SPECIAL ITEM

ITER: CONCEPT DEFINITION

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ABSTRACT. Early in 1988, an agreement was reached by the European Community, Japan, the United States of America and the Soviet Union to jointly conduct conceptual design activities for the International Thermonuclear Experimental Reactor (ITER) until the end of 1990, under the auspices of the International Atomic Energy Agency. Since May 1988, participants from the four Parties meet regularly in Garching, Federal Republic of Germany, to carry out the design work. On the basis of the investigation results obtained so far, a concept for ITER has been defined which incorporates the maximum possible flexibility of the device and allows a variety of plasma configurations and operating scenarios to be adopted. For the technology experiments, with a full breeding blanket, the device can be operated typically with a plasma carrying a current of 18 MA at a major radius of 5.5 m. For the plasma physics experiments, the device can be configured with a thinner shield, if required, and it can produce a plasma of 22 MA with fully inductive operation and higher currents under limited technical conditions. A list of important specific physics and technology research and development tasks for ITER has been prepared and these tasks are being implemented.

1. INTRODUCTION

The technical success in magnetic fusion research and the progress in international collaboration in this field have created conditions which encourage a high level multinational effort aimed at designing, and possibly implementing, the world's first fusion reactor. This has been reflected in recent summit conference discussions, which have resulted in an agreement to design the International Thermonuclear Experimental Reactor (ITER). Under the auspices of the International Atomic Energy Agency (IAEA), participants from the European Community, Japan, the United States of America and the Soviet Union meet in Garching, Federal Republic of Germany, to carry out the design work. Supported by home teams and Research and Development (R&D) activities, the ITER design team will complete its initial work by the end of 1990. At that time there will be sufficient design and cost information to provide the basis for a decision on whether to proceed with the construction of ITER.

The history of specific preparations for ITER began in 1987 when, in response to the summit initiatives, the Director General of the IAEA invited the four Parties to Vienna to discuss enhanced international collaboration in fusion research. As a result, groups were formed to develop proposals on how to proceed, both technically and organizationally. A guiding document called the 'Terms of Reference' was worked out and now constitutes the basis for the present activities. Work began after the four Parties accepted the invitation of the Director General to participate in the design work for ITER, under the auspices of the IAEA and in accordance with the Terms of Reference [1].

In the Terms of Reference, the integrated international design activity shall:

- define a set of technical characteristics of ITER and subsequently carry out the work necessary to establish its conceptual design;
- define future R&D needs and prepare cost, manpower and schedule estimates for the realization of such a device;
- define the site requirements for ITER and perform a safety and environmental analysis;
- carry out in a co-ordinated manner specific validating R&D work supportive of the design activities.

The activities are divided into two phases: a definition phase, lasting from May to November 1988, and a design phase, lasting from November 1988 until December 1990. In the definition phase, a set of ITER technical characteristics has been developed and are documented in a definition phase report, which also contains a plan for the supportive R&D activities.

The design phase began as soon as the definition phase report was approved. A final report will be submitted at the end of 1990.

The conceptual design activities are directed and managed by (1) the ITER Council (IC), which is responsible for the overall direction of the activities and supervises their execution, and (2) the ITER Management Committee (IMC), which is responsible for the direct management of the activities. An ITER Scientific and Technical Advisory Committee (ISTAC) advises the IC on scientific and technical matters.

In accordance with the overall design schedule, about 40 professionals from the four Parties work together at a technical site of the Max-Planck-Institut für Plasmaphysik in Garching over a period of several months each year. The design work to be conducted at the participants' home sites will involve the contribution of about 80 man-years of each Party for the whole period. The joint work of the definition phase of ITER began on 2 May 1988 in Garching. The design team was organized in a matrix form comprising project units, which have the role of co-ordinating the design activities, and design units, which conduct the detailed design. This organization reports to the IMC, which has overall executive responsibility. Communication with home support teams is expedited through the use of electronic mail and other computer links. To assess the present situation in specific areas of physics and technology relevant to ITER, fifteen specialists' meetings were held in Garching. Also, a number of short term visits by experts greatly contributed to the ITER activities.

On the basis of the guidelines outlined in the Terms of Reference, the definition phase report, presenting the results of the ITER activities for the period of May to September 1988, has been prepared [2], and highlights of the work have been presented at the IAEA Nice Conference in October 1988 [3–7].

A summary of the definition phase report is given in the following sections.

2. ROLE OF ITER

ITER is intended to be an experimental thermonuclear tokamak reactor with the aim of testing the basic physics performance and the technologies essential for future fusion reactors. It is envisaged as a main fusion machine of the next generation, following the present large experiments such as JET, JT-60, T-15 and TFTR. The specific technical objectives of ITER are elaborated in Section 3. In successfully accomplishing these objectives, ITER will provide the database necessary for the design and construction of a demonstration fusion power plant.

The ITER design will rest on extensive new design work, supported by new physical and technological results, and on the great body of experience built up over several years from previous national and international reactor studies. Conversely, the ITER design process should provide to the fusion community valuable insights into what key areas need further development or clarification in moving forward towards practical fusion power.

3. OBJECTIVES OF ITER

The overall objective of ITER is to demonstrate the scientific and technological feasibility of fusion power. ITER will accomplish this objective by demonstrating controlled ignition and extended burn of a deuterium and tritium plasma, with steady state as an ultimate goal, by demonstrating technologies essential for a reactor in an integrated system, and by performing integrated testing of the high heat flux and nuclear components required in the practical utilization of fusion power.

To reach this objective, ITER will be operated in two phases:

— A physics phase, focused mainly on the achievement of the plasma objectives.
— A technology phase, devoted mainly to engineering objectives and the testing programme.

The plasma physics objectives, i.e. to demonstrate controlled ignition and extended burn of deuterium–tritium plasmas, with steady state as an ultimate goal, will be reached through:

— Inductive plasma operation under conditions of controlled burn;
— Extension of the burn pulse towards the steady state, using non-inductive current drive;
— Provision of current carrying capability as required for ignition experiments in which the predictions regarding plasma confinement are studied; this capability has to be at least about twice the maximum current in the present generation of large tokamaks;
— Provision of plasma parameters sufficient for the technology phase with a Q-value of about 5.

The engineering and testing objectives are: to validate design concepts and to qualify engineering components, to demonstrate the reliability and maintainability of the reactor systems, and to test the main nuclear technologies (blanket modules, tritium production, extraction of high-grade heat appropriate for the generation of electricity). Specifically, it is hoped that the device can provide: (1) testing conditions with an average neutron wall loading of about 1 $MW \cdot m^{-2}$; (2) a neutron fluence of about 1 $MW \cdot a \cdot m^{-2}$, but with the design allowing for a higher neutron fluence, in the range of 3 $MW \cdot a \cdot m^{-2}$; (3) a tritium breeding blanket with which a breeding ratio as close to unity as possible can be achieved; and (4) an overall availability of at least 10%, but with the aim of reaching a level of 25% and providing continuous operation for a period of one to two weeks.

4. OVERVIEW AND PHYSICS

4.1. Overview

The choice of the ITER reference parameters is based on work conducted over a period of several months. Tentative physics and engineering guidelines were adopted early in 1988 to provide the framework for the initial definition phase. These guidelines and the overall technical philosophy have evolved considerably in the course of the work. The final definition phase parameters should not be considered as being fixed; rather, they are likely to change over the design period.

In the summer of 1988, the current scientific database and design guidelines were re-examined in some detail. Attention was given not ouly to the present status of physics but also to possible improvements or extensions which could have a significant favourable impact on the design. One of the key requirements for achieving ignition in ITER, assuming reasonable extrapolations of the present database, is a plasma current in the range of 20 MA.

In connection with this requirement, a detailed analysis of the technical feasibility of a 20 MA machine was conducted. Although the study revealed some difficult engineering tasks, the general conclusion was that a 20 MA machine with 5.8 m major radius and a full blanket is possible, with reasonable engineering assumptions. The study also showed that it should be possible to change the blanket and shield configuration in the period between the end of the physics phase and the start of the technology phase, and indicated the potential for limited operation of the machine at higher plasma currents should this prove necessary for ignition. As will be shown later, this 'flexibility' in the use of the basic machine main-frame has proven most important.

In parallel with the physics and engineering studies, scoping studies for a broad range of concepts were carried out, using systems codes. Since today's scalings of the energy confinement time can be characterized by the individual dependences on plasma current (I) and aspect ratio (A), it was found convenient to express the various possibilities in I–A space. This provided a common basis for discussion of the overall concept possibilities and the sensitivity of the design concepts to various technological and physical assumptions. Examples of concepts that were examined are the 20 MA device mentioned above, a device with a high aspect ratio and a long Ohmic pulse, a large device capable of igniting under 'L-mode' confinement, a device with an advanced magnet design, and a device optimized with regard to the achievement of steady state rather than ignition. However, except for devices operating in the range of 20 MA with an aspect ratio of about 3, all concepts were considered

TABLE I. TYPICAL OPERATING PARAMETERS OF ITER

	Physics phase	Technology phase
R (m)	5.8	5.5
a (m)	2.2	1.8
A	2.6	3.1
k_{95}	1.9	2.0
q_{psi} (95%)	3.2	3.1
g	1.9	3.0
I_p (MA)	22.0	18.0
B (T)	5.0	5.3
P_{fus} (MW)	1000	820
P_n ($MW \cdot m^{-2}$)	1.0	0.9
P_{CD} (MW)	—	90
Q	—	9.1
n_e (10^{20} m^{-3})	1.1	0.8
T (keV)	10	18
τ_E (s)	3.1	1.8
H_{SO}/H_{RL}	1.7/0.8	1.6/0.7
H_G/H_{T10}	1.8/1.7	1.5/1.4
τ_{burn} (s)	>200	Steady state

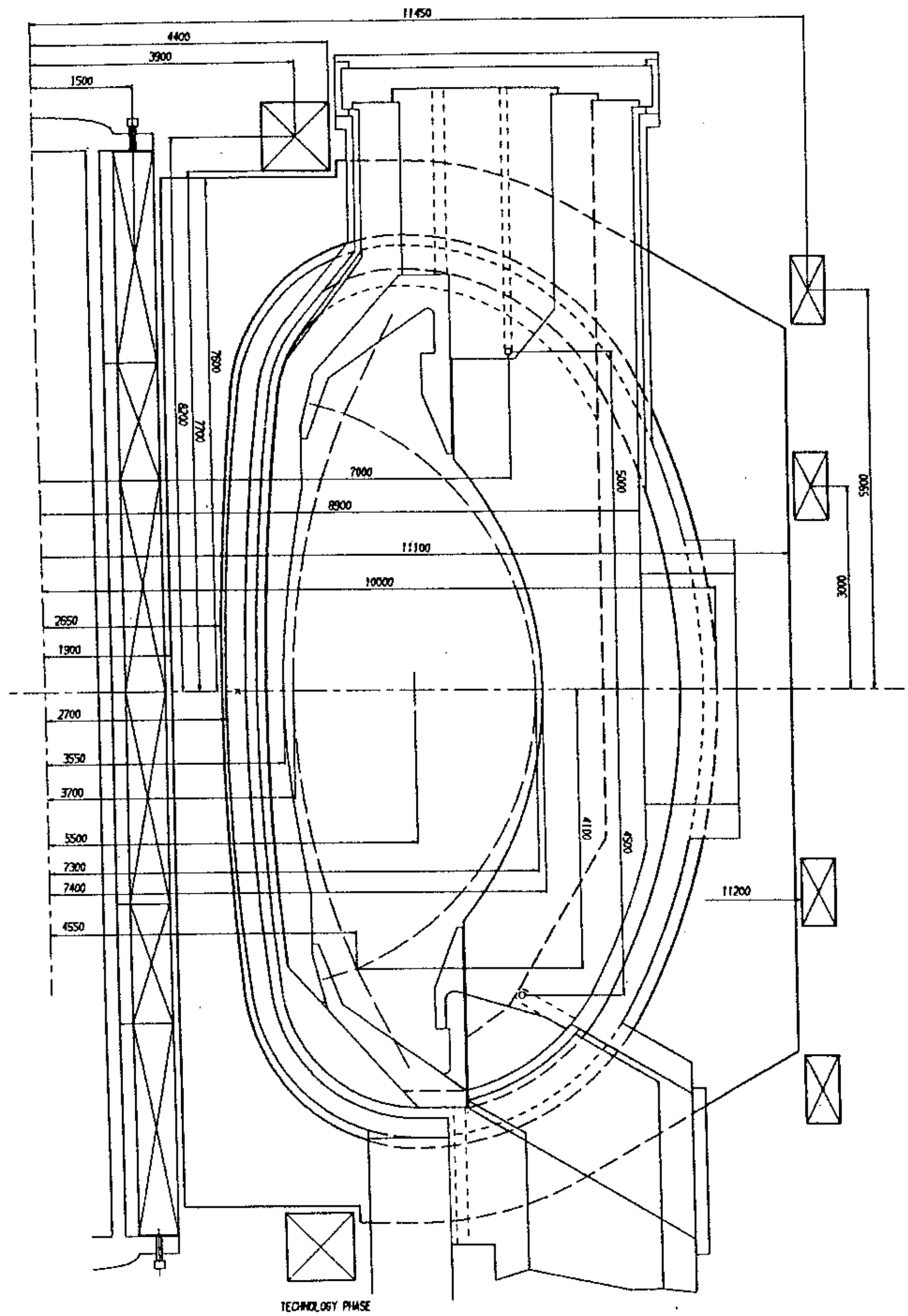

FIG. 1. ITER configuration in the technology phase, elevation view.

unsuitable because they were found to be (1) too large, costly and difficult to build, (2) deficient in plasma performance and unable to meet the Terms of Reference, or (3) not adequately based on the physics database or on proven technological practices.

On the basis of the outcome of the physics, engineering and system studies for a machine in the range of ~ 20 MA, the final device parameters were selected with a view to reducing the size and cost of the device while optimizing the performance for the physics and technology phases. Recognizing the different goals of the physics phase ($Q = \infty$) and the technology phase ($Q \geq 5$, with high fluence, a breeding blanket and nuclear testing), a different set of plasma parameters (and subsequently in-vessel configurations) was chosen for each phase. These parameters are shown in Table I. In the technology phase, a full blanket configuration will be employed in conjunction with an 18 MA, 5.5 m major radius plasma, as shown in Fig. 1. In the physics phase, a thinner shield will be employed because of the lower fluence expected, and a 22 MA plasma with 5.8 m major radius is proposed.

These parameter choices correspond to a deliberate attempt to synchronize the demands on confinement for the two phases and, in the technology phase (when ignition is not a necessity), to alleviate the problems associated with energy removal, recirculating power and tritium consumption. To further accommodate the divergences in confinement predictions, the possibility of limited operation is being studied, with some shifts in physics risks at currents of up to 25–28 MA in the physics phase.

A design approach has been adopted which allows the blanket and shield as well as the other components inside the vacuum vessel to be changed while leaving all other machine components fixed. This provides considerable experimental flexibility in meeting the individual requirements of the physics and technology phases. It may also prove beneficial in introducing advanced features and new capabilities. Thus, the proposed design is flexible and 'robust', enables the desired ignition performance in the physics phase, provides reasonable steady state operation in the technology phase and is feasible from an engineering point of view.

4.2. Physics basis and performance predictions

4.2.1. Introduction

Significant progress has been made in the last decade with respect to improvements of both the plasma parameters attained in tokamaks and the understanding of the physical processes in tokamak plasmas. This progress includes the achievement of reactor grade ion temperatures of 10–30 keV at low to moderate plasma densities, with high heating power (10–30 MW) in the largest tokamaks (JET, JT-60 and TFTR), and the attainment of energy confinement times close to 2 s in JET, with moderate heating levels (≤ 7 MW) in the H-mode and confinement times of ~ 0.5 s in the L-mode. The fusion ignition parameter $n_i(0)T_i(0)\tau_E$ has been increased in the present generation of large tokamaks by a factor of about 30 over the values achieved in the late 1970s, to 2.5×10^{20} $m^{-3}\cdot keV\cdot s$. Progress in understanding tokamak physics has also been substantial. For example, the experimental beta limit has been found to be in close agreement with theoretical predictions, and the physics of the mechanisms for plasma heating and current drive also appear to be well understood. On the basis of this significant progress, the specifications and guidelines for the ITER design have been developed. While strong emphasis has been placed on the physics credibility, the design guidelines also take into account the requirement that ITER should be able to take advantage of possible further improvements in tokamak physics before and during its operation.

Six physics areas have been assessed for the ITER design:

(1) Plasma performance and operational limits (including energy confinement, fast ion confinement, density limits, MHD limits and plasma performance);
(2) Plasma equilibrium and control as well as poloidal field configuration;
(3) Current drive and heating;
(4) Power and particle exhaust conditions;
(5) Disruptions;
(6) Operations and diagnostics.

Ignited operation and driven steady state burn with a sufficiently high neutron wall loading will require adequately high levels of energy as well as alpha particle confinement and stable operation at sufficiently high plasma density and beta. The realization of adequate levels of confinement and beta requires a high plasma current which can best be achieved with large plasma elongation. The design of the poloidal field system for the control and shaping of such largely elongated plasmas requires detailed physics analyses of the interaction of the plasma with the plasma control and shaping system. For steady state operation, an efficient non-inductive current drive system will be required. A part of the current drive system will be used for heating during ignited operation (≤ 50 MW). The exhaust of power and helium ash as well as the control of plasma contaminants from the plasma facing components are also very important issues. The high levels of plasma current and stored thermal energy expected in ITER make plasma disruption one of the most important design issues. Also, a preliminary operation plan and a preliminary layout of the diagnostic systems have to be developed. For each of these areas, assessments of the present state of the physics knowledge and of the potential improvements have been performed. These assessments have then been used to develop the physics guidelines for the ITER design.

Regarding a number of questions in connection with the ITER physics guidelines, the present knowledge is still incomplete. Because of these questions, further tokamak experiments and theoretical analyses will have to be performed. A set of ITER physics R&D requirements has been formulated which identifies the additional information needed for the design of ITER. The R&D requirements are discussed later.

4.2.2. *Plasma performance and operational limits*

For projections of the plasma performance of ITER, it is necessary to determine both the parameters required for adequate energy and alpha particle confinement and the operational constraints, such as those imposed by MHD stability limits and the density limit.

Energy confinement is one of the most important issues for ITER since, without an adequate level of energy confinement, ITER will not achieve the required level of plasma performance. The level of plasma current required for adequate energy confinement is one of the major factors that determine the size of the device. An assessment of the energy confinement data from present tokamak experiments has been carried out. This assessment has emphasized data from the present generation of large tokamaks (TFTR, JET, JT-60 and DIII-D) as well as data from older experiments (ASDEX, DIII, JFT-2M, T-10, etc.).

Generally, in tokamaks the energy confinement is divided into confinement with Ohmic heating and confinement with auxiliary heating. With intense heating (the appropriate regime for ITER), two general regimes of confinement have been observed: the *L-mode* with significantly degraded confinement compared to Ohmic confinement, and the *H-mode*, with improved energy confinement compared to the L-mode. A great deal of important work regarding the theory of tokamak transport has been done, and many of the mechanisms are well understood, but none of the theories has yet been validated to the point that it could be used for predicting the confinement in ITER to a sufficient degree of precision. A number of empirical scalings have been used to fit the present L-mode confinement data, but it is not rational to choose one empirical scaling from among the many candidates on the basis of the present data. Optimization of the predictions for energy confinement in ITER should include consideration of these scalings.

A figure of merit has been developed, of the form $n\tau_E T \approx (I \cdot A^x/R^y)^z$, where $x \sim 0.0$–0.5, $y \sim 0.0$ and $z \sim 2$, for some scalings, and $x \sim 1.2$, $y \sim 0.3$ and $z \sim 3$, for a number of other scalings. This type of figure of merit illustrates that, for one type of scaling, the confinement can be strongly improved by an increase of the aspect ratio at constant current. For the other class of scalings, the plasma current is much more important than the aspect ratio. As far as the dependence on aspect ratio is concerned, the present database is limited because most of the experiments are done in a range around $A = 3$.

While no sufficiently precise scaling has been developed for energy confinement with the H-mode, a comparison of H-mode data with L-mode scalings indicates that a reasonable fit can be obtained by using a constant multiplier of about two times the L-mode scaling. Although the H-mode has an improved confinement compared with the L-mode, it also has some disadvantages. H-mode operation in present experiments is often accompanied by increased particle confinement, especially for impurities. This often results in a radiative collapse of the discharge. In addition, the width of the scrape-off layer is very short. H-mode operation is also usually accompanied by MHD oscillations in the outer plasma layers (edge localized modes, ELMs). While large amplitude ELMs can destroy the good confinement of the H-mode, small amplitude ELMs, in some conditions, lead to only a limited degradation of the energy and particle confinement, allowing steady state H-mode operation.

In addition to H-mode operation, other techniques have been used to improve the confinement in a number of experiments. These include the use of divertors (JET, DIII-D), control of the density profile (TFTR, Alcator-C, etc.), and the use of plasma heating techniques with energy deposition in the plasma centre (JET, TFTR, etc.).

From a general assessment of the present status of energy confinement in tokamaks, it is recommended that the energy confinement time in ITER be estimated by the prescription of $\tau_E = f \times \tau_E$, where $f \sim 2$ for the H-mode and $f \sim 1.5$ for 'enhanced' L-mode confinement.

The major issue with alpha particle confinement is the loss of fast alpha particles to the first wall, which is due to ripple induced loss processes, and the subsequent potential for high peak heat loads on the first wall. Preliminary estimates of the peak heat loads that can be tolerated imply that the losses of fast alpha particles should be kept below a few per cent. Assessments for ITER indicate that the dominant loss process is diffusion due to collisional scattering and collisionless stochastic scattering. One specific requirement for the minimization of alpha particle losses is that the ripple in the plasma region with fast alpha particles be kept below the threshold for stochastic losses. Although, for q (95%) $\sim$ 3 and A $\sim$ 3, minimizing the fast alpha particle losses generally translates to a requirement that the outboard edge ripple be $\leq$1–1.5%, it is actually the ripple profile as well as the safety factor profile and the alpha particle birth profile that are important, and a detailed assessment must be carried out for each coil design. High-q operation strongly increases the ripple losses and it is therefore anticipated that this is not desirable with respect to alpha particle losses.

While the exact value of the beta of fast alpha particles is not defined yet, the limit due to fishbones may be estimated to be in the range of 4% if there is strong shear near the q = 1 surface. This would limit the central temperature to 30 keV or less for $\langle n_e \rangle \sim 10^{20}$ m^{-3}.

Assessment and analysis of the density limits in present experiments indicate that the density limit is due to a cooling of the plasma edge, which induces a shrinking of the current channel and eventually destabilizes the m/n = 2/1 tearing mode. Edge radiation by impurities plays an essential role, but other loss channels, in particular heat conduction and, indirectly, particle transport, may also be important. On the basis of this picture for ITER, the guideline adopted is that densities are accessible for which the power radiated from the hot plasma for $r \leq r(q=2)$ does not exceed 70% of the total heating power. Murakami–Hugill and Greenwald scalings can provide useful points of comparison.

Both experimental evidence and theoretical analyses show that the beta limit in tokamaks globally follows the Troyon scaling, $\beta_{max}(\%) = gI_p(MA)/a(m)B(T)$. The values of the factor g for which stable and close to steady state operation have been obtained are $g \leq 2.7$ in DIII-D, $g \leq 2.5$ in JFT-2M and $g \leq 2.3$ in ASDEX. Higher g-values can be reached in transient states. The corresponding q-limits are q (95%) $\sim$ 2.2 in DIII-D and JFT-2M. ASDEX has mainly operated with q (95%) > 3 where the beta limit is usually non-disruptive. In DIII-D, the maximum value of g is practically independent of the elongation k = b/a for $k \leq 2$.

For $q_I = 5(1+k^2)a^2(m)B(T)/2R(m)I_p(MA) > 1.8$ and $1.5 \leq k \leq 2$, ideal MHD theory predicts stability up to g = 3–3.5 for optimized profiles of pressure and toroidal current. For larger elongation, such as k = 2–2.5, the value of g tends to decrease and β_{max} tends to saturate as a function of I_p.

The fact that steady state experiments show lower beta limits than predicted by ideal MHD theory may be due to the observed pressure profiles which, even though broadened at low q, are still more peaked than the optimum profiles.

For q(0) < 1, higher betas can be reached with higher triangularity. Also resistive and kinetic effects, which have not yet been fully analysed, may affect the beta limit.

From these considerations, the following conclusions for ITER have been derived: Plasma elongation beyond k = 2.5 will degrade the beta limit. For 2 < k < 2.5, the present theoretical analyses suggest that beta is almost constant with respect to plasma current. However, it is possible that more detailed optimization could enhance the dependence of β on k in this region. For performance predictions in ignition studies, $g \leq 2.5$ and q (95%) $\geq$ 3.0 should be used. However, it is desirable that the device have the technical capability to allow operation with g up to 3.0 and q (95%) down to 2.1. For steady state operation, $g \leq 3.0$ (for performance predictions) and $g \leq 3.3$ (desirable technical capability) has been recommended, taking credit for the fact that, in this case, current profile control is possible; the q-limits are as above. The limitation of q (95%) $\geq$ 3.0 for performance predictions is also based on the consideration that confinement tends to deteriorate with low-q operation.

Setting the criteria for successful achievement of ignited burn and steady state operation requires estimating the energy and alpha particle confinement, including the constraints imposed by MHD stability and density limits. Using the large variety of empirical scalings developed to describe the present tokamak confinement data, the ignition requirements of a potential ITER design have been evaluated, without taking into account engineering design constraints to be met in the actual machine design (Fig. 2). In the evaluation, all the confinement requirements expressed by

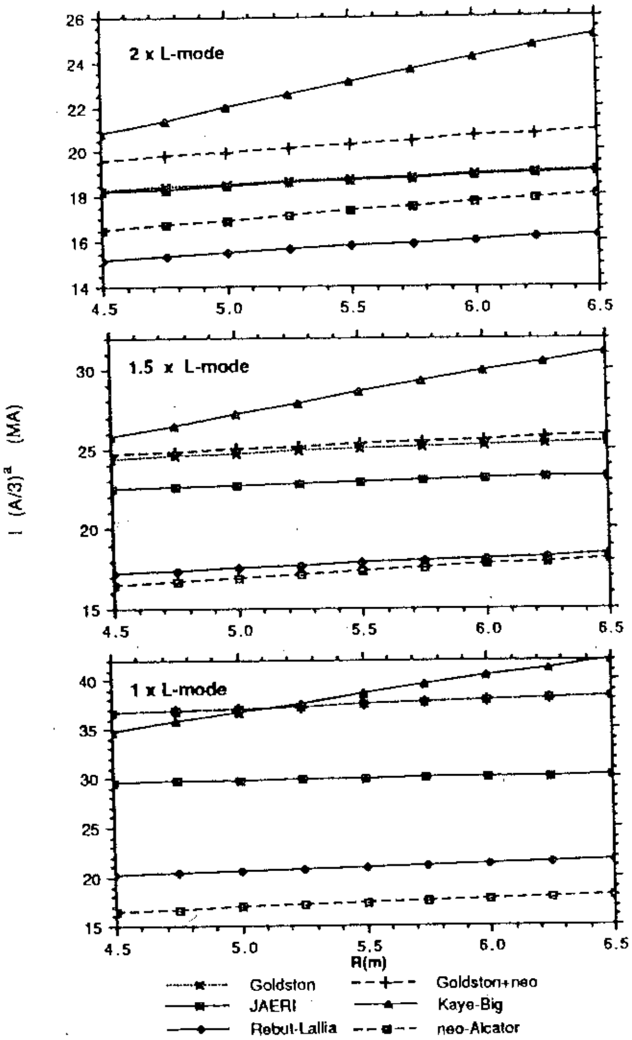

FIG. 2. Confinement requirements expressed in terms of I $(A/3)^a$, needed for ignition with 1, 1.5 and 2 times the commonly used L-mode scalings for energy confinement.
I is the plasma current, A is the aspect ratio, and a = 1.25 for neo-Alcator, a = 1.37 for Goldston + neo, a = 0.67 for Rebut–Lallia, a = 0.5 for JAERI and a = 0.82 for Kaye–Big.

various scalings have been converted into a common term of I $(A/3)^a$ for convenience of comparison. These studies indicate that, while the different empirical scalings vary widely, the requirements for plasma current (with an aspect ratio ~3) generally fall into the range:

17–22 MA	H-mode
22–27 MA	'enhanced' L-mode
> 30 MA	L-mode

Since ITER will operate at high power levels and is designed with a divertor, H-mode operation will be difficult to avoid. However, the H-mode may not be suitable for achieving ignition because of problems with impurity retention and other effects. Therefore, provision must be made to provide operational flexibility to raise the plasma current from 18 MA to at least 25–27 MA, if such plasma currents should prove necessary to reach ignition.

With the plasma current required to be in the range of 18–22 MA or greater, the plasma volume will be large, and operation at the beta limit will result in a very high fusion power. Heat exhaust considerations dictate that the ignited plasma will have to be operated well below the beta limit, for example by reducing the density. The beta limit is, however, an important constraint for current drive, since the current drive power can be reduced by operating at higher temperatures and lower densities than optimum for ignition.

4.2.3. Axisymmetric magnetics

One of the major improvements in tokamak physics and engineering in the 1980s has been the successful operation of largely elongated plasmas, with elongations up to k = 2. This has allowed the design of tokamaks with very high currents, and therefore improved confinement and MHD stability performance, with relatively modest increases in plasma size and cost. For instance, INTOR was designed for a current of 6 MA and a major radius of 5.2 m with a k of 1.6, whereas ITER is being designed with a current of 18 MA and a major radius of 5.5 m with a k of 2.

However, the design of a poloidal field (PF) system to shape and control such a highly elongated plasma from the beginning of the discharge through plasma burn and shutdown requires a detailed understanding of the interaction of the plasma and the PF system. Specified magnetic configurations for ITER have been computed using 2-D free-boundary equilibrium calculations that include the geometry and location of PF coils with prescribed flux linkages and plasma conditions. The PF system must be designed to provide the plasma control necessary to accommodate the ranges of plasma shapes and parameters (e.g. pressure, current and profiles) that will occur during the entire plasma pulse and in each of the different proposed operating scenarios. Analyses of axisymmetric stability have also been performed with rigid-body stability models in combination with an eddy current analysis of the surrounding structures. More detailed analyses of the time evolution of the equilibrium, flux consumption and plasma movement will be carried out during the design phase, with more sophisticated codes incorporating energy transport and a self-consistent model for the interaction of the plasma and the magnetic fields from the PF coils.

4.2.4. *Heating and current drive*

To meet the Terms of Reference, ITER must be designed to allow both ignited and steady state operation. Steady state current drive clearly represents a more demanding requirement than heating to ignition, and it is this requirement that determines the choice of heating and current drive systems. Four methods of non-inductive current drive have been considered for ITER: lower hybrid slow waves (LH), neutral beams (NB), electron cyclotron waves (EC) and ion cyclotron waves (IC). The neoclassical bootstrap effect may supply as much as 30–40% of the total plasma current.

Given the high currents required for adequate confinement and the expected current drive efficiencies, a Q-value of no more than 8–12 can be anticipated for ITER. It will require about 100 MW of auxiliary power to support a plasma current of around 20 MA. This is about twice the heating power required to reach ignition. The optimum steady state operating points are at densities somewhat below those envisioned for ignited operation ($(0.5–1.0) \times 10^{20}$ m^{-3}) and at higher temperatures (typically $\langle T \rangle \sim 20$ keV).

For LH waves there is a very solid database for non-inductive current drive. In addition to steady state current drive, LH waves should prove useful for startup assist and current ramp-up, transformer resetting and profile control. However, given the present ITER design and the projected steady state operating points, LH waves cannot be used to drive current in the plasma core. The wave accessibility constraint necessitates the choice of a comparatively high $N_{\parallel}$. Electron Landau damping then causes the wave to be absorbed before it reaches the plasma core. Hence, LH heating is most useful for driving current in the outer portion of the plasma. Another method must then be used to drive current non-inductively in the plasma core.

For the ITER baseline, three scenarios are being considered for driving the current in the plasma core: NB, EC or IC schemes. In each case, LH waves would be used to drive current on outer flux surfaces, and bootstrap effects would also be taken into account when computing the non-inductive current drive requirements. A typical scenario would have about 50% of the current driven by NB, EC or IC waves in the plasma core, with 20% of the current supplied by LH waves and the remaining 30% arising from the neoclassical bootstrap effect.

Also, there is a good database for IC non-inductive current drive. Hence, it must presently be considered the primary option for driving current in the plasma core. NB heating is well suited for central current drive because the NB current drive efficiency increases with plasma temperature, while penetration of the neutral beams to the plasma core can be achieved if neutral beams with energies in the range 0.7–2.0 MeV can be developed. The major physics issue for NB current drive in ITER is the possibility of kinetic instabilities associated with the injection of beam ions at velocities higher than the local Alfvén velocity. Another issue is a more accurate determination of the enlargement of the stopping cross-section for neutral beams due to multistep processes.

Recent theoretical studies have indicated that EC heating at upshifted frequencies is capable of driving current in the core of reactor grade plasmas with figures of merit, $\langle n_e \rangle$ IR/P, of about (0.2–0.3) $\times 10^{20}$ A$\cdot$W$^{-1}\cdot$m^{-2}. While this is about one-half of that for NB, LH or IC heating, it is still high enough to yield interesting steady state EC current drive scenarios for ITER. The major physics issue for EC current drive in ITER is the need for a convincing demonstration and characterization of EC current drive on a large tokamak between now and the end of the ITER design period.

Because of the low resistance of the vacuum vessel and the structure and magnet requirements, plasma breakdown is required for a toroidal electric field of ~ 0.3 V$\cdot$m^{-1}. EC heating will be useful in ITER for allowing breakdown with such a low voltage and it will also be useful for startup assist and for current profile control. Hence, an EC system with a power level in the range of 10 MW is recommended for ITER in the event that EC heating is not chosen to drive current in the core of the ITER plasma. A frequency as low as 100–120 GHz could be used for startup, which is somewhat lower than the frequency required for current drive.

There is a good database for IC plasma heating. Theory predicts that the current drive efficiency will increase with the plasma temperature to the point where it will be comparable to the efficiency achieved with NB heating for ITER parameters. However, the database for IC current drive is very restricted and must be improved before the end of the ITER design period. The generation and maintenance of H-mode plasmas as well as impurity generation from the antenna are other issues that must be resolved. Frequencies in the range $\omega = \omega_{CD} - 2.5\omega_{CD}$ ($f \sim 40$–100 MHz) are currently being considered. For bulk heating, $\omega = 2\omega_{CT}$

would be employed, while a somewhat higher frequency would be used for current drive to allow interaction with electrons. The propagation of IC waves to the plasma core has been demonstrated in IC heating experiments.

4.2.5. *Power and particle exhaust*

Evidence from ASDEX, JFT-2M, JT-60, DIII-D and JET, together with modelling data, show that a single-null or a double-null poloidal divertor is presently the most credible concept for the exhaust of the plasma power and helium ash and also for the control of impurities. Predictions for the anticipated ITER conditions indicate that, for edge densities above $\sim 5 \times 10^{19}$ m^{-3}, substantial localized recycling will occur within the divertor, thereby ensuring a relatively low plasma temperature at the plate, good retention of eroded material (e.g. $\sim$95%) and the provision of wall recycling conditions in the scrape-off region that are compatible with the attainment of H-mode confinement. Nevertheless, the ITER divertor plasma conditions will be very demanding with respect to power loading, the erosion of plate material and the pumping of helium exhaust gas.

Modelling calculations, which are in general agreement with the sparse data that are available and the existing data themselves indicate that the energy flow scrape-off layer in ITER will be narrow, especially during H-mode operation. For typical ITER parameters, with neutron wall loads of 1 MW$\cdot$m^{-2}, the expected peak divertor power loads will exceed 5 MW$\cdot$m^{-2} even if the divertor plate is inclined at the highest practical angle to the magnetic surfaces (about 20°). Conditions during the H-mode are also complicated by the presence of ELMs, which can cause substantial fluctuations in the peak heat loads.

The analyses also indicate that the peak temperature of the plasma at the divertor plate increases in a strongly non-linear manner with decreasing density in the scrape-off plasma. It is predicted that operation at low $\bar{n}_e$ (e.g. $\leq 10^{20}$ m^{-3}) or with strongly peaked density profiles can raise the divertor plasma temperature to such an extent that the sputtering rate of high-Z plate materials would be unacceptable. This has led to the initial choice of low-Z (e.g. carbon or carbon based) materials, at least for the beginning of operation during the physics phase. The divertor plate material for the technology phase will be chosen on the basis of the performance of carbon during the initial physics phase and with consideration of the data available from other experiments using carbon and other materials such as tungsten. The predicted erosion rates of low-Z materials increase as the temperature of the divertor plasma increases. The net erosion rates are high even when allowance is made for redeposition. There is thus a strong incentive to reduce the divertor plasma temperature either by increased edge radiation (which is not compatible with low-Z impurities) or by operation at increased density in the scrape-off plasma (which is in conflict with the requirements for efficient current drive). Other divertor configurations with the potential for stronger recycling should also be considered. It is also envisaged that the peaking of both power load and erosion can be reduced by sweeping the plasma across the plate.

The plasma flux amplification at the plate, which is a consequence of localized recycling, decreases with increasing plasma temperature and hence the pumping requirements for neutral gas are commensurately increased with low density operation, i.e. they are well in excess of 100 m$^3\cdot$s^{-1}. There is some uncertainty with respect to the pumping requirements because the helium ion transport into the divertor is yet insufficiently characterized.

The critical nature of these issues emphasizes the need for improved and experimentally validated models as well as experimental data for helium/impurity ion transport and the reduction of erosion by redeposition. Optimization of the working conditions of the divertor plate and of power and helium exhaust is one of the most important and demanding issues of the ITER design.

4.2.6. *Disruptions*

Plasma disruption is also a very important event in ITER. It can lead to high pulsed heat loads on the plasma facing components and to electromechanical shocks on the components and structures surrounding the plasma. A definite quantification of the consequences of a major disruption is not yet possible because the phenomenon has not been studied in enough detail. It is clear, however, that nearly all of the plasma kinetic energy is lost over a time of typically 0.1–3 ms. The energy deposition profile of ITER is not well known at the moment, but it is anticipated that a large fraction of the energy will be exhausted to the divertor plates. Current quench is expected to be fast, with current quench rates of about 1 MA$\cdot$ms^{-1}, or even larger if plasma position control cannot be made efficient enough to reduce plasma/wall contact to a minimum. During current quench, a large part of the poloidal magnetic field energy is resistively dumped into the plasma and lost onto the first wall, mainly by radiation. Because of the corresponding heat load it is anticipated that protection of the first wall by the use of graphite tiles over the whole surface will be required. Moreover, high electromechanical forces, in particular in the vacuum vessel, are predicted, especially when the current decay is accompanied by fast vertical plasma motion. Fast current quench is also known to lead to the generation of runaway electrons having energies of up to 100 MeV and carrying up to 50% of the original plasma current. These electrons, when impinging on the first wall, cause very high localized heat loads. The main conclusion of the analysis performed for ITER is that it is extremely important to develop ways to reduce the frequency of occurrence of disruptions and to soften current quench. Moreover, intensifying research to obtain a detailed characterization of disruption phenomena is a prerequisite for being able to take proper account of disruptions in the ITER design.

4.2.7. Diagnostics

Plasma diagnostics have to provide both detailed information on the behaviour of H–D and burning D–T plasmas and the data needed for plasma control. Plasma diagnostics must also be compatible with the working conditions of ITER. The main issues include compatibility with neutron and gamma radiation as well as operation with tritium, reliability, remote maintenance, lifetime, and adequacy for long steady state pulses. For the H–D phase, extrapolation of the existing diagnostic techniques will be possible. For ignition studies, alpha particle diagnostics have to be developed. During the ignition studies and in the technology phase, all diagnostics will have to be radiation resistant and capable of withstanding high heat fluxes or else they will have to be shielded. All of the diagnostics must have much higher standards of reliability than they have in present experiments.

5. CONCEPT OF ITER

5.1. Reactor configuration and containment structure

5.1.1. Introduction

ITER, as a nuclear plant, consists of many systems providing the necessary functions of the reactor:

(1) The basic machine, where the plasma is contained and heated, consisting of the main semi-permanent systems and including toroidal field (TF) and poloidal field (PF) coils, the vacuum vessel and the cryostat.

(2) The in-vessel replaceable components, such as the first wall, breeding blanket, guard limiters, divertor cassettes, and antennas for heating and current drive, which are contained inside the vacuum vessel.

(3) The external (with respect to the basic machine) systems, providing cooling, fuelling, tritium processing, heating and current drive, pumping, power supply, remote handling and control of the reactor.

(4) The various systems on the plant site.

ITER, as an integrated system, must permit physics investigations over a wide range of plasma parameters. It should be designed to operate with different plasma parameters and blanket/divertor geometries in both the physics phase and the technology phase. The approach to component design must take into account both requirements and provide the necessary lifetime and reliability of the reactor.

5.1.2. Reactor configuration

The basic machine is supported by structural legs extending from a base mounting slab. To provide thermal isolation of the cryogenic and elevated temperature machine components, two sets of legs are provided — one for the TF coils and one for the vacuum vessel and its internal components. All concepts include the same coil set, established in the reference design, which consists of a configuration of 16 TF coils, wedged at the inboard face and tied together with an intercoil structure. The PF coil set consists of a central solenoid with three pairs of ring coils. In the configuration studies the use of two vacuum containments was assumed — an external cylindrical containment surrounded by a concrete structure enveloping the whole torus, and an internal toroidal containment providing a separate plasma outer zone, with the nuclear components being located in the inner zone. Direct access to the inner zone from the outside is possible by means of ports between the two containments.

The inner vacuum containment (henceforth referred to as the vacuum vessel) is within the bore of the TF coils and lies outside the nuclear components, i.e. the first wall, the blanket and the bulk shielding. Since this chamber is the primary tritium boundary, double containment welding shall be used at all joints. This approach allows for continuous monitoring of the space between the containments for detection of possible tritium leaks.

5.1.3. *Vacuum vessel and ports*

The vacuum vessel is an assembly of 16 segments with parallel ends, hence called the parallel segments, and 16 wedged shaped pieces. The parallel segments are preassembled into the TF coils and are thus installed concurrently with the coils. The wedged pieces are installed between, and connected with, the parallel segments to complete the torus. To increase the structural rigidity required to withstand the atmospheric and eddy current loads (especially those generated during a plasma discharge), the shield structure is partially integrated into the vacuum vessel. Since the shield will have an operating temperature higher than that of the vacuum vessel, the necessary thermal expansion joints are incorporated into the combined structure. Interchangeable blanket modules with different thicknesses must be considered in order to meet the flexibility requirements placed on the device. These modules are installed after completing the assembly of the vessel. They are lowered into position through the upper port and connected with the vessel by attachments provided on the inside wall. Active plasma control coils, one upper coil and one lower coil, are installed in segments through the ports and are electrically connected outside the vacuum vessel.

The parallel vessel segments have long and thin metallic inserts that bridge the electrically insulated thick wall sectors which provide the required electrical resistance of the assembled vessel. The parallel sector is an assembly of two halves, preassembled in a shop to ensure the quality of the required seals, joints and electrical insulation, especially those surrounding the resistive metallic inserts. If required by the design option selected, magnetic inserts for reducing the field ripple would also be installed on the parallel segments. Once assembled, the parallel sector is placed in the bore of the TF coil for installation into the reactor assembly.

The wedged sectors include the access penetrations necessary at the interface between the reactor heating and current drive systems, the nuclear test ports, the machine diagnostics, the fueller and the assembly and maintenance equipment. These penetrations are made by using rigid walls tightly connected with the outer containment structure and include the necessary flexible mechanical elements which allow relative movements between the two vessels.

5.1.4. *Biological shield and cryostat concept*

The torus is surrounded by a concrete biological shield which also serves as a building structure, a vacuum boundary for the machine cryostat, and a kind of secondary containment. The lower and equatorial ports in the vacuum vessel penetrate this shield, such that access during shutdown is possible from the annular torus hall. This torus hall contains in-vessel maintenance equipment, test module hardware, fuellers, beam ducts and diagnostic support systems. The upper ports can be accessed by retracting a rolling shield (normally deployed during machine operation), which permits ex-vessel remote maintenance of the torus and removal of in-vessel components by cranes for transfer to maintenance/storage hot cells.

The overall height of the torus building is determined by the dimensions of the tallest torus component (a vacuum vessel segment in the reference design), since this component must be installed/removed with vertical clearance above the torus. When extra space is required for the crane, the roof trusses and the torus support structures and for access to the lower service stations, the height of the torus building will generally not be less than 2.6 times the tallest component.

5.2. Assembly and maintenance

5.2.1. *General requirements*

For assembly and maintenance, all machine components and systems are classified into three categories.

Those in the first category — the semi-permanent components — are assembled initially and should not be removed except in the case of failure. This category includes the superconducting coils and the vacuum vessel/shield structure, which are the main components of the basic machine.

The second category comprises in-vessel components, such as the breeding blanket, which can be removed during the lifetime of the machine for maintenance or to permit phased operation.

The third category comprises those in-vessel components which have a short lifetime or which undergo frequent damage, i.e. mainly the plasma facing components such as the divertor plate, the guard limiters and the first wall protection tiles. These components should be replaced as rapidly as possible to minimize reactor down-time.

With regard to assembly and maintenance, the most important design requirements for ITER are the following:

— Fully remote maintenance of the device should be aimed at, but with provision for hands-on maintenance where possible;
— It should be possible to operate the device in different phases with various blanket and divertor geometries;
— Maintenance of components with a short lifetime or a high failure rate should be possible without moving other components or disturbing the reactor's internal and external environment.

These requirements influence the design of each of the components, their integration into the ITER machine and the assembly procedures. Because of the necessity of performing all operations remotely, very simple assembly procedures with high reliability must be applied.

The guiding principles for maintenance design are: to minimize waste, to minimize contaminated space, and to enable simple and rapid maintenance with a minimum of remote handling equipment and without sophisticated mechanisms.

5.2.2. *Assembly scenario and procedure*

A modular design of the semi-permanent and internal components of the machine will help to meet the maintenance requirements. Modularity, as determined by the number of TF coils, means that the assembly procedures are developed for only one sector and then applied in the other sectors until the torus is completed. The advantage of such a modular approach is that, in the case of extensive repair operations, only one sector has to be dismantled to have access to all components within this sector. However, modularity cannot be applied for the PF coils. Therefore, special care must be taken to position these coils and to integrate them into the device so that they can be installed and removed without disturbing the TF coils or other components.

The assembly/disassembly process for the vacuum vessel depends largely upon the welding method — U-type lip sealing or thick plate welding. A vacuum vessel of the first type could be easily disassembled in sectors by cutting the lip seal. However, the use of a U-type lip seal would lead to void formation in the vacuum vessel structure, which may result in a reduction of the effective shielding capability. When thick plate welding is applied, such void formation is avoided and the vacuum vessel becomes a more robust structure. However, in this case, when a failed vacuum vessel has to be replaced, it is necessary to cut the thick plates, which could ultimately necessitate replacement of the entire vacuum vessel. Mechanical analysis of the electromagnetic forces resulting from a plasma disruption is required in either case.

5.2.3. *Maintenance scenario and procedure*

Provision of vertical, horizontal and oblique access is considered for the maintenance of the in-vessel components, i.e. the divertor, guard limiters, first wall protection armour tiles and blanket. The large elongation of the plasma and the layout of the PF coils necessitates top loading of some in-vessel components, mainly the inboard blanket segments and the guard limiters.

The blanket generally has two or three segments per machine sector. In the case of three segments, first the two side segments are installed in the vacuum vessel and then the central segment is inserted between the side segments. The sequence is reversed for retraction. The method of attaching the blanket segments to the vacuum vessel should be carefully studied, considering the high mechanical loads to be transmitted during plasma disruption. For blanket replacement, the required movements are vertical and horizontal, and an overhead crane with high positioning accuracy is required. Also, some remote handling equipment for movement in the horizontal direction is needed.

Mechanically attached armour tiles and guard limiters could be used as rapidly replaceable components for the inboard surface. These can be replaced without moving other reactor components.

Guard limiters are protruding sections of the first wall. They can be located either on the inboard surface or on the outboard surface and are intended to help localize damaged areas of the wall, thus facilitating repair work.

Maintenance of the first wall armour tiles is to be performed inside the vacuum vessel by using remote manipulators such as an articulated boom. Easy detachment of the armour tiles, rapid identification of the damaged parts (there are several thousand tiles) and mechanical attachment of the tiles to the first wall are the main factors in this procedure.

The maintenance method for the divertor is a determining factor in the configuration of the reactor and the building. Several options have been studied; the following two options are representative of them.

The first option is to use a divertor cassette assembly which can be replaced horizontally or obliquely from the outside. This assembly would consist of the heat absorbing surface, the service piping and a section of the shielding in the desired extraction path. For the lower cassette, a section of the vacuum pumping duct would be included in the assembly.

The second option is to handle the divertor plates from inside the plasma chamber. In this concept, only the heat absorbing surfaces would have to be replaced; however, a part of the first wall near the divertor may also have to be included. Maintenance and configuration schemes based on this concept have been conceived and evaluated.

The divertor maintenance method is based on the utilization of a horizontal port where a remote manipulator can be placed for removal and insertion/attachment of the required components. Both the upper and lower divertor parts can be served in this way.

This method minimizes radioactive waste, simplifies the building/reactor interface, eases contamination control problems during divertor exchange and simplifies the design of the top access port. A principal concern with this method, for which further in-depth analysis is required, is the feasibility of a manipulator which will have to manoeuvre 45–90° toroidally (depending on the number of maintenance ports) around the vacuum vessel and which will have to support the heavy divertor while performing the positioning and attachment work. A method for positioning and attaching the divertor plate in the vacuum vessel should also be developed.

All maintenance operations on the internal, plasma facing components will take place inside containment casks, under vacuum or in an inert gas atmosphere. This is necessary to avoid contamination of the plasma chamber and the release of tritium and activated dust to the outside of the reactor. However, a cask is not considered to be appropriate for the large components such as the blanket for which scheduled maintenance will be required only once or twice during the reactor life. Simple containment methods such as the use of plastic covers will be considered for these components.

5.2.4. *Concluding remarks*

In the design phase, quantitative and detailed comparisons of the advantages and disadvantages of the candidate options will be made in order to come to a decision about the appropriate maintenance option.

An important point is the selection of the maintenance scheme for the divertor, i.e. either maintenance from the outside with a cassette assembly or in-vessel maintenance. The key issues to be investigated are: reliable attachment/detachment mechanisms of in-vessel components to/from the inner wall of the vacuum vessel, development of an in-situ maintenance procedure and of equipment for in-vessel components, and development of simple equipment for containment.

5.3. Magnets

5.3.1. *Introduction*

The TF and PF magnets constitute the core of the basic machine. Since ITER is to have both long pulse ignition and steady state operation, the magnets must be superconducting in order to achieve the necessary performance (field and current density) and to minimize power consumption.

5.3.2. *System requirements*

The magnet system must satisfy requirements which enable the tokamak to achieve a certain minimum performance capability and which are at the same time compatible with the requirements of the other machine systems. These requirements, which have to be taken into account in the design, are as follows:

Central plasma field	>5.3 T (R = 5.5 m)
Field ripple at the plasma edge	$<1.5\%$
Steady state operation	hours
Magnetic capability	>250 V·s
Radius of the OH magnet	<1.87 m
Inner radius of the TF magnet	>1.9 m

The general specifications, which are tentatively based on these requirements for magnet design, are listed in Tables II and III.

5.3.3. *Design concept*

A number of design options for the TF and PF magnets have been considered, with a view to satisfying the system requirements in compliance with the design criteria. The following key design concepts have been presented in the definition phase. More detailed work, including mechanical, thermal and electrical analyses, will be done in the design phase.

(a) Electrical design

Several conductor and coil design concepts for TF and PF magnets have been studied. For the TF and central solenoid magnets (PF1 to PF3) the highest field is around 12 T. Therefore, a superconductor based on Nb_3Sn has been selected with a critical current density of up to 800 A·mm^{-2} at 12 T, 4.2 K and zero strain.

TABLE II. GENERAL SPECIFICATIONS OF TF COILS

Items	Reference values	Allowable range
He inlet temperature (K)	4.5	3.6–4.5
He inlet pressure (bar)	10	5–10
He outlet pressure (bar)	>5	>5
Nuclear heating ($mW \cdot cm^{-3}$)	1	<5
Insulator dose (rad)	2×10^9	5×10^8 to 5×10^9
Neutron fluence, >0.1 MeV ($n \cdot cm^{-2}$)	1×10^{19}	1×10^{19}
Copper disintegrations per atom (dpa)	5×10^{-4}	5×10^{-4}
Plasma disruption	$40\ T \cdot s^{-1}$ for 20 ms	—
Poloidal cycles	<100 000	<100 000
Charging cycles	<100	<100
Cooldown cycles	<20	<20
Design peak field (T)	12	11–12
Overall current density ($A \cdot mm^{-2}$)	13–14	13–14
Rated current, I_{op} (kA)	30–40	16–45
Critical current at operating conditions (kA)	$>1.6I_{op}$	$1.6–2I_{op}$
Temperature margin (K)	>0.5	>0.5
Pressure drop (bar)	<3	<8
Dump voltage (kV)	<20	<20
Dump time constant (s)	<10	5–10
Hot-spot temperature (K)	<150	<150

TABLE III. GENERAL SPECIFICATIONS OF PF COILS

Items	Reference values	Allowable range
He inlet temperature (K)	4.5 (Nb_3Sn)	3.6–4.5
	4.5 (NbTi)	3.6–4.5
He inlet pressure (bar)	10	5–10
He outlet pressure (bar)	>5	>5
Design peak field (T)	12 (OH coil)	12–13
	<7 (EF coil)	<7
Rated current, I_{op} (kA)	30–40	30–40
Critical current at operating conditions (kA)	$>1.6I_{op}$	$1.6–2I_{op}$
Temperature margin (K)	>0.5	>0.5
Pressure drop (bar)	<3	<3
Dump voltage (kV)	<20	<20
Hot-spot temperature (K)	<150	<150

For the equilibrium magnets (PF4 to PF6), a NbTi superconductor may be used because the magnetic field requirement is less than 7 T. The concept of forced-flow cooling by supercritical helium has been selected for all magnets because of the mechanical and electrical insulation requirements. To protect the magnets in the event of an accident, a terminal and ground voltage of up to 20 kV has been specified for them. Furthermore, cable-in-conduit type PF conductors and cable-in-conduit or monolithic type TF conductors have been tentatively selected.

(b) Structural design

On the basis of a preliminary stress analysis, two possible TF support concepts (wedging of the inner magnet legs or bucking onto the central solenoid) have been considered. There is general agreement that the two concepts offer similar machine capabilities for the same radial configuration. It has been decided to use the wedging method as the reference concept and the bucking method as an alternative.

The structural area required to sustain the magnetic force has been analysed. The inner intercoil structure has a strong effect on the out-of-plane deflection of the vault. The typical stress values in the vault region are <700 MPa, which is close to the allowable stress of 800 MPa. The outer intercoil structure must be very stiff because of the tension and shear loading generated by hoop and overturning forces. A coil case cross-section of 1.1 m $\times$ 1.1 m in the outer TF coil leg is selected as a reference geometry.

For the central solenoid magnets, both layer winding and pancake winding have been considered. With pancake winding, extra radial space is required for the pancake joints, the current lead joints and the piping connections, but in this case high mechanical stiffness in the axial direction is expected. On the other hand, when layer winding is applied, extra space is needed in the axial direction for the joints and connections, which reduces the mechanical stiffness but makes it possible to fully utilize the radial space. The winding method for the central solenoid magnets, which are operated with a cyclic loading of ~ 12 T, will be selected in the design phase on the basis of more detailed analyses including fatigue considerations.

(c) Thermal design

Preliminary calculations of the AC losses for the TF and PF magnets have been carried out. The results show that the average heat load is around 3 kW during nominal operation; an extra heat load of 100 kW at 4 K is required for the plasma swing of ± 15 cm at a frequency of 0.3 Hz. Since this heat load is of the same magnitude as the total heat load during nominal operation at 4 K, it is concluded that this part of the design may not be practical and should be revised in order to reduce the heat load to 10 or 20 kW by reducing the frequency or by using some other means.

A preliminary design concept of the cryogenic system has been developed, including a cryogenic pump. The total heat load at 4 K should be less than 100 kW, corresponding to a refrigerator electrical power of 50 MW at room temperature. This capacity of the cryogenic system enables the magnets (weight $\approx$ 10 000 tonnes) to be cooled within 300 h. An exact estimation of the pressure drop is necessary because the pumping power is part of the heat load of the cryogenic system.

A concept of double thermal shielding, composed of an 80 K shield and a 5 K shield, has been proposed in order to decrease the heat load in the 4 K region, to permit stable operation and to cool down the coil system. The TF coil case can be designed as a first thermal shield, cooled to around 5 K by the refrigerator in order to avoid excessive heat to the winding pack. The shield of the TF coil case is essential in the high field region in which high radiation heat is generated. In addition, an 80 K thermal shield plate should be installed between the coil case and the plasma vacuum vessel which can operate at up to 450 K.

5.4. Poloidal field system

The plasma equilibrium of ITER is controlled by the PF system, which consists of passive and active systems for fast vertical displacement, a slow control system with superconducting poloidal coils for plasma equilibrium, and a disruption control system. These systems have been studied with a set of parameters similar to the reference set and it will be easy to apply them to the reference device.

5.4.1. Vertical position control system

A control system for vertical instabilities poses constraints on both the plasma configuration and the design of the first wall/blanket and the vacuum vessel. Rapid vertical displacement caused by the instability of a vertically elongated plasma is initially restrained by magnetic fields due to eddy currents induced in the passive structure; then the active coils installed inside the vacuum vessel are excited to restore the magnetic field. Thus, the essential parameters are the stability margin and the growth rate of instabilities.

The stability margin is given by the plasma configuration and the structure; it should be higher than about 0.5 so that it is possible to cope with uncertainties and approximations in both the plasma modelling and the design of the in-vessel structure. The growth time of instabilities is given not only by the configuration but also by the time constant of the eddy currents in the structure and should be longer than the time constant of the active feedback system. To cope with the practical constraints of a realistic power supply system and an active control coil system, the natural growth time should be more than about 10 ms.

Because of the requirement of a largely elongated plasma, vertical position control with a rather ideal stabilizing structure set close to the plasma surface has been investigated. A saddle type structure is set on each segment of the outboard first wall/blanket. For an elongation of 2.5 at the separatrix magnetic surface, the stability margin is 0.2 and the growth rate is 4 ms with 48 segments of the first wall/blanket, and 0.34 and 8 ms, respectively, with 16 segments. Thus, it is very difficult to control such a largely elongated plasma in a reactor.

A more detailed study, assuming a plasma elongation of 2.2 at the separatrix magnetic surface or 2.0 at 95% of the flux surface, has been performed with realistic in-vessel structures. A suitable design of the plasma control system for this elongation can be achieved by optimizing the shape of the copper or aluminium saddle loops attached to each outer blanket and by minimizing the gaps between neighbouring blankets. It would appear that a realistic limit for elongation is about 2.2 at the separatrix magnetic surface.

Further investigations, especially for the active control system and the interface between the passive stabilizing loops and the blanket segments, are needed.

5.4.2. *Poloidal coil system for slow plasma control*

The sizes of the central solenoid coil and the outermost PF coils are important in evaluating the machine size. Optimization of the central solenoid coils has been studied. The maximum obtainable volt-second is strongly increased when the maximum magnetic field increases from 10 T to about 12 T, and it is weakly increased when the maximum field increases from 12 T to 14 T. The achievable volt-second actually decreases when the field increases to more than 14 T because of the large mass of structural material. Therefore, it is reasonable to employ a maximum field of 12 T for the central solenoid coils.

The radial position of the outer legs of the TF coils affects the poloidal magnetic field energy. Assuming about 1% TF ripple, a large reduction of the PF energy is obtained by increasing the number of TF coils from 12 to 14 or from 14 to 16, but a small reduction of the PF energy is expected when the number of TF coils is increased from 16 to 18. Therefore, it is reasonable to employ 16 TF coils for the PF system in the design of the magnets and the power supply.

Optimization of the number, position and cross-section of the PF coils located outside the TF coils has been studied with the plasma parameters of the initial engineering study, i.e. a current of 20 MA, a major radius of 5.8 m, a minor radius of 2 m, a double-null divertor, an elongation of 2 and a triangularity of 0.3–0.5 at the 95% flux surface, an internal inductance of 0.6–1.0 and a poloidal beta of 0.1–1.0 in the initial basic device with 16 TF coils and a 2 m radius for the central solenoid coils. The criteria used to assess the configurations are the controllability of the null location, the shape of the separatrix line, the current density, the circuit energy, the magnetic field levels and the forces on the PF coils.

The minimum number of independent PF coil sets is six, i.e. three central solenoid coil sets, one divertor coil set at the bottom and top of the device, and two coil sets outside the outer legs of the TF coils. The estimated error field from the PF coils is less than 0.001 T in the initial plasma region at breakdown. The error field from eddy currents in the reactor structures, especially in the vacuum vessel, is dominant. The total volt-seconds is 280–290, which is sufficient for 20 MA operation. The coil concept can also cope with flexible and extended capability, as discussed in Section 7. The maximum circuit energy is 12 GJ in this case, but it can be reduced to 8 GJ by reducing the radius of the top and bottom coils of the outermost coil sets by about 2 m.

A similar concept is also suitable for the reference parameters in the physics phase, i.e. a plasma current of 22 MA, a plasma major radius of 5.8 m and a plasma minor radius of 2.2 m. This concept should be improved for the reference parameters in the technology phase, i.e. a plasma current of 18 MA, a major radius of 5.5 m and a minor radius of 1.8 m, because the coupling between the PF coil system and the plasma is not favourable for operation with the small plasma radius. The estimated total volt-seconds is in the range of 230–260 V·s; this is marginal for the 22 MA inductive operation in the physics phase and the 18 MA inductive operation with the small radius in the technology phase.

Further study is required, especially for operation with the small radius, and for different operation scenarios, power supply systems and plasma control systems.

5.4.3. *Electromagnetic effects of disruption*

The engineering consequences of disruption are mechanical loads resulting from the interaction of induced currents with magnetic fields, heat loads in the TF and PF coils and resistive elements of the vacuum vessel, overvoltages in resistive elements, in-vessel components and other structures, and induced currents in the PF coils, especially in the divertor coils. The simplest disruption model with a static plasma was employed for the preliminary assessments: a plasma current of 20 MA, a current decay time of 20 ms and no vertical displacement. A study was also made for a vertical displacement of 1 m, which is larger than the displacement of 0.6 m laid down in the physics guidelines. The reference value of the vacuum vessel resistance is 40 $\mu\Omega$.

The induced current in the vacuum vessel is about 15 MA and the decay time is about 40 ms with the 40 $\mu\Omega$ resistance. The decay time is a weak function of the one-turn resistance in the range of a few tens of micro-ohms. The maximum current density is about 1 $\mathrm{MA \cdot m^{-2}}$. The peak value of the radial force is about 90 MN. The vertical force due to asymmetry of the vacuum vessel is 10 MN, which is about 30% of the vessel weight. Another $\pm$32 MN is expected from the 1 m vertical displacement.

The out-of-plane forces in the TF coils do not increase significantly with disruption. The maximum change of the magnetic field is about 40 $\mathrm{T \cdot s^{-1}}$ in the inner legs of the TF coils, with a decay time constant of about 40 ms.

The induced total current in the PF coils is 11 MAT, but the induced current reduces the total coil current in all coils except the divertor coil. The maximum value is about 0.8 MAT, which is less than 10% of the initial current in the divertor coil.

These results suggest that the machine can accommodate the current disruption effects. However, the forces in all parts of the vacuum vessel and the in-vessel components are complex and will have to be extensively studied, especially with regard to vertical plasma displacement.

TABLE IV. TORUS VACUUM PARAMETERS

Volume ($\mathrm{m^3}$)	1100
Surface area ($\mathrm{m^2}$) (projected)	800
Surface material	Graphite
Initial pump-down pressure (600 K, mbar)	4×10^{-7}
Pre-shot base pressure (mbar)	4×10^{-5}
Leak rate[a] ($\mathrm{mbar \cdot L \cdot s^{-1}}$)	10^{-5} to 10^{-6}
Effective pumping speed ($\mathrm{m^3 \cdot s^{-1}}$)	
range	350–700
nominal	500
Required downpipe conductance ($\mathrm{m^3 \cdot s^{-1}}$)	1000–1500
Total operating pump speed required ($\mathrm{m^3 \cdot s^{-1}}$)	1000–1500

[a] Maximum tolerable system in-leakage rate, based on oxygen.

5.5. Primary vacuum system

The primary vacuum system requirements are summarized in Table IV. The effective pumping speeds for helium at the divertor exit, predicted to be in the range of 350–700 $\mathrm{m^3 \cdot s^{-1}}$, are considerably larger than those reported for INTOR (100 $\mathrm{m^3 \cdot s^{-1}}$). The duct diameters, particularly in the divertor region, must be as large as possible, and this necessitates a trade-off with the coil shielding. For these pumping speeds, compound cryopumps (CCPs) are the preferred alternative. The main technical uncertainties concern the definition of the most suitable helium pumping surface, the effect of dust on valves undergoing frequent redeposition during pump regeneration, and the effect on tritium processing when cryotrapping with Ar or SF_6 is used.

5.6. Current drive and heating systems

The primary objective of the current drive and heating (CD&H) design activities is to satisfy the physics requirements for bulk plasma heating to ignition and for steady state non-inductive operation. The CD&H systems should also provide plasma startup assist (gas breakdown, plasma preheating, current initiation), profile control and current ramp-up. The design of current drive is the most demanding task and determines the required power. Provided the necessary technology can be developed, any of the systems studied could bring ITER to ignition, but the philosophy for CD&H is that the same systems will be used for heating and for current drive. Therefore, four heating methods are under consideration: electron cyclotron waves (EC), ion cyclotron waves (IC), lower hybrid waves (LH) and neutral beams (NB).

The systems have been combined in three scenarios for study: (1) NB, LH and EC, (2) EC and LH, and (3) IC, LH and EC. In all scenarios, LH waves will be used to drive current in the outer region of the plasma and to assist in ramp-up; EC waves will be used to initiate the plasma and provide for disruption control. In the core of the plasma the current will be driven by NB, EC or IC. Scenarios (1) and (3) will provide about 100 MW and scenario (2) will provide about 150 MW to the plasma, which is expected to drive a current of about 13 MA in steady state operation. Also, 5 MA of bootstrap current is expected, which would give a total current of 18 MA.

The first scenario, using LH, NB and EC, is the most credible one, given the present physics database. The second scenario, where EC waves would drive the current in the core of the plasma, is the preferred option because of the perceived simple plasma/torus interface, but an improved database would be needed for it to be viable. The third scenario, where IC waves would drive the current in the core of the plasma, is an attractive option because of its potential for high efficiency and relatively simple technology, but, again, an improved database is required. All systems are technically challenging, but the design of the antennas/launchers for LH and IC heating will require special effort because they must be integrated with the first wall. The viability of the NB option is dependent on the successful development and demonstration of negative ion neutral beams with a nominal energy of 1 MeV, which will require a significant improvement in technology. In addition to the necessary improvement of the physics database for the EC method, high power, high frequency sources will have to be developed.

5.6.1. *Electron cyclotron system*

Two EC systems are being considered: a startup and profile control system with a nominal frequency of 140 GHz, and a current drive/heating/profile control system with a nominal frequency of 180–200 GHz. For the scenario with EC current drive, it is not clear whether the higher frequency is consistent with startup and profile control. Therefore, it may be necessary to use the startup system at 140 GHz to augment the current drive system.

The total power required for startup and profile control is estimated to be 10–20 MW. Several development programmes are under way to produce a 1 MW gyrotron at the desired nominal frequency of 140 GHz. The critical design issues for this application include the plasma–machine interface and the development of reliable gyrotrons.

The design of the EC current drive system is more demanding since the nominal frequency is between 180 and 200 GHz and the total power required is nominally 130 MW. Tangential access to the current drive system will be necessary for launching power at about 35° from perpendicular. The critical design issues include: the development of an EC source at the desired frequency with a reasonably high power rating so as to minimize the number of sources required, the use of RF windows which can reliably isolate the EC source(s) from the plasma and tritium, and the use of RF components to make this large and complex system practical.

5.6.2. *Ion cyclotron system*

The IC system is being considered for central current drive and heating. This would require about 100 MW of RF power at a frequency in the range of 35–100 MHz. The critical design issue for the IC system is the antenna(s). The options being considered for the IC antennas include all-metal loop antennas of the type presently in use in large devices and waveguide antennas with a developed radiating surface. Preliminary studies show that antennas with a surface of ~ 20 m^2 will be required. This large antenna surface with many radiating elements will be effective in coupling the desired power (nominally 100 MW) to the plasma and launching a well tailored power spectrum at long wavelengths that will couple effectively to a plasma with a narrow scrape-off layer and drive current efficiently. If multiple antennas are used, they must be arranged in one or two continuous arrays. The antennas must be close to the plasma and therefore they have to be treated as plasma facing components. The RF sources, components and transmission lines are relatively well developed and no major development programmes will be needed.

5.6.3. *Lower hybrid systems*

The LH system is being considered for current drive, heating, current ramp-up, current profile control and possibly current initiation. The total power required is about 50 MW and the frequency is in the range of 4–8 GHz. A final choice of the frequency will be made in 1989, on the basis of the theoretical, modelling and technical studies now in progress. No major technology step is required for the development of RF generators and transmission lines. A critical issue is the launcher design. The present baseline design employs a 'grill' launcher such as is used in ongoing programmes on large tokamaks. The launcher must be in contact with the plasma and therefore it has to be treated as a plasma facing component. Also, it must be possible to move the launcher so that it can cover the plasma edge.

5.6.4. Neutral beam system

Neutral beams are being considered for driving current in the plasma core and for heating. The deuterium beam energy chosen for the concept is in the range of 0.45–1.6 MeV, which appears to be consistent with the physics requirements. The required power is estimated to be between 50 and 100 MW, and the neutral beams should be injected along a tangent to the magnetic axis of the machine in order to achieve maximum current drive efficiency. It must be possible to adjust the profile of the driven current in the core of the plasma with the NB system; this in turn sets requirements on the beamline geometry and/or the beam energy. The number of ports and the port size should be minimized.

A concept for a 100 MW NB system has been developed for ITER. A typical beamline will include the following elements: negative ion plasma generator, DC accelerator, neutralizer, ion deflection system, particle dumps and vacuum system. The technology for all these elements will have to be improved. The efficiency of the gas and electrical systems and the current density of the plasma generator should be improved by a factor of two to four. The required NB energy is a factor of five to ten above the level demonstrated on large tokamaks. Two approaches will be considered: a DC accelerator based on positive ion technology and an electrostatic quadrupole (ESQ) accelerator. Although a gas cell neutralizer would be adequate, a plasma neutralizer will be investigated because it may improve the overall system efficiency from 45% to 60%. Particle dump technology will have to be investigated to ensure that high energy particles can be reliably absorbed; this may require the application of direct energy recovery techniques. Finally, a continuously working vacuum pump must be developed so that the beamlines can operate continuously for up to two weeks.

5.7. Plasma facing components

Because of the required flexibility of the different plasma scenarios and, additionally, the present understanding of plasma edge physics, a wide range of possible operating parameters for plasma facing components (PFC) have to be considered, concerning mainly the nominal heat fluxes and particle energies as well as the disruption heat loads and the frequency.

The operating conditions for PFC will first be established during the initial physics phase and it will then be possible to optimize the design of these components.

5.7.1. Plasma facing armour material

Carbon based materials — preferably carbon fibre composites (CFC) — will be used for the plasma facing armour, hoth on the divertor plates (DP) and in the major areas of the first wall (FW), at least during the physics phase. The reasons for the use of carbon based materials are their unique thermomechanical properties and their generally good performance in present large tokamaks. The favourable aspects of using carbon based materials include (i) the elimination of high-Z impurities; (ii) the superior disruption resistance, with a predicted loss of material per typical disruption of 0.1 mm and 0.01 mm for DP and FW, respectively; and (iii) the lower sensitivity of the DP sputtering erosion to the particle energies compared with high-Z material such as tungsten (the predictions for most of the ITER plasma scenarios do not favour the use of tungsten for DP armours).

There are, however, also a number of critical issues associated with the use of carbon based materials for the plasma facing armour:

(a) The main drawback is the high nominal gross erosion rates for the divertor plates. There are indications that the net erosion rates could actually be reduced by two orders of magnitude through redeposition and sweeping of the separatrix and through the use of materials with less chemical erosion (by hydrogen and oxygen). Radiation enhanced sublimation is estimated to limit the peak DP and FW temperatures to 1500 and 2000°C, respectively.

(b) The retention of hydrogen and other gaseous impurities seems to be high at temperatures below 1000°C, in particular with irradiation, while higher temperatures lead to reduced impurity content and increased outgassing. Therefore, baking of the armour at 350°C is mandatory.

(c) Radiation damage as a result of swelling caused by a neutron fluence of 1–3 $MW \cdot a \cdot m^{-2}$ is expected to limit the lifetime of the armour.

(d) Water and/or air ingress into the plasma chamber with carbon materials above 1000°C represent potential safety hazards.

5.7.2. *First wall*

The following features have been selected for the FW:

- The FW panel is integrated with the removable blanket segments so that a box-like structure is achieved;
- The structural material of the blanket is solution annealed type AISI 316 austenitic steel cooled by water at ≤ 1 MPa and ≤ 100°C under normal operation.

The following FW design options are being studied in more detail:

- Poloidal or toroidal coolant channels with or without double containment of the coolant;
- Mechanical attachment or bonding of 15–20 mm thick armour tiles to the FW structure, with cooling of the tiles by radiation or conduction.

For the different FW concepts proposed so far, allowed nominal peak heat fluxes limited by cyclic thermal fatigue of the steel structure have been estimated: for typically 3×10^4 burn cycles at 1 $MW \cdot m^{-2}$ neutron wall load, the peak heat fluxes are about 0.8 and 0.2 $MW \cdot m^{-2}$ for the radiative and the conductive armour tiles, respectively.

5.7.3. *Divertor plates*

The DP of the basic concept consists of a 5–10 mm thick armour brazed onto a water cooled heat sink of copper or molybdenum alloy. The preliminary results for the thermomechanical performance of the DP indicate:

- Static peak heat fluxes of up to 10 $MW \cdot m^{-2}$ may be marginally acceptable, considering the limits for armour temperatures and thermal stress of the heat sink;
- Peak heat fluxes of 15–20 $MW \cdot m^{-2}$ may be permissible when the separatrix is swept within the limits acceptable for the coil system.

The major critical issue for the DP seems to be the limited lifetime of the carbon armour:

- In the physics phase, after about 100 major disruptions the plates may have to be exchanged;
- In the technology phase (assuming very few disruptions), for five DP replacements the net erosion would have to be about a factor of one hundred lower than predicted for the case without redeposition and separatrix sweeping.

TABLE V. ALLOWED AVERAGE NEUTRON WALL LOAD ($MW \cdot m^{-2}$)

	Ignited plasma	Driven plasma	
	$Q \approx \infty$	$Q \approx 10$	$Q \approx 5$
Divertor heat flux — with sweeping	1–1.3	0.7–0.9	0.4–0.5
First wall heat flux — at 3×10^4 pulses	1.4–1.7	1.2–1.5	1–1.2

5.7.4. *Neutron wall load limits*

Table V presents the allowed average neutron wall loads at which the limits of the FW and DP heat fluxes for different ITER plasma scenarios are not exceeded. The range of values is given for the worst case and for the average case of radiated power to the FW.

Table V illustrates that the performance of ITER is critically constrained by the capacity of the DP, in particular for the case of a driven plasma. Possible solutions for the problems in connection with the DP are discussed in Section 8.

5.8. Blanket and shield

A blanket is required for producing the tritium necessary for ITER operation, particularly during the technology phase. The test modules and sectors for the blanket options of power reactors are considered in the later sections. Emphasis is placed on blanket concepts which minimize the risks connected with the operation of ITER and the R&D problems of which can be largely resolved before the start of detailed design. Less emphasis is placed on reactor relevance.

5.8.1. *Blanket options*

All blanket options must be compatible with the basic machine configuration and maintenance scheme. The blanket will be integrated with the FW in a box-type construction, with separate coolant paths for the FW. The poloidal sectors of the blanket will be removed via vertical lifts. Generally, the coolant headers will be located in the upper part of the reactor.

Because of the desired achievement of a tritium breeding rate (TBR) of 1, as well as the limited inboard space and the need for penetrations, the local TBR obtained with the various blanket options must be relatively high. This implies the use of neutron multipliers such as beryllium and lead.

Stainless steel type 316 has been recommended as the reference structural material for the FW and blanket. The main reasons for this are a better database and ease of fabrication. Backup materials include cold-worked austenitic steel (e.g. prime candidate alloy, PCA) and magnesium stabilized steels. The latter are of interest because of reduced long term activation and thus may have some advantages with regard to waste management. However, such steels have a higher short term activation which increases the afterheat problems.

The reference coolant for ITER is low temperature, low pressure water.

A number of blanket options for ITER have been considered. Three concepts, which are believed to have the potential to meet ITER's goals, have been selected for more detailed studies, namely the aqueous salt concept, the solid breeder concept and the lithium–lead concept. The first option emphasizes design simplification and has limited reactor relevance. The other two concepts are supported by a reasonable worldwide R&D programme; also they have more reactor relevance and are judged to have acceptable risks.

In the aqueous salt concept, lithium bearing salts (e.g. LiOH, $LiNO_3$) are put in the water coolant to achieve tritium breeding. Typically, significant amounts of beryllium are required for neutron multiplication. The water flows through pebble beds of beryllium balls or between beryllium plates.

In the solid breeder concept, a solid lithium ceramic (e.g. Li_2O, Li_2ZrO_3, $LiAlO_2$ and Li_4SiO_4) is used for tritium breeding. As with the first concept, beryllium is mostly used as neutron multiplier. The beryllium (~75%) can be mixed with the ceramic (~25%) in the form of small spheres or else a plate type arrangement can be used.

In the lithium–lead concept, a eutectic (17Li–83Pb) is used which is cooled by water. The eutectic may be in a solid or a liquid state during fusion burns. The eutectic material is heated and during off-times it is directed out of the reactor to recover the bred tritium.

Each of the three blanket concepts has its particular advantages and disadvantages:

Aqueous salt concept

Advantages: Simpler design, easier transition from non-breeding to breeding, demonstrated tritium recovery from water, tolerance of power variations.

Disadvantages: Large tritium inventory in water, limited reactor relevance, possible enhanced stress corrosion of steels, possible radiolysis and electrolysis of the water, and production of ^{14}C if $LiNO_3$ is used.

Solid breeder concept

Advantages: Reactor relevance, potentially low tritium inventory, no tritium in the main water coolant, much reduced chemical reaction potential.

Disadvantages: Performance at high lithium burnup, less tolerance to power level variation, more complicated design to ensure thermomechanical parameters, separate purge stream required.

Lithium–lead concept

Advantages: No beryllium is required; there is a lower potential of chemical reactions when the material is in solid form; low cost material, and reasonable tolerance of power variations.

Disadvantages: Necessity of tritium containment/recovery; large tritium inventory in solid form; low melting temperatures and volume change in phase transition, heavy weight and production of polonium.

5.8.2. *Shield concepts*

The shield is integrated with the vacuum vessel. Because of structural requirements, the vacuum vessel itself is a thick component (tens of centimetres) and thus has an important shielding function. The shield and the vacuum vessel are considered as semi-permanent structures, i.e. replacements of these components are not foreseen.

The primary consideration for the nuclear design of the shield is the protection of the superconducting TF coils. Another consideration is the doses to plant workers, but this is less important since totally remote maintenance is specified.

The nuclear analysis of the shield and vacuum vessel concentrates on the response of insulators in the TF magnet and on nuclear heating in the coil. For epoxy based insulators, doses of 5×10^8 to 5×10^9 rad are considered. Nuclear heating of up to a few tens of kilowatts is also considered; the limit depends on the design of the TF coil and the allowed cooling power of liquid helium. Furthermore, the fast neutron fluence on the superconductor must be considered. For Nb_3Sn, fluences of up to 10^{19} n·cm^{-2} (at a spectrum typical of a TF coil in a fusion reactor) are possible; this is usually not a constraint for the magnet design. Finally, radiation damage of the copper stabilizer must be taken into account; this depends on the details of the magnet design and the operating scenario.

The preferred material for the shield is stainless steel (type 316) together with water. Within the constraints of the reference design there is space for 85 cm SS on the inboard shield, from the TF magnet to the FW. For a graphite covered FW (2 cm thick) and a thin inboard blanket (5–20 cm), preliminary recommendations have been made regarding the radial distribution of water and steel in the shield and the vacuum vessel. Typical nuclear response parameters for the TF coils are insulator doses of $\sim 2 \times 10^9$ rad and nuclear heating rates of 10–15 kW (assuming a thicker outboard shield/blanket). In principle, the shield thickness could be reduced; however, this can only be done by permitting higher nuclear doses and higher heating in the coil or by using substantial amounts of a heavy material such as tungsten. Tungsten has the disadvantages of a high cost, high afterheat and difficult fabrication.

Preliminary consideration has also been given to the major penetrations of the shield for the vacuum pumping and plasma heating systems. The typical cross-section of the port is 1 m × 3.5 m, for which a minimum shielding of 40–50 cm near the back of the shield and 70 cm or more in the region close to the plasma will be required.

5.9. Fuel cycle

The ITER fuel cycle includes a set of processes for the recovery and reuse of tritium and other hydrogen isotopes in a closed cycle. The process elements of the fuel cycle are associated with three main functions — fuelling and plasma exhaust processing, tritium recovery from the blankets and tritium processing, and processing of various liquid and gaseous streams and solid wastes. The key elements of the fuel cycle which were addressed during the definition phase are summarized below.

5.9.1. Fuelling

Fuelling will be based on gas puffing, probably in the upper divertor region. For the construction of the fast valves which can provide the required fuelling rate, only a modest extrapolation of the valve designs for current large tokamaks is required.

Pellet injection will be used for edge density profile control and for assistance in fast density ramp-up. Typical pellet injector characteristics are:

Velocity	1.5–2.0 km·s^{-1}
Repetition rate	1.0–3.0 Hz
Pellet radius	0.2–0.5 cm

Ultra-high velocity (>2 km·s^{-1}) pellet injectors may be provided if the need for them is demonstrated by experiments. At present, however, there is no technology for the development of a high velocity, high repetition rate, continuous pellet injector; also, there are inherent limits which are due to the maximum rate of D–T pellet acceleration. Pellet injection at ultra-high velocity will probably require jacketing of the pellets and the use of a superheated propellant — factors which make the design of a reliable continuous pellet injector extremely difficult. Furthermore, for the plasma dimensions and conditions of ITER, it is unlikely that pellet penetration to the centre can be achieved. If this is required, alternative approaches may have to be found. For one such mechanism — compact toroid injection — a proof of principle testing may be performed.

5.9.2. Tritium recovery from breeder blankets

For the three blanket concepts proposed for detailed consideration in the ITER design phase, key issues of tritium recovery have been identified.

In principle, for tritium extraction from an aqueous lithium salt blanket it is possible to use the technology applied in heavy water reactors. Elements of the extraction system which require further development include the suppression of radiolysis or, after radiolysis, gas/water separation and/or recombination, as well as separation of the lithium salt. With high local tritium concentrations (>100 Ci·L^{-1})* in the extraction system, special attention will have to be paid to local confinement and to maintenance procedures. The pre-concentration step for tritiated water is somewhat energy intensive (up to 25 MW). However, this is low grade energy (<200°C) and may be partially derived from waste heat from the cooling of other plant systems.

* 1 Ci = 3.7×10^{10} Bq *or* 37 GBq.

For the ceramic breeder, the technology of tritium recovery in the purge gas is straightforward, consisting of a combination of catalytic oxidation and drying or cold trapping. However, because of the need to process the collected, highly tritiated water ($>50\ L\cdot d^{-1}$, $>1000\ Ci\cdot L^{-1}$), the design alternatives of shifting tritium to the gaseous phase have to be carefully considered, i.e. electrolysis, vapour phase catalytic exchange (VPCE), or liquid phase catalytic exchange (LPCE). Earlier studies have generally assumed that a low leakage, low inventory electrolysis cell would be available, but the water processing rate in ITER will exceed that obtainable with the cells currently being developed. If pre-swamped molecular sieve dryers were used in place of cold traps, a larger volume of more dilute tritiated water would be recovered; the highest degree of safety may be afforded by feeding the recovered water directly to LPCE or VPCE systems.

Tritium recovery from a lithium–lead blanket is also expected to be technically feasible. However, a database for tritium behaviour in the breeder under the proposed operating conditions still has to be developed. Since tritium recovery is a batch operation, performed at elevated temperatures ($>400^{\circ}C$) with large accumulated inventories (up to 2 kg) and potentially high tritium partial pressures (>10 bar), special attention must be given to inventory segregation and to containment of the process lines and extractor elements where permeation losses can occur.

5.9.3. External tritium recovery

In the ITER definition phase, the necessity of extracting tritium from a large variety of solid waste materials has been identified as having a potentially large impact on the layout of the plant and facilities, especially the hot cells. High priority must be given to the development of treatment processes for graphite tiles and the dust accumulated on the FW armour because of the large amonnts of tritium involved (100–1000 g) and the potentially large quantities of material to be treated because of the relatively short lifetime of the armour. Other solid wastes that pose significant problems with regard to tritium recovery include stainless steel, which may have to be subjected to vacuum degassing, and beryllium, for which a stable oxide layer may have to be removed.

5.9.4. Plasma exhaust processing

Impurity separation from fuel flows by using a cryogenic molecular sieve is a proven approach which can be considered to be available for ITER. It can also be expected that advanced extraction techniques, using Pd/Ag membranes, will be proven within the time span of the design work for ITER. Tritium recovery from impurities will require further development of an electrolysis cell or the use of an equivalent reduction process. This is also anticipated in the ITER R&D programme.

For isotope separation the proven technique of cryogenic distillation will be used.

5.10. Plant systems

5.10.1. General description

A series of plant systems and services is required for the operation of ITER in support of the reactor. Since the cost of these systems and services may account for 50% of the total cost, it is important to analyse their capacities, major characteristics, sizes and costs as well as their inherent problems.

The reactor building is sized to house the reactor and to provide space for assembly and maintenance. The building layout and size will be worked out in detail during the design phase. The rooms and the equipment must be so constructed that the largest diameter PF magnets can be transported.

ITER may contain up to several kilograms of tritium and about 1 GCi of activation products, and the total radioactivity release inside the reactor building in the case of the most severe accident may be several megacuries. Therefore, the reactor must have a containment which prevents the release of radioactivity to the atmosphere and retains its leaktightness in all anticipated accidents. An alternative is a reactor building equipped with a reliable filtered vent confinement system which can be used in an accident to relieve the overpressure by passing the gases through the filters and which ensures that radioactivity releases to the environment are kept at permissible levels. It is also possible to combine these two approaches.

The tritium processing building should be adjacent to the reactor hall. The main tritium processing equipment, various collection and storage tanks and air detritiation facilities are placed in this building.

The main treatment and processing system will have a triple boundary arrangement: the system boundary, the 'glove box' boundary and the confinement area boundary. All lines carrying tritium should be double-walled.

The power supply for components should include all specialized power supplies for the heating and current drive systems, magnets, cooling systems, cryogenic systems, etc.

The electrical power distribution system includes high, medium and low voltage equipment, a switchyard, transformers, breakers, reserve diesel generators, and auxiliary and general electrical systems.

The systems for gas, fluid and heat supply include supply and storage facilities for helium, nitrogen and compressed air; facilities for auxiliary steam generation; steam, hot water and cold water distribution facilities; heating facilities for primary cooling water system startup, and conventional heating installations.

It is assumed that liquid helium at a temperature of 4 K and liquid nitrogen at a temperature of 80 K will be produced on the site. For normal operation of the superconducting magnets a helium liquefier/refrigerator of 70 kW is required. This capacity has to be increased to 100 kW if a plasma swing at a frequency of 0.1 Hz is necessary. The required helium should be stored as a liquid and as a gas (at 18 bar), and the nitrogen should be stored as a liquid.

The cooling water system consists of water and possibly steam circuits which remove heat from the primary heat transport system. The total capacity of the system is about 1.7 GW. The circulation of the secondary loop water is possible with or without cooling towers, depending on the site location.

If required, the system will include a turbogenerator as well as condenser and feedwater heating facilities.

Liquid and solid waste handling systems are required. The liquid waste handling subsystem consists of collecting drains, sumps, holding tanks, pumps, filters, ion exchange beds and possibly evaporators. Solid wastes should be compacted and low level wastes may be stored on the site in special structures.

The site services will include a factory for winding of the superconducting coils if their transportation appears impractical. There will also be premises for administration and offices for medical and social care, a rail and road network, fire station garages, a petrol station and site security services.

5.10.2. Plant layout

For the ITER plant, an area of at least 1 km by 1 km will be necessary. The location of the reactor building relative to the other plant structures will be determined by the requirements to minimize the potential release of tritium and activated materials to the environment, to reduce power losses and to meet certain technical constraints connected with the length of the communication system.

It is necessary to provide direct access from the reactor building to the maintenance building, the tritium processing building and the building containing the heat removal system and auxiliary systems which may become contaminated. The reactor building is the central element of a block of buildings with controlled access. Other services and buildings will be arranged around this block as required, depending on practical and cost considerations.

6. SCOPING STUDIES AND RATIONALE FOR CONCEPT SELECTION

The ITER reference parameters have to be chosen such that the device is capable of achieving the joint objectives of ignition and steady state performance. With the present database, especially for energy confinement, it is difficult to optimize the design on the basis of a single set of physics parameters. The plasma performance for ignition may be quite different from that required for nuclear testing. Therefore, the parameters for ITER must be selected with a view to achieving the most acceptable compromise, permitting a broad range of operational flexibility to deal with the physical uncertainties and joint objectives, without incurring unreasonable expenses.

6.1. Overview of the design space and ignition capability

Since the energy confinement time scalings under consideration depend to various degrees on the plasma current and the aspect ratio (i.e. the plasma size), the choice of the device parameters is best described in the plasma current–aspect ratio (I–A) space.

Figure 3 shows a typical example, calculated for reasonable physics and engineering assumptions regarding the highest field at the TF coil, the distance between the TF coil and the inner plasma edge, the safety factor at the plasma edge, the plasma temperature, and a limited neutron wall load. The curves for the key parameters traverse the design space from top right to bottom left. Therefore, moving to the lower right corner of the I–A space leads to larger, more expensive, higher powered devices. Conversely, but subject to constraints on the radial build, moving to the top left reduces the size, cost and power to more manageable levels.

The enhancement factors for the energy confinement time τ_E needed for ignition are varied over the I–A space according to the scaling law. Goldston scaling

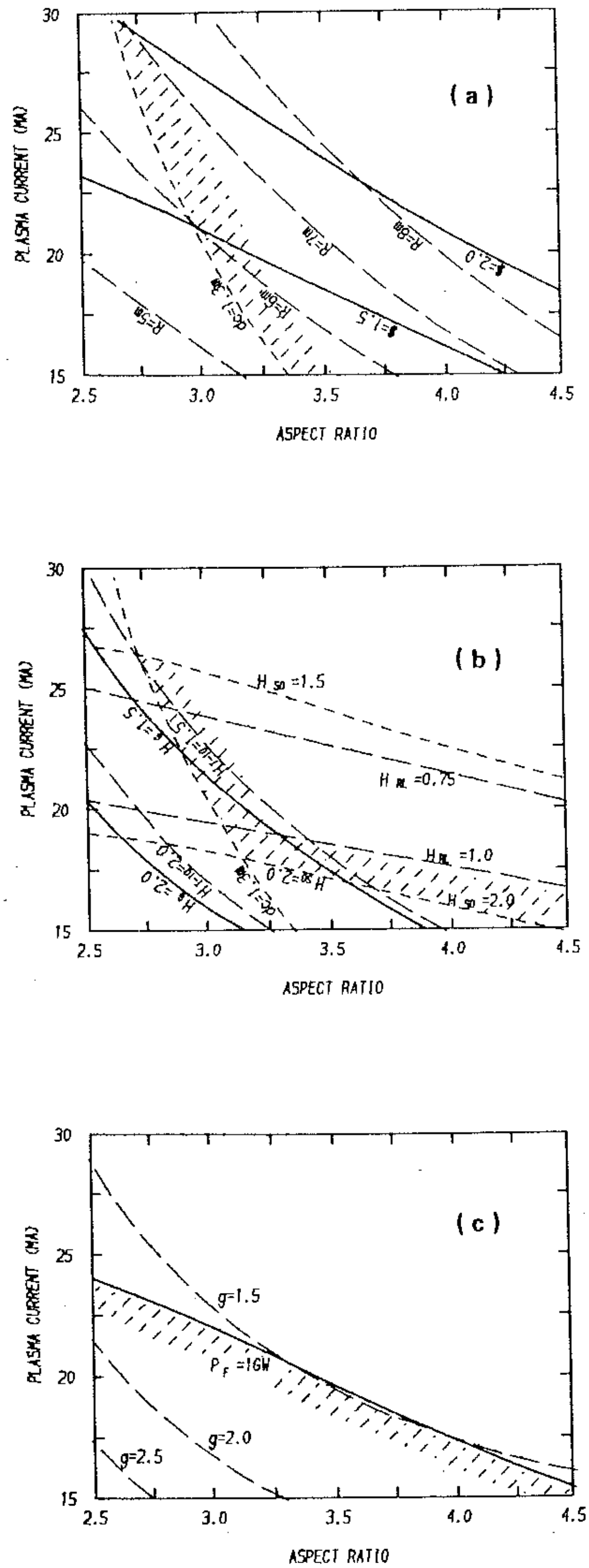

FIG. 3. Main parameters of I–A space at 1 MW·m^{-2}. (a) Geometry and cost, (b) enhancement factors and (c) fusion power and g.

and T-10 scaling favour devices of high aspect ratio rather than devices of high current for cost reduction. Rebut–Lallia scaling and Shimomura–Odajima scaling favour devices with high current and with low aspect ratio, to reduce enhancement factors; this leads to lower cost. Assuming that the enhancement factors required for ignition are at most 2.0 for the Goldston, Shimomura–Odajima and T-10 scalings, the left boundary in Fig. 3(b) is set by the curve for the Shimomura–Odajima scaling. The low aspect ratio side of this area is restricted by requirements regarding the radial build of the coils.

To attain ignition with a reasonable enhancement factor requirement (around 1.8 for the Goldston, Shimomura–Odajima and T-10 scalings) in a technically feasible device with an acceptable power level (less than 1 GW) and reasonable costs, it is considered that the parameters I ~ 20 MA and A ~ 3 represent the best compromise.

6.2. Current drive capability over the design space

The current drive performance of a given device is somewhat more difficult to express than the ignition performance since there is much greater flexibility for current drive in the available operating space. For instance, the current as well as the density and temperature can be varied widely in a given device to obtain reasonable values of neutron wall load ($P_n > 0.8$ MW·m^{-2}), current drive power ($P_{CD} < 100$ MW) and energy multiplication factor ($Q > 5$), with acceptable values of the enhancement factor.

Figure 4 illustrates this flexibility over the temperature–plasma current (T–I) operating space for a device with I = 20 MA and A = 3. At a given plasma temperature the lower limit of the plasma current is established by the need to achieve sufficient wall load and the upper limit is determined by the constraint on current drive power. Q-values of 7–9 are achieved within the region between these limits. The required enhancement factors are at least 1.5 for the Shimomura–Odajima scaling and at least 1.7 for the Goldston scaling. If lower enhancement factors were obtained, the operation point could be shifted to a lower plasma temperature and/or a higher plasma current. However, in this case, more fusion power would be produced and a higher current drive power would be required because of the reduced efficiency. The neutron wall load and the total heat load to the divertor would then pose serious problems.

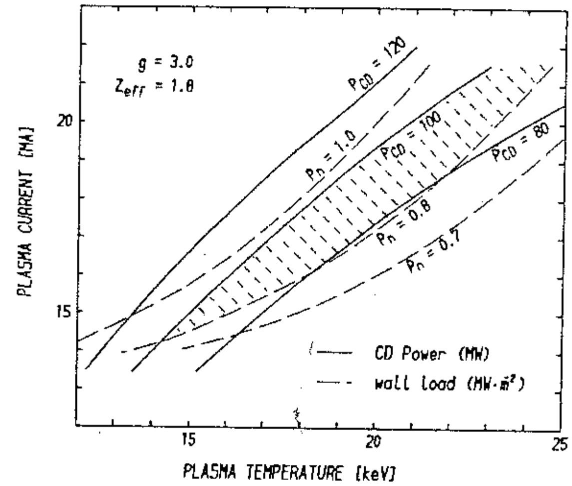

FIG. 4. T–I space for a device with I ≈ 20 MA and A ≈ 3.0. The operating window is bounded by the contours of P_{CD} = 100 MW and P_n = 0.8 MW·m^{-2}.

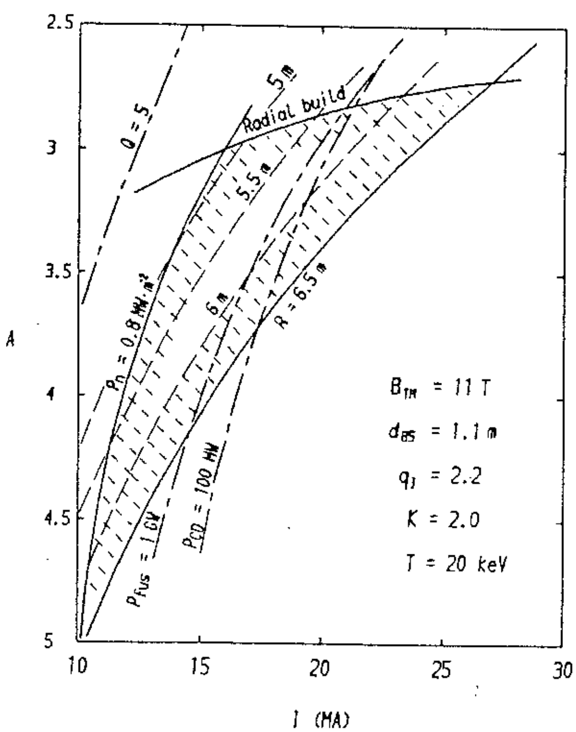

FIG. 5. Acceptable design space on the I–A plane, with the plasma temperature fixed at 20 keV.

Figure 5 shows schematically the impact of such limits for power, Q and wall load on the available region of the I–A design space. Again, the upper limit is determined by the constraint regarding the radial coil build. The left boundary is set by the minimum neutron wall load required for testing, or by the required Q (Q > 5), and the right boundary is set by the maximum fusion power or current drive power. These left and right boundaries change as the temperature is varied and, at 14 keV, the Q > 5 and P_{CD} = 100 MW curves intersect, at a value of the wall load of approximately 0.8 MW·m^{-2} and a device radius of approximately 5 m. This temperature is similar to the lower operational limit for the larger device. The major difference compared with a smaller plasma is that Q is more marginal, and thus a device with parameters of I ~ 20 MA and A ~ 3 appears to be reasonable for steady state operation as well as for ignition.

6.3. Background for machine selection

If in the design work emphasis is placed mainly on achieving ignition with the most pessimistic scaling, then ignition would have to be in the L-mode. For this, the Goldston scaling is the most restrictive scaling under consideration. This would lead to a device with R = 9.3 m, a = 2.8 m, A = 3.3 and I = 30 MA. The cost of such a device would be more than twice that for a device with R = 5.8 m, a = 2.0 m, A = 2.9 and I = 20 MA, which is used here as a reference.

If emphasis is placed primarily on steady state operation for testing in the technology phase, it is possible to choose a device which is just large enough that in current driven operation it can achieve a wall load of 1.0 MW·m^{-2} with a current drive power of <100 MW, while still maintaining sufficient volt-seconds for inductive operation. Such a device would have R = 5 m, a = 1.7 m, A = 2.9 and I = 16 MA, and a cost saving of about 18% would be possible. However, the enhancement factors required for ignition would be >2 for the Goldston, Shimomura–Odajima and T-10 scalings, implying a greater risk of non-ignition.

If emphasis is placed on reliable purely inductive long-pulse operation instead of purely steady state operation for testing, it is possible to choose a device with a large aspect ratio and enough volt-seconds for a plasma burn of several thousand seconds. The typical parameters would be R = 6.3 m, a = 1.4 m, A = 4.5, I = 12 MA, and such a device would be 13% cheaper. Concerning ignition performance, however, there is increased uncertainty, since for the Goldston and T-10 scalings the ignition performance is better but for the other scalings it is considerably worse. There is also some concern about extrapolating the present scalings to such a high aspect ratio because no experimental data are available.

The low aspect ratio region of parameter space is difficult to reach with conservative magnet design. If a more advanced magnet design is employed and the plasma current is kept at 20 MA, the device can be reduced to R = 5.2 m, a = 2.1 m and A = 2.5. Such a device would be cheaper, were it not that the advanced magnet design could lead to increased cost. The enhancement factors required for ignition and steady state operation are similar to those of the reference device for the Shimomura–Odajima and Rebut–Lallia scalings but somewhat worse for the other scalings.

Before selecting the device parameters, a more detailed study of the I–A space with I = 20 MA and A = 3, which appears to be the most reasonable I–A space, was performed in an attempt to reduce the size of the device without seriously affecting its performance. It was found that reasonable ignition performance can still be achieved for a device with I = 17.5 MA and R = 5.5 m, with a cost saving of about 10%. With extended plasma operation modes (see later), the ignition performance of such a device is improved. Furthermore, the device would have a reasonable steady state operation space, with a comfortable Q-value of about 10.

6.4. Design choice and characteristics

For the ignition goal in the physics phase, a major concern is the uncertainty regarding energy confinement. It seems to be necessary to choose the highest plasma current and the largest size to obtain the best performance. For the technology phase, major concerns are to achieve reasonable values for the wall load, pulse length, current drive power, Q, total neutron fluence and tritium consumption. Considering the different goals in the physics and technology phases, different sets of plasma parameters have been chosen for each phase. Thus, changes of the in-vessel components may be required between the phases.

In the physics phase the baseline plasma is chosen to have a larger plasma size and a higher plasma current than in the technology phase, i.e. R = 5.8 m, a = 2.2 m and I = 22 MA. In the technology phase the baseline plasma is chosen to have R = 5.5 m, a = 1.8 m and I = 18 MA. These parameters should satisfy the demands on confinement in the two phases and alleviate the problems associated with energy removal by reducing the fusion power in the technology phase when ignition is not a necessity. The major design parameters in the two phases are shown in Table VI.

TABLE VI. TYPICAL PERFORMANCE PARAMETERS OF ITER

	Physics phase	Technology phase	
R (m)	5.8	5.5	5.5
a (m)	2.2	1.8	1.8
A	2.6	3.1	3.1
k_{95}	1.9	2.0	2.0
q_{psi} (95%)	3.2	3.1	3.1
g	1.9	2.0	3.0
I_p (MA)	22.0	18.0	18.0
B (T)	5.0	5.3	5.3
P_{fus} (MW)	1000	880	820
P_n ($MW \cdot m^{-3}$)	1.0	1.0	0.9
P_{CD} (MW)	—	—	90
Q	—	—	9.1
n_e (10^{20} m^{-3})	1.1	1.3	0.8
T (keV)	10	10	18
τ_E (s)	3.1	2.7	1.8
H_{SO}/H_{RL}	1.7/0.8	2.0/1.0	1.6/0.7
H_G/H_{T10}	1.8/1.7	1.8/1.8	1.5/1.4
τ_{burn} (s)	>200	>600	Steady state

The physics baseline plasma can be operated fully inductively and plasma burn can be maintained for several hundred seconds. For the technology baseline plasma of 18 MA, the inductive volt-second consumption is reduced, but the volt-second contribution of the equilibrium coil may decrease, depending on the design of the PF coil system. Overall, it is estimated that an extra 30 V·s will be available in the technology phase for additional plasma burn, if necessary, without non-inductive current ramp-up assist.

7. PERFORMANCE FLEXIBILITY OF ITER

Performance flexibility is essential for enhancing the capability of the device, providing possibilities for the introduction of advanced features and new capabilities, optimizing the plasma performance and accommodating the divergences in physics predictions. Therefore, in each phase the device should be optimized and designed so that it has not only the capability of standard operation but also the capability of extended operation and flexibility especially with regard to plasma size, current and operation scenario.

7.1. Phased operational scenario

The operation of ITER will be carried out in three phases, a H–D phase, a D–T physics phase and a technology phase.

When the complete heating and current drive system is installed, it will be possible to perform extensive plasma experiments, especially on confinement, steady state operation with non-inductive current drive, heating, operational limits, disruptions, as well as power and particle balance and exhaust in a wide range of operational parameters, with favourable capabilities regarding diagnostics, space and maintainability in H–D plasmas. The capability of the device can be tested in particular regarding the confinement, the power and particle exhaust conditions and the performance of the plasma facing components, and a favourable operational range for the subsequent D–T phase can be developed.

When the tritium system is installed, it is possible to study the pumping system and the shielding, ignition with purely inductive drive, steady state operation with non-inductive current drive, and various other operating modes. In this D–T physics phase, relatively few pulses of ignition and steady state operation with intense neutron flux will be required and the total neutron fluence may be less than 0.01 $MW \cdot a \cdot m^{-2}$. Thus, a thin shield can be employed and full advantage can be taken of the maximum capabilities of the device regarding plasma size, plasma current and flexibility.

In the technology phase, breeding blankets will be installed and, consequently, the plasma size and current will be reduced. In this phase, reliable operation in a steady state and/or in a long pulse mode should be possible. Also, the engineering objectives will be studied and testing will be performed. The total neutron fluence will probably be 1–3 $MW \cdot a \cdot m^{-2}$ with about 1 $MW \cdot m^{-2}$ flux. Plasma operation as developed in the physics phase should still be reliable in the technology phase when the plasma current is reduced and the plasma facing components are optimized.

To have optimum conditions for each phase, the device should be flexible, especially regarding the plasma size and plasma control. The design of in-vessel components should not be frozen at an early phase and thus it should be possible to replace them, and the performance of the PF system should be flexible. Because of the uncertainty with respect to the choice of the optimum method for current drive, the design of the access ports should be flexible, including the possibility of tangential ports. The plasma facing components and the divertor throats will be optimized for particle and heat control and with respect to plasma disruption; uncertainties in this regard will be clarified in the early phase of operation. Therefore, the plasma facing components should be easily replaceable. Thus it is possible to develop the design of the in-vessel components and the plasma operation in an early phase and to optimize the design for the technology phase.

7.2. Extended operation

Extended operation with a larger plasma size and a higher plasma current is especially important to enhance the confinement capability of ITER in the physics phase. This was studied with the reference technology phase parameters of ITER, i.e. an inductive plasma current capability of 18 MA with total volt-seconds of 250, a major radius of 5.5 m, a minor radius of 1.8 m, an elongation of 2 and a safety factor of 3 at 95% of the flux surface.

The in-vessel components can be replaced to change the plasma size, and it is reasonable to have a larger plasma size and a higher plasma current in the physics phase, with a minor radius of about 2.2 m, an aspect ratio of about 2.6, an elongation of about 1.9 and a plasma current of 22 MA, in the same vacuum vessel with thinner shielding blankets. A TF ripple of 1.5% is reasonable since a larger TF ripple is acceptable because of the lower aspect ratio, and the higher heat flux due to the larger TF ripple is acceptable because of the small number of operations at full ignition and steady state. The reduced thickness of the inboard shield leads to a higher neutron flux and heat load to the magnet. Although only a part of the inboard shield near the midplane is reduced, the increased heat load is serious and means of reducing the risks should be carefully studied, including the provision of a thicker shield. Neutron damage is not serious because of the very low fluence in the physics phase. Therefore, extended operation can be normally employed in the physics phase.

By making additional assumptions, such as a smaller distance between the separatrix and the plasma facing components, non-inductive current ramp-up assistance and/or bootstrap current contribution, and a small number of operation cycles, a higher plasma current and a larger plasma size can be accepted. Typical parameters are a plasma current of 28 MA, a minor radius of 2.25 m, an elongation of 2, a safety factor of 2.7 and a saving of 50 $V \cdot s$ with the non-inductive method. In this example, only a 25% increase in coil current and no significant increase in PF energy are required because of the large plasma size and the small aspect ratio. Some of the PF coils must have a larger cross-section (a few tens of a per cent) than the coils required for basic operation. Hence, operation with a larger plasma size and a higher plasma current seems possible with little additional effort and cost. The possibility of operation at a higher coil current by limiting the number of pulses should also be investigated.

7.3. Flexibility of plasma operation

7.3.1. Plasma parameters

A wide range of plasma configurations will have to be studied, in order to optimize the plasma performance, the power and particle exhaust conditions and the layout of the plasma facing components. Such studies should include the following: use of double-null, single-null and semi-double-null divertors, distance between X-point and divertor plates, distance between separatrix surface and first wall, elongation, triangularity, aspect ratio, beta, safety factor and profiles. The possibility of a radiation cooling mode, which is desirable in a power reactor, with high density and high beta, should also be studied. By careful design of the PF system and the plasma facing components, a reasonably extended operation range in addition to the standard operation will be possible.

7.3.2. Plasma operation scenario

ITER should be designed not only for ignited operation but also for steady state operation. Other useful operating modes should also be studied.

For steady state operation, important design constraints are the neutron wall load required for nuclear tests (the minimum value being 0.8 $MW \cdot m^{-2}$), the Q-value ($Q = P_{fusion}/P_{current\ drive}$) which should be about 5 or higher, and the maximum allowable current drive power, which is assumed to be about 100 MW. Furthermore, the heat load on the divertor plates and the first wall must not be too high; the total fusion power must be less than about 1 GW so that the sum of the alpha particle and current drive power does not exceed 300 MW.

The results are sensitive to the plasma profile, the bootstrap current distribution and the current drive scheme. The following rather optimistic assumptions are made: $T = 1 - r^2$, $n = (1 - r^2)^{0.5}$, $I_{bootstrap} = 0.3 I_{plasma}$, maximum Troyon factor = 3.0, and current drive with 1 MeV neutral beams. Therefore, the following assumptions are probably optimistic.

The required enhancement factor of the energy confinement for steady state operation with additional high heating power is similar to that for ignited operation because the plasma parameters are optimized not only for confinement but also for non-inductive current drive. The design space is limited mainly by the minimum wall load and the maximum allowable current drive power because the fusion power is low in a high temperature plasma with favourable current drive efficiency or the current drive power is high in low temperature plasma with favourable fusion power. The minimum plasma major radius is about 5 m with a plasma current of about 15 MA and an aspect ratio of about 3. In ignited operation, the required enhancement factor of the energy confinement time is >2 for both the Shimomura–Odajima scaling and the Goldston scaling. In steady state operation, the enhancement factor is 1.8 and 1.5, respectively, for the two scalings. The Q-value is about 6 and the operational space is strongly limited. Therefore, such a small device is not suitable for ignition or for steady state operation at high Q.

Another typical design point is a high aspect ratio device, for example with a major radius of 6.3 m, an aspect ratio of 4.5 and a plasma current of 12 MA. In this case, the required enhancement factor for Goldston scaling is about 1.3 for both steady state and ignited operation, and a very long pulse operation (3000 s) is possible with purely inductive current drive because of the large central solenoid coils. However, in this case the size of the device is very large because the permissible TF ripple is very low at this high aspect ratio. The required enhancement factor of the energy confinement time for Shimomura–Odajima scaling is >2 even in steady state operation, and the Q-value is about 6.

In the reference ITER device, the steady state operation space is quite satisfactory. The Q-value is in the range of 8–10 with an enhancement factor of the energy confinement time of about 1.5 for Shimomura–Odajima scaling. Therefore, the reference ITER device is suitable for both ignition studies and steady state operation.

The combination of inductive and non-inductive current drive methods makes it possible to study various useful operating scenarios, e.g. a very long pulse mode with a high density plasma, control of the current profile, quasi-steady-state operation and high current operation. Also, this type of flexible operating scenario can be studied in experiments performed in ITER.

8. TECHNICAL ISSUES

The feasibility issues critical to the conceptual design have been included in the R&D efforts for ITER. Some of these issues have received special attention during the definition phase, particularly those for which combined and co-ordinated efforts of different specialists are needed.

Power and particle exhaust in ITER is an example of such extremely demanding physics and engineering tasks. On the basis of the present data from large, high power experiments such as JET, DIII-D, TFTR and JT-60, and from analyses based on the best present understanding of the appropriate physics and engineering issues, it has been found that the use of a poloidal divertor represents the most credible concept for handling the large heat loads and providing adequate helium removal. Low-Z materials such as graphite appear to be the most suitable materials for the divertor plate in the physics phase (with a total operation of 10^4–10^5 s); however, erosion by sputtering and disruption will probably pose serious problems. The gross erosion rate of these materials in the technology phase (when 10^7–10^8 s of operation are needed) is expected to be too large to ensure a reasonable lifetime for the divertor plates. The intended solution for this problem, namely to operate the poloidal divertor in a high recycling regime (for which it would be appropriate to use high-Z material for the divertor plate), is now questionable because of the expected high fusion power, the narrow scrape-off layer, and the additional heating power and the lower density required for optimum operation with current drive.

The present performance prediction for the ITER divertor is based mainly on computational models, since no existing experiment has provided conditions sufficiently close to those of ITER. The models indicate that the plasma temperature at the plate will probably be too high to reduce the sputtering, but small changes in the parameters or in the physics assumptions for the models could also lead to plasma conditions permitting the use of high-Z materials or to sheath potentials above the sputtering thresholds for the best available materials. The physics used in the models has to be improved and compared with data from present divertor experiments, which puts more emphasis on measurements of plasma parameters in divertors and on data analysis with models. High-Z divertor plates should also be tested in operating tokamaks with divertors. The physics and the numerical algorithms used for the models must be improved. Thus, it is important to improve the ability of predicting the plasma conditions in the divertor.

In addition to the problems of sputtering and proper plasma conditioning in the divertor, there are other technical problems which must be considered and resolved:

- Erosion caused by plasma disruption (preliminary estimates indicate that the divertor plate lifetime is limited to about 100 disruptions);
- Quantitative prediction of erosion/redeposition;
- Adequate helium exhaust rate;
- Recycling enhancement at the divertor by changing its configuration (e.g. increasing the baffling);
- Detailed analyses, taking into account the combined effects of erosion/redeposition, thermomechanical response of the structure and neutron irradiation effects.

The peak static heat loads on the divertor plates in ITER are expected to be as high as 10 $\mathrm{MW \cdot m^{-2}}$. Such heat loads should be marginally acceptable, considering the limits on the surface temperatures of the graphite tiles and on the lifetimes determined by fatigue due to thermal stress in the heat sink material (e.g. copper and molybdenum alloys). Higher heat loads (due to plasma power excursions, vertical plasma movement, misalignment of tiles, etc.) would lead to higher surface temperatures and increased erosion due to radiation induced sublimation as well as to shorter lifetimes determined by fatigue. For these reasons, consideration is being given to sweeping the plasma separatrix and thus sweeping the regions of peak heat loads and erosion across the divertor plates. Preliminary analysis suggests that application of such sweeping techniques will permit higher heat loads (up to 15 $\mathrm{MW \cdot m^{-2}}$ and possibly even 20 $\mathrm{MW \cdot m^{-2}}$). This would also distribute more evenly the erosion and increase the lifetime of the plate by a factor of two to three.

If the plasma temperature at the divertor plate cannot be made low enough to allow the use of high-Z divertor plates, then other approaches to the problem of heat and particle removal have to be studied. These approaches include schemes with non-conventional divertor plates (e.g. liquid metal films, liquid metal droplets, solid balls), alternative pumping schemes, active techniques for improving the divertor plasma parameters, such as electrical biasing or ergodization of the edge layer, different divertor configurations and the use of new materials. Each of these schemes needs to be more carefully assessed during the design phase.

While the power and particle problem for ITER is very challenging, the problems in connection with reactor scale experiments with substantially increased power loads are much more serious. Solution of these problems is a necessary step on the way to a tokamak reactor.

9. OPERATIONS AND EXPERIMENTAL TESTING PROGRAMME

The experimental plan of ITER is divided into two phases — the physics phase and the technology phase. These phases are distinctly different from each other in the experimental requirements and operating modes. Nonetheless, there will be a significant amount of engineering testing in the physics phase, particularly with regard to operation of the basic device, as well as a small amount of nuclear component testing. Similarly, some limited physics testing can be expected to continue in the technology phase. Thus, there is no sharply defined point in time that marks the transition from physics testing to technology testing. The physics goals and many significant technology goals can be achieved in ten years of operation.

9.1. Physics phase

Before the actual start of operation in the physics phase, there will be a significant amount of commissioning tests of the tokamak components and of the supporting systems, such as TF and PF magnets, magnet power supplies, magnet protection systems, cryogenic systems, vacuum and fuelling systems, plasma heating systems, maintenance equipment, and reactor control and instrumentation systems.

The various operating phases of ITER are summarized in Fig. 6. In the first three years of the physics phase, there will be H and D–^{3}He plasmas, with either zero or low neutron activation levels in the structure of the device. This permits hands-on maintenance and verification tests of remote maintenance procedures.

In the first two years of the physics phase, work will concentrate on establishing the operational conditions in H plasmas through a sequence of Ohmic discharges; this will be followed by operation at the full capability of the magnetic field and plasma heating systems. Approximately 6000 shots are expected for this period. The third year (about 1500 shots) will be devoted to optimizing the plasma performance, in preparation for D–T operation in the physics phase.

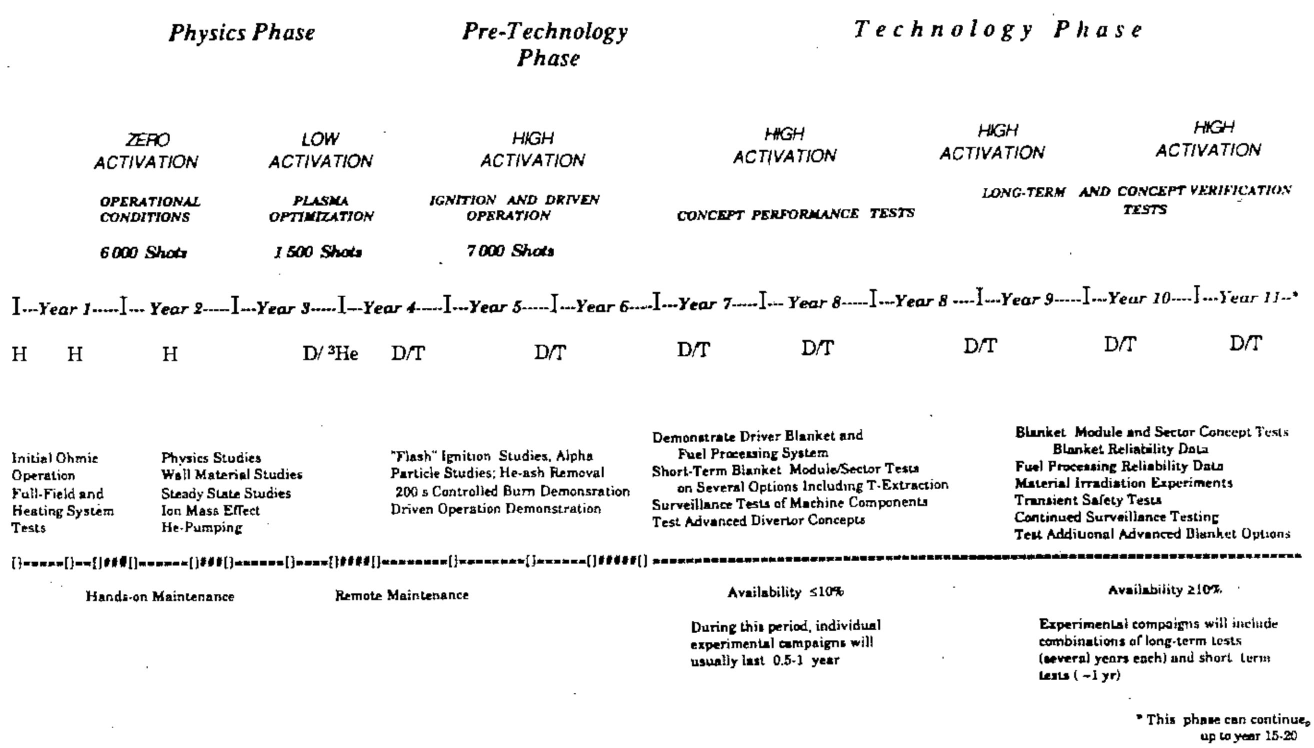

FIG. 6. Possible operating phases for ITER.

During this initial three-year period, the integration of the plasma heating systems into the tokamak system is demonstrated, the functioning of the plasma control systems under plasma transients is checked and the performance of the high heat flux components (divertor, first wall) is tested. In addition, the response of in-vessel components (vacuum vessel and shield) to plasma operations, including disruptions, is checked.

During D–T operation in the physics phase (lasting approximately three years with 7000 shots) the main aim is to achieve ignition and to study plasma operation with current drive. In this phase, the operation of the plasma tritium processing system will be verified. If the tritium producing ('driver') blanket can be installed, preliminary tests of its performance will be carried out. Also some initial blanket tests (neutronics, liquid metal flow, etc.) will be conducted. This is referred to as the 'pre-technology phase' in the last part of the physics phase.

Throughout the physics phase the neutron fluence will be quite low, ≤ 0.05 MW·a·m^{-2}, resulting in damage levels to the first wall of ≤ 0.5 dpa.

9.2. Technology phase

The technology phase will have several stages. In the first stage, a period of about three years will be devoted to concept verification tests of various power reactor blanket options; each of these tests will usually last one-half to one year. It will be necessary to demonstrate the operation of the tritium producing blanket. Also expected are tests of advanced divertor and first wall concepts as well as surveillance testing of the tokamak in-vessel components to check their performance. The overall availability in this period is expected to remain at modest levels ($\sim$10%). Here, availability means the actual operation at significant flux levels during one year (in per cent), not counting the time for experimental modifications and maintenance. Success in this stage would be a major milestone in the development of fusion energy because operation of all major reactor components in an integrated reactor system would have been demonstrated.

After successful completion of the first operation stage, an extended period of technology testing (for example 7 years) could follow, the main tasks of which would be concept verification and long term tests of the blanket options selected earlier. This period of testing should prove particularly cost effective and thus attractive if steady state operation at high availability and/or high neutron flux would already be possible. For nuclear testing, it has been shown in a number of studies that a fluence of at least 1 MW·a·m^{-2} is required and that higher values ($\sim$3 MW·a·m^{-2}) would be very useful. Therefore, ITER is being designed to withstand 3 MW·a·m^{-2}; fluences of that order are theoretically possible if higher fluxes ($\geq$1 MW·m^{-2}) and/or higher availabilities (25–30%) can be achieved. The data obtained in this period should be sufficient to select with confidence a blanket concept for devices to come after ITER. It is also expected that useful information on materials damage will be provided as well as reliability data on the blanket, plasma facing components and tritium processing systems. In addition, some safety related transient tests will be carried out; tests of a more severe kind will be done near the end of the operating period.

Regarding nuclear component testing, the average neutron wall load should be no less than 0.8 MW·m^{-2}. Steady state plasma operation is highly desirable. If a pulsed mode of operation is necessary, then it is desirable to have a burn time of 1–3 hours (minimum of 600 s). The dwell time should not exceed 70–200 s. It is necessary to have continuous operation (in steady state or pulsed mode) of 1 to 2 weeks at very high availability ($\sim$100%). The space required for blanket tests includes eight modules (1 m $\times$ 2 m $\times$ 0.5 m) and two outboard poloidal sectors, assuming that most of the different tests to be performed by the four parties can be combined.

The specific details and the exact operating schedule will depend on several key decisions, to be made later during the design and physics phases. The first major decision, to be made at the beginning of detailed engineering design, is whether to proceed with the baseline design or to go on to extended design, allowing for higher physics performance. At the end of the hydrogen part of the physics phase (3 years), a decision has to be made on whether to use the baseline concept or an extended concept in the D–T physics period. At the start of the technology phase, it has to be decided which operating mode should be used in the first years, i.e. pulsed operation or steady state operation. At present, it is assumed that the baseline design (permitting more room for shielding) and steady state operation will be utilized for the long term testing period of the technology phase. During the design phase, more thorough cost/benefit analyses of the testing programme will be made.

10. SAFETY AND ENVIRONMENT

One of the important ITER objectives is to demonstrate the potential of safe and environmentally acceptable operation of a power producing fusion reactor. In the light of this, it has been decided to adopt passive safety as an ultimate design goal for ITER. This implies recognizing the unique characteristics of the fusion process and making this knowledge an integral part of the design work. It is also necessary to define the radiation protection targets and site requirements, as well as the safety and environmental analyses to be performed. Much of this work can only be done at a later stage, after the design of the device has been developed in some detail. In the definition phase, a basic philosophy for the safety of the device has been established.

Design targets have been proposed as a first safety and environmental guideline and the status of the critical issues has been reviewed. These issues are: calculational models for environmental impact, accident analysis, waste management and handling of activated components.

10.1. Proposals of design targets

The international criteria for radioactivity differ significantly, especially in the area of waste management. Thus, exact definitions of the legal requirements for safety and environmental protection cannot be formulated in the conceptual design phase of ITER. The ALARA (As Low As Reasonably Achievable) principle has been generally adopted. Design targets have been proposed for normal effluents, exposure of workers and maximum accidental releases, considering the range of national criteria, previous design studies and the ALARA principle. The following proposals have been made.

Normal effluents (public exposure)

Dose to the maximum exposed individual

Atmospheric:	50 Sv·a^{-1} (5 mrem·a^{-1})
Liquid:	50 Sv·a^{-1} (5 mrem·a^{-1})
Total:	100 Sv·a^{-1} (10 mrem·a^{-1})

Exposure of workers

General dose:	10 mSv·a^{-1} (1 rem·a^{-1})
Dose to radiation workers:	25 Sv·a^{-1} (2.5 mrem·h^{-1})
Dose to non-radiation workers:	5 Sv·h^{-1} (0.5 mrem·h^{-1})

Maximum accidents (public exposure)

Dose to the maximum exposed individual
100 mSv (10 rem).

Although generally the approach of probabilistic risk assessment has not been legalized in the participating countries, there is a trend to adopt this approach as an element of accident assessment. Most countries have considered probabilistic risk assessment in some form in their national design studies. Preliminary discussions have led to the proposal of initial values for limiting the consequences as a function of off-normal event frequency.

Although the concept of passive safety is not accounted for in the legal requirements, the opinion prevails that exploiting passive features ('passive safety') for the protection of the public and the workers is an appropriate goal. It has been recommended to the designers that the problems connected with possible accidents should be solved by passive mechanisms and that feasibility studies should be performed.

10.2. Evaluation of radioactivity doses

Calculational models for evaluating the environmental impact (doses) of HTO are fairly well established, but for HT there are still large uncertainties since the behaviour of HT is still being studied. The global conversion factor, CF, for released HT to HT converted to HTO by soil contact, is estimated to be about 10%. This has to be confirmed or a more exact value has to be found in the near future on the basis of calculations because of the strong impact on dose calculations. An improved model should take into account the deposition of HT on the soil and the re-emission of HTO from the soil, utilizing recent experimental data, as well as the uptake of tritium by the vegetation and incorporation as organically bound tritium.

Recent experiments indicate that the radiation quality factor (relative biological effectiveness) for tritium beta radiation is higher than 1.0. The actual value of the quality factor to be used in dose calculations should be further investigated. For the ITER dose commitment calculations, this factor is tentatively proposed to be 2.

10.3. Accident analyses

Off-normal events, which could lead to exposure of the public or of workers, are being analysed, keeping in mind the goal of passive safety. A list of significant events has been set up. In anticipated order of risk, these events are:

- Major rupture of the divertor or the first wall cooling pipes inside the vacuum vessel (LOCA type);
- Major failure of vacuum vessel elements, vacuum ducts and pumps, heating and fuelling devices (LOCA type);
- Total LOCA (major cooling system failure outside the vacuum vessel);
- Major disruption;
- Major tritium system failure;
- Major rupture of blanket cooling pipes (LOCA type);
- Magnet current lead break or major quench;
- Total LOFA of the whole device.

These events and their consequences should be taken into account in the design work, i.e. passive mechanisms should be used to the largest extent possible to cope with them. As soon as the design has been sufficiently developed, these analyses should be performed, at least in a deterministic way. Additionally, probabilistic analyses should be made if they are of use in concept selection and if adequate failure data are available.

10.4. Waste management

Because of differences in the national criteria, the area of waste management for ITER remains undefined. Wastes could include high level wastes (HLW) or perhaps only low level wastes (LLW), depending on the regulations of the country where the reactor is located.

If commercially available steels (austenitic and martensitic) are used as structural materials, there will be wastes from the first wall, divertor and blanket which may or may not qualify for shallow land burial (SLB) according to United States regulation 10 CFR 61. At least part of these components will have to be considered as medium level wastes or even high level wastes, depending upon national regulations.

The masses and dimensions of the first wall and blanket segments, for which intermediate storage will be necessary, are large and also large hot cells for detritiation and treatment (cutting, compacting, conditioning) will be required.

In some countries the final tritium levels in activated components (e.g. 1 Ci per tonne after detritiation) would necessitate their geological disposal according to present regulations.

Further assessments are needed to define a waste management strategy for ITER and on this basis to assist in the choice of design solutions and materials.

10.5. Handling of activated components

To minimize radioactivity release from reactor components such as the first wall, blanket and divertor, it is desirable to use baking before opening the vacuum chamber and to keep the temperatures low during maintenance.

The presence of activated erosion products inside the vacuum vessel (mainly dust from the first wall and the divertor surface materials) has to be considered in the development of maintenance strategies. The use of intervention containment with inert gas or vacuum during maintenance operations can reduce or avoid the contamination of the reactor hall by activated dust and tritium.

The maintenance equipment must be qualified for earthquakes according to the legal requirements and the physical specifications of the selected site.

11. RELIABILITY AND AVAILABILITY

It is the task of the reliability and availability programme for ITER to promote the use of highly reliable and maintainable components and systems and thereby to further strengthen and ensure the desired availability of the device.

The actually achieved availability of the present generation of large fusion devices, such as TFTR, T-10, JT-60 and JET, is far below the availability aimed at in the next-generation devices such as ITER. The present devices have the understanding of the physics as their primary goal, while ITER will also have technology oriented goals. The technology operations will for the first time involve an activated and nuclear environment for both operation and maintenance. Therefore, the need for reliable components and systems is apparent.

The availability goals as established in the ITER Terms of Reference relate to three specific targets. First, the overall availability of ITER in the technology phase must be at least 10%. Second, in the period of its highest reliability, ITER should reach availability levels as high as 25%. Third, ITER will be required to operate at very high availability (continuous operation) for periods lasting one to two weeks.

Generally, it can be said that these goals are ambitious. The first and secoud goals are interpreted to mean that the respective percentages include all planned and unplanned downtimes of the device. It is clear that, to achieve these goals, it will be necessary to improve the reliability of the components and also to develop approaches and methods by which the planned downtimes can be limited.

The reliability and availability activities will be performed mainly in the technology phase of ITER. It is assumed that in the physics phase considerable operating experience and understanding will be derived and that means will be found to preclude plasma oriented failures causing plant shutdown. The physics phase will also be used to determine how to avoid or strictly limit major plasma disruptions and how to handle plasma–wall interactions in a manner that ensures the proper functioning of all components. It is recognized that even with such knowledge, it may be necessary to provide for periodic (and perhaps frequent) replacement of some components. In component design, care is taken to avoid wear-out failures during operation. Those components which have high failure rates or whose failure rates are uncertain will be replaced regularly or will be periodically inspected. The operating and testing programme foresees refurbishment periods during which the necessary replacements can be made.

The reliability and availability programme will include studies of the failure mode, effects, criticality and analysis (FMECA studies) for key components considered to contribute most to the overall unavailability, unreliability or uncertainty of the plant. Pertinent definitions of the reliability and availability nomenclature have been developed, reliability related general design guidelines for the component designers have been compiled and a very preliminary plant evaluation has been performed. This initial evaluation has utilized data for the mean system failure rate and the mean downtimes after failure obtained from earlier studies of national designs. Obviously, new data must be derived for ITER in the course of the design work. However, the preliminary assessment provides an early indication of the key components for which reliability studies appear warranted.

The results of the plant evaluation have been used to develop a priority ranking for the key components contributing most to the overall unavailability of the plant. These components are: the neutral beam system, PF magnet system, TF magnet system, RF system, vacuum vessel, fuel handling equipment, plasma vacuum pumping system, heat transport system, first wall, limiter, blanket and shield.

In the design phase, FMECA studies will be performed for these key components. In addition, the early reliability and availability assessments of all other components will be reviewed and updated as necessary. Finally, the results of the component assessments will be used to update the overall plant evaluation.

12. COST

The cost of the device is an essential input for machine selection. However, since design specifications, construction procedures and plans for the operation phase are not yet available, no meaningful estimate of the total cost of the ITER device or project can be made at this stage. Rather, for the selection of the device, the existing cost models within the parametric analysis codes have been used. No attempt has been made to develop one single cost model for the definition phase studies. In the cases where the cost has been used as a guideline for device selection, no major conflict in the tendency has been observed between the various models.

However, this situation is unsatisfactory. Thus, in this phase, considerable effort has been put into laying the groundwork for developing a cost estimate of the ITER project. A cost-centre breakdown has been prepared. This defines and allocates all the cost components of the design, identifying those persons in the existing team which are responsible for providing cost information and defining as carefully as possible the boundaries between the different cost centres to ensure that there are no overlaps. At this stage, such a breakdown can only be preliminary and will be updated in the course of the design work.

Any future cost estimate will also depend on the organization of this multinational construction project with regard to sharing out the work between the participants while adequately protecting the intellectual and commercial rights, and with regard to the share of responsibility between the central co-ordinating team and the industry that provides the hardware. These issues will be addressed in the design phase.

13. RESEARCH AND DEVELOPMENT

Extensive fusion R&D activities are being carried out in the national programmes of various countries. Obviously, a great deal of essential information will come from these R&D activities. It is natural to assume that this information will be made available for the ITER design activities either directly or indirectly. The Terms of Reference indicate that R&D should focus on issues critical to a conceptual design that meets the ITER objectives. Regarding the schedule, it is assumed that the main milestones of R&D will be achieved in 1990 at the latest and that the essential R&D supporting the ITER objectives and design should be completed by 1993.

TABLE VII. ITER RELATED PHYSICS R&D TASKS

PH-1	Power and helium exhaust conditions
PH-2	Helium radial distribution in high temperature tokamak discharge
PH-3	Viability of a radiative edge
PH-4	Sweeping of the divertor target load
PH-5	Characterization of low-Z materials for plasma facing components
PH-6	Characterization of high-Z materials for plasma facing components
PH-7	Characterization of disruptions
PH-8	Disruption control
PH-9	RF plasma formation and preheating
PH-10	RF current initiation
PH-11	Scaling of volt-second consumption during inductive current ramp-up in large tokamaks
PH-12	Alpha particle losses induced hy the toroidal magnetic field ripple
PH-13	Compatibility of plasma diagnostics with ITER conditions
PH-14	Steady state operation in enhanced confinement regimes (H-mode and 'enhanced' L-mode)
PH-15	Comparison of theoretical transport models with experimental data
PH-16	Control of MHD activity
PH-17	Density limit
PH-18	Plasma performance at high elongation
PH-19	Alpha particle simulation experiments
PH-20	Electron cyclotron current drive
PH-21	Ion cyclotron current drive
PH-22	Impact of Alfvén wave instability on neutral beam current drive
PH-23	Proof of principle of fuelling by injection of field reversed compact toroids

TABLE VIII. ITER RELATED TECHNOLOGY R&D TASKS[a]

		EC	Japan	USSR	USA
Blanket					
BB1	Ceramic breeder	*	*	—	*
BB2	LiPb breeder	*	—	*	—
BB3	H_2O/Li solution breeder	*	—	—	*
BM	Beryllium	—	—	—	*
BS	Structure material	*	—	*	*
Plasma facing components					
PC1	Low-Z material	*	*	*	*
PC2	High-Z material	—	*	*	—
PC3	First wall test	*	*	—	*
PC4	Divertor test	*	*	*	*
Magnet					
MT	Toroidal coil	*	*	*	—
MP	Poloidal coil	*	*	—	*
MI	Insulation material	*	*	*	*
MS	Structural material	*	—	*	—
MA	Radiation tolerant magnet	—	—	—	*
MC	Cryogenics	—	*	*	—
Fuel cycle					
FC1	Fuelling	*	*	*	*
FC2	Pumping	*	*	—	—
FC3	Fuel purification	*	—	—	—
Heating/current drive					
HD1	Source: EC: 150–250 GHz	*	—	*	*
HD2	Source: LH: 6–8 GHz	*	—	—	—
HD3	Source: NB: negative ions	*	*	*	*
Maintenance					
RH1	Component qualification	*	—	—	—
RH2	In-vessel operation demonstration	*	*	—	—

[a] Asterisks indicate that proposals from the Parties have been received.

The specific physics R&D needs for ITER are listed in Table VII. The tasks are divided into two groups: the first group (PH-1 to PH-12) covers a number of crucial design related physics R&D issues on which additional information is urgently needed to permit, in 1990, the confirmation of the technical working assumptions on which the ITER concept is based. The second group (PH-13 to PH-22) covers general issues of plasma performance (energy confinement, operational limit, burn control, long pulse operation) which are crucial for reaching the ITER objectives but which are not sufficiently covered by the ongoing fusion programmes. The contributions from the Parties to each task are still under discussion.

The specific technology R&D needs for ITER are listed in Table VIII. They cover issues which are considered essential in proving that the main components of the machine can achieve the required performance. Major emphasis is put on achieving the main milestones in 1990, but it should be understood that most of the tasks will be completed by 1993. The tasks are divided into six areas: blanket, plasma facing components, magnets, fuel cycle, heating and current drive, and maintenance. For each task, detailed proposals from the Parties have been received; an assessment of these proposals indicates that the balance of effort between the technical areas is generally appropriate and corresponds to the need of the programme.

The budget of 120 million US dollars until the end of 1990, envisaged in the ITER Terms of Reference, was clearly not sufficient to finance both the technology tasks and the physics tasks. Since most of the physics tasks can be executed in the ongoing programmes through a change of emphasis, it has been decided to include in the ITER R&D budget only the technology tasks specific to the design; this choice does not imply any priority status of either the physics tasks or the technology tasks.

TABLE IX. MAIN PARAMETERS FOR ITER

BASIC DESIGN PARAMETERS

	Physics phase	Technology phase
Major radius, R (m)	5.8	5.5
Minor radius, a (m)	2.2	1.8
Aspect ratio	2.6	3.1
Elongation, k (95%)	1.9	2.0
Triangularity, δ (95%)	0.4	0.4
Safety factor, q_{psi} (95%)	3.2	3.1
Plasma current, I_p (MA)	22.0	18.0
Toroidal field on axis (T)	5.0	5.3
Plasma volume (m^3)	1040	700
Total fluence ($MW \cdot a \cdot m^{-2}$)	<0.01	3

OPERATION MODE

	Physics phase	Technology phase	
	Inductive	Hybrid	CD
Burn time (s)	>200	~ 600	Continuous
Energy multiplication factor, Q	>30	~ 10	9.1
Troyon coefficient, g	1.9	2.0	3.0
Effective charge, Z_{eff}	1.8	1.8	1.8
Wall loading ($MW \cdot m^{-2}$)	1.0	1.0	0.9
Fluence ($MW \cdot a \cdot m^{-2}$)	—	1.0	1.0
Electron density (10^{20} m^{-3})	1.1	1.3	0.8
Average temperature (keV)	10	10	18
Energy confinement time, τ_E (s)	3.1	1.7	1.8
Power for CD/heating (MW)	—	88	90
Fusion power (MW)	1000	880	820

TABLE IX. (cont.)

SYSTEM PARAMETERS	
Toroidal field coils	
Number	16
Conductor	Nb_3Sn
Maximum field (T)	12
Average cable current density ($MA \cdot m^{-2}$)	40
Inner leg overall current density ($MA \cdot m^{-2}$)	13
Poloidal field system	
Total volt-seconds required (phys./tech.)	280/250
Conductor:	
inner coils	Nb_3Sn
outer coils	NbTi
Maximum field at coil (T)	
inner coils	12
outer coils	8
Breakdown electric field ($V \cdot m^{-1}$)	0.3
Technique for steady state current drive (TBD)	NB, EC, IC, LH
Total injected power (MW)	~100
Mode of impurity control	double-null/semi-double-null poloidal divertor
Mode of fuelling	gas puffing, pellets
Material/protection of first wall	austenitic SS/graphite
Surface material of divertor plate	C, W (W alloy), Mo alloy
Structural material of shield	SS
Structural material of blanket	SS
Tritium breeding ratio	~1.0
Consumption of tritium (at 25% availability)	~12 kg per year
Nominal pumping speed for He and D–T ($m^3 \cdot s^{-1}$)	500
Helium refrigeration requirement at 4 K (kW)	100

14. CONCLUSIONS

On the basis of the above considerations, it can be concluded that the physics and technology objectives of ITER outlined in the Terms of Reference can be met with a reasonable degree of certainty. The technical approach has sufficient flexibility to provide a reasonable possibility of ignited operation, driven steady state burn and appropriate nuclear technology testing in the same basic machine. Such a machine seems to be technically feasible, although some complicated technical and physical problems exist. Careful analysis of these problems will be carried out in the design phase. Emphasis will be placed on improving the reliability of the design and reducing the size and cost of the device. To reach these aims, special attention will be given to design areas in which device improvements can be expected or which present particularly difficult problems.

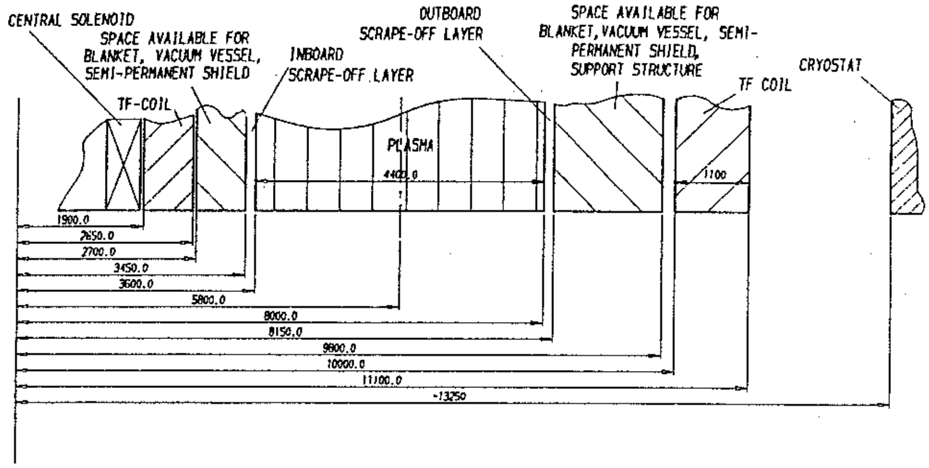

FIG. 7. Radial build of basic configuration in the physics phase.

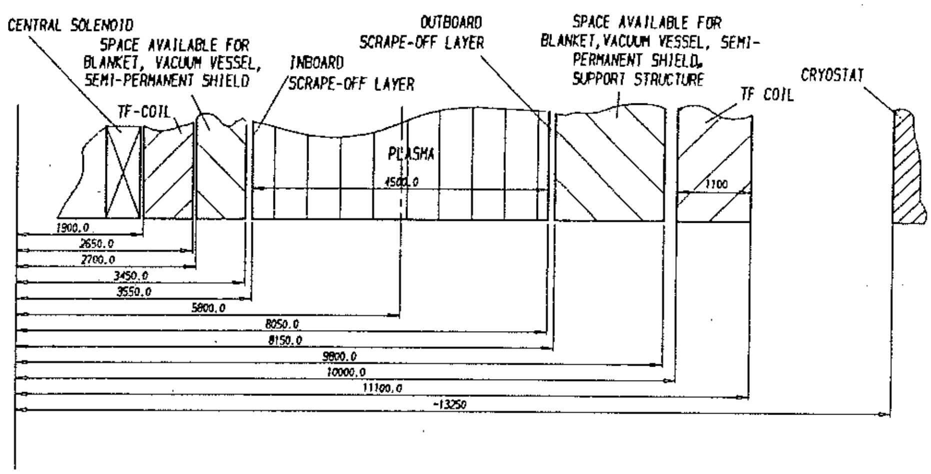

FIG. 8. Radial build of extended configuration in the physics phase.

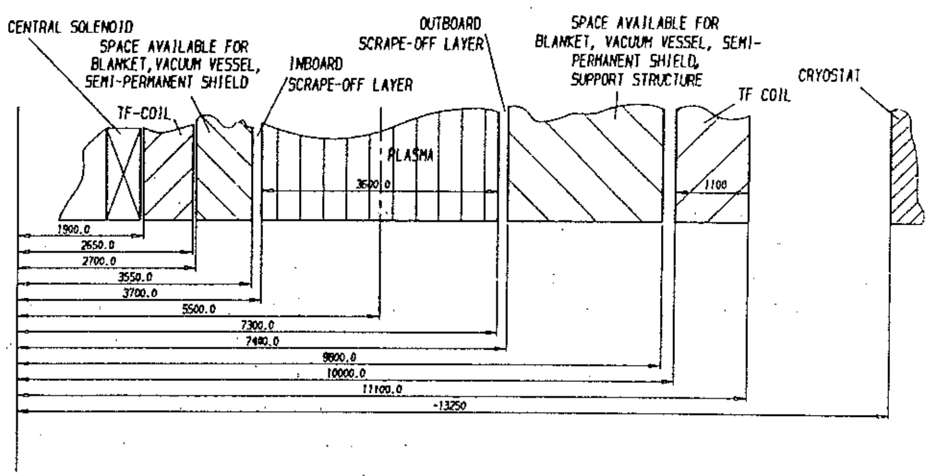

FIG. 9. Radial build of configuration in the technology phase.

Thus, a number of high leverage design options will be studied during the design phase, and a continuous selection process will be applied until a single set of technical characteristics of ITER is defined at the end of the design phase.

The ITER parameters which have been selected during the definition phase and which will serve as initial guidelines in the design phase are shown in Table IX. The radial structure of the basic configuration and the extended configuration are shown in Figs 7–9.

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Appendix

DEFINITION PHASE CONTRIBUTORS

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