# Overview of ITER-FEAT — The future international burning plasma experiment

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**Abstract.** The focus of effort in ITER EDA since 1998 has been on the development of a new design to meet revised technical objectives and a cost reduction target of about 50% of the previously accepted cost estimate. Drawing on the design solutions already developed, and using the latest physics results and outputs from technology R&D projects, the Joint Central Team and Home Teams, working together, have been able to progress towards a new design which will allow the exploration of a range of burning plasma conditions, with a capacity to progress towards possible modes of steady state operation. The new ITER design, whilst having reduced technical objectives from those of its predecessor, will nonetheless meet the programmatic objective of providing an integrated demonstration of the scientific and technological feasibility of fusion energy. The main features of the current design and of its projected performance are introduced and the outlook for construction and operation is summarized.

## 1. Introduction

The motives for developing fusion as an energy source lie in its attractions as a possible large scale contributor to the energy mix in the second half of this century, with a virtually inexhaustible fuel supply, good safety characteristics and an acceptable environmental impact. These incentives have been driving the world fusion research programme since its inception. Continuing population growth and the growing economic aspirations of all humanity, combined with the increasing international concern over the potential climatic threat from dependence on fossil fuels, reinforce the case for providing a range of practical energy options for sustainable energy supply. Establishing the fusion energy option can make a critical contribution to the welfare of future society.

After the impressive progress in recent years in bringing the fusion research programmes to the threshold of reactor conditions in both physics and technology, the imperatives for future progress in fusion are now:

(a) In physics, to move across the threshold into fusion conditions that current machines cannot access, in particular to reach the point at which energetic $\alpha$ particles become the main source of plasma heating and the principal determinant of plasma behaviour;

(b) In technology, to combine and test key features of fusion reactor technology in reactor relevant conditions;

(c) In terms of public acceptance, to demonstrate in practice the favourable safety and environmental characteristics of fusion.

## 2. Summary of progress of ITER to 1998

The ITER project has its origins in the common recognition within the leading fusion communities worldwide of the need for a next step experiment with the programmatic objective of demonstrating the scientific and technological feasibility of fusion energy for peaceful purposes [1]. Building on the performance advances of leading machines and a wide database from both small and large machines, ITER has the core of a working fusion reactor and is thus designed to embody the next step machine that serves the imperatives stated in the Introduction.

The technical conditions of a burning plasma experiment themselves demand the use of advanced fusion technologies. In addition, the integration of burning plasma physics with fusion technologies will be an essential step on the strategic path towards establishing the fusion energy option. In enabling, in one device, full exploration of the physics issues, as well as a proof of principle and testing of some key

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technological features of possible fusion power stations, ITER would provide the basis for the subsequent design of the first demonstration fusion power station. That would demonstrate the reliable generation of electricity, before a prototype power plant could be envisaged for commercial use.

The ITER collaboration was set up to provide its Parties (Euratom, and the Governments of Japan, the Russian Federation and the United States of America) with the option to make the next step within the framework of global collaboration in which participants could pool their accumulated scientific and technological expertise, share the burden of costs and secure a degree of political commitment consistent with the scope and timescale of the task.

Six years of joint work under the EDA Agreement [1] yielded a mature design, cost estimate and safety analysis — the ITER 1998 design [2] — that was supported by a body of validating physics and technology R&D. The 1998 design met the detailed objectives that had been set for it in 1992, focusing on plasma ignition (plasma power amplification, $Q = \infty$) in reference inductive operation, with margins in physics and technology to allow for unqualified design concepts, whilst satisfying the cost target originally set for it.

At that point, the Parties negotiated a three year extension to the original EDA (the USA terminated its participation in 1999) in order to prepare for a decision to build. At the same time, in view of financial pressures, the Parties undertook a review of the detailed technical objectives in order to explore the scope for cost savings that might be possible whilst still serving ITER's overall programmatic objective.

## 3. Revised guidelines for ITER design

The revised guidelines for ITER [3] require in terms of plasma performance

(a) To achieve extended burn in inductively driven plasmas at $Q > 10$ for a range of scenarios, whilst not precluding the possibility of controlled ignition;
(b) To aim at demonstrating steady state operation through current drive at $Q > 5$.

In terms of engineering performance and testing, the new design should

(1) Demonstrate availability and integration of essential fusion technologies;
(2) Test components for a future reactor;
(3) Test tritium breeding module concepts, with a 14 MeV average neutron power load on the first wall $\geq$0.5 MW/m$^2$ and an average neutron fluence $\geq$0.3 MW a/m$^2$.

The new design should aim for a cost target of about 50% of the costs of the 1998 ITER design.

## 4. Convergence to the new design point

As a first approach to identifying designs that might meet the revised objectives, system codes were used in combination with costing algorithms to establish possible feasible design points for further analysis. The systems approach combined a detailed plasma power balance and boundaries for the window of plasma operating parameters, providing the required range of $Q$ for the DT burn, with engineering concepts and allowable limits. The four key parameters — aspect ratio (plasma major/minor radius), peak toroidal field, plasma (cross-section) elongation and flux available to drive an inductive burn — are intimately linked, allowing options in the systems analysis to be characterized principally by the aspect ratio, in addition to the device size. The access to the plasma (e.g. for heating systems) and allowable elongation (simultaneously constrained by plasma vertical position and shape control, and by the necessary neutron shield thickness) are functions of aspect ratio.

On this basis, the system studies indicated a domain of feasible design space, with aspect ratios in the range 2.5–3.5 and a plasma major radius around 6 m, able to meet the modified requirements, with a shallow cost minimum across the range.

In order to provide a basis for rigorous exploration and quantification of the issues and costings, representative options that span an appropriate range of aspect ratio and magnetic field were selected for further elaboration and more comprehensive consideration. With this more tangible appreciation of the key issues, combined Joint Central Team/Home Team Task Forces were able to converge progressively towards a preferred outline design point taking the following as guiding principles:

(a) To preserve, as far as possible, physics performance and margins against the revised targets, and the scope for experimental flexibility, within the cost target and relevant engineering constraints;

**Table 1.** Main parameters and dimensions of the ITER plasma

| | |
|---|---|
| Total fusion power | 500 MW (700 MW) |
| $Q$ (fusion power/auxiliary heating power) | ≥10 |
| Average 14 MeV neutron wall loading | 0.57 MW/m$^2$ (0.8 MW/m$^2$) |
| Plasma inductive burn time | ≥300 s |
| Plasma major radius | 6.2 m |
| Plasma minor radius | 2.0 m |
| Plasma current, $I_p$ | 15 MA (17.4 MA) |
| Vertical elongation at 95% flux surface/separatrix | 1.70/1.85 |
| Triangularity at 95% flux surface/separatrix | 0.33/0.49 |
| Safety factor at 95% flux surface | 3.0 |
| Toroidal field at 6.2 m major radius | 5.3 T |
| Plasma volume | 837 m$^3$ |
| Plasma surface | 678 m$^2$ |
| Installed auxiliary heating/current drive power | 73 MW (100 MW) |

(b) To exploit the recent advances in the understanding of key physics and engineering issues drawn from the results of the ITER voluntary physics programme and the large technology R&D projects (Section 6);

(c) To maintain the priority given to safety and environmental characteristics, using the principles, analyses and tools developed through the ITER collaboration up to the present time.

The resulting configuration for the new design of ITER [4] represents an appropriate balance of the key technical factors, the cost target and conservative energy confinement scaling.

## 5. Parameters and plasma performance of the new ITER design

The main parameters and overall dimensions of the ITER plasma are summarized in Table 1. The figures show the parameters and dimensions for nominal operation. The numbers in brackets represent maximum values under specific limiting conditions, and their implementation may require, in some cases, additional capital expenditure. The cross-section of the tokamak is shown in Fig. 1, and a cutaway view of the tokamak and the subsystems in the cryostat is shown in Fig. 2. The performance is discussed in more detail elsewhere [5–7].

### 5.1. Inductive operation

The reference operating scenario for inductive operation is the ELMy H mode (i.e. high confinement mode with edge localized MHD modes present), and the rules and methodologies for projection of plasma performance to the ITER scale are those established in the ITER Physics Basis (IPB) [8], which has been developed from broadly based experimental and modelling activities within the magnetic fusion programmes of the ITER Parties.

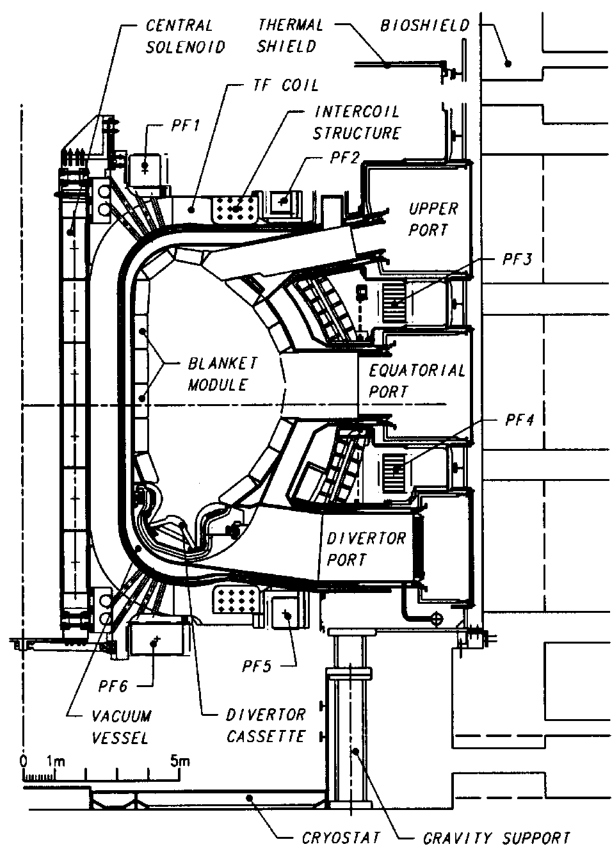


**Figure 1.** Cross-section of the ITER tokamak.

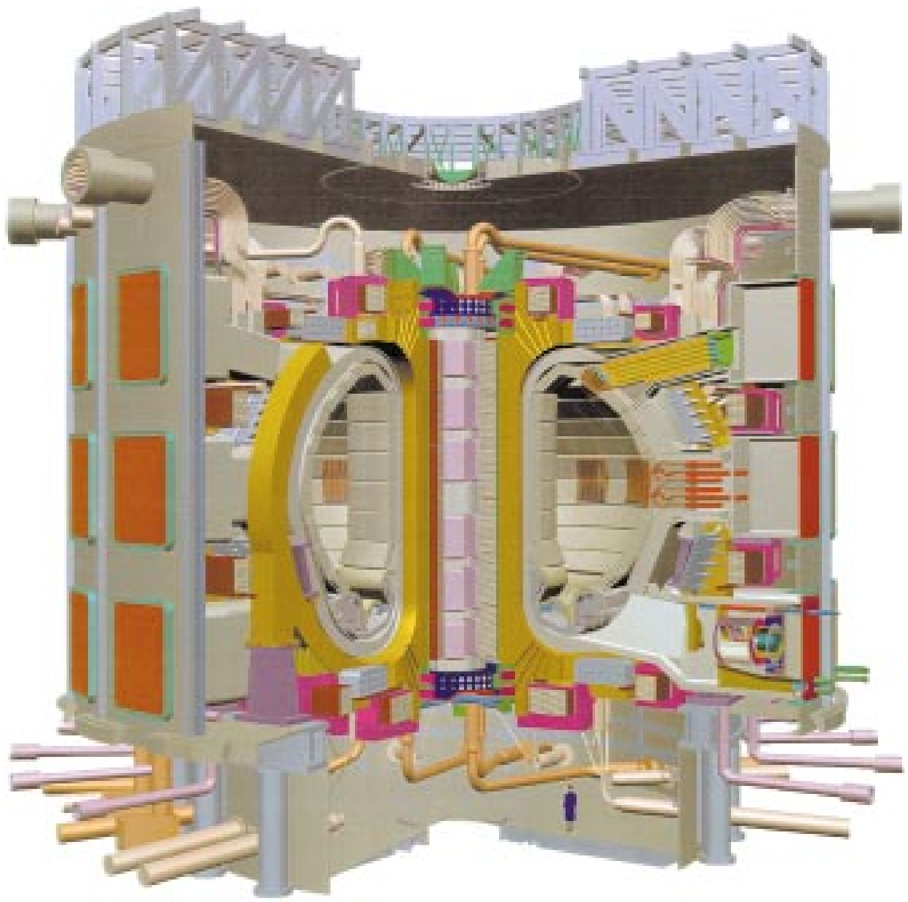

**Figure 2.** Cutaway view of ITER.

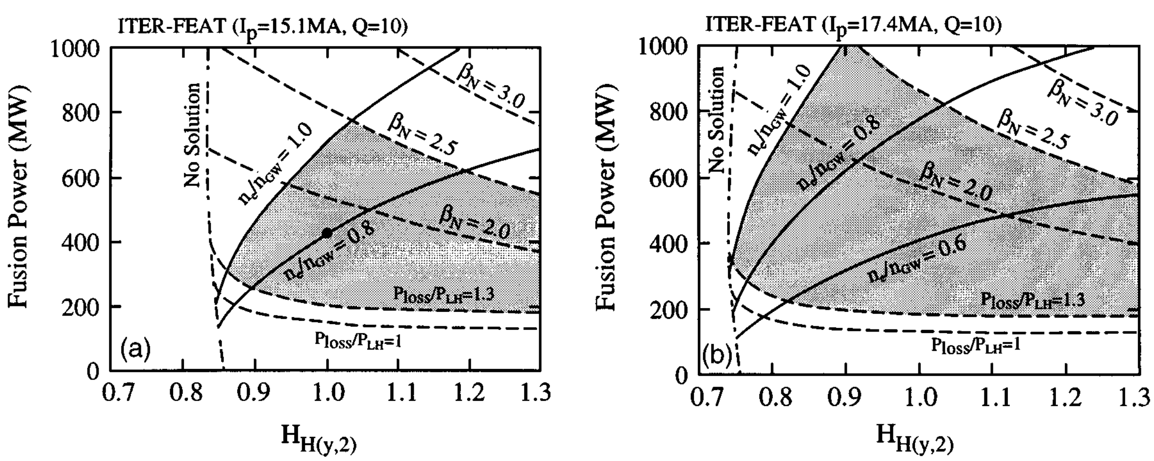


**Figure 3.** (a) The $Q = 10$ domain (shaded) for $I_p = 15.1$ MA ($q_{95} = 3.0$). The (b) $Q = 10$ domain (shaded) for $I_p = 17.4$ MA ($q_{95} = 2.6$).

The key limiting factors for inductive operation are normalized $\beta$ ($\beta_N = \beta(\%)a(\mathrm{m})B(\mathrm{T})/I(\mathrm{MA})$), the density in relation to the Greenwald limit ($n/n_{GW}$, where $n_{GW}(10^{20}\ \mathrm{m}^{-3}) = I(\mathrm{MA})/\pi a(\mathrm{m})^2$) and the L–H mode transition power threshold (where $P_{L\text{–}H} = 2.84 M^{-1} B_T^{0.82} \bar{n}_e^{-0.58} R^{1.00} a^{0.81}$ in units of MW, amu, T, $10^{20}$ $\mathrm{m}^{-3}$ and m). A view can be formed of the range of possible plasma parameters at which $Q = 10$ by analysing, with flat density profile across the plasma, possible operational domains in relation to the above limiting factors, for given values of $Q$, plasma current and confinement enhancement factor $H_H$, as illustrated in Figs 3(a) and (b). (Confinement time and $H_H$ are defined by

$$\tau_{E,th}^{IPB98(y,2)} =$$

$$0.0562 H_H I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_x^{0.78}$$

where the units are s, MA, T, MW, $10^{19}$ m$^{-3}$, amu and m.)

It is evident from Figs 3(a) and (b) that:

(a) For operation at a safety factor at the 95% flux surface, $q_{95} = 3$, the fusion output power from the new ITER design is in the region of 200–700 MW (at $H_{H(y,2)} = 1$), corresponding to a mean separatrix neutron flux (mean neutron wall loading) of 0.23–0.80 MW m$^{-2}$, so that the device retains a significant capability for technological studies, such as tests of tritium breeding blanket modules.

(b) The margin in the H mode threshold power (at $H_{H(y,2)} = 1$) is significantly greater than the predicted uncertainty derived from the scaling.

(c) The device has the capability of $Q = 10$ operation at $n/n_{GW} \approx 0.7$ and $\beta_N \approx 1.5$ (when $H_{H(y,2)} = 1$).

The results also illustrate the flexibility of the design, its capacity for responding to factors which may degrade confinement while maintaining the goal of extended burn $Q > 10$ operation, and, by the same token, its ability to explore higher $Q$ operation as long as energy confinement times consistent with the confinement scaling are maintained. For instance, operation at a range of $Q$ values is possible and values as high as 50 can be attained for nominal parameters if $H_{H(y,2)} \approx 1.2$ in an improved confinement mode, for example, reversed shear (the normalized rate of change of safety factor perpendicular to the flux surface), shallow shear with an internal transport barrier or, as presently observed, if operation at lower $q_{95}$ ($\approx$2.6) can be sustained without confinement degradation [5].

Ignition can be achieved, after a few seconds pulse with 73 MW of auxiliary power, with $I_p = 17$ MA, $n/n_{GW} = 0.8$, either for a period limited to about 40 s during the buildup of helium impurity in the plasma with a helium particle confinement time $\tau_{He}/\tau_E = 5$ and $H_{H(y,2)} = 1$ or, for as long as the burn flux allows, if the $H_H$ factor were improved by 10%.

### 5.2. Steady state operation

Steady state operation can be regarded as an ultimate goal of the tokamak development programme. Coherent and complete scenarios with supporting databases for possible modes of steady state operation do not yet exist. The next step experiment should thus be capable of exploring the requirements for steady state operation. It must also have the built-in flexibility to exploit new developments in the fusion programme as they arise. In ITER it is likely that a variety of candidate steady state modes of operation will be investigated and it is therefore essential that the requisite tools for the control of plasma geometry and radial variations (profiles) of key parameters are available.

On-axis and off-axis current drive capabilities will enable plasmas with shallow or negative shear configurations to be sustained, in the latter regime simultaneously maintaining the central safety factor well above unity, while the minimum safety factor is held above two. ITER is designed with a poloidal field system capable of controlling the more highly shaped plasmas characteristic of high poloidal beta $\beta_p$ operation, and with methods to allow reliable long pulse operation at high beta, including techniques for the stabilization of neoclassical MHD tearing modes (using electron cyclotron current drive) and resistive wall MHD modes (using correction coils).

For the new ITER design, possible operational scenarios are being considered for steady state operation in line with some present experiments and able to provide $Q = 5$, for example, high currents (12 MA) with monotonic $q$ and shallow shear, and modest currents (9 MA) with negative shear. High current steady state operation requires all the current drive power (100 MW) available for ITER, but the requirements on confinement ($H_H \approx 1.2$) and beta ($\beta_N \approx 3$) are modest. Low current steady state operation requires more challenging values of confinement improvement: $H_H \approx 1.5$ and $\beta_N \approx 3.2$–3.5. Performance predictions for these modes of operation are much less certain than those for inductive operation, with a larger power to the divertor. In particular, the operating space is sensitive to assumptions about current drive efficiency and plasma profiles.

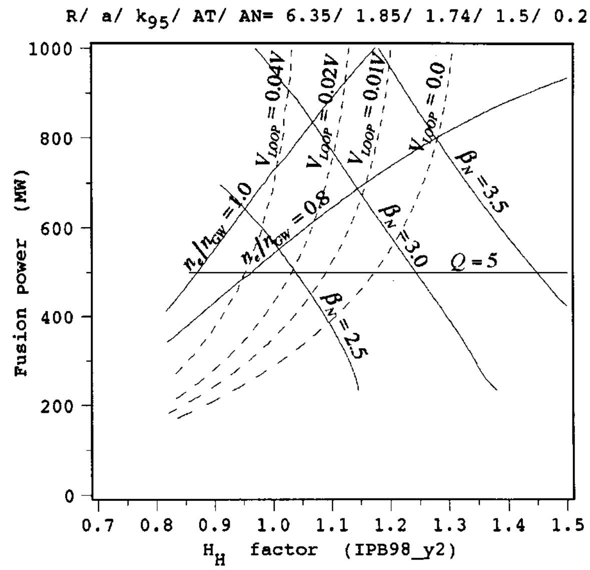


**Figure 4.** Operation space for hybrid (long pulse) and steady state operation. $I_p = 12$ MA and $P_{CD} = 100$ MW. (AN and AT are profile indices.)

**Figure 5.** Central solenoid model coil facility, showing outer coil module insertion into the cryostat.

### 5.3. Hybrid operation modes

Hybrid modes of operation, in which a substantial fraction of the plasma current is driven, in addition to the inductive part, by external heating and the bootstrap effect, leading to extension of the burn duration, appear to be a promising route towards establishing true steady state modes of operation. This form of operation would be well suited to systems engineering tests.

The analysis of the operation space, in terms of fusion power versus confinement enhancement factor, indicates that, for a given value of fusion power (and hence $Q$), as the confinement enhancement factor, $H_{H(y,2)}$, increases (simultaneously decreasing plasma density and increasing $\beta_N$), the plasma loop voltage falls towards zero. For example (Fig. 4), operation with $V_{loop} = 0.02$ V and $I_p = 12$ MA, which corresponds to a flat-top length of 2500 s, is expected at $H_{H(y,2)} = 1$, $Q = 5$, $n_e/n_{GW} = 0.7$ and $\beta_N = 2.8$. This suggests that the ITER design permits a long pulse mode of operation at $Q = 5$ as an approach to steady state operation.

## 6. ITER technology and engineering

### 6.1. R&D basis

The overall philosophy for the ITER design has been to use established approaches through detailed analysis and to validate their application to ITER through technology R&D projects, including fabrication of full scale or scalable models of key components.

Significant efforts and resources have been devoted to the seven large R&D projects [9–16]. These have focused on the key components of the basic ITER machine, by building model central solenoid and toroidal field (TF) coils, a model vacuum vessel, blanket modules and a divertor cassette, and by demonstrating the remote maintenance systems for in-vessel components. Technology R&D issues for the new design of ITER are largely the same as for the 1998 design. These major projects are all expected to meet their objectives for the EDA: the major developments and fabrications have been completed and tests are continuing to demonstrate their performance margin and/or to optimize their operational use.

The technical output from the R&D validates the technologies and confirms the manufacturing techniques and quality assurance incorporated in the ITER design, and supports the manufacturing cost estimates for key cost drivers. For example, two of these R&D projects, which have already achieved their expected results, are shown in Figs 5 and 6. The former shows the central solenoid outer

**Figure 6.** Divertor remote handling test platform, showing the cassette toroidal mover in remote operation.

module being placed outside the inner module, already installed in the vacuum chamber at the test facility in JAERI, Naka, where the complete coil has undergone a comprehensive test programme under conditions well beyond those required for ITER operation [10]. The latter shows a top view of the divertor remote handling test platform at ENEA, Brasimone [15].

The implementation of major joint technology projects offers insights for a possible future collaborative construction project. Valuable and relevant experience has already been gained in the management of industrial scale, cross-Party ventures. The successful progress of these projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

### 6.2. Design modifications

Whilst the new design of ITER [17–22] uses, as far as possible, technical solutions and concepts previously developed and qualified for the 1998 ITER design, the changes in overall scale and in some physics requirements (e.g., more plasma shaping) and the pressure to preserve the plasma performance capacity and flexibility, whilst approaching the 50% cost savings target, have induced some significant changes in the design features.

In addition, data from technology R&D, in particular the seven large R&D projects, have enabled changes in design criteria associated with a better knowledge of the available margins.

Changes to the engineering features of the design have been influenced by the unwillingness to compromise with physics extrapolation so as to provide enough margins in the physical parameters and physics related systems, for example in plasma size, fuelling, and heating and current drive, for instance:

(a) The in-vessel backplate has been eliminated, thus allowing the largest possible plasma volume within the reduced overall size of the tokamak.
(b) The higher plasma shaping, introduced to ensure the achievement of the plasma performance targets, has necessitated the use of a segmented central solenoid and enhancements in the stability control system.
(c) Maintaining the size of port access requires some reduction in the size of the intercoil mechanical structure.

Design changes outside the vessel also balance the general pressure to reduce the dimensions of and simplify the ITER systems on cost grounds against the need to maintain the projected level of performance. In the magnet system, the segmentation of the central solenoid to provide increased control of plasma triangularity, led [17] to the adoption of a wedged support of the TF coils (their number is reduced to 18) and to modifications in the global mechanical structure. Other changes include a poloidal field coil configuration quasi-symmetrical about the equatorial plane.

In the divertor system, a V shaped configuration of the target and divertor floor was adopted [19] as well as a large opening between the inboard and outboard divertor channels to allow an efficient exchange of neutral particles between them. These choices provide a large reduction in the target peak load, without adversely affecting the helium removal.

The reduction in the size and cost of ITER has led to a simplified building and plant layout, and the main remote handling systems have also had to adapt to the general reduction of scale.

A major focus of continued design effort is improvement in the manufacturing processes (with their feedback on design) in order to approach as closely as possible the target of a 50% saving in direct capital cost from the 1998 ITER design.

## 7. Safety considerations

Safety considerations of the new ITER design [23] remain largely unchanged from the 1998 design. Thus, the favourable evaluation of ITER's safety and environmental characteristics remains valid. Indeed, with a longer initial non-nuclear phase of operations now foreseen for the new design [6], it will be possible to have a more precise evaluation of the plant characteristics for nuclear operation.

Informal contact has been made with the regulatory authorities of the ITER Parties, to prepare for possible licensing actions and with an aim also to develop an international consensus on the safety principles for fusion so that the experience with ITER can be generalized for application beyond the host country.

The target for the current phase of ITER is to provide a Generic Site Safety Report (GSSR), which will document the safety assessment of the new design, as part of the final output of the ITER EDA. The GSSR is also intended to provide a basis from which to start preparing regulatory submissions for siting, subject to the further site specific design adaptations and host country specific safety assessments that will be needed to obtain regulatory approval for construction.

## 8. Planned construction and operation costs

The project cost estimate for the eight year construction of the new design is to be based on an industrial cost analysis undertaken by firms of the Parties in the second half of 2000. Pending such an analysis, a simple re-scaling exercise, based on the cost analysis of the 1998 ITER design, indicates an overall reduction to about 56% of the estimated direct capital costs of the 1998 design. The scope to approach closer to 50% will be better understood only after the Parties' industrial experts have had the opportunity to study and estimate procurement packages which incorporate expected improvements in the design and fabrication process. These are now the most important areas of activity for aligning capital costs more closely to the 50% target — US $\$2.9 \times 10^9$ (January 1989 value), a figure roughly equivalent to Euro $3.5 \times 10^9$ (January 2000 values), Y. $4.20 \times 10^{11}$ and US $\$3.9 \times 10^9$ when inflation adjusted for each Party.

The operating costs for the 20 year operating life of ITER are highly dependent on the cost of electricity, the salaries of the estimated 200 professionals and 400 support personnel, and the cost of the divertor high heat flux component replacements and general maintenance expenses, most of which may vary quite substantially amongst the potential host sites for ITER. Simple scalings from the operating cost estimates for the 1998 ITER design suggest an indicative annual figure of about 5% of the capital cost over the first ten years of ITER operation, which represents a saving of almost 50% compared with the 1998 ITER design.

## 9. The impact of ITER and the future outlook

### 9.1. Benefits of the ITER collaboration

The ITER co-operation to date, in combination with the continuing general progress in fusion research, has brought its Parties and the world fusion development programme to the point at which they are technically ready and able to proceed to construction of a next step tokamak device that bridges the strategic gap between the present generation of large tokamak experiments and a first demonstration fusion power reactor.

Sharing costs and pooling expertise has allowed the Parties jointly to undertake tasks that would be beyond the financial and/or technical capacity of each individually, as witnessed in the seven large R&D projects. In the process, the Parties have together developed a mature and wide ranging capacity for successful focused international joint work, including co-operative problem solving, such as in the efficient co-ordination of the fusion physics programme to establish and extend the physics basis of ITER.

The success of the ITER EDA collaboration demonstrates the feasibility and underlines the desirability of aiming for a joint implementation of ITER in a broadly based international collaborative framework: it supports the Parties' declared policy interests to pursue the development of fusion through international collaboration.

### 9.2. Need for a new organization

The ITER EDA Agreement does not commit the Parties to joint construction. Such a move requires new decisions at the highest levels of government following negotiations amongst those interested in participating in the full realization of ITER.

The current ITER Parties started, in spring 2000, non-committal exploratory discussions as precursors to formal negotiations on a joint implementation of ITER. Critical issues to be settled between the Parties' 'Explorers' include:

(a) Establishment of a legal framework for joint implementation that properly reflects various necessary considerations, for instance to provide the focus needed for effective and accountable project management, while ensuring the inclusiveness needed to sustain necessary levels of support and commitment from the wide range of disparate interests throughout the participating countries.
(b) Settlement of the linked issues of siting, cost sharing and task allocation, in equitable ways, with regard to siting. Site offers should be presented around spring 2001, and there are at present efforts being made to promote interest in potential sites in Europe, Canada and Japan.

Obviously, the domestic fusion research and development programme of each Party should allow for full and effective participation in ITER construction and operation in ways that

(1) Assure the technical success of the project,
(2) Ensure a permanent knowledge of the project available throughout the programme,
(3) Stimulate sustained interest from home institutions to participate.

### 9.3. Parallel technical work

During this period of approach to possible joint implementation, further technical work will still be required to enable an efficient start of ITER construction when decided. The main factors are:

(a) Adaptation of the design to the characteristics of (a) potential site(s) and its (their) regulatory environment, and a formal review of its (their) completeness (a necessary step in quality assurance);
(b) Preparation of licensing applications by a closer (possibly formal) dialogue with the host regulators;
(c) Continuation of physics R&D in order to benefit from future experimental results in present devices, and movement from technology R&D towards more manufacturing R&D, except in ongoing development in a few specific areas, such as heating and current drive systems, as well as NbTi superconducting coil winding tests, to confirm operational margins;
(d) Preparation of technical specifications for procurement of hardware on the critical path of the construction schedule.

The Parties, in their exploratory discussions, are considering a new possible framework for their collaboration in technical work after July 2001 and after the end of the EDA Agreement. This framework should maintain the good co-operation between the Parties enjoyed during the EDA and provide for an organization strong enough to preserve the coherence of the project, in the light of requests for design changes linked to specific characteristics of potential sites. There would be a considerable advantage if the organization for this phase already resembled that thought appropriate for the ITER construction phase. The Explorers should take full advantage of this interim period to increase confidence in the Parties' capacity to build and operate a successful ITER.

## 10. Summary and conclusions

(a) In 1999 a four Party Working Group concluded unanimously that *"the world program is scientifically and technically ready to take the important ITER step"* [24]. The progress of ITER EDA in the last two years, combined with the continuing flow of scientific and technological data from existing experiments, has sustained this view.

(b) The design of ITER, which now meets the revised detailed objectives established in 1998 of a cost saving target approaching 50%, still satisfies the overall programmatic objective of ITER.

(c) The lower costs of the new design make it possible for participants to benefit from the sharing of costs and the pooling of expertise that joint implementation allows, whilst maintaining a good balance in the domestic programme of each Party.

(d) The success of the joint activities among the ITER EDA Parties demonstrates the feasibility and underlines the continued desirability of aiming for a joint implementation of ITER in a broad based international collaborative framework. The key tasks for the fusion community are now to confirm, within the programme planning, the strategic priority to proceed with ITER in an international collaboration as the centrepiece of the world fusion energy development programme, to determine, with other potential participants, the overall terms of an international framework for joint construction and operation, and to prepare the necessary consequential adaptations of the programme organization.

## References


[1] ITER EDA Agreement and Protocol 1, ITER EDA Documentation Series No. 1, IAEA, Vienna (1993).
[2] ITER Final Design Report, Cost Review and Safety Analysis (FDR) and Relevant Documents, ITER Documentation Series No. 14, IAEA, Vienna (1999).
[3] ITER Council Proceedings: 1998, ITER EDA Documentation Series No. 15, IAEA, Vienna (1999).
[4] Technical Basis for the ITER-FEAT Outline Design, ITER EDA Documentation Series No. 19, IAEA, Vienna (2001).
[5] Shimada, M., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/05, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[6] Shimomura, Y., et al., Nucl. Fusion **41** (2001) 309.
[7] Murakami, Y., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file EX5/1, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[8] ITER Physics Basis, Nucl. Fusion **39** (1999) 2137–2638.
[9] Salpietro, E., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/21, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[10] Tsuji, H., et al., Nucl. Fusion **41** (2001) 645.
[11] Nakahira, M., et al., Nucl. Fusion **41** (2001) 375.
[12] Dänner, W., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/18, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[13] Mazul, I.V., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITER/6, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[14] Kakudate, S., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/23, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[15] Maisonnier, D., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/24, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[16] Murdoch, D.K., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/13, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[17] Huguet, M., et al., Nucl. Fusion **41** (2001) 1503.
[18] Ioki, K., et al., Nucl. Fusion **41** (2001) 265.
[19] Janeschitz, G., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITER/3, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[20] Bosia, G., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/14, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[21] Fujisawa, N., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/01, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[22] Ladd, P., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/12, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[23] Gordon, C., et al., in Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001) CD-ROM file ITERP/16, and http://www.iaea.org/programmes/ripc/physics/fec2000/html/node1.htm.
[24] ITER Council Proceedings: 1999, ITER EDA Documentation Series No. 17, IAEA, Vienna (1999).


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