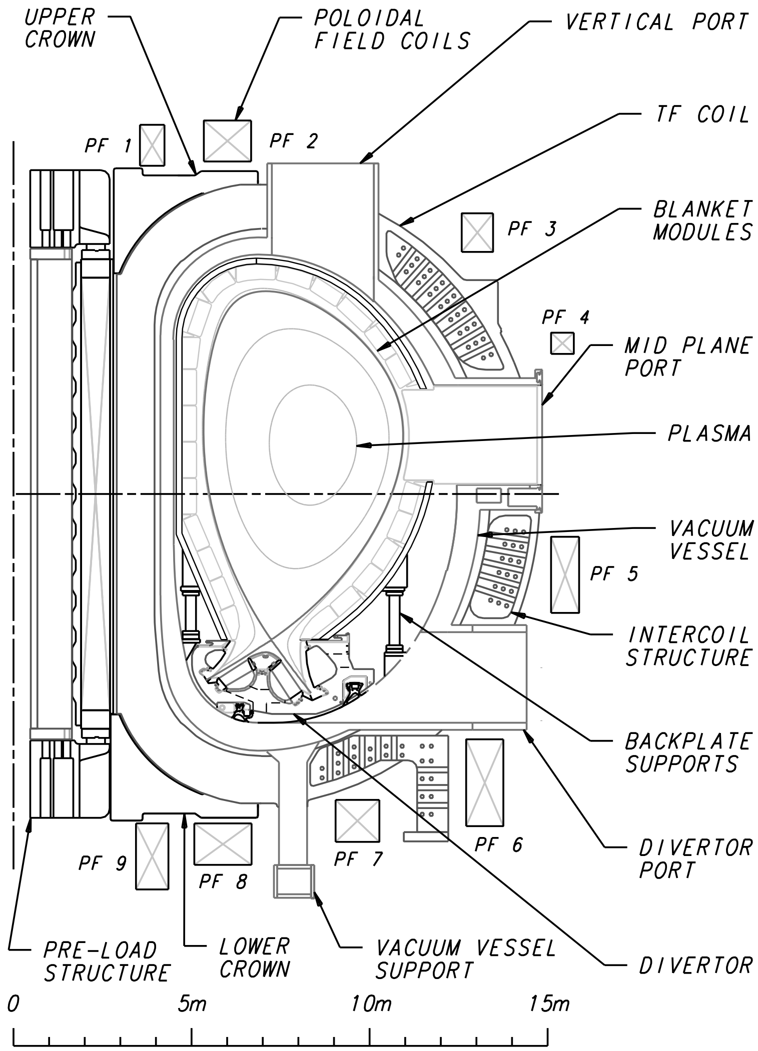

Figure 1. A cross-section of the ITER tokamak.

ports bear the weight of the magnet system together with the vacuum vessel and internals.

The in-vessel components comprise modular, removable components, including blanket modules (mounted on a double-walled permanent backplate attached to the vessel), divertor cassettes, and port plugs such as the limiter, heating antennas, test blanket modules, and diagnostics sensors. These components absorb most of the radiated heat from the plasma and protect the vessel and magnet coils from excessive nuclear radiation by absorbing the neutron power from the fusion reactions. The divertor absorbs most of the α-particle power exiting the plasma, exhausts the helium ash and limits the concentration of impurities in the plasma. The modular components are designed to be readily and safely maintainable by a practical combination of remote handling and hands-on techniques.

Materials for the in-vessel components must be able to operate under the simultaneous influence of different life-limiting factors, such as neutron irradiation, hydrogen atmosphere, dynamic stresses, cyclic mode of operation, thermal loads, and water-cooling environment. To achieve good performance and adequate availability of the whole machine, the in-vessel components have to remain highly reliable throughout the design lifetime. The design solutions typically involve fabricating components from a combination of different materials, which serve as armour (carbon-fibre composites, tungsten or beryllium), heat transfer (copper alloys) and mechanical structure (stainless steel) bonded together. Ease of fabrication, good weldability, resistance to corrosion, good strength and fatigue resistance, adequate ductility and fracture toughness after neutron irradiation are essential requirements both for the materials and for the bonds between them.

The heat deposited in the internal components and the vessel is rejected to the environment via the tokamak cooling-water system, which is designed to preclude releases to the environment of tritium and activated corrosion products. The tokamak is housed in a cryostat, with thermal shields between the hot parts, and the magnets and support structures which are at cryogenic temperature. The cryostat is installed in an underground pit, inside a building of minimum height. Auxiliary systems and facilities are housed in galleries around the tokamak pit, in the tokamak building, and in other buildings and structures laid out on a site of approximately 70 hectares. Table 2 summarizes the design features and materials choices of the ITER systems.

4. The approach to ignition in ITER

Ignition of a D–T plasma is defined as a stationary process in which the power produced by the α-particles released from the fusion reactions is sufficient to maintain the temperature of the plasma. ITER is designed to achieve controlled ignition and extended burn with an inductively driven plasma operation scenario which follows the familiar sequence established in present tokamaks using the well-characterized 'ELMy H-mode' of operation (see below).

The key elements of this reference mode of planned operation are well established and have been successfully tested in current machines. Nonetheless, the step change from the parameters of current machines to those of ITER means that projections of ITER performance must depend on extrapolation from current experience of many different plasma phenomena. The main issues include energy confinement; edge parameters and transitions between L- and H-modes (low and high modes of confinement); applicable values of β (the ratio of plasma pressure to magnetic pressure) and particle density; and divertor operation and fuelling. Each of these issues has been studied in a collaborative framework of expert groups, which bring together the specialists from the Parties' fusion programmes. The Physics Basis for ITER thus draws on extrapolations from extensive, critically analysed databases derived from experiments around the world, and on scaling and modelling techniques that have been evaluated in experiments.

(*a*) *Energy confinement and performance*

Global energy-confinement scaling is the recommended method for extrapolation of overall confinement performance in ITER. This uses a statistical analysis (log-linear regression) of an extensive H-mode database compiled from results from 10 different tokamaks to yield the overall scaling function, denoted by ITER/H-97, which

Table 2. *ITER main engineering features and materials choices*

superconducting toroidal field coils (20 coils)	
superconductor	Nb_3Sn in circular high nickel content (INCO 908) jacket in grooved radial plates
structure	pancake wound, in welded steel case wind; react and transfer technology
superconducting central solenoid (CS)	
superconductor	Nb_3Sn in square INCO 908 jacket
structure	layer wound, 14 layers, 4 conductors in-hand wind, react and transfer technology
superconducting poloidal field coils (PF 1–9)	
superconductor	NbTi in square stainless-steel conduit
structure	double pancakes, typically 2 conductors in-hand
vacuum vessel	
structure	double-wall welded ribbed shell, with internal shield plates and ferromagnetic inserts
material	stainless steel 316 LN structure, SS 304 with 2% boron shield, SS 430 inserts
first wall/blanket (basic performance phase)	
structure	armour-faced modules mechanically attached to toroidal backplate
materials	Be armour copper alloy heat sink stainless steel 316 LN structure
divertor	
configuration	single null, 60 solid replaceable cassettes
materials	W and C plasma facing components copper alloy heat sink stainless steel 316 LN structure
cryostat	
structure	ribbed cylinder with flat ends
maximum inner dimensions	36 m diameter, 30 m height
material	stainless steel 304L
heat transfer systems (water-cooled)	
heat released in the tokamak during nominal pulsed operation	2200 MW at $\sim$4 MPa water pressure, 150 °C
cryoplant	
nominal average He refrigeration/liquefaction rate for magnets and divertor cryopumps (4.5 K)	120 kW/0.25 kg s^{-1}
nominal cooling capacity at 80 K	510 kW
fuelling	
gas puffing	200 Pam^3 s^{-1} for D,D–T@0.1 MPa
pellet injection	3–10 mm pellets at up to 0.5 km s^{-1} 50 Pam^3 s^{-1} for T_2 pellets @1–10 Hz

Table 2. *Cont.*

additional heating and current drive	
total injected power	100 MW
candidate additional heating and current drive (H&CD) systems	electron cyclotron, ion cyclotron, lower hybrid, neutral beam from 1 MeV negative ions
electrical power supply	
pulsed power grid demand	
active/reactive	650 MW/500 Mvar
steady-state power grid demand	
active/reactive	230 MW/160 Mvar

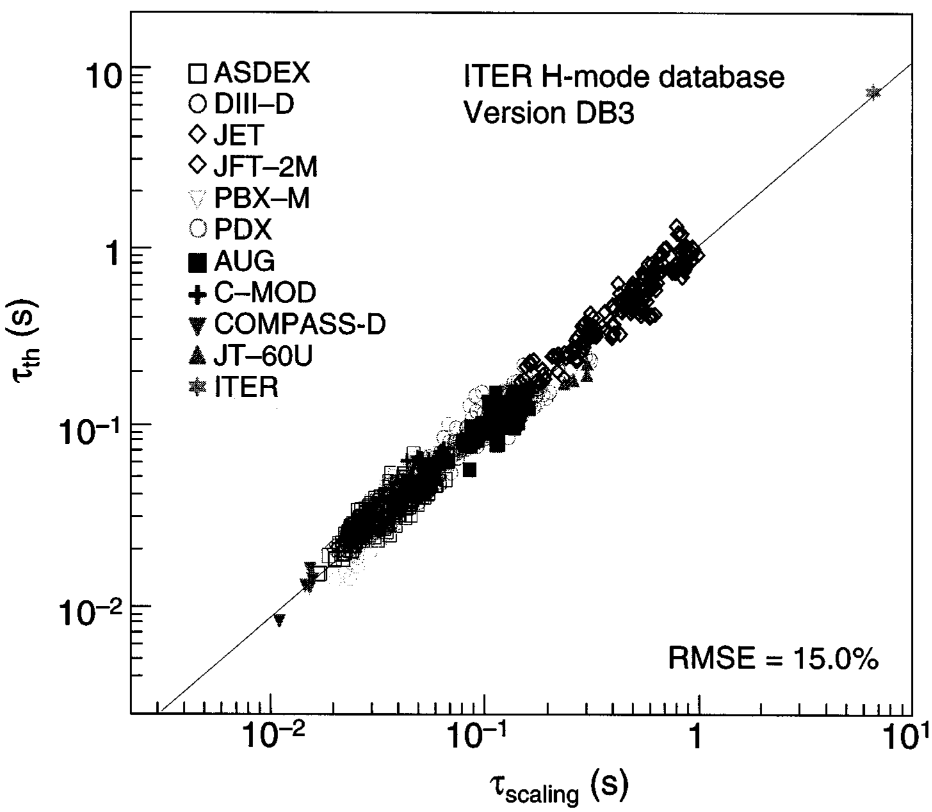

Figure 2. Observed ELMy H-mode confinement times versus ITER/H-97 scaling law.

assumes the plasma thermal energy confinement time τ_E be a power law function of engineering and global plasma parameters such as size, plasma current, magnetic field, etc.

The fit of observed ELMy H-mode confinement times to the recommended scaling function and its extrapolation to ITER are illustrated in figure 2.

The global scaling approach is complemented and supported by an approach in which experimental studies are made of plasma having all major non-dimensional parameters except that related to size (ρ^*, the ion Larmor radius normalized to plasma radius) as close as possible to ITER plasmas. This approach is used successfully in other fields, such as wind-tunnel experiments in turbulent fluid dynamics, in which the results of similar experiments can be linked through their non-dimensional parameters even though the underlying physical process is not known explicitly. Extrapolation to ITER from these experiments will require only extrapolation in ρ^*. The choice of ITER parameters has been confirmed in dedicated studies ('ITER Demonstration Discharges') in JET and DIII-D which have the same dimensionless parameters but different engineering plasma parameters and which are found to have the same dimensionless energy confinement times $\Omega_{\mathrm{i}}\tau_{\mathrm{E}}$ (where Ω_{i} is the ion gyrofrequency).

Both the empirical scaling and dimensionless approaches indicate an expected confinement time for ELMy H-mode operation of about 6 s. This confinement is sufficient to allow sustained ignition at about 1.5 GW fusion power, taking into account loss mechanisms and fuel dilution from helium and added impurities.

The issue of ITER performance projections has stimulated significant progress in the development and testing of local transport model of both first-principles and semi-empirical type. But the point has not yet been reached at which these models reliably predict overall plasma performance in the present machines; thus they cannot be relied upon for extrapolation of overall performance in ITER. More work is needed in testing against experiments and in establishing confidence limits. Nonetheless, when renormalized, the models yield some valuable and consistent results concerning more detailed features of plasma dynamic behaviour such as temperature and density profiles.

ITER performances are thus projected from present experiments by combining

the global thermal energy confinement time, as outlined above;

normalized temperature and density profiles, as observed in ITER Demonstration Discharges, mapped on magnetic surfaces deduced from plasma equilibrium in actual geometry;

1–$1\frac{1}{2}$-dimensional transport codes, including heating sources, radiation, and a divertor model

The resulting performance projections are summarized in figure 3, which shows projected fusion power output as a function of variations of confinement in relation to the reference scaling law (shown as the factor H_{H}) for given values of β and particle density. Limiting values for these parameters are established from present experiments. A plausible maximum of 2.5 for normalized β_{N} ($\beta_{\mathrm{N}} = \beta(\%)aB/I$ m T MA^{-1}) reflects the consequences on long timescales of large resistive neoclassical islands grown on rational-q surfaces of the plasma (which may probably be counteracted by a feedback of localized current drive). The particle density operating range is expressed relative to the empirical 'Greenwald' value n_{GW}; ($n_{\mathrm{GW}} \times 10^{20} = I/(\pi a^2)$ MA m^{-2}); the ratio cannot be too high without conflicting experimentally with good confinement (although it may be noted that a value of 1.5 has been achieved to date only with deep pellet fuelling), and cannot be too low without leading to conductive power losses (P_{loss}) below the power threshold for the L- to H-mode transition. The figure shows that even a shortfall of 20% in the expected confinement performance is acceptable, but possible larger values of β_{N} cannot be beneficial unless he density can be pushed to higher values than n_{GW}.

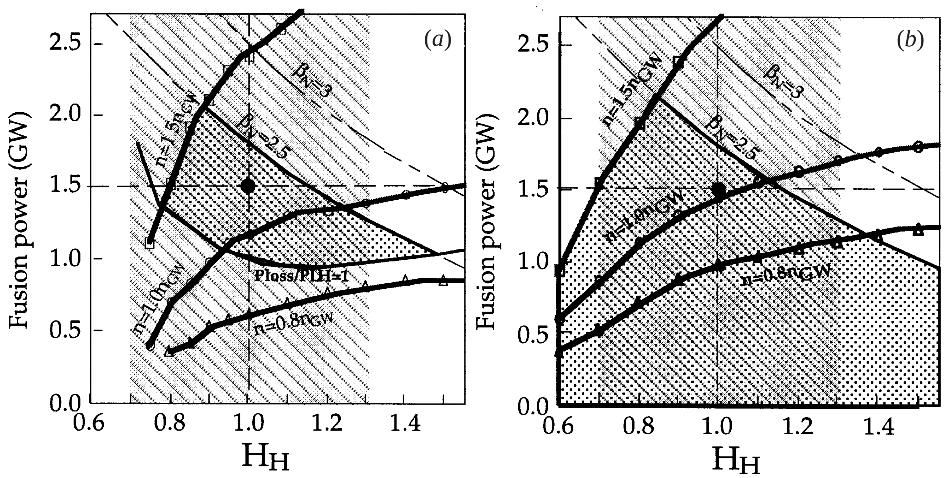

Figure 3. Fusion power domains of ITER @21 MA as a function of confinement (H_H factor) and limiting parameters values: L–H-mode power threshold ($P_{loss}/P_{threshold} > 1$), particle density in relation to Greenwald value ($n/n_{GW} < 1.5$) and normalized β value ($\beta_N < 2.5$). The dotted area indicates domains satisfying the three limits; the dashed area shows the 95% confidence level for confinement around the reference log-linear scaling.

Figure 3*b*, which presents the same analysis in the case of driven burn with 100 MW of auxiliary power, shows that the domain of operation with $Q \geqslant 10$ spans an even wider range of uncertainties. There is thus a high level of confidence that ITER will fulfil its programmatic objective in the reference mode of operation.

(*b*) *Edge physics*

The physical mechanisms occurring at the edge of the plasma in ITER—the boundary layer of *ca.* 10 cm width in ITER between the high-temperature plasma core and low-temperature divertor region plasma—are being addressed following a similar combination of database development and analysis, empirical extrapolation and theory-based modelling and simulation. Here the recent availability of increasingly sophisticated and accurate measurements of plasma temperature, density and magnetohydrodynamics (MHD) activity in the edge region of present tokamak plasmas has provided new understanding of how the physics of edge region affects the operational characteristics of the core and divertor regions. The conductive power flux (P_{loss}) crossing the plasma boundary controls the edge values; a minimum value ($P_{threshold}$) allows to achieve the transition from L- to H-mode of confinement. This is accompanied by the build-up of a pedestal in plasma edge pressure, which may be limited by localized instabilities—edge limiter modes (ELMs). The current understanding, which is still evolving, can be embodied in a so-called plasma edge operational space diagram (figure 4), in which the projected density and temperature of the ITER edge region are shown to be correlated with the temperature and density needed for attainment of H-mode confinement and ideal-MHD edge presssure gradients (manifested as type I ELMs), and avoidance of the resistive MHD instabilities (type III ELMs) and excessive edge impurity radiation loss that would otherwise compromise energy confinement and stable divertor operation.

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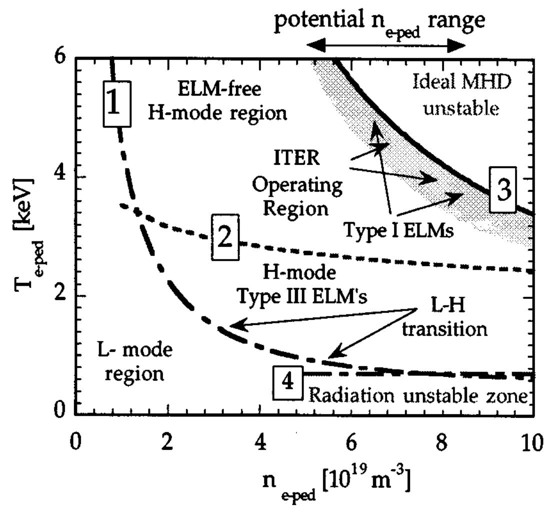

Figure 4. Plasma edge operational space diagram for ITER indicating how the edge temperature is limited, as a function of density, by ideal MHD (3) at high values and by radiation (4) at low values. The required temperatures for a transition from L- to H-modes (1) and from type III to type I ELMs (2) are also indicated. The grey area indicates the region where ITER is projected to operated.

The projected ITER edge operation conditions (n, T at the pedestal high) lie along the ideal MHD ballooning instability limit contour in the upper right corner of the diagram, at $n \sim 8 \times 10^{19}\ \mathrm{m}^{-3}$, $T \sim 4\ \mathrm{keV}$, in the region where type I ELMs are present, and where type III ELMs and excessive edge radiation losses are avoided.

(*c*) *Power and particle control and divertor performance*

Achieving the ITER performance goals depends on successful power and particle control and exhaust with the ITER divertor and pumping and fuelling system. Power exhaust in ITER will be accomplished by radiating most of the alpha and auxiliary heating power (of 3–400 MW total) to the first wall and divertor chamber walls, with radiation losses due to hydrogen and intrinsic impurities supplemented by small, controllable amounts of added recycling impurities (Ne or Ar). The additional impurities will be added to limit the divertor target power and plasma temperature to acceptable levels for as long as an adverse impact on the overall plasma core energy balance can be avoided. According to modelling from current experiments, the ITER divertor will operate in partly detached mode to allow the heat flux to the divertor target to be limited about 50 MW (5–10 MW m^{-2}) without degrading confinement, while still providing adequate He exhaust (*ca.* 5×10^{20} He atoms s^{-1}). The amount of injected impurity will be dynamically controlled to obtain simultaneously optimal 'partial detachment' from the divertor and plasma core performance. Figure 5 shows a schematic illustration of the ITER power and particle control system and the division of power flowing to the first wall and divertor.

The complexity of physics processes and interactions in the divertor area has called for extensive collaboration action by the Parties' experts in divertor physics, which has encompassed database development, transport and divertor code develop-

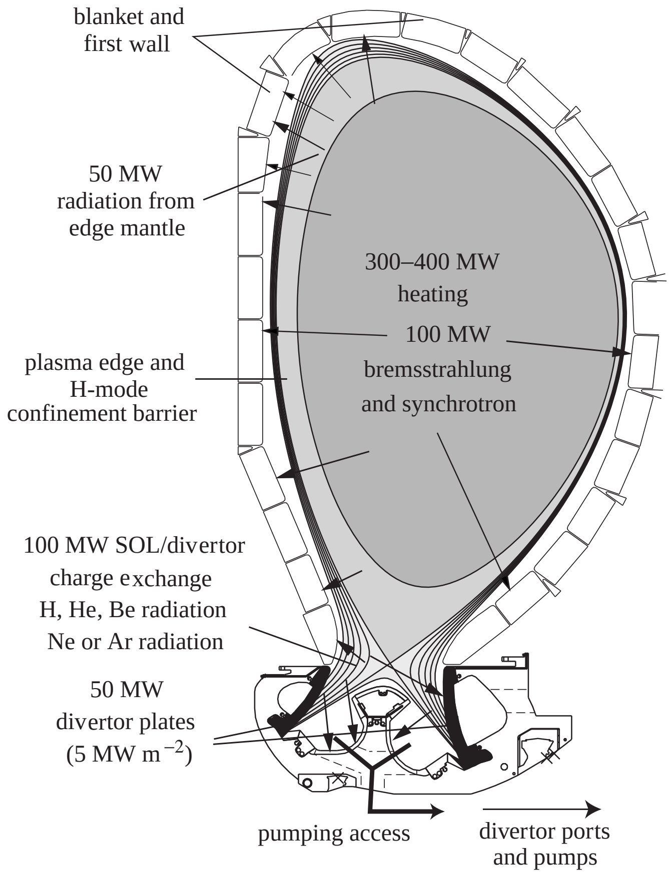

Figure 5. ITER power and particle control system.

ment and validation and ITER-orientated experiments in existing tokamaks, notably ASDEX Upgrade, DIII-D, JET, C-Mod and JT-60U. On the basis of such contributions, the selected design approach for divertor operation is now well-established and favourable in several respects: low impurity level in the main plasma, low neutral density in the main chamber, well-distributed radiation, low peak power on the divertor target plates, and adequate He pumping.

(*d*) *Advanced operating modes*

The poloidal field system is able to control reduced current steady-state scenarios with higher plasma elongation and triangularity within the reference in-vessel components. The use of heating during current ramp up will make it possible to create a hollow (reversed central magnetic field shear) current profile at the start of the current flat top of the pulse; such profiles can then be sustained by non-inductive means. The flexibility thus provided by the PF and by the heating and current drive systems will allow ITER to operate in a number of alternative scenarios, including the 'advanced' modes which use reversed (or negative) central magnetic field shear and are orientated towards steady-state operation having a large component of bootstrap current. Such modes of operation are now being widely studied in vari-

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ous pulsed tokamak experiments, with promising results. If future experiments will establish that these regimes can be controlled and maintained with similar ion and electron temperatures providing at the same time the optimal pressure profiles, with suitable values for other plasma characteristics such as β and helium exhaust, and with adequate density and confinement properties at the smaller plasma current, then ITER would be able to explore steady-state operation with total plasma current in the range 12–15 MA and with high bootstrap current fraction maintained by 100–150 MW of plasma heating and current drive with 1–1.5 GW of fusion power. This approach is the primary candidate for ITER steady-state scenarios.

(*e*) *Physics conclusions*

ITER will be a major step from existing tokamaks (e.g. factor *ca.* 10 in plasma volume and *ca.* 100 in power handling). Although each of the necessary key plasma parameters, such as temperature, density and confinement time, has already been achieved separately, ITER's projected performance will represent significant extrapolations of simultaneous performance. However, the achievements of present tokamaks and the coordinated collation of results and analysis covering wide ranges of parameters have provided a comprehensive basis that allows confidence in projections. Accordingly, ITER should achieve its set objectives—demonstration of controlled ignition and sustained burn of D–T plasmas, with steady state as an ultimate goal.

Nevertheless, uncertainties must inevitably remain in such extrapolations. Some will be reduced through further experiment and analysis, but many may only be resolved through construction and operation of a machine such as ITER. Indeed the domains of uncertainty can be seen as setting the agenda for ITER's operating programme in order to serve its programmatic objective and to lay the foundations for designing a demonstration fusion power station.

In addition, optimizing ITER operation and exploitation will depend on continuing wide-ranging programmes of complementary experimental studies addressing the areas of uncertainty in the performance projections so as to enhance understanding of features of ITER's expected operations and to optimize the detailed design of components and diagnostics and to plan an efficient operating programme. Key areas of interest for continued study include further analysis of applicable β and density values and of the L–H-mode power threshold, divertor operations with ITER-like approach of high power radiation with impurities and limited power densities at the target plates, characterization of advanced modes of tokamak operation and their potential applicability to ITER, and further experience and analysis of plasmas with significant proportions of α-particle heating.

(*f*) *Engineering assessment of ITER*

The main challenge of ITER engineering design has been to address a wide range of major interrelated technical challenges and to combine the solutions in each area to provide a coherent, integrated system. The overall policy for engineering has been to use established approaches and well-characterized materials and to validate their application to ITER through detailed analysis, manufacturing and testing of large/full-scale models and prototypes of the critical systems. Of particular importance has been the unprecedented combination of superconducting coils, plasma-facing components for a full-power reactor, and advanced remote handling capability to provide satisfactory availability and the flexibility appropriate to an experimental device.

As the designs have evolved to maturity over the period of the EDA, they have been supported by extensive technology R&D to validate the key aspects, including development and qualification of the applicable technologies and development and verification of industrial level manufacturing techniques with related quality assurance (QA). In particular, seven large, multi-party projects with significant industrial content have been pursued, each devoted to one of the main ITER engineering systems. The projects are designed to establish the chosen technologies and all major manufacturing and fabrication processes at levels/scales that allow direct application to actual construction. Results flowing from these projects influence the design development and assessments. Similarly, industrial studies undertaken primarily for costing purposes have served also to influence the choices of manufacturing processes and to confirm manufacturing feasibility.

Two of the projects are directed towards developing superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. The Central Solenoid Model Coil Project and the Toroidal Field Model Coil Project drive the development of the ITER full-scale conductor including strand, cable, jacket/conduit and terminations, and integrate the supporting R&D programmes on insulators; joints; conductor AC losses and stability; Nb_3Sn conductor 'wind, react and transfer'† processes; and quality assurance. In each case the Home Teams concerned are collaborating to produce relevant scale model coils and associated mechanical structures and the required testing facilities. The total planned production of 29 t of Nb_3Sn strands, from seven different suppliers throughout the four Parties, has been produced and qualified. The strands have been jacketed, the coils have been wound and the difficult heat treatment to react the superconducting alloy at 700 °C for about 250 h has proceeded successfully without degrading the mechanical properties of the jacket. Figure 6 shows coil windings for the model CS after heat treatment.

Three Projects focus on key in-vessel components, including development and demonstration of necessary fabrication technologies and initial testing for performance and assembly/integration into the tokamak system.

In the Vacuum Vessel Sector Project, the main objective is to produce a full-scale sector of the ITER vacuum vessel with the necessary control of welding distortions within required tolerances to achieve dimensional accuracy. Each 18° sector weighs about 220 t, including as much as 20 t of weld material. The key technologies have been established and two full-scale vacuum vessel segments (half-sectors) have been fabricated in Japanese industry to tolerances of 2.5 mm using a range of welding techniques including tungsten inert gas (TIG), metal active gas (MAG) and electron beam (EB), and the two half-sectors have been welded together. Figure 7 shows a completed half-sector ready to be installed into the jig for attachment to its mate.

The in-vessel components that face the plasma must operate reliably and predictably in the demanding operating conditions of ITER operation, including extreme heat fluxes and neutron fluence, electromagnetic forces from VDEs and disruptions and internal temperature gradients. The systems must also be amenable to remote

† Nb_3Sn is a brittle material with strain-dependent performance; it is therefore necessary to wind the conductor in its jacket before forming the Nb_3Sn phase by reaction at high temperature. Its electrical insulation is done later when it is transferred to its final location in the coil.

Figure 6. ITER CS model coil after heat treatment.

maintenance and replacement; for this reason a modular approach is adopted. In general the components must incorporate appropriate combinations of armour, heat-sink and stuctural materials and the development and validation of practical material bonding and fabrication techniques.

The Blanket Module Project is aimed at producing and testing full-scale modules of primary wall elements and at demonstrating prototype integration in a model sector and the Divertor Cassette Project aims to demonstrate that a divertor can be built within tolerances and to withstand the thermal and mechanical loads imposed on it during normal operation and during transients such as ELMs and disruptions. For both projects the materials technologies have been established and validated and prototype modules have been manufactured for characterization and testing.

The last two of the Large Projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in the vacuum vessel in tritium-contaminated and activated conditions on reasonable timescales. These technologies should provide the flexibility needed for ITER to pursue its scientific and technical goals with acceptable repair times and availability while satisfying stringent safety and environmental requirements. In this area, full-scale tools and facilities should be developed, and their testing extended on a long period of time in order not only to check the right procedures, but also to optimize their use in detail and minimize the intervention time.

The Blanket Module Remote Handling Project is aimed at demonstrating that the ITER Blanket modules can be replaced remotely. The procedures have already been

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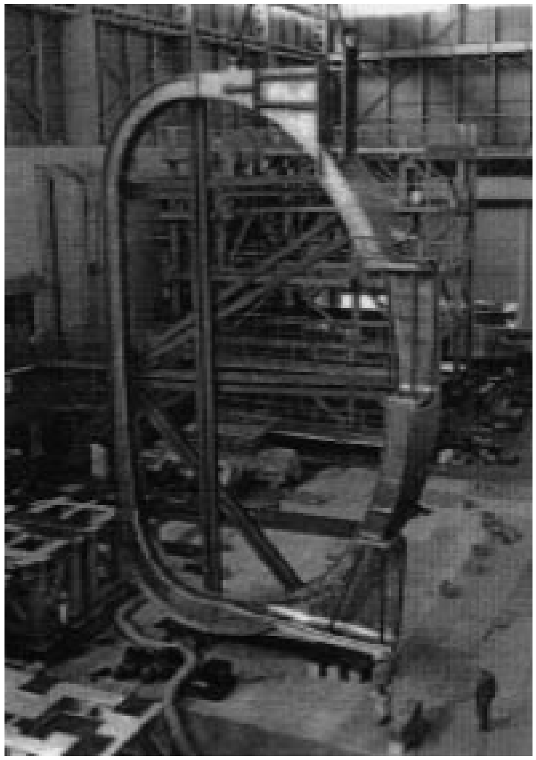

Figure 7. Half-sector prototype of the vacuum vessel.

successfully demonstrated at about one-fourth scale so as to reduce the risk/cost for the development of full-scale equipment. Work is now in progress on full-scale demonstration. In the Divertor Remote Handling Project, the main objective is to demonstrate that the ITER divertor cassettes can be removed remotely from the vacuum vessel and remotely refurbished in a hot cell. This involves the design and manufacture of full-scale prototype remote handling equipment and tools and their testing in a Divertor Test Platform to simulate a portion of the divertor area of the tokamak and a Divertor Refurbishment Platform to simulate the refurbishment facility. Construction of the necessary equipment and facilities is complete and testing is in progress.

The technical output from the Seven Large R&D Projects has direct importance in validating the technologies and related manufacturing techniques and QA for ITER and in supporting the manufacturing cost estimates for key cost drivers. The activities are foreseen as continuing during the EDA extension to further the prototype component testing and/or to optimize their operational use. Their performance also offers insights for a possible future collaborative construction activity. Already much valuable and relevant experience has been gained in the management of industrial-scale, cross-party ventures. The successful progress of the projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

The maturity of the systems designs, supported by the information flow from the related R&D, has enabled preparation of realistic assembly procedures and overall

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project scheduling and has enabled the Parties' industries to undertake thorough cost studies around more than 80 'procurement packages', each representing a plausible potential supply contract. The estimates so derived are in line with targets set at the start of the EDA. The resulting schedule and cost estimates thus provide a firm basis for Parties' consideration of the possible future development of the project.

5. Conclusions and observations

Over the past six years, the ITER collaboration has built efficiently on the impressive achievements of the Parties' fusion development programmes over the past two decades and delivers to the Parties what they sought at the start of the EDA—a detailed, complete and fully integrated engineering design of ITER and all technical data necessary for future decisions on the construction of ITER. The design as described in the FDR serves ITER's programmatic objective to demonstrate the scientific and technical feasibility of fusion energy for peaceful purposes. The design is feasible from engineering and technological point of view; ITER can be manufactured to specifications and will be capable of meeting its operating objectives. The information from the EDA and from the supporting physics and technology R&D is now available to all Parties for use in their domestic fusion programmes or in some continued international collaboration.

The experimental results from the present generation of tokamak experiments and related advances in understanding have been analysed and compared within the ITER framework to provide a reliable database for ITER Physics projections (ITER 1999*b*). The reference operating scenarios, based on well-established and characterized modes of operation, give high levels of confidence in extrapolation to ITER parameters. Such projections show potential for ignition across a wide operational domain and assurance of extended burn in line with the technical objectives set. ITER stands squarely on the path to ignited fusion reactors.

By its scale and power rating, ITER will represent the core of a working power reactor. ITER will have the experimental capability to address the key issues needing resolution in order to define and to optimize the design of the first working fusion power reactor, including the possibility to explore advanced modes of tokamak operation and to prove their potential for reactor application.

The technologies used in the design have been qualified. They are available not only for ITER as defined but also in a broader perspective for almost any project of a fusion device. Large multi-party projects have demonstrated the manufacturing feasibility of the design for the key components. Prototypes and large-scale models are under test to confirm operating performance and to understand operating margins. The mature state of the engineering design has enabled rigorous and realistic costing studies of plausible 'procurement packages' by the Parties' industries. Work undertaken at the plant system level to understand and confirm the operational performance of components working together as integrated systems has established the coherence of the parts with the whole across all the main functional domains.

The EDA has proved to be an efficient and successful international collaboration which has respected the principal of equality of the Parties and has stimulated significant progress in fusion physics and technology by providing a clear focus, discipline and concrete targets for joint endeavour. Each Party, having access to the complete set of results, will realize a high leverage of its financial contribution should it choose

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to use the results in its domestic programme. The progress on major R&D projects which integrate industrial contributions from all Parties provides confidence in the possibility of joint construction of ITER in an international framework.

Bringing ITER to fruition through continued international collaboration would provide for all participants both the breakthrough into the regime of ignition and long burning plasmas and the integrated demonstration of key fusion technologies. As such, ITER offers a unique opportunity for the Parties to make a significant next step towards realizing the promise of fusion as a source of useful energy on Earth.

This paper is a report of work done in the framework of the International Thermonuclear Experimental Reactor Engineering Design Activities conducted by the European Atomic Energy Community, Japan, Russian Federation, and the United States under the auspices of the International Atomic Energy Agency.

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Discussion

R. S. Pease (*West Ilsley, Newbury, UK*). Is superconducting technology necessary for ITER? Do we have adequate experience of this technology as applied to fusion, for a big investment like ITER?

R. Aymar. The superconducting technology is an absolute required technology for ITER. Long pulses (1000 s or more) make this requirement inevitable, from a practical and economic point of view (in particular, compared to a resistive magnet, besides the power required, the radial build of the TF magnets is much smaller with superconducting conductors). The experience acquired with this technology in fusion research is limited (larger in other areas) and no exceptional design/operational failures have been encountered. But, for ITER, a large R&D programme is ongoing to add more manufacturing experience, and more knowledge of operational margins.

D. C. Robinson (*UKAEA Fusion, Culham Science Centre, Abingdon, UK*). After six years of engineering design activity, what, from an engineer's or a designer's view is the most important for physicists to address in the near future?

R. Aymar. In my present assessment, the most important question to be addressed in the near future by physicists is related to the possibility of achieving a core average density above (say 1.5 times) the empirical Greenwald limit. This is a need not only for ITER in the reference scenario, but also for advanced modes, and equally true for future reactors. The solution to this problem is probably related to the possible separation of edge physics and core physics (L–H-mode transitions) and to the possible fuelling inside the ELM's concerned boundary thickness (high field pellet launch). This answer does not imply that no other issue remains to be classified and/or quantified.

G. H. Wolf (*Institute for Plasma Physics, Jülich, Germany*). I would like to address the question of divertor target materials and the problem of tritium burial. Does Dr Aymar consider solutions other than carbon-based materials like tungsten, and related to this, does he consider sawteeth as a necessary mechanism for purifying the core plasma from accumulation of medium to high Z impurities?

R. Aymar. For the first part of the question, I think carbon-based (CFC) materials are inevitable at the strike point of the plasma on the divertor target. Only this material can resist the large pulsed power from ELMs and from the thermal phase of disruptions. However, it is necessary to limit as much as possible the use of carbon in interaction with the plasma, because of the co-deposition of tritium with carbon eroded from the first-wall surfaces, which is difficult to remove and builds up tritium in the machine, a safety concern. For the second point, I refer to the comment of Dr Hawryluk.

R. J. Hawryluk (*Princeton Plasma Physics Laboratory, USA*). I would like to comment on Dr Wolf's question regarding sawteeth. In TFTR, sawteeth were suppressed and high impurity accumulation was not observed in supershots. Sawteeth are not required to avoid impurity accumulation; however, the scaling of the turbulent pressure for impurity has not been established.

R. Aymar. I agree with this comment and its extrapolation to ITER.

M. G. Haines (*Imperial College, London, UK*). Will ITER have control of plasma flow (toroidal and poloidal rotation) to influence internal thermal barriers, onset of H-mode, and occurrence of locked MHD modes due to error fields?

R. Aymar. At the present time, the ITER design considers only the limited capability of plasma rotation induced by neutral beam injection, corresponding to 50 MW at 1 MeV. Error fields are prescribed to be limited to smaller than damaging values, by the use of control coils (three sets of four coils).

C. Windsor (*UKAEA Fusion, Culham Science Centre, Abingdon, Oxfordshire, UK*). I would like more details on the full-scale vessel prototype sections, in particular the tolerance, defect frequency, and expected lifetime?

R. Aymar. The full-scale vessel section was built to establish manufacturing processes (welding, NDT, etc.) and QA associated, in order to provide the high tolerance required for a fully welded torus structure. In order to make credible the possibility of replacing a sector of the ITER machine, after operation and activation of the walls, the large poloidal welds have been made with automatic (remote-handled) welding (cutting) tools. The tolerances achieved are well inside the specifications; reports on results achieved are available from the ITER/EDA final documentation.

M. Keilhacker (*JET Joint Undertaking, UK*). Earlier this year the ITER Director and his team were confronted with the fact that one of the partners would no longer participate in the ITER design, that had been worked on for six years, or longer if one includes the conceptual design phase. The ITER Council responded by setting up a Special Working Group. It would be helpful if Dr Aymar could tell us about the remit of this group, and its consequences for ITER.

R. Aymar. The ITER programme has the broad objective of demonstrating the scientific and technological feasibility of fusion energy. In 1992 a Special Working Group defined a set of detailed scientific and technical objectives, which were endorsed by the Council at the time. The scientific objective was for controlled ignition lasting about 1000 s, and at least 1 MW m^{-2} average wall loading, in the form of 14 MeV neutrons. A secondary objective was to aim for steady-state operation using non-inductive current drive. The technical objectives related to the testing of nuclear components. In particular, blanket modules breeding tritium, relevant to a high-temperature power plant, should operate at a fluence of not less than 1 MWy m^{-2}.

The new Working Group, set up by the ITER Council in February 1998, was working to reduce these objectives in such a way that the ITER cost would be reduced significantly, say by 50%. The group reporting back to the council in early July 1998, recommended a Q value in a burning plasma of not less than 10, but which would not preclude the possibility of ignition, and a power at the boundary of not less than 0.5 MW m^{-2}. The objective of steady-state operation remained but with a Q value in the region of 5. The technical objective on fluence was reduced to 0.3 MWy m^{-2}. It was agreed that if this objective were achieved with a pulsed plasma, then the number of pulses should be limited to a few tens of thousands. The proposals were accepted by the Council, who noted the view of the Working Group that these objectives appeared compatible with a cost reduction of *ca.* 50%. The Group was also asked to quantify the consequences that the reduction would have on the future development of a fusion power plant.

H. Bruhns (*European Commission, Brussels, Belgium*). To say that the reference ITER design was not accepted is not correct. The domestic assessments we have made accepted the design as it stands. The ITER Engineering Design Report was the first comprehensive fusion power plant design. It integrated all the physics and technological aspects into a machine that could be built. The question is now to exploit the design that we have to the maximum.

M. Keilhacker. This is certainly true, and nobody doubts it. As Dr Aymar said, the various assessment groups fully endorsed the design, and commented positively on the excellent work done. The ITER design process has focused experimental, theoretical and technological work in a way which would never have been possible otherwise. However, for cost reasons there is no longer a global acceptance of the reference ITER design.

H. Bruhns. ITER, as described in the ITER Final Design Report, has a robust physics basis with some margins in operational aspects, which give high confidence that the device will achieve its goals. This is the prime conclusion of the European domestic ITER FDR assessment. Is, in the light of current considerations to reduce the size of ITER, the issue of ignition an essential design consideration?

R. Aymar. While in inertial fusion the self-burn of a pellet is of unconditional necessity, in magnetic confinement fusion the issue of ignition is losing relevance when the operation of a reactor is considered in more detail, since control of the plasma by a corresponding power input is necessary. A reasonable target for a device will be a high Q (e.g. $Q \sim 50$) but not ignition, i.e. $Q = \infty$. Also, it is frequently believed that achieving 'ignition' or near-ignition in magnetic and inertial fusion would indicate the same advanced progress towards a fusion power plant. This is the case for magnetic fusion but not necessarily so for inertial concepts. In lay discussion, these aspects are frequently not well understood.

H. Bruhns. Does a smaller ITER evolve from the present design, or is there a fundamentally new design effort and corresponding cost, time and manpower necessary?

R. Aymar. Essential elements in the overall workload of the ITER EDA were to establish design criteria, design tools, to install mechanisms for system integration and 'horizontal' design aspects like safety; these investments are directly applicable to the design of a smaller ITER. Furthermore, the design effort for specific components, to a large extent, remains valid even in a smaller or modified ITER.

K. Lackner (*Tokamak Physics Division, Garching, Germany*). We should look at other factors that have changed since 1992, besides the financial factors. One is the advent of the 'advanced scenarios' of plasma control. The 1992 ITER design had a steady-state option, but this was primarily to be achieved by improving the efficiency of the current drive. With the advanced scenarios we have a totally different paradigm with the possibilities for the steady state determined by what you can control.

R. Aymar. It is not yet documented whether the new advanced modes have any real advantage over the present ITER inductive mode. We have shown that we can run the reference ITER for over 10 000 s (2.8 h), which is close to the practical requirement for a power plant of around 8 h. To be efficient, the new modes require a thermal barrier at a large radius and a plasma pressure profile as flat as possible. Besides the control of this barrier location, the essential issue of particle control (density fuelling in the centre, and helium exhaust) is presently without answer.

P. Vandenplas (*Laboratory for Plasma Physics, Royal Military Academy, Brussels, Belgium*). I would like to add an important 'nuance' to what Professor Lackner has said concerning the advanced modes. We may have to wait many years, perhaps even 10, before we know if these potentially promising advanced modes have a chance of being implemented in steady state. So even a reduced ITER (at a cost of 50–60%) has to be thought of as a 'classical' long-pulse tokamak having, however, the capability of implementing these advanced modes if proven successful on a long timescale.

R. Aymar. Professor Vandenplas makes a valid point on the reference scenarios of ITER.

J. G. Cordey (*JET Joint Undertaking, UK*). If we are to have another three years of design studies for the ITER-LITE, is it not possible that, before we start, we can get a commitment from the partners that they will actually build a device at the end of it?

R. Aymar. I could not agree more! I have done my best, but it is difficult to get governments to commit funds before they have to.

J. Sheffield (*Energy Technology Programs, Oak Ridge National Laboratory and the Joint Institute for Energy and the Environment, University of Tennessee, USA*). Dr Aymar made an interesting point that being clear whether the Greenwald limit on density was a hard limit was very important to both the ITER and to the usefulness of the power plant. Many people believe that advanced operating modes (with performance better than the ITER reference performance) are also essential. In this regard, it is useful to look at ITER-EDA and ITER-LITE in regard to performance against different modes of operation:

Reference mode	ITER-EDA ignites, ITER-LITE $Q \sim 10$.
Better mode	ITER-EDA larger margin, ITER-LITE $\rightarrow Q = \infty$.
Worse mode	ITER-EDA $Q \sim 4$–10, ITER-LITE $Q \rightarrow 2$–5?

At the lower end the output of the two machines may be mainly on remote handling; long-pulse, high-power density, divertor operation; radiation hardened diagnostics; heating and fuelling systems; and some $\gg$JET/TFTR D–T physics. At the middle (reference performance) LITE does say 80–90% of EDA. At better than reference LITE does 95% of EDA. It may be that the modes in which LITE is not very useful and the EDA is very useful are a very small part of the possible operating space.

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