ITER: CONCEPT DEFINITION*

Presented by
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Abstract

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On the basis of the results of the investigation carried out since May 1988, an ITER concept has been defined which incorporates the maximum possible flexibility and allows a variety of plasma configurations and operating scenarios. For technology experiments, with a full breeding blanket, the machine can be operated typically with a plasma of 18 MA at a major radius of 5.5 m. For the plasma physics experiments the same machine can, if required, be configured with a thinner shield and produce a plasma of 22 MA with fully inductive operation and higher currents under limited conditions. A list of important ITER specific physics and technology R&D tasks has been developed. Implementation of these tasks is now under way.

1. INTRODUCTION

The activity of the International Thermonuclear Experimental Reactor (ITER) is conducted under the auspices of the International Atomic Energy Agency jointly by Euratom, Japan, the Union of Soviet Socialist Republics and the United States of America. About ten experts from each of the four participating parties gathered in May 1988 at the technical site, the Institut für Plasmaphysik, Garching, and started work on defining the ITER concept. During the early phase of the design work an ITER concept is to be developed together with an R&D plan relevant to ITER design which will be carried out by the four parties.

In the 'Terms of Reference' for the ITER activity [1] it is stated that the objectives of ITER are to demonstrate controlled ignition and extended burn of deuterium and tritium plasma, with steady state as an ultimate objective. Guidelines are given for ITER design, i.e. confinement capability by specifying plasma current, inductive capability for a pulse length of a few hundred seconds, Q values for steady state operation and a neutron wall loading of about 1 MW/m^2. It is also stated that the ITER device should be based on the scientific and technological database that is expected to be available to support a decision, at the end of the design phase, i.e. at the end of 1990, to proceed to engineering design and construction.

On the basis of the results of the investigation made on possible configurations and performances of ITER machines, an ITER concept has been defined; this paper describes the details of that concept.

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2. PHYSICS BASIS [2]

Crucial physics areas for the ITER design are:

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

For each of these areas an assessment of the present state of the physics knowledge and of the improvement potentially possible has been performed. From these preliminary assessments some initial conclusions have been drawn and key issues identified. In the H-mode the global energy confinement time is roughly twice the L-mode value. Some enhancement of the energy confinement over 'normal' L-mode values is achievable with centrally peaked heating, divertor operation or some degree of profile control.

The analyses of fast alpha particle confinement indicate that the ripple in the plasma edge should be below 1.5% to keep the stochastic and collisional ripple losses at acceptable levels. The initial conclusion regarding the density limit in present experiments is that it appears to be due primarily to disruptions following modifications of the current profile caused by excessive radiation losses.

The Troyon scaling for the beta limit (beta = gI/aB) is a reasonable characterization of the beta limit that can be expected in ITER. It is recommended that for ignition studies and ignited operation $g \leq 2.5$ be used, that for steady state operation $g \leq 3$ be used with $q(\Psi = 95\%) \geq 3$ and that $g = 3$–3.3 with $q(\psi = 95\%) \geq 2.1$ be used for extended operation.

On this basis it has been recommended that the plasma current in ITER should be approximately 20 MA with an aspect ratio of about 3 to have a reasonable degree of assurance for ignited operation and driven steady burn operating in the H-mode or at least in an improved L-mode.

On the basis of assessments of the performance of candidate systems for heating and current drive, three major current drive options for the central plasma have been identified: high energy neutral beams ($E \sim 1$ MeV), electron cyclotron waves and ion cyclotron fast waves, all supplemented by lower hybrid current drive in the outer portions of the discharge. The power required for steady state operation is anticipated to be of the order of 100 MW. A limited amount of electron cyclotron wave power is required to assist in initiating the discharge at low induced electric fields ($\sim$0.3 V/m) and, if viable, for local profile control. It also appears credible to include the bootstrap current in the analyses of the reference scenarios.

The recommended configuration for power and particle control is a poloidal divertor. However, the development of a credible concept for a long integrated operation time will require substantially more physics and engineering work.

Disruption data from the present generation of tokamaks have been assessed and characterized. The initial conclusions are that the thermal quench time can be as short as 100 μs; the maximum current decay rate is assumed to be about 1 MA/ms. The minimization of disruption effects is a key design issue.

3. SCOPING STUDIES

The desirable device parameters for ITER were surveyed using four different overall systems codes — SUPERCOIL/SCAN, TRESCODE, TETRA and SYS — as well as more simplified analysis. ITER reference parameters needed to be chosen such that the device should be capable of achieving the joint objectives of ignition and steady state performance. Since today's typical scalings for energy confinement time depend to differing degrees on plasma current (I) and aspect ratio (A), an investigation was made to scope out the device parameter choice in I–A space, with fixed assumptions on peak fields (e.g. B_{tmax}), safety factor (q_{cyl}) and distance between plasma and TF coil on the inboard side (d_{BS}), amongst others. These are the key parameters determining machine size for given I and A and are related by the equation

$$a = [IA^2 q_{cyl}/5B_{tmax} + d_{BS}]/(A - 1)$$

Further restrictions were that the neutron wall loading be kept at 1 MW/m^2 and that the central solenoid have an adequate inductive burn time without exceeding coil stress limitations.

The variation over I–A space of the enhancement factors for τ_E needed for ignition differs according to scaling law. Goldston and T-10 scalings favour devices of high aspect ratio and high current, whereas for Rebut–Lallia and Shimomura–Odajima scalings high current is the main way to reduce these factors.

The best compromise position from the point of view of attaining the ignition goal at a reasonable cost and power level in a technically feasible device is seen to be a device with parameters near I = 20 MA, A = 3. It seems that the compromise region of design space for ignition performance also produces devices with sufficient flexibility for current driven operation. Therefore the region around I = 20 MA, A = 3 was studied in further detail to find ways of reducing device size and cost while optimizing performance for both the physics and the technology phases. In view of the different goals for the physics ($Q \simeq \infty$) and technology phases ($Q \geq 5$), a different set of plasma parameters was chosen for each phase. The change between the phases might require changing in-vessel components. These parameters were chosen to symmetrize the demands on confinement for the two phases and to alleviate the problems associated with energy removal by reducing fusion power in the technology phase when ignition would not be a necessity.

4. ITER CONCEPT

On the basis of the results of the assessment of the relevant scientific and technical data and of the investigation of possible configurations and performances of ITER machines, an ITER concept has been defined prudently as a machine which has the capacity to operate with a variety of plasmas in order to meet both the technology and the physics objectives assigned to ITER.

For technology experiments and tests the machine can be operated typically for a plasma of 18 MA at a major radius of 5.5 m with the full tritium breeding blanket installed. For the plasma physics experiments the same machine, but with a thinner shield blanket instead of the breeding blanket, can produce a plasma of 22 MA by fully inductive operation or a larger plasma current under some conditions.

The layout and major parameters of the basic ITER device [3, 4] are shown in Fig. 1 and Table I. The toroidally and poloidally segmented in-vessel components (tritium breeding blanket, first wall and divertor) are contained and supported within a combined vacuum vessel and outer shield. There are large penetrations at the top of the vessel for removal of major in-vessel components, with subsidiary access or additional heating ports on, above and below the equator.

The 16 TF coils use $(NbTi)_3Sn$ type superconductor at 4–5 K. The centring forces are taken by a vault structure formed from the noses of the coil cases. The superconductor uses forced flow cooled conductors with glass fibre reinforced epoxy resin insulation and bonding. The PF coils, apart from the vertical stability control coils which are normal conducting and situated within the vacuum vessel, are superconducting and lie outside the TF coils. The straight solenoid is self-supporting and contains three up–down symmetric coil sets. There is an upper and a lower 'divertor' coil and two pairs of outer equilibrium coils.

Maintenance of all basic device components is designed to be fully remote. Provisions for hands-on maintenance will be made wherever possible. The main biological shielding for the machine lies outside the PF coils, with the vacuum vessel shielding providing the minimum shield necessary for the coils.

Low temperature, low pressure water coolant and 316 L austenitic stainless steel as a structural material have been selected for first wall and blanket designs. The first wall being considered for the early phase of operation is a protected wall of carbon based tiles, mechanically attached or bonded to the heat sink. Proposed first wall designs are anticipated to be operable up to neutron wall loads of $\sim 1.4\ MW/m^2$ in ignition conditions and at $\sim 1.0\ MW/m^2$ in driven operation ($Q \geq 5$).

Divertor plates with carbon based protection and Cu, Mo or V heat sink material are predicted to work at heat fluxes of 3–6 MW/m^2 for physics experiments. Plates with high Z material protection could operate for technology testing with a longer lifetime if low temperature, high density plasma edge conditions could be realized. Sweeping of the separatrix will ease, to some extent, the technical problems in designing the divertor plates.

TABLE I. ITER MAJOR PARAMETERS

	With breeding blanket installed (for technology experiments)		Without breeding blanket installed (for physics experiments)	
	Steady state operation performance	Full inductive operation performance		Extended operation performance
Major radius (m)	5.5	5.5	5.8	5.8
Minor radius (m)	1.8	1.8	2.2	2.25
Aspect ratio	3.1	3.1	2.6	2.6
Elongation (95% flux surface)	2.0	2.0	1.9	2.0
Troyon factor	2.9	2.0	1.7	1.5
Safety factor	3.1	3.1	3.2	2.8
Toroidal field on-axis (T)	5.3	5.3	5.0	5.0
Plasma current (MA)	18	18	22	25
Neutron wall loading (MW/m^2)	1.1	1.3	1.0	0.98
Electron density ($10^{20}\ m^{-3}$)	0.72	1.2	1.0	0.96
Ion temperature (keV)	18	10	10	10
Required confinement time (s)	2.1	2.5	3.1	3.3
Inductive flux (V·s)	235	235	260	260
Fusion power (MW)	850	1000	1000	1000

For supplying tritium to the ITER machine itself, three options for blanket design are under consideration, i.e. solid breeder, LiPb eutectic breeder and lithium salt solution breeder.

The vacuum vessel, inboard blanket and radiation shield will result in an insulator dose of about 1×10^9 rad (1×10^7 Gy) for a first wall neutron fluence of 3 $MW \cdot a/m^2$ and a total nuclear heating in the magnets of ~ 13 kW. During the physics experiments the inboard blanket shield can be a thinner one, permitting an additional space of up to 15 cm with greater nuclear heating for operation of a larger plasma size.

An important objective of testing is the extraction of high grade heat from reactor relevant modules and sectors. Testing space will include four to eight modules with a volume of $\sim 1\ m^3$ each, and two full outboard blanket sectors.

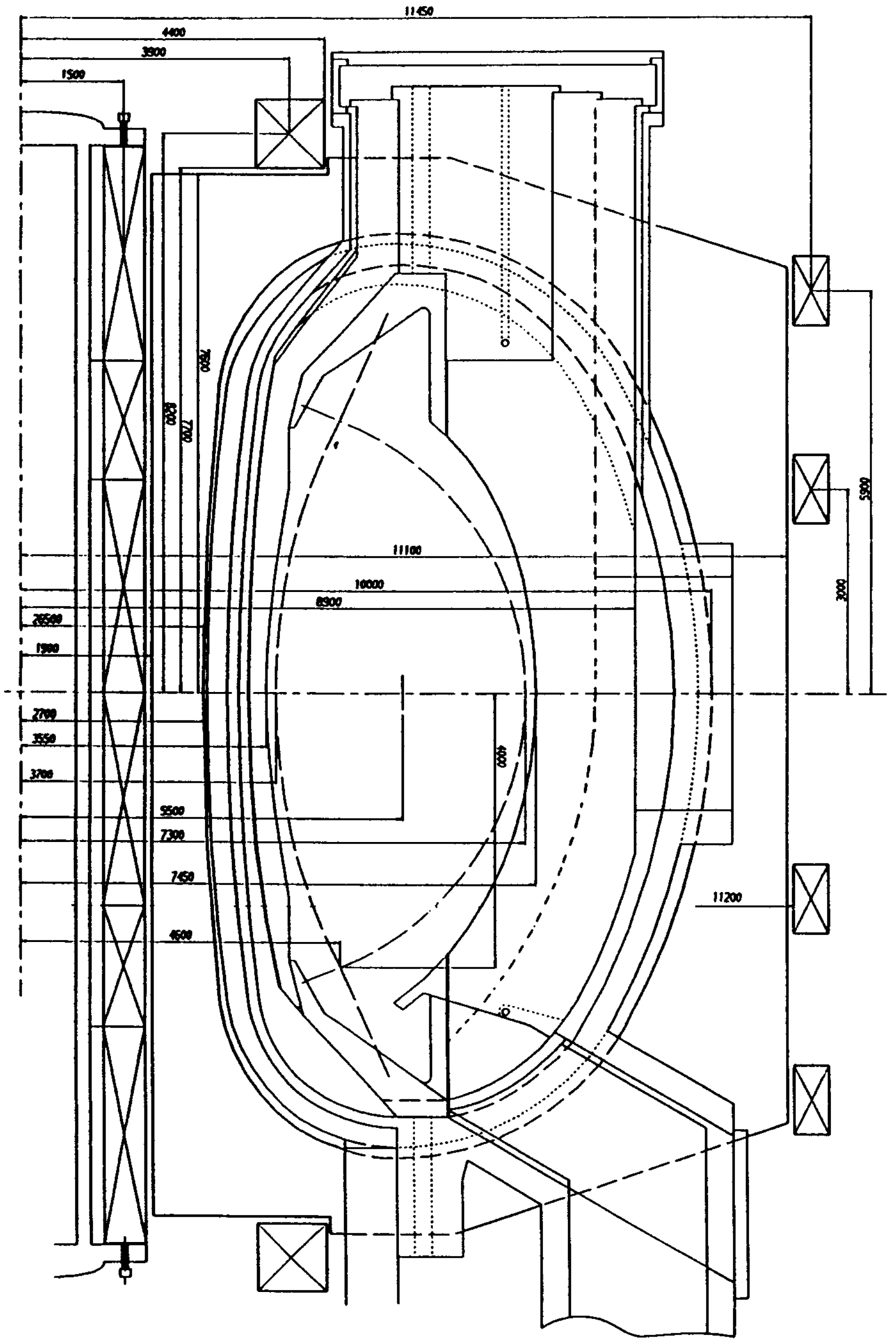

FIG. 1(a). Vertical cross-section of ITER for the technology phase.

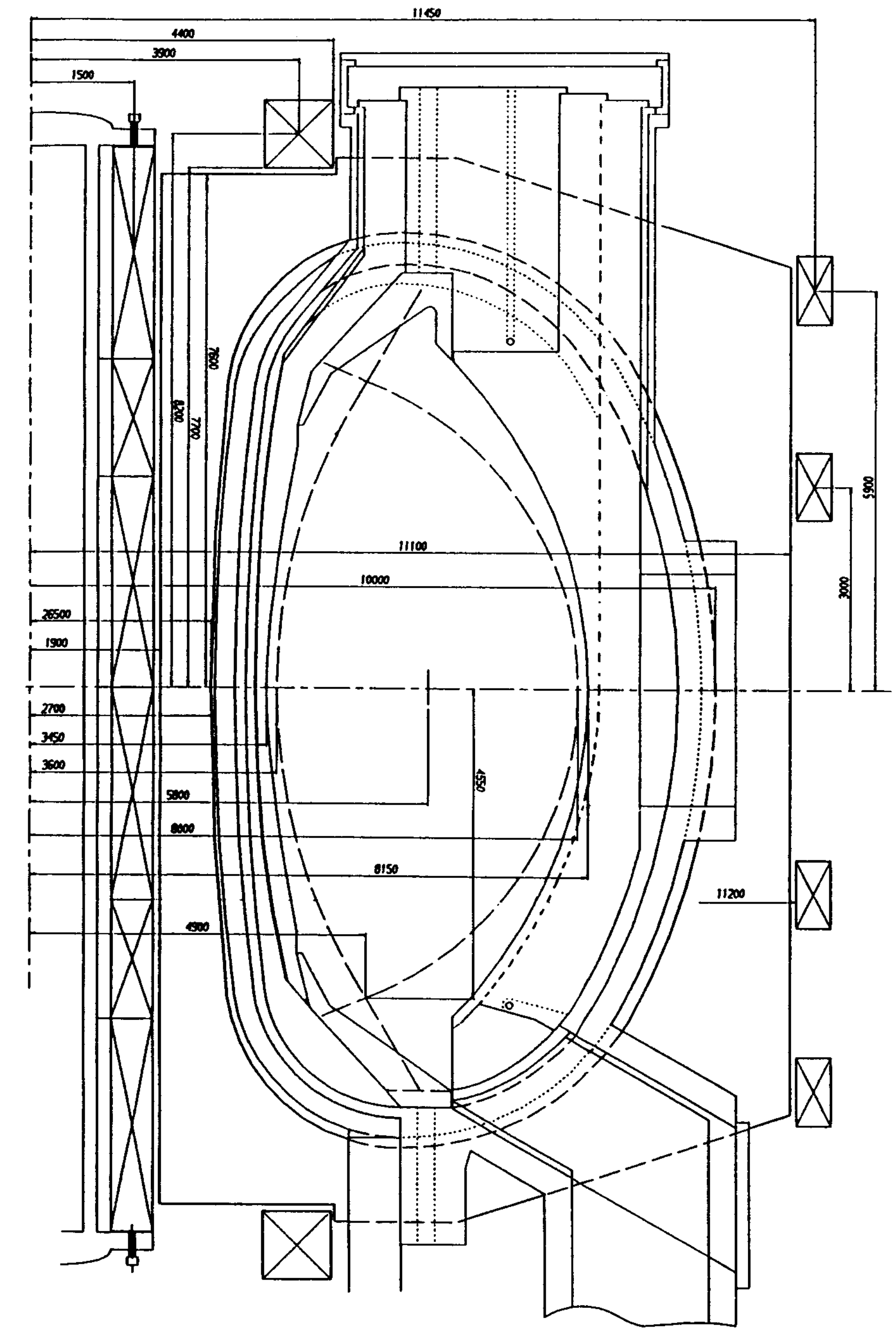

FIG. 1(b). Vertical cross-section of ITER for the physics phase.

The basic fuelling mechanism will be gas puffing. Pellet injection (~2 km/s) is planned for density ramp-up and profile control. The torus exhaust pumping speed required is in the range 350–700 m^3/s. Compound cryopumps are a possible option.

5. PERFORMANCE FLEXIBILITY OF THE ITER CONCEPT [5]

The ITER design should be as flexible as possible to provide access for the introduction of advanced features and new capabilities. At the present state of ITER design this flexibility is being considered primarily with regard to physics uncertainties. During an early phase of physics experiments in ITER operations, an ignition mode with pure inductive drive and a steady state burn mode with non-inductive current sustainment will be studied. In this DT physics phase a relatively small number of ignition mode and steady state mode operations with intense neutron production are envisaged, and neutron fluence is expected to be less than 0.01 $MW \cdot a/m^2$. After the installation of breeding blankets ITER should be reliably operated in a steady state and/or a long pulse mode and devoted to engineering objectives and testing programmes. Therefore the machine should have flexible and extended capabilities especially with regard to plasma size, plasma current, configuration (elongation, triangularity, different divertor configurations, etc.) and operation range (beta, safety factor, etc.). In order to achieve these capabilities the structures surrounding the plasma should be replaceable and the poloidal field system should be flexible.

Extended operations with a larger plasma size and higher plasma current are especially important to enhance the confinement capability of ITER in the physics phase. Operation has been studied for a plasma current of 22 MA, major radius of 5.8 m, minor radius of 2.2 m, elongation of 1.9 and safety factor of 3.2 at the 95% flux surface and toroidal field ripple of about 1.5%. Assuming a bootstrap current of 5 MA or a 50 V·s saving with non-inductive current drive, it seems possible to produce in the same machine an extended plasma, having a plasma current of 28 MA, minor radius of 2.25 m, aspect ratio of 2.58, elongation of 2.0, safety factor of 2.5 and toroidal field ripple of about 2%. Further studies of the engineering implications of such a high current will be made in 1989.

The device is also suitable for the steady state mode of operation and in that case the plasma current should be around 15–18 MA with a reasonable Q value (≥ 5), current drive power (~100 MW), neutron wall loading (~1 MW/m^2) and confinement capability.

6. R&D PLAN

Extensive fusion R&D activities are being carried out in the national programmes of various countries. It is obvious that a great deal of information essen-

TABLE II. ITER RELATED PHYSICS R&D TASKS

PH1	Power and helium exhaust conditions
PH2	Helium radial distribution in high temperature tokamak discharge
PH3	Viability of a radiative edge
PH4	Sweeping of the divertor target load
PH5	Characterization of low Z materials for plasma facing components
PH6	Characterization of high Z materials for plasma facing components
PH7	Characterization of disruptions
PH8	Disruption control
PH9	RF plasma formation and preheating
PH10	RF current initiation
PH11	Scaling of volt-second consumption during inductive current ramp-up in large tokamaks
PH12	Alpha particle losses induced by toroidal magnetic field ripple
PH13	Compatibility of plasma diagnostics with ITER conditions
PH14	Steady state operation in enhanced confinement regimes (H-mode and 'enhanced' L-mode)
PH15	Comparison of theoretical transport models with experimental data
PH16	Control of MHD activity
PH17	Density limit
PH18	Plasma performance at high elongation
PH19	Alpha particle simulation experiments
PH20	Electron cyclotron current drive
PH21	Ion cyclotron current drive
PH22	Impact of Alfvén wave instability on neutral beam current drive
PH23	Proof of principle of fuelling by injection of field reversed compact toroids

tial to ITER will be generated from these R&D activities, and therefore one should naturally assume that such information will be made available for the ITER design activities directly or by other, indirect means.

In the 'Terms of Reference' for ITER R&D it is stated that the level of effort of ITER related R&D should stay at about US $10 million per party per year. Clearly all the tasks needed for ITER cannot be supported within the budget of US $120 million up to the end of 1990. Therefore, only technology tasks specific to

TABLE III. ITER TECHNOLOGY R&D: SUMMARY OF TASKS AND PROPOSALS FROM THE ITER PARTIES

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

ITER design have been included in the budget, and for the physics tasks the ITER activities should assume voluntary contributions.

A list of specific physics R&D needs for ITER design has been developed (Table II). Tasks PH1 to PH13 cover a number of crucial design related physics R&D issues on which additional information is urgently required to permit, in 1990, the confirmation of the technical working assumptions on which the ITER concept is based. The rest of the tasks cover general issues of plasma performance (energy confinement, operational limit, burn control and long pulse operation), crucial for ITER to enable it to reach its objectives, but not sufficiently covered by the ongoing fusion programmes.

A plan for ITER technology R&D has been developed. In the plan the R&D tasks are divided into six areas: blanket, plasma facing components, magnets, fuel cycle, heating and current drive, and maintenance. The tasks and the work proposed by the ITER participating parties for each task are summarized in Table III.

It should be noted that the R&D plan for ITER has been produced on the basis of the information available at a preliminary stage of the ITER design activity, and that it will be reviewed and modified as the need arises.

7. CONCLUSION

A described above, it is concluded that a machine concept that makes possible flexible and extended operation would satisfy the objectives assigned to ITER, although a number of identified technical problems need to be resolved. The present guidelines for both the physics and the engineering of ITER will continue to be developed during the ITER conceptual design, and the design activities will continue to aim at the production of an integrated ITER design to meet the required goals.

It should be emphasized here that there are many crucial design related R&D issues in both the physics and the technology areas on which additional information is urgently needed, and the activities of the fusion community working on these technical issues should be encouraged.

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