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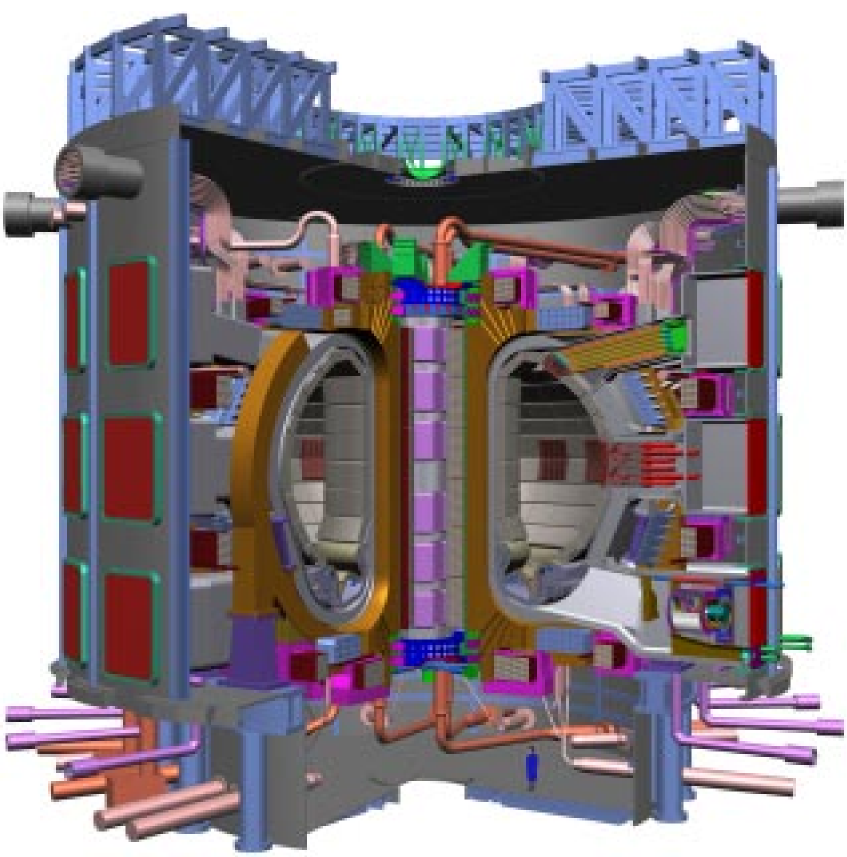
Summary ITER Final Design Report (July 2001)



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1 Introduction

This document presents a summary of the technical basis for the ITER Final Design Report (presently in Draft form). This technical basis is itself supported by detailed technical documentation which is composed of "living" internal documents whose content evolves with progress in the design details.

The Final Design Report is the culmination of collaborative design and supportive technical work by the ITER Joint Central Team (JCT) and Home Teams (HTs) under the terms of the ITER EDA Agreement.

The ITER Parties, working through a special working group (SWG) constituted under the terms of the ITER EDA Agreement, reviewed and compared two possible strategies for meeting the programmatic objective of demonstrating the scientific and technological feasibility of fusion, based on:

- an ITER-like machine, capable of addressing both scientific and technological issues in an integrated fashion, and
- a number of complementary lower cost experiments each of which would specialise on scientific/technological issues.

With regard to the second strategy, the SWG[1] found that the complex non-linear interactions between α-particle heating, confinement barriers and pressure and current profile control, and their compatibility with a divertor can be addressed only in an integrated physics/technology step such as an ITER-type experiment, capable of providing long burn in conditions in which α-particles are the dominant source of plasma heating. A satisfactory understanding of these physics/plasma/technology interactions is essential to any reactor-oriented fusion development programme. Moreover the SWG expressed the unanimous opinion that the world programme is *"scientifically and technically ready to take the important ITER step."* This viewpoint was subsequently endorsed by the Parties through the ITER Council.

The SWG specified technical guidelines and objectives fitting the programmatic background of the above conclusions, which were also endorsed by the Parties. In summary the revised performance specifications for ITER, adopted by the parties in June 1998, require:

- to achieve extended burn in inductive operation with $Q \geq 10$, not precluding ignition, with an inductive burn duration between 300 and 500s, a 14MeV average neutron wall load $\geq 0.5 MW/m^2$, and a fluence $\geq 0.3 MW/m^2$.
- to aim at demonstrating steady state operation using non-inductive current drive with $Q \geq 5$.

In addition, the device should:

- use as far as possible technical solutions and concepts developed and qualified during the EDA, and
- cost about 50% of the direct capital cost of the 1998 ITER Design, and particular attention should be devoted to cash flow.

[1] SWG report to the ITER Council on Task #2 Result, ITER Meeting 10-3-1999 ROM Attachment 5

To identify designs that might meet the revised objectives, a task force involving the JCT and the HTs met during 1998 and 1999 to analyse and compare a range of options for the design of such a device. Using system codes to consistently relate the main plasma parameters, physics, engineering design constraints and costs, representative options spanning an appropriate range of aspect ratio were selected for further elaboration and consideration. This led, at the end of 1999, to a single configuration for the ITER design with parameters considered to be consistent with limitations and cost targets, yet fully meeting the objectives with sufficient margins.

In January 2000, the ITER Meeting (Tokyo) *"accepted the ITER-FEAT Outline Design Report, taking note of the TAC Report and recommendations and agreed to transmit the report to the Parties for their consideration and domestic assessment"*. The Parties assessments were overwhelmingly positive in their endorsement of the outline design, and the process of assessment by the Parties offered the opportunity to further tune the design to find the best compromise to simultaneously match the aims of each Party. The design was subsequently approved by the governing body of ITER in Moscow in June 2000, recognising it as a single mature design for ITER consistent with its revised objectives.

With the further development, summarised in this report, on the technical basis for the final design of ITER, the result is a robust design which confers a high degree of confidence that it will meet its objectives. All information needed to take a decision on construction is now available. After the completion of Explorations, the Parties negotiators should agree on a preferred site to allow specific site adaptation, and a text for the construction agreement ready to sign.

The progress towards the proposed design rests on:

1. physics understanding: the ITER Physics Basis[2] (IPB) plus new results of voluntary R&D;
2. R&D results in technology[3] development since 1992, which have provided qualified solutions by testing models after their manufacture: they have demonstrated feasibility through clearly identified manufacturing processes;
3. a consensus across Parties on safety principles and criteria for limiting consequences to the environment, and results of analysis on all possible, even hypothetical, accidents with regard to their consequences.
4. a cost target: a cost analysis has been established by industries of all Parties for manufacturing which is probably not yet fully optimised towards a reduced cost; this would be the outcome of "manufacturing R&D", needed anyway to achieve reliable production.

The key requirements to achieve Q >10 in inductive pulsed mode of operation according to the IPB can be summarised as:

1. a plasma current sufficient to provide adequate plasma energy confinement,
2. a large enough plasma density and a plasma energy confinement, good enough to achieve $Q \geq 10$,

[2] Nuclear Fusion 39 (1999) 2137-2664
[3] To be Published in Fusion Engineering and Design in July 2001. A short review is attached at the end of this document.

3. a reliable power exhaust and impurity control in a single-null divertor configuration, while at the same time considering the limits imposed by various instabilities on plasma design parameters such as safety factor, normalised beta, elongation, triangularity, and He ash impurity content after transfer of α-energy to the thermal plasma.

With regard to steady state operation modes, the data presently in hand does not possess the coherence across the present experiments which is required to make it the design basis for nominal performance. However, there does not appear to be any crucial conflict regarding designs based on H-mode physics to exploit whatever operational modes future progress will establish, if efficient and flexible current drive systems are available with enough power.

2 Design Overview

ITER is a long pulse tokamak with an elongated plasma and a single null poloidal divertor (Figure 2.1-1 to Figure 2.1-5 and Table 2.1-1 to Table 2.1-3). Nominal inductive operation produces a fusion power of 500 MW for a burn length of 400 s.

The major components of the tokamak are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside a toroidal vacuum vessel. The magnet system comprises toroidal field coils, a central solenoid, external poloidal field coils, and correction coils. The centring force on toroidal magnets is reacted by the central solenoid. The TF coil cases are used to support the external PF coils. The vacuum vessel is a double-walled structure supported on the toroidal field coil. The magnet system together with the vacuum vessel and internals are supported by gravity supports, one beneath each sector.

Inside the vacuum vessel, the internal, removable components, including blanket modules, divertor cassettes, port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics sensors, absorb most of the radiated heat from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The divertor exhausts the helium ash and limits the concentration of impurities in the plasma. The other vessel internals are chosen so that they do not contribute unacceptably to the concentration of impurities in the plasma. The shielding blanket design does not preclude its later replacement on the outboard side by a breeding blanket.

The heat deposited in the internal components and the vessel is rejected to the environment via the tokamak cooling water system (comprising individual heat transfer systems) which is designed to preclude releases of tritium and activated corrosion products to the environment. Some parts of these heat transfer systems are also used to bake and hence clean the plasma-facing surfaces inside the vessel by releasing impurities. The tokamak is housed in a cryostat, with thermal shields between the hot parts and the magnets, and support structures which are at cryogenic temperature.

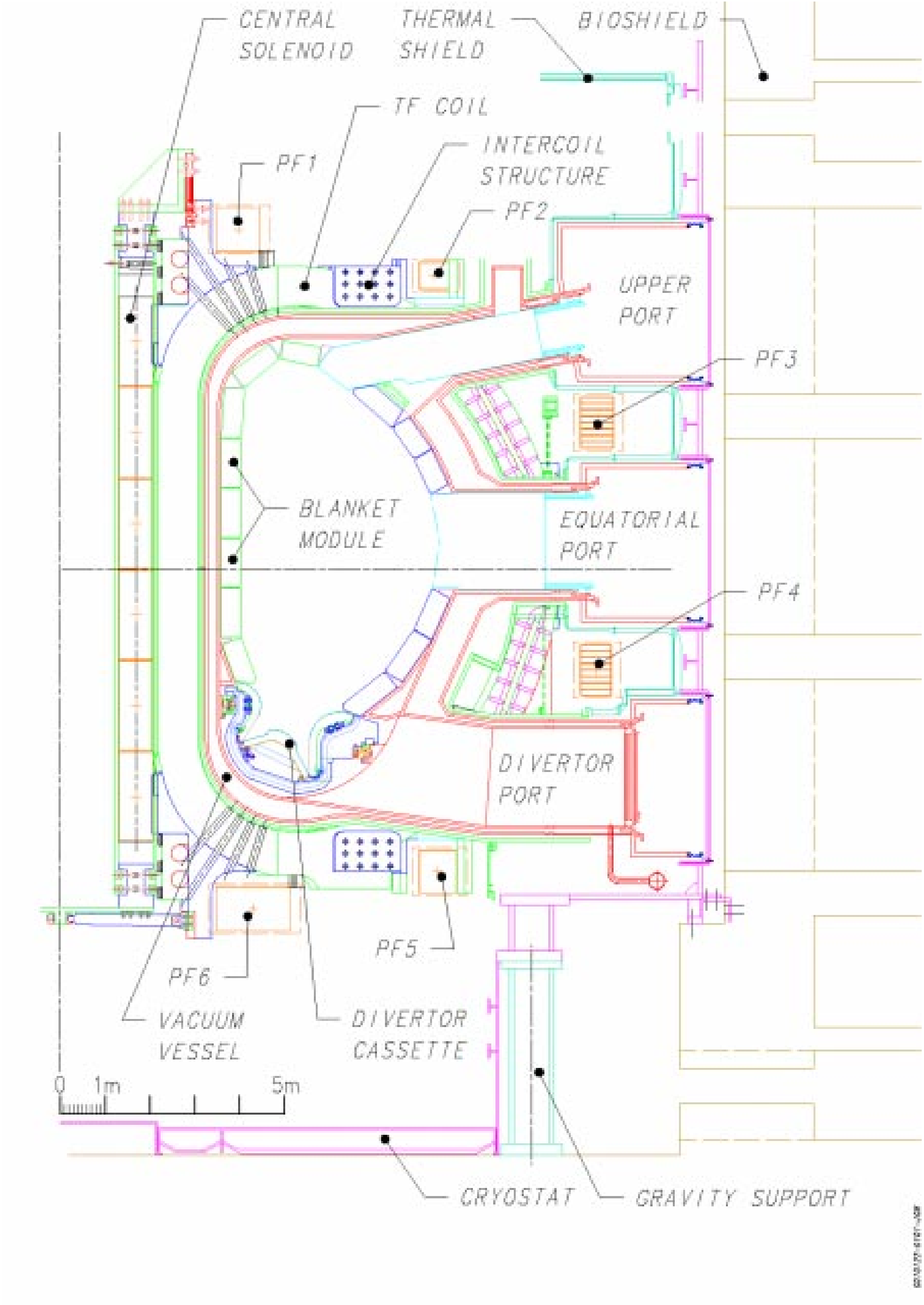


Figure 2.1-1 ITER Tokamak Cross-section

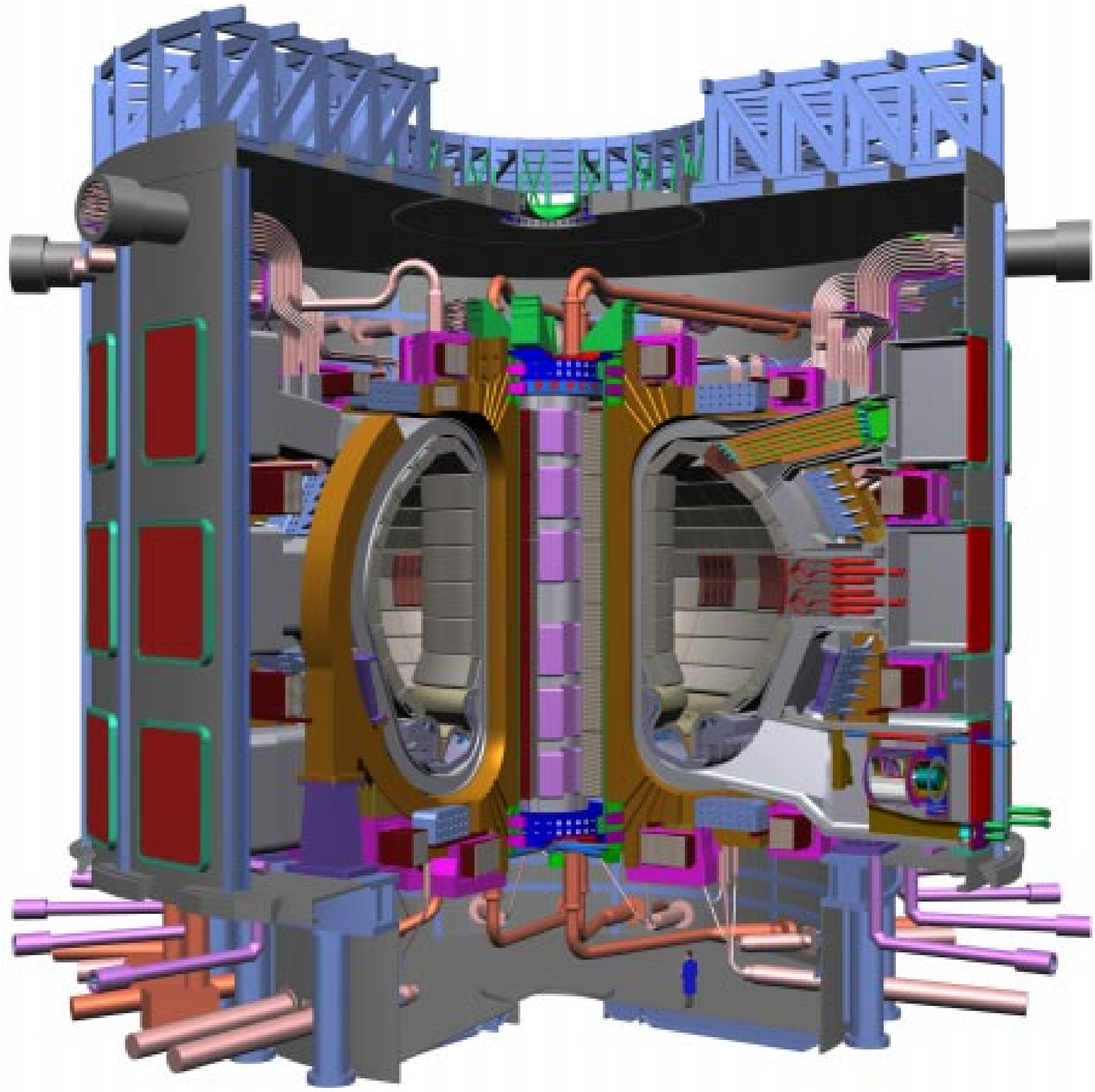


Figure 2.1-2 ITER Tokamak Cutaway

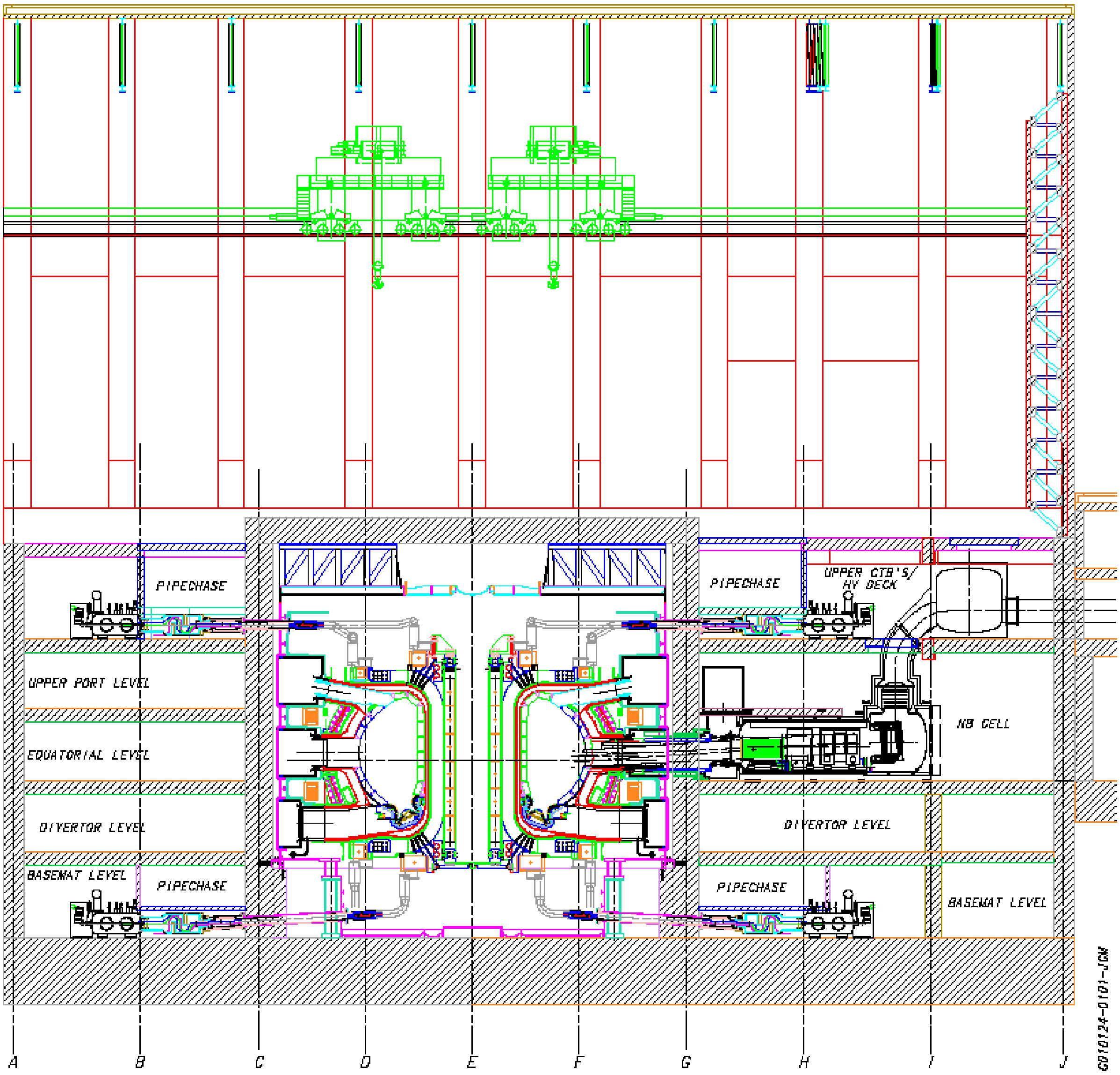


Figure 2.1-3 Cross-section NS Through the Tokamak Building

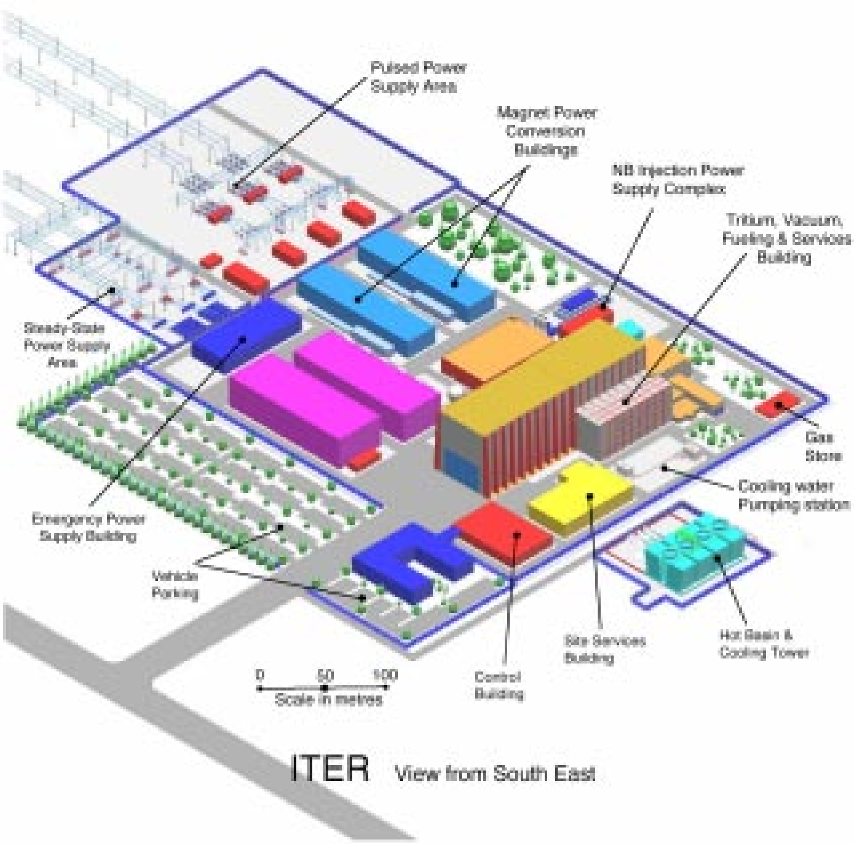


Figure 2.1-4 Generic ITER Site View

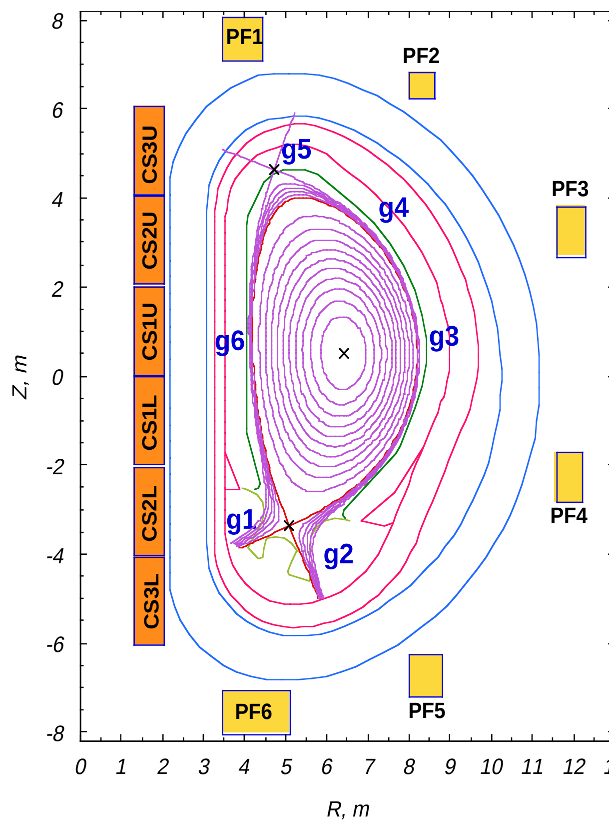


Figure 2.1-5 ITER Nominal Plasma Configuration

The tokamak fuelling system is capable of gas and solid hydrogen pellet injection. Low-density gaseous fuel will be introduced into the vacuum vessel chamber by a gas injection system. The plasma will progress from electron-cyclotron-heating-assisted initiation, in a circular configuration touching the limiter, to an elongated divertor configuration as the plasma current is ramped up. After the current flat top (nominally 15 MA for inductive operation) is reached, subsequent plasma fuelling (gas or pellet) together with additional heating for ~ 100 s leads to a high Q DT burn at 500 MW. With non-inductive current drive from the heating systems, the burn duration can be extended to ~ 3600 s, or longer. In inductive scenarios, before the inductive flux available is consumed, the burn is terminated by reducing the fuelling to rampdown the fusion power, followed by current rampdown and plasma termination. The inductively driven pulse has a nominal high energy multiplication burn duration of 400 s, and the pulse repetition period may be as short as 1800 s. Plasma control is provided by the poloidal field system, and the pumping, fuelling (D,T and impurities such as N_2, Ar) and heating systems, based on feedback from diagnostic sensors.

With regard to safety and licensing issues, the current design focuses on confinement as the overriding safety function of equipment, other functions being recognised as being required to protect confinement. A "lines-of-defence" methodology is used to obtain the required level of safety while balancing the functional requirements of systems and components. The number and quality of the lines of defence then depend on the inventory at risk.

Successive barriers are provided for tritium (and activated dust). These include the vacuum vessel, the cryostat, active air conditioning systems, with de-tritiation and filtering capability in the building. Confinement and effluents, normal as well as accidental, are filtered and detritiated, in such a way that their release to the environment is as low as reasonably achievable (ALARA).

Table 2.1-1 Main Plasma Parameters and Dimensions

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

Table 2.1-2 Main Engineering Features of ITER

Superconducting toroidal field coils (18 coils)	
Superconductor	Nb_3Sn in circular stainless steel (SS) jacket in grooved radial plates
Structure	Pancake wound, in welded SS case, wind, react and transfer technology
Superconducting Central Solenoid (CS)	
Superconductor	Nb_3Sn in square Incoloy jacket, or in circular Ti/SS jacket inside SS U-channels
Structure	Pancake wound, 3 double or 1 hexa-pancake, wind react and transfer technology
Superconducting poloidal field coils (PF 1-6)	
Superconductor	NbTi in square SS conduit
Structure	Double pancakes
Vacuum Vessel (9 sectors)	
Structure	Double-wall, welded ribbed shell, with internal shield plates and ferromagnetic inserts
Material	SS 316 LN structure, SS 304 with 2% boron shield, SS 430 inserts
First Wall/Blanket (421 modules)	(Initial DT Phase)
Structure	Single curvature faceted separate FW attached to shielding block which is fixed to vessel
Materials	Be armour, Cu-alloy heat sink, SS 316 LN str.
Divertor (54 cassettes)	
Configuration	Single null, cast or welded plates, cassettes
Materials	W alloy and C plasma facing components Copper alloy heat sink, SS 316 LN structure
Cryostat	
Structure	Ribbed cylinder with flat ends
Maximum inner dimensions	28 m diameter, 24 m height
Material	SS 304L
Tokamak Cooling Water System	
Heat released in the tokamak during nominal pulsed operation	750 MW at 3 and 4.2 MPa water pressure, ~ 120°C
Cryoplant	
Nominal average He refrig. /liquefac. rate for magnets & divertor cryopumps (4.5K)	55 kW / 0.13 kg/s
Nominal cooling capacity of the thermal shields at 80 K	660 kW
Additional Heating and Current Drive	
Total injected power	73 MW initially, 100 MW nominal maximum
Candidate systems	Electron Cyclotron, Ion Cyclotron, Lower Hybrid , Negative Ion Neutral Beam
Electrical Power Supply	
Pulsed Power supply from grid	
Total active/reactive power demand	500 MW / 400 Mvar
Steady-State Power Supply from grid	
Total active/reactive power demand	110 MW/ 78 Mvar

Table 2.1-3 Heating and Current Drive Systems

	NB	EC (170 GHz)	IC (~ 50 MHz)	LH (5 GHz)
Power injected per unit equatorial port (MW)	16.5	20	20	20
Number of units for the first phase	2	1	1	0
Total power (MW) for the first phase	33	20	20	0
The 20 MW of EC module power will be use either i) in 2 upper ports to control neoclassical tearing modes at the q = 3/2 and q = 2 magnetic surfaces, or ii) in one equatorial port for H&CD mainly in the plasma centre.				

2.1 Operation Scenarios and Operation Phases

The design of ITER needs to be able to cope with various operation scenarios. Variants of the nominal scenario are therefore considered for extended duration plasma operation, and/or steady state modes with a lower plasma current operation, with H, D, DT (and He) plasmas, potential operating regimes for different confinement modes, and different fuelling and particle control modes. Flexible plasma control should allow the accommodation of "advanced" plasma operation based on active control of plasma profiles by current drive or other non-inductive means.

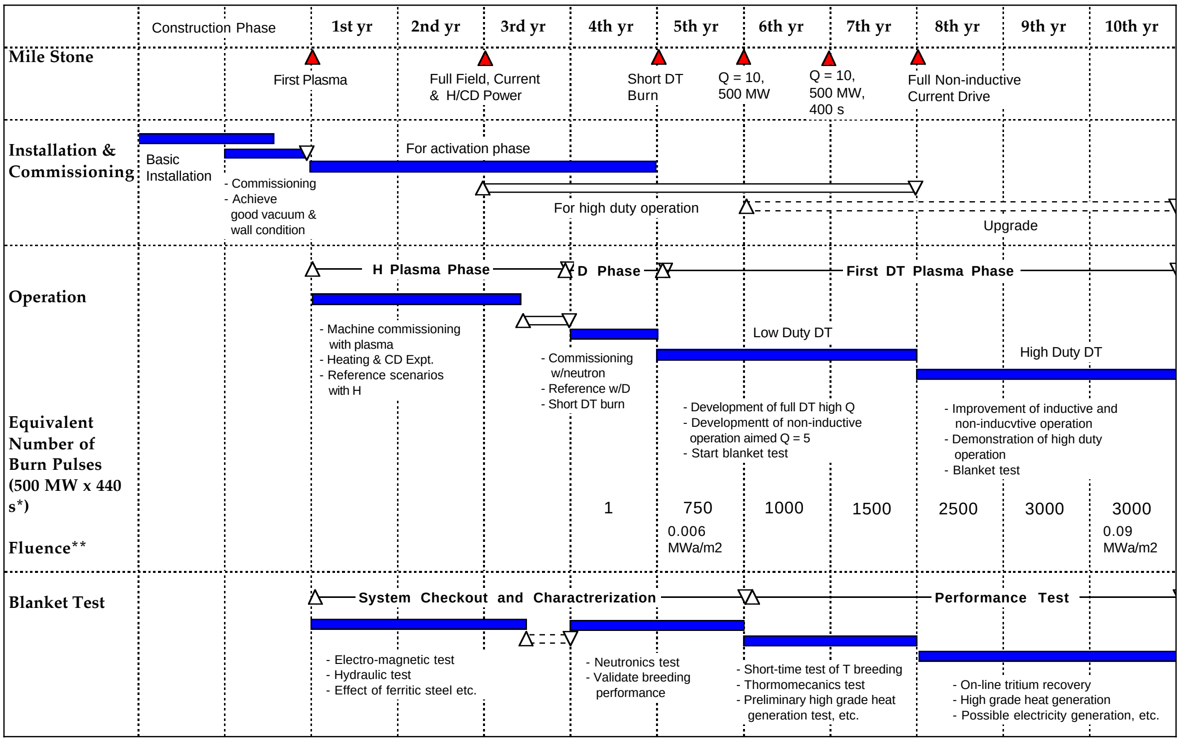
Four reference scenarios are identified for design purposes. Two alternative scenarios are specified for assessment purposes to investigate if and how plasma operations will be possible within the envelope of the machine operational capability assuming a reduction of other concurrent requirements (e.g. pulse length).

Design scenarios

1. Inductive operation I: 500 MW, Q=10, 15 MA operation with heating during current ramp-up
2. Inductive operation II: 400 MW, Q=10, 15 MA operation without heating during current ramp-up
3. Hybrid, operation
4. Non-inductive operation I: weak negative shear operation

Assessed scenarios

5. Inductive operation III: 700 MW, 17 MA, with heating during current ramp-up.
6. Non-inductive operation II: strong negative shear



* The burn time of 440 s includes 400 s flat top and equivalent time which additional flux is counted during ramp-up and ramp-down.
** Average Fluence at First Wall (Neutron wall load is 0.56 MW/m2 in average and 0.77MW/m2 at outboard midplane.)

Figure 2.1-1 Initial Operation Plan

The ITER exploitation during its lifetime is divided into successive phases.

H Phase

This is a non-nuclear phase, mainly planned for full commissioning of the tokamak system in a non-nuclear environment where remote handling is not mandatory. The discharge scenario of the full DT phase reference operation can be developed or simulated in this phase. The peak heat flux onto the divertor target will be of the same order of magnitude as for the full DT phase. Characteristics of electromagnetic loads due to disruptions or vertical displacement events, and heat loads due to runaway electrons, will be basically the same as those of the DT phase. Studies of the design-basis physics will significantly reduce the uncertainties of the full DT operation.

Some important technical issues cannot be fully tested in this phase because of smaller plasma thermal energy content and lack of neutrons and energetic alpha-particles.

The actual length of the hydrogen operation phase will depend on the merit of this phase with regard to its impact on the later full DT operation, in particular on the ability to achieve good H mode confinement with large enough plasma density.

D Phase

Characteristics of deuterium plasma behaviour are very similar to those of DT plasma except for the amount of alpha heating. Therefore, the reference DT operational scenarios, i.e., high *Q,* inductive operation and non-inductive steady state operation, can be simulated. Since tritium already exists in the plasma, fusion power production at a significant power level for a short period of time without fully implementing cooling and tritium-recycle systems, which would be required in the subsequent full DT phase, could therefore also be demonstrated. By using limited amounts of tritium in a deuterium plasma, the integrated commissioning of the device is possible. In particular, the shielding performance can be checked.

DT Phases

During the first phase of DT operation the fusion power and burn pulse length will be gradually increased until the inductive operational goal is reached. Non-inductive, steady-state operation will also be developed. DEMO-relevant test blanket modules will also be tested whenever significant neutron fluxes are available, and a reference mode of operation for that testing will be established.

The second phase of full DT operation, beginning after a total of about ten years of previous operation, will emphasise improvement of the overall performance and the testing of components and materials with higher neutron fluences. This phase should address the issues of higher availability of operation and further improved modes of plasma operation. Implementation, and the programme, for this phase should be decided following a review of the results from the preceding three operational phases, and assessment of the merits and priorities of programmatic proposals.

A decision on incorporating tritium breeding during the course of the second DT phase will be decided on the basis of the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance. Such a decision will depend on the R&D completed during the first phase.

3 Plasma Performance

According to the conclusions of the ITER Physics Basis, from broadly based experimental and modelling activities within the fusion programmes of the ITER Parties, the regime retained for nominal inductive operation of ITER is the ELMy-H-mode confinement regime.

In this regime, plasma turbulent heat conduction across the magnetic surfaces drops dramatically in a thin transport barrier layer just inside the magnetic separatrix. This layer is commonly observed to undergo successive relaxations called edge localized modes (ELMs). The interest in ELMy H-modes follows from experimental observations that show that this mode reduces transport throughout the plasma core. The standard working hypothesis, supported by many observations, is that H-mode occurs when the power transported across the separatrix (P_{loss}) exceeds a threshold value ($P_{L\text{-}H}$).

From the statistical analysis of confinement results obtained in all previous devices, an expression of the energy confinement time has been established as a function of plasma parameters, verified through three orders of magnitude, and expressed as

$$\tau_{E,th}^{IPB98(y,2)} = 0.0562 H_H I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_x^{0.78} \qquad \text{(rms err. 0.145)}$$

where the units are s, MA, T, $10^{19}m^{-3}$, MW, m and amu, and where H_H is a scalar which can be used to represent either how close the actual value observed in one experiment is from the average, or a level of inaccuracy. This expression will only be valid in H-mode, that is when $P_{loss} > P_{L\text{-}H}$, where

$$P_{\text{L-H}} = 2.84 M^{-1} B_{\text{T}}^{0.82} \bar{n}_{\text{e}}^{0.58} R^{1.00} a^{0.81} \qquad \text{(rms err. 0.268)}$$

where the units are MW, amu, T, $10^{20}m^{-3}$, m.

3.1 ITER Plasma Current and Size

Assuming $P_{loss} > P_{L\text{-}H}$, and using the previous expressions for τ_E and $P_{L\text{-}H}$, one can derive the relationship between the plasma parameters and the capability to achieve a given value of Q = P_{fusion}/P_{aux}, which can be formulated approximately as

$$\left[\frac{H_H I_{MA} \frac{R}{a}}{X}\right]^3 = \frac{Q}{Q+5}$$

with X ~ 50-60, a slowly varying function of parameters. This relation provides an easy basis for I_{MA} = 15MA, R/a = 3.1 if Q = 10, X = 55 and H_H = 1.

Expressing now I_{MA}.R/a = 5.B_T.a/q.f, where f is a function of aspect ratio, increasing with triangularity, δ, and mostly with plasma elongation κ, it is obviously important to increase the value of f, decrease the value of q, and compromise between B_T and size.

However, there are limiting values for f and q: too large an elongation provides a condition where the vertical stability of plasma position cannot be assured practically, and q below 3 is limited by the occurrence in a large volume near the plasma axis of “sawtooth relaxation” (an instability which periodically destroys the confinement in this volume) and there is an increasing susceptibility to instability as q=2 is approached.

A large aspect ratio allows a larger value for B_T on the plasma axis for a given maximum B on the TF coil conductor (B~l/R), an important constraint for superconductors, but access to the plasma for heating and maintenance becomes limited by smaller openings, and plasma shape control becomes more difficult. A compromise is therefore needed.

3.2 Plasma Confinement Extrapolation

Experiments have shown that once an ideally stable equilibrium is assured by externally applied shaping fields, the plasma response to auxiliary heating and fuelling is governed by the spontaneous appearance of a fine scale turbulence.

Profiles of plasma parameters (Figure 3.2-1) are the consequence of transport properties which are governed by turbulence, the characteristic scale of which is much smaller than the device size. The physical processes prevailing depend on dimensionless variables, built from density, temperature and magnetic field values, mainly ρ* = ion gyration radius/minor plasma radius, β = plasma pressure/magnetic pressure and ν = collisionality.

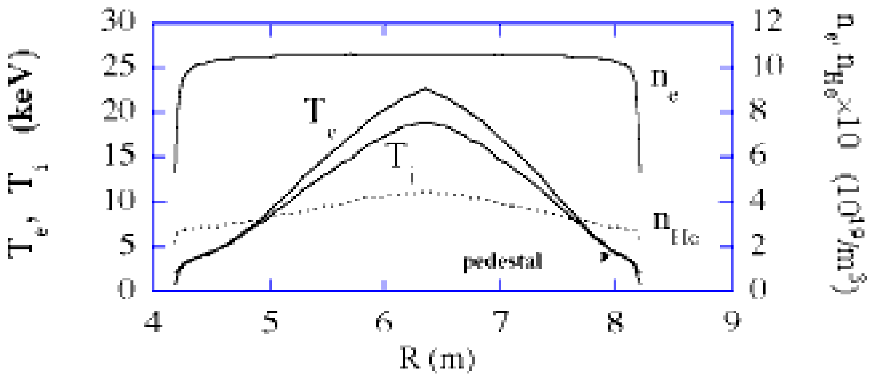


Figure 3.2-1 Profiles of Electron Temperature (T_e), Ion Temperature (T_i), Electron Density (n_e), Helium Density (n_{He})

In this respect, experiments having identical non-dimensional parameters, but differing magnetic field, density and temperature, have been shown to have the same non-dimensional energy confinement time defined by $\Omega_{ci}.\tau_E$, (where Ω_{ci} is the plasma ion cyclotron frequency). Therefore, present experiments have been used to simulate ITER discharges, which reduces the problem of extrapolation to that of a single parameter ρ*.

Accordingly, codes have been built to model plasma evolution, in particular ITER. These codes take into account the magnetic configuration in detail, assuming constant density and temperature on a magnetic surface, adjusting the thermal diffusivity in such a manner that the global energy confinement time computed by the code is constrained to be equal to the global scaling relation, and adapting its spatial profile to provide temperature profiles close to those observed in ITER demonstration discharges.

These results, confirmed by all the previous experiments of very different size, provide confidence in the performance of confinement ITER will achieve.

3.3 H Pedestal and ELMs

The transport barrier which occurs just inside the magnetic separatrix in H-mode provides a thin layer, where the pressure increases sharply (a large radial gradient is established) and at its inner edge a pedestal is formed where density and temperature values serve as boundary conditions for the core profiles. Pedestal temperatures can be very important if the core temperature gradient is constrained (a fact not always observed in present experiments) to lie near a marginal upper value. Even if this is not the case, the energy content of the pedestal is generally not negligible compared to the core energy content (in fact about one third). Moreover it has a scaling that differs from the core global scaling, and is not definitely agreed yet.

ELMs appear as a pseudo-periodic relaxation of the pressure gradient, due to an instability which depends on the detailed shape of the magnetic surface near the separatrix (under the global influence of the separatrix curvature variation, triangularity and magnetic shear). As the ELMs' periodicity becomes smaller, their amplitude increases and the energy removed from the pedestal by each ELM becomes large and, as it is deposited onto the divertor targets, leads to their strong erosion. The physical phenomena involved are not understood in quantitative terms at present.

3.4 Internal Confinement Barrier

In some conditions (not completely understood or controlled until now), another confinement barrier might occur inside the core, and limit even more the turbulent heat conduction across the plasma. This barrier again requires for its existence a threshold in the power crossing it. It provides a steep pressure gradient and occurs usually in a region where the magnetic shear is very weak (as at a minimum of q). This internal barrier, if its existence and stability can be controlled on a long time scale, will lead to better confinement performance, which can be the more interesting the larger can be its radius. In addition, because a toroidal current is driven by the pressure gradient (the "boot-strap" current), this internal barrier is considered an important feature for possible steady-state tokamak operation, where the toroidal current, driven by non-inductive drive methods from auxiliary power, has to be minimised.

3.5 β limits; Non-axisymmetric Perturbations; Islands

Because fusion power production scales as $\beta^2 B^4$, there is an incentive to operate at the highest value of β allowed by plasma stability. For simple, monotonic q profiles, characteristic of inductive operation, the MHD stability limits $\beta_N = \beta/(I/aB)$ to values $< 4\, l_I \simeq 3.5$.

However, numerous experiments have shown the appearance of modes, no longer rigorously axisymmetric, which change the topology of the magnetic field in the vicinity of low order rational magnetic surfaces (q = 1.5, 2). These modes lead to islands which grow from a small seed width to a much larger width when β_N reaches values much lower than 3.5. The observed limit in present experiments is around 2.5 but it may decrease with ρ^* by a factor ~ 3. Nevertheless, the stabilisation of these "neoclassical tearing modes" (NTMs) has been achieved by a localised plasma current addition, driven from electron cyclotron waves on the specific magnetic surfaces, a method to be applied in ITER.

The existence of small-amplitude non-axisymmetric error fields produced by residual asymmetries in the magnetic coil positions, can lead to the development of large magnetic islands, again on low order rational magnetic surfaces, and subsequently to disruptions. Therefore, these error fields should be eliminated by appropriate correction coil currents which produce a controlled small amplitude helical field.

Disruptions are abrupt uncontrolled events, involving a rapid cooling of the plasma. Growing, large amplitude, islands overlap and lead to complete ergodisation of the magnetic field lines, thus a large heat flow occurs along field lines to the boundary walls, cooling the plasma and leading to an influx of impurities through interaction with the walls. This is followed by a rapid decrease of the plasma current; its decay time depends on the amount of losses through impurity radiation, the faster decay corresponding to the lower plasma temperature. Simultaneously, electrons can be accelerated to large energies by the electric field associated with the decrease in current in these low temperature plasmas, and lead to large runaway currents, if the confinement of energetic electrons is not limited by the magnetic fluctuations which may remain from the previous field ergodisation phase.

3.6 Divertor and Power Exhaust

The magnetic field configuration in Figure 2.1-5 shows closed nested magnetic surfaces with increasing inside volumes from the plasma axis until a separatrix occurs, outside of which magnetic surfaces are open. The particles diffusing out of the plasma through the separatrix flow along the field lines until they hit a target, localising the plasma contact with the wall to a large distance from the plasma along field lines (a few times the torus major circumference).

Along these field lines, the power flow is very high and if it were not for the possibility of volume power losses, the power density on the target (even taking into account its inclined position and the flux expansion due to a smaller B_p) would be too large for the capability of heat removal and the surface material temperature. This power should remains below 10 MW/m^2 on average. With no power losses, the temperature gradient along the field lines remains small, the pressure constant, and the plasma temperature at the target very high: this is the so-called “attached plasma” divertor operation, not acceptable for ITER.

On the contrary, if the plasma density is high enough at the separatrix, the possibility of radiation losses from impurities, and from ionisation of a large neutral density built in front of the target, provides a new situation, the so-called “detached plasma”. Towards the divertor target, the pressure along field lines decreases, the plasma density increases significantly and the plasma temperature at the target becomes very low (a few eV): the power crossing the separatrix becomes distributed by radiation (and charge exchange neutrals) onto the much larger surface of the divertor side walls, and the power density to the divertor target can remain inside reasonable limits.

In the latter conditions, the impurities, which are removed from the target by erosion and ionised by the plasma, contribute to the radiation losses, and thus to the decrease of the plasma temperature. Because this erosion increases with the particle energy impinging on the target, the process in itself may be self-regulating, as modelled in the case of a carbon target material. Moreover, these impurities are mostly stopped from flowing upwards along the field lines and entering the plasma by the hydrogen flow towards the target. This is the main function of the divertor, to screen the plasma from impurities due to plasma-wall interaction.

Another function of the divertor is the control of plasma density, and in particular the removal of the helium reaction product, the density of which should remain small (a few % of the electron density) in order not to dilute the reacting ions D and T, for a given electron density.

These helium particles are born at 3.5 MeV. They should become thermalised in the plasma at some keV, and provide, mostly by interaction with the plasma electrons, the heat source needed (in addition to the auxiliary heating power) to keep the plasma temperature constant by compensating for its power losses. This helium ash should not be lost to the boundary at high energy, through the action of specific instabilities or because of a large ion gyration radius and too large a magnetic field ripple (along field lines) due to the discreteness of the TF coils. This last source is minimised in ITER by ferromagnetic inserts, installed in the shadow of each TF coil, while the first one should not be detrimental in ITER according to the present understanding (with the probable values of He pressure and its gradient).

Due to the flat density profile and the small scale turbulence present in the core, the He ions when created in the plasma volume are driven to the boundary, then flow into the divertor, where the high neutral particle density allows an easier pumping at high pressure ($\simeq$ 1 Pa).

Together with the He, a large flow of D and T are pumped, and the plasma density is the result of this outwards flow against an inward flow coming from gas fuelling near the separatrix and/or solid (D or T) pellet periodic injection (a few tens of mm^3 at a few Hz). These pellets should be able to reach the plasma core, well inside the H-barrier: it will be the source of particles, which will allow a density gradient to be built; where there is no source, the density is flat.

It is generally observed that the plasma density is experimentally limited on average across the plasma width by the so-called Greenwald limit ($n_{GW} = I_{MA}/\pi a^2 \times 10^{20}/m^3$). The fusion power being quadratic with the density, it is important, if possible, to provide a peaked density profile, for a given average.

4 Functional Role of Systems

The preceding tokamak physics issues are linked with the hardware systems necessary to be installed in ITER, and with their functional requirements and implementation.

4.1 Magnets

The plasma is confined and shaped by a combination of magnetic fields from three origins: toroidal field coils, poloidal field coils and plasma currents. Aiming in ITER at steady-state operation, all the coils are superconducting; copper coils would require too large an electric power to be acceptable for ITER as well as for a future reactor.

4.1.1 Toroidal Field Coils

The nested magnetic surfaces are able to confine a plasma pressure equivalent to a few atmospheres, with a density 10^6 times smaller that in the atmosphere ($n = 10^{20}/m^3$, $T \approx 10$ keV). For an average β of 2-3%, the toroidal magnetic field value on the plasma axis is 5.3T, which leads to a maximum field on the conductor ≤ 12 T. Because of this high field value, Nb_3Sn is used as superconducting material, cooled at 4.5K by a flow of supercritical helium at ~ 0.6 MPa. The total magnetic energy in the toroidal field is around 40 GJ, and leads to very large forces on each coil which are restrained by a thick steel case to resist circumferential tension ($\approx$ 100 MN) and by constructing a vault with the inboard legs of all 18 coils (the large centripetal forces are due to the 1/R variation of the toroidal field). The compressive stress levels inside this vault are large, and therefore the side surfaces of each coil should match one another as perfectly as possible.

The coils are linked together (Figure 4.1-1) around their lengths (except to provide access to the vacuum vessel through ports) by specific bolted structures, and by two compression rings made of unidirectional fibre glass, which provide an initial radial force on each coil (2 x 30 MN) in the absence of the current in the coils.

This very robust assembly is provided to resist the toroidal forces induced by interaction of the TF coil current with the transverse poloidal fields from plasma and poloidal field coils. These forces produce a distribution of torque around the axis of revolution proportional to the magnetic flux crossing unit length of the TF coil (the net torque is thus 0). These local forces are pulsed, and therefore fatigue is a concern for the highly stressed structural steel of the coils. These forces, due to the highly shaped plasma, are largest across the inboard coil legs, mostly at their curved parts, where they are resisted by the friction between coil sides (under high compression) and by specific keys.

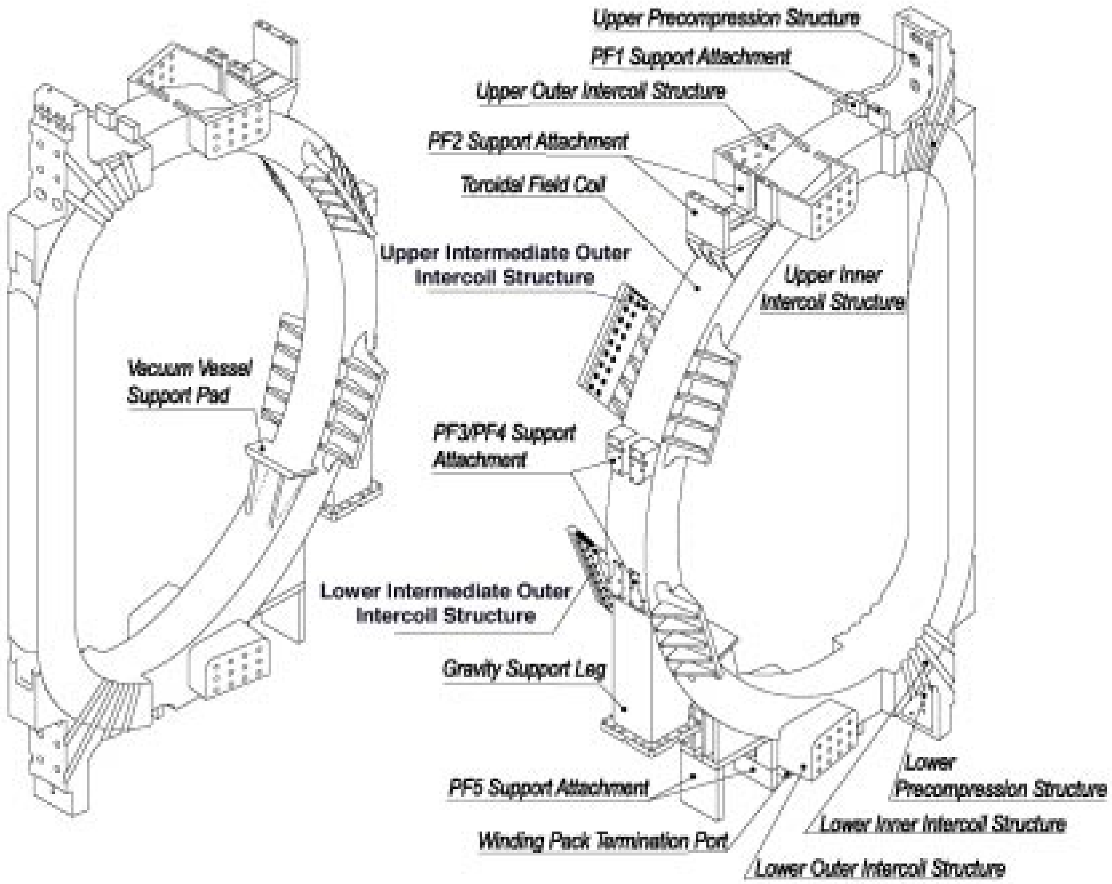


Figure 4.1-1 TF Coil Structure

4.1.2 Poloidal Field Coils

The plasma shape is controlled by the currents distributed inside the six modules of the central solenoid (CS) and the six large coils placed outside the TF coils. All these axissymetric coils use superconductors cooled by a flow of supercritical helium at 4.5K and 0.6 MPa.. Nb_3Sn is used in the CS modules, NbTi in the PF coils, where the maximum field value is lower than 6T.

The magnetic configuration provided by these currents is such that the plasma toroidal current will experience a vertical force as soon as its centre is displaced vertically, and this force will increase with the displacement: the plasma with its elongated shape is in a vertically unstable equilibrium.

Stabilisation of the plasma vertical position can be achieved in the following way. First, any plasma movement, associated with small changes of its energy content, induces eddy currents in any axisymmetric conducting surface surrounding the plasma, i.e. the two-shell vacuum vessel, which reacts to slow down the displacement. These conducting surfaces are shaped in order that the current distribution can provide a neutral equilibrium position, near the plasma centre of gravity, for most expected plasma energy changes, so as to minimise the sources of instabilities.

Second, using a feedback position control system, the currents in the 4 large poloidal field coils will be changed by feeding them with a fast power supply , in an antisymmetric way, across the plasma equatorial plane. These changes provide an additional radial magnetic field and lead to a restoring force on the plasma towards its controlled position.

Moreover, the plasma shape can be similarly feedback controlled, by an appropriate action on each coil voltage by its own distinct power supply. The gaps (Figure 2.1-5) between the plasma boundary and the walls are measured at six critical positions, and brought back to a prescribed value after an excursion due to a plasma internal change (loss of or change in current distribution/internal inductance, l_I, loss of plasma thermal energy).

In the inductive scenario, the plasma current is generated by the magnetic flux change inside the toroidal plasma annulus. This flux swing is largely realised by the CS coil, which will see a complete inversion of field from +13.5 T to - 12 T in the central modules. The flux consumption during plasma initiation and current increase will be reduced by providing a few MW of plasma heating through electromagnetic waves, in order to secure a sufficient flux variation available to sustain the current flat top during at least 400s at a plasma current of 15 MA.

After the plasma current is set up inductively, a non-inductive scenario can follow, in which the plasma current flat top is extended towards steady state by driving the current externally. This will be achieved by high energy (l MeV) beams of neutral D^0 injected at a small angle to field lines, which become ionised along their path across the magnetic field. This can also be achieved by toroidally propagating electromagnetic waves (at ion and electron cyclotron frequencies, or at the lower hybrid frequency), in addition to the “bootstrap” current linked to the plasma radial pressure gradients. The preceding sources of different radial current distributions deposit large amounts of power at specific locations in the plasma, and this has to be done in a way compatible with the necessary plasma pressure profiles and their allowable rates of change. A complete scenario for steady state operation in ITER with Q=5 is yet to be consistently developed. Nevertheless, the non-inductive current drive systems provided in ITER should be able to accommodate the steady state operational requirements (for over 2000 s).

4.1.3 Error Field Correction Coils

As mentioned previously, the need to correct imperfections in the magnetic field symmetry, due to the imperfect positioning of the TF, CS and PF coil currents, requires the use of “correction coils”, able to provide a helical field of a few 10^{-5} times the TF value. The Fourier components of toroidal and poloidal modes are n = 1 in the toroidal direction, and a distribution between m = 1, 2 and 3 in the poloidal direction. These coils are composed of 3 sets of six coils, around the torus, above and below the TF coils. The same coils can be used to stabilise possible resistive-wall modes, which happen to have the same geometry as the error fields to be corrected, but a much faster time variation. These coils counteract the MHD instabilities which are not stabilised by the conductive walls, on the longer time scale associated with the wall resistance.

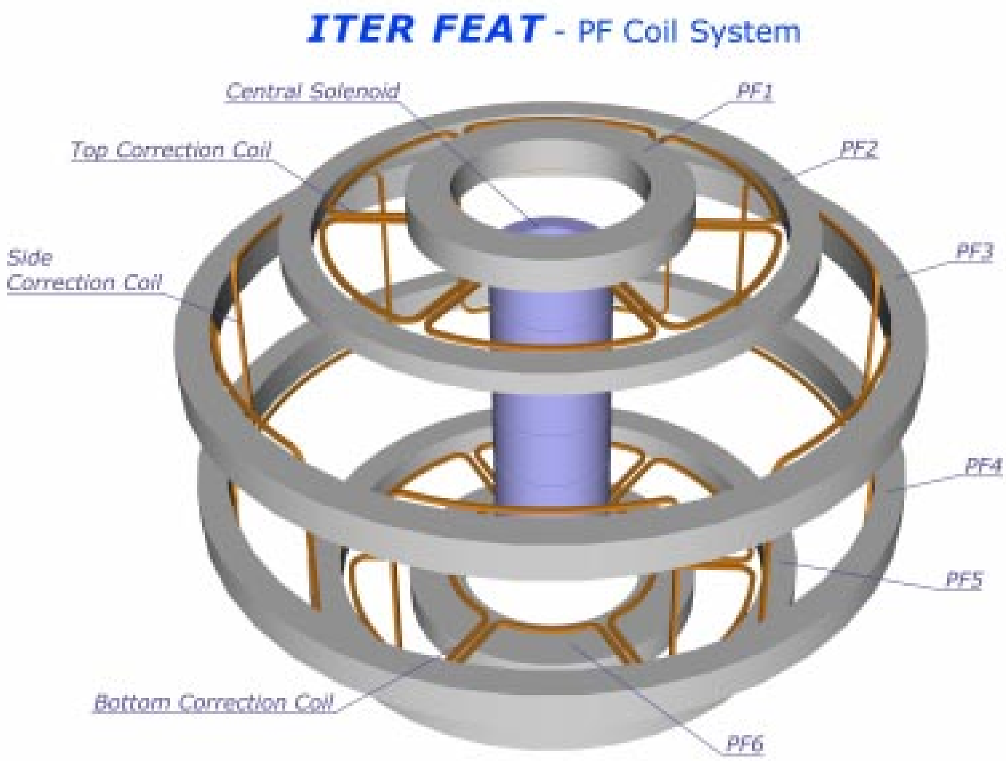


Figure 4.1-2 ITER Error Field Correction Coils

4.1.4 Superconducting Coil Protection

The superconductor of all coils should be protected against local overheating, should the coil current continue to flow after a local transition from superconducting to normal conducting state due to an off-normal local energy dump. In this case, after identification of a resistive voltage across the coil terminals increasing with time, an external resistor is switched in, dumping rapidly a large part of the coil magnetic energy. The time constant of this fast emergency discharge should be small, in order to minimise the energy dissipated into the coil and to limit its local temperature increase, but there is a minimum value for this time constant due to the maximum voltage through the coil terminals and the induced current (and related forces) in conducting material magnetically coupled with the coil. One example of this limit comes from the forces applied to the vacuum vessel due to the large poloidal current induced in the vessel shells by the fast discharge of all TF coils.

In addition, all these coils should be protected against the heat coming from their surroundings. Therefore, a large cryostat puts all the coils in a vacuum good enough to limit convection, and a thermal shield, cooled at about 80 K by a flow of helium, is provided between the coils and hot parts to shield against their radiation. The geometry of this thermal shield is obviously complex, but the avoidance of radiation hot spots is compulsory to limit the already large amount of power to be removed from the coils at 4.5K. This permanent heat load (13.5 kW) due to radiation, and conduction through supports, adds to the non-ideal efficiency of the circulation pumps feeding the supercritical helium in each coil.

4.1.5 Superconducting Coil Cryogenic Cooling

In addition, there is a large pulsed heat load on the coils from two origins: the neutron flux produced by the fusion reaction and attenuated by the blanket and vessel shields, and eddy currents induced by any field change in the coil superconductor and steel cases during the operational scenario of the plasma pulse, and even more during a disruption ending the pulse.

However, the cryogenic plant, producing the cold helium through expansion of a high pressure flow in turbines with brakes, is essentially a steady state system. Therefore, between the coils and the cryogenic plant, an energy storage should be present to buffer the large pulsed loads, and to control the transfer time of this energy to the cryogenic plant.

The energy storage is mainly provided by the large steel mass of the TF coil cases, and by the temperature variation of the liquid helium bath which cools the supercritical helium flow through heat exchangers. The extra energy dumped into the coils at 4.5 K during a nominal pulse amounts to 19 MJ; and a plasma disruption can add a further 14 MJ. Due to the assumed duty cycle, the time average load on the cryogenic plant (all users) amounts to 41 kW.

4.2 Vessel and In-vessel Systems

4.2.1 Neutron Shielding

The 14 MeV neutrons, i.e. 80% of the fusion power produced, escape from the plasma. This power should be transferred to a water coolant, and subsequently to the environment, by their collisions with the materials present around the plasma (mostly steel and water) in the blanket modules and in the vacuum vessel. The neutron power, not absorbed in these two shields, will be dumped in the TF coil structure at very low temperature, and should be absolutely minimised.

In addition to inelastic collisions, the neutrons will be absorbed by some nuclei, which will become activated and later radiate energetic γ rays according to their specific properties. Neutrons, not absorbed in the radial thickness of the blanket, VV and magnets, or leaking through gaps, will be absorbed outside and induce activation in the cryostat, a process which should be limited as far as possible, to allow human access, in case of an unexpected need for repair. Therefore, the shielding thickness (and attenuation efficiency by optimising the volume ratio between steel and water) has been carefully chosen, and its variation along the poloidal length optimised, to match the above two goals.

The thickness distribution between blanket and vessel comes from the need to allow rewelding of the vessel inner shell until the ITER end of life. This requires a low enough (around 1 appm) helium content (due to n, α reactions) in the vessel steel material. Accordingly, the blanket thickness is set at 45 cm, and gaps maintained as small as practicable.

4.2.2 Blanket Modules

The blanket is divided into two parts: a front part separable from a back one (Figure 4.2-1). The back part of around 30 cm thickness is a pure shield made of steel and water. The front part, the “first wall”, includes different materials: 1 cm thick Be armour, 1 cm Cu to diffuse the heat load as much as possible, and around 10 cm of steel. This will become the most activated and tritium-contaminated component of ITER. It could be in contact with the plasma in off-normal conditions, and thus can suffer damage from the large heat locally deposited, and may have to be repaired or changed.

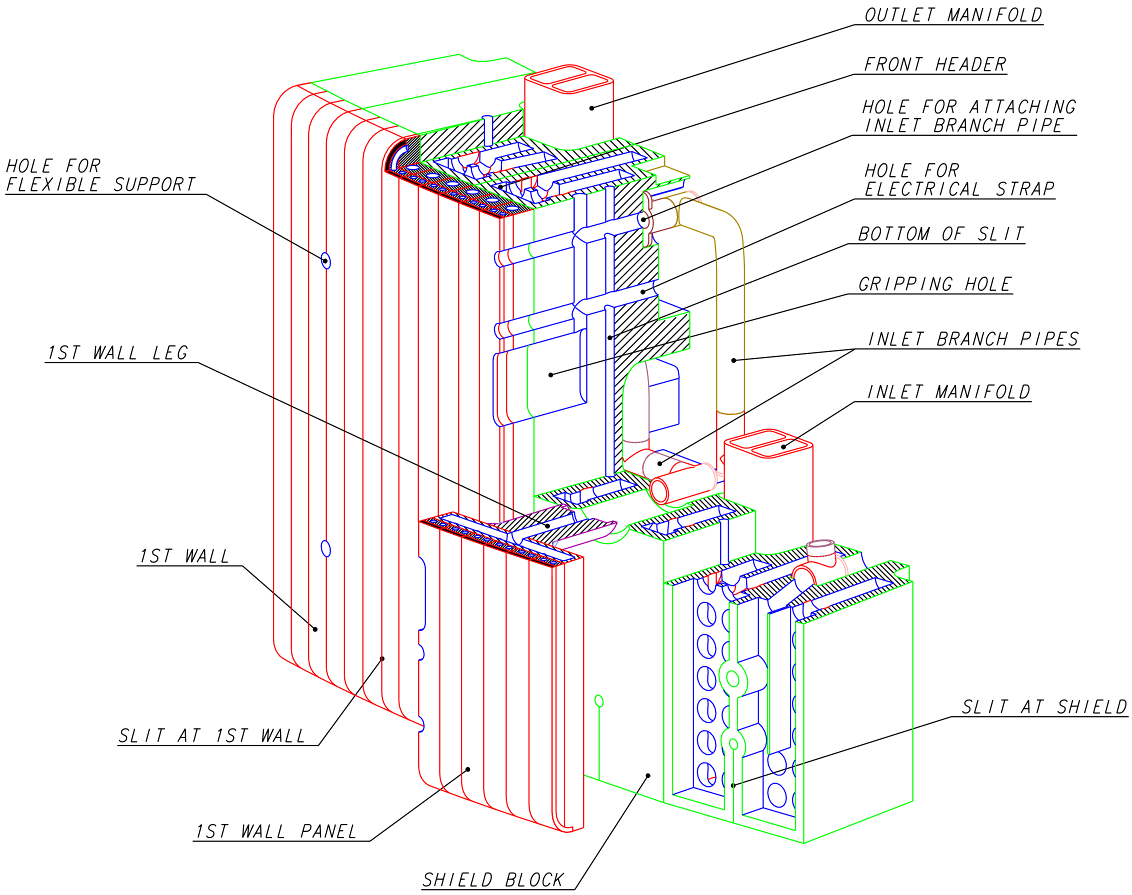
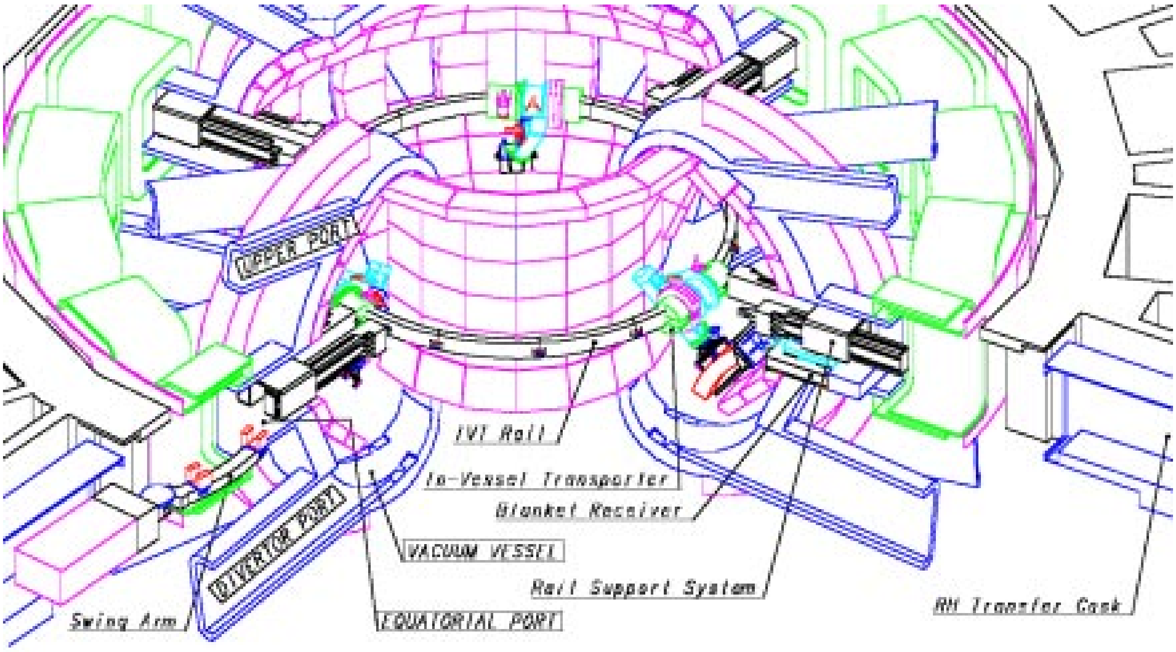


Figure 4.2-1 Blanket Module

4.2.3 Blanket maintenance

In order to allow a practical method of maintenance, this blanket has to be built in modules (~ 420 in total) with a maximum weight of 4.5 t (each with about 1.5 m^2 facing the plasma). Each module is attached to the vessel by 4 flexible links stiff radially but pliant against toroidal or poloidal motions. This flexibility is required because, across the blanket thickness, the absorbed power density decreases sharply and, whereas the water cooling redistributes the heat progressively toward a uniform temperature, at the end of the pulse the front part becomes necessarily the colder part. Thus the blanket module suffers an alternating bowing effect during each plasma pulse. The toroidal and poloidal loads (i.e. during a disruption) acting on the module are therefore reacted by additional key elements provided with sufficient compliance gaps.

The maintenance of a blanket module is done by first removing it from the vessel. A vehicle, equipped with an end effector, moves along a toroidal rail which is deployed toroidally along the vessel centreline, from a specific cask attached to an equatorial port door of the vessel (see Figure 4.2-2). The end effector is able to cut the connection to the water pipe feeders, to unbolt the module, and to bring it to an equatorial maintenance door. There it will be transferred into a cask, and from there, to the hot cell for repair or replacement. The cask operates by docking and undocking automatically to the ports of the vessel and of the hot cell, avoiding any atmosphere contamination to the environment. The same procedure is used for removal of any equipment installed in any equatorial or upper port of the vessel, i.e. heating launcher, diagnostics, or tritium breeding test blanket.



From Drawing 23.182.1

Figure 4.2-2 Blanket Maintenance Concept

4.2.4 Divertor

The same modularity philosophy and maintenance procedure are used for the divertor. The cassettes (54 in total) are removed from the vessel at three lower ports, to which they are brought by a toroidal mover on specific rails, to which the cassettes are finally attached.

Besides providing shielding of the vessel, the modular cassettes (Figure 4.2-3) support the divertor targets, and very high heat flux components, built with high conductivity armour of carbon fibres and tungsten.

These armour materials can be eroded by the plasma particles, mostly during short pulses of high heat loads, associated with ELMs or plasma disruptions. This erosion process not only will call for replacement from time to time of the worn out divertor targets, but also will create dust, and in particular tritiated carbon dust. Studies are going on to define the best way for removal of this dust, mostly to limit the tritium inventory inside the vessel, and the possibility of metallic dust (Be, W) reaction with hot water during an accidental in-vessel water leak, which could lead to hydrogen formation.

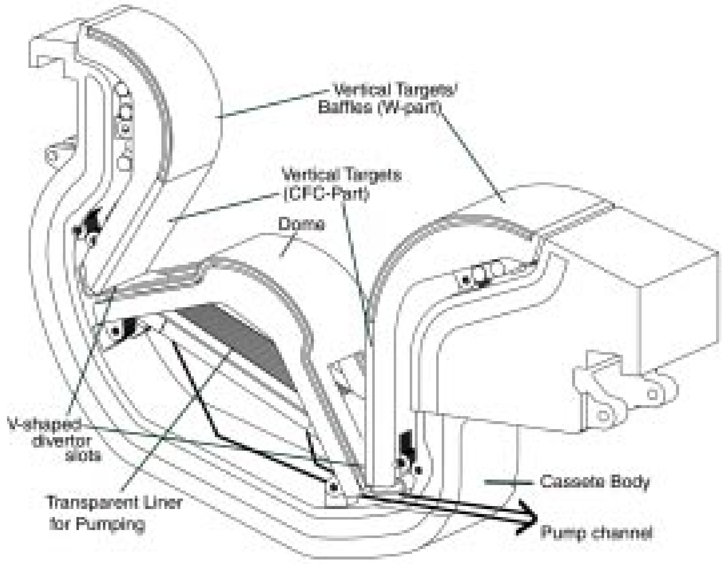


Figure 4.2-3
Divertor Cassette

4.2.5 In-vessel Component Water Cooling

Each divertor cassette is separately cooled by water, with feeder pipes connecting to the manifold outside the vessel and cryostat. Two, sometimes three blanket modules are similarly fed by separate pipes installed on the plasma side of the inner shell of the vacuum vessel. This arrangement leads to handling a large number of small size pipes, but allows the identification of possible water leaking modules or cassettes from tests done from outside the cryostat, a crucial procedure to be able to localise the leaks in vacuum.

The pressurised coolant water input is maintained around 100°C permanently; the output temperature during a pulse at nominal fusion power will be around 150°C. At the end of a pulse, control valves allow to short circuit the large heat exchanger to the heat rejection system, in order not to cool the in-vessel components below 100°C. During standby, the coolant flow will be reduced, using a different pump, to 10% of the flow during the pulse, and the flow in the heat rejection system reduced to 25% of its normal value. From time to time, after a maintenance period, the pressurisation will be increased, and the water coolant used to heat and bake the in-vessel components to 240°C.

4.2.6 Cryogenic Pumps

At the divertor port level, cryogenic pumps operating at 4.5 K are installed, which have the capacity to pump hydrogenic atoms as well as helium by adsorption and condensation. These pumps are equipped with a closing valve. The pumping performance can be varied and the condensed gases removed by heating the pumping panels to 80 K and pumping them with a roughing pump when the valve is closed. For long plasma pulses, this procedure will be carried out in sequence through all the installed cryogenic pumps, one after another, in order to limit the amount of hydrogen in each pump below the deflagration level in case of an accidental ingress of oxygen, which corresponds to pumping 200 Pam^3s^{-1} of DT for 450 s.

4.2.7 Vacuum Vessel

The vacuum vessel has a multiple role, namely:

- to provide a boundary consistent with the generation and maintenance of a high quality vacuum, necessary for limiting impurity influx into the plasma;

- to support the in-vessel components and their resultant mechanical loads;
- to participate in shielding against neutrons, and to remove the corresponding power during a pulse, and moreover to remove the after-pulse decay heat of all in-vessel components in case of there being no other coolant available;
- to provide a continuous conductive shell, fitting tightly to the plasma, for plasma stabilisation;
- to provide all access to the plasma through ports, for diagnostics, heating systems, pumping, water piping, etc…;
- to provide the first confinement barrier for tritium and activated dust with a very high reliability.

All these functions are essential, and they require a very robust vessel mechanical design analysed for stresses in all possible normal and accidental conditions. The vessel is built with two shells linked by ribs and fitted with shielding material.

To ensure cooling reliability, two independent water loops are used which can remove by natural convection the decay heat from all in-vessel components (if they are not cooled directly). The vessel water temperature is maintained at 100°C (at 200°C during baking of the in-vessel components), limiting to 50°C its difference with the in-vessel component cooling temperature.

4.2.8 Vacuum Vessel Pressure Suppression

In the case of a water pipe rupture inside the vessel, the pressure will increase, but be limited below 0.2 MPa by the opening of rupture disks and communication with a large enclosed volume located above the vessel and half-filled with water, in which the steam will be condensed (the vacuum vessel pressure suppression system - VVPSS). Simultaneously, liquid water condensed in the vessel will be driven into drain tanks located at the bottom of the tokamak building.

4.3 Mechanical Loads and Machine Supports/Attachments

Part of the technical difficulties in the ITER design are due to the large mechanical loads which are applied to the various components.

The mechanical loads acting on ITER fall into four categories.

1. Inertial loads which are due to gravitational and seismic accelerations,
2. Kinetic pressure loads due to coolant pressure and atmospheric pressure,
3. Thermal loads,
4. Electromagnetic loads, which are usually a strong design driver, either static (as in TF coils) or dynamic, acting on the magnet and on all conducting structures nearby due to fast or slow transient phenomena such as plasma disruptions and VDEs.

The support scheme of tokamak components must be designed to minimise the reaction of each support to loads on the component. In interconnecting components, a proper load path must be chosen to maximise the stiffness associated with the load path itself. The support hierarchy is schematically drawn in Figure 4.3-1. All core components of the machine are attached to the TF coil cases.

The support schemes for the magnet and the vacuum vessel must allow for their changes in temperature from the time of assembly, i.e. the radial shrinkage of the magnet and radial growth of the vessel, and provide adequate resistance to seismic and disruption forces. All supports of the machine core are flexible in the radial direction and stiff in all others.

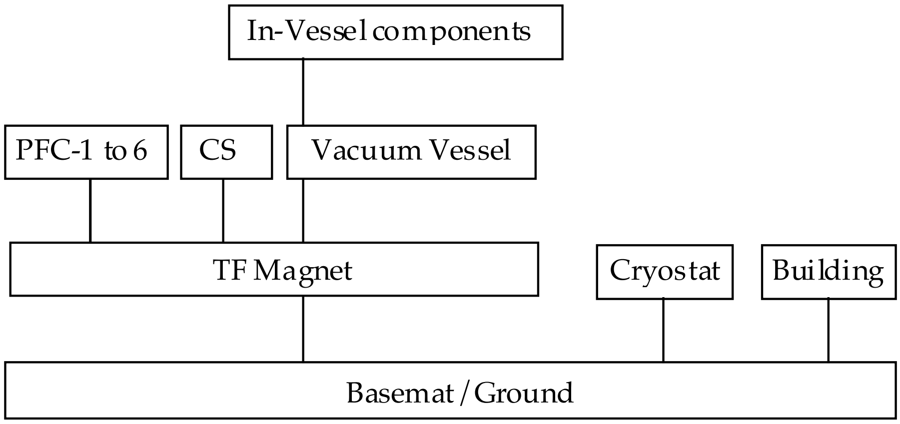


Figure 4.3-1 Schematic of Supports Hierarchy

4.3.1 Seismic Loads

Earthquakes simultaneously produce vertical and horizontal random ground motions which are statistically independent; the horizontal ones having the most important consequences. Even if the ground peak horizontal acceleration is a small fraction of gravity, a seismic event is in fact, in many cases, the most demanding loading condition, in particular for the interface structures (e.g. supports). Under horizontal excitations with a relatively broadband spectral content in the range 1-10 Hz, resonances occur in component motions. The tokamak global structure exhibits then oscillatory modes which involve horizontal shearing as well as rocking motions

In addition, the relative distance between components must be maintained (between vessel, TF coils, thermal shield etc…). Normalised seismic conditions of 0.2 g ground acceleration (at high frequency, 33 Hz) have been applied to ITER and found acceptable. In case of larger values, the use of horizontal seismic isolators below the building basemat has been shown to be effective at lowering the peak acceleration to acceptable values, at the expense of larger displacements.

4.3.2 Electromagnetic loads

The restraint of the loads occurring on the TF coils, either static in-plane from the toroidal field itself, or out-of-plane cyclic due to their interaction with the poloidal field crossing them, and the consequence of a fast emergency TF energy discharge on the VV stress level, which provides a limit to operating conditions, have been described above.

Other important electromagnetic loads are associated with transient phenomena which are consequences of changes in plasma current, internal energy or position. They act on the poloidal field coils and all conductive structures close to the plasma (blanket modules, divertor cassettes, VV).

For slow transients (time scales longer than those associated with these structures), there are negligible induced currents, and thus no net force on the PF/CS magnet assembly as a whole. In each PF and CS coil, vertical forces are reacted through the TF coil structure (the shortest path) and radial ones by the development of a toroidal hoop stress inside each coil.

For fast transients, such as plasma disruption or loss of vertical position control (vertical displacement event, VDE), significant currents are induced in conducting structures, and their interaction with the toroidal or poloidal magnetic field develops significant forces and stresses.

In the case of disruption, the load severity is the larger the shorter is the assumed current quench duration (lowest plasma temperature after the thermal quench). In the case of a VDE, load severity will depend on how large is the plasma displacement across the destabilising poloidal field, without a decrease of the plasma current. Again, in all these cases the forces developed between the coils and the vessel are restrained through the shortest path through the TF coil structure, taking advantage of the direct link between these components.

Detailed numerical studies, under conservative assumptions, of all these important events, have led to the following conclusions:
- the plasma control system will be capable of maintaining the plasma vertical position for all nominal plasma disturbances including minor disruptions; therefore VDEs should occur only during a major disruption or a failure of the control system;
- during a major disruption, the plasma will move inward radially and upward vertically, but vertical forces will be much smaller than for a downward VDE, which could occur only in the absence of control; if a "killer pellet" can be used, it will trigger plasma quench early during its motion, and thus limit the loads.

The global mechanical structure of ITER is strong enough to resist the most conservative assumed case: a combination of loads which can occur simultaneously, being trigged by one single event, i.e. an earthquake for example.

4.4 Fuel Cycle

For delivering 500 MW of total fusion power, about 0.1 g of tritium will be burnt every 100 s. But, if the maximum throughput of fuelling and pumping are used to satisfy the divertor operational conditions, more than 25 g of tritium will be injected into the vessel during the same 100 s, and removed by the cryogenic pumps.

The need is obvious to process the pumped gases on line, to remove impurities and separate the tritium, and to store it for recycling. This fuel cycle is shown in Figure 4.4-1. It includes first a permeator to separate impurities from hydrogen in line with the pumping exhaust. Then, the impurity flow is processed before sending it to atmosphere, with an ALARA (as low as reasonably achievable) content of tritium. The hydrogen flow is processed to separate the different masses, by isotope separation through cryogenic distillation. This part of the plant is optimised to minimise the tritium inventory as far as possible, compatible with the isotope separation ratio required (not very high) and the global throughput. For nominal pulses (< 450 s), the fuel cycle does not operate as a steady state, online system. The outlet stream of hydrogen isotopes from the permeator feeds a buffer storage tank, before being processed on a longer timescale by the isotope separation system. For longer pulses, on the contrary, steady state operation could be reached using a direct feed from the permeator output stream.

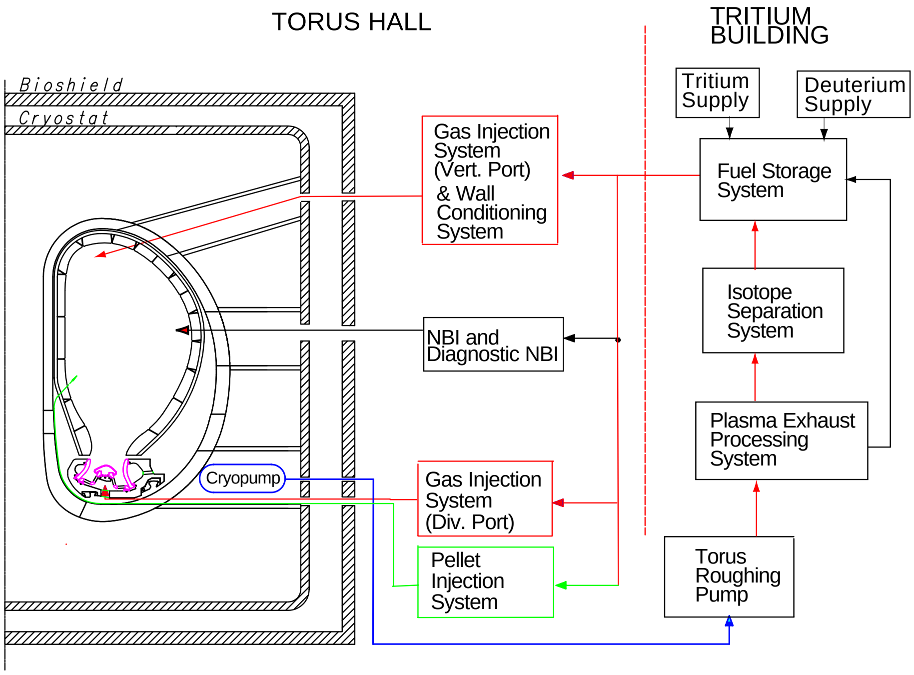


Figure 4.4-1 Block Diagram of Fuel Cycle

Segregation of tritium-containing equipment in separated structures, with limitation of the local inventory and robust confinement barriers, is appropriate for safety reasons. The storage of D_2, DT and T_2 is achieved in many parallel canisters, and adsorbed on ZrCo beds, which can deliver rapidly the required flow for plasma fuelling. Their tritium content is measured by calorimetry with around 1% accuracy.

Tritium accountability in the fuel cycle is an important issue, because a part of the tritium injected in the plasma can remain in the vessel trapped by co-deposition with carbon dust. This tritium in-vessel inventory should remain below a ceiling, fixed presently at 450 g, and measures for dust removal are under study, as already mentioned.

In addition, a small amount of tritium, adsorbed on all in-vessel surfaces, will be progressively desorbed, and recovered partially in oxidised form, from detritiation systems installed to limit the tritium content in the in-vessel atmosphere during maintenance, or in the hot cell atmosphere during component repair. The resulting tritiated water will be processed to reinject the tritium into the fuel cycle.

4.5 Tokamak Building

The buildings should provide the volumes and controlled atmosphere required for ITER assembly and operation. In addition, the tokamak building is important for its contribution to safety, by the following means.

First, a biological shield of borated concrete is provided around the cryostat to limit the radiation levels outside the pit to values insignificant for the activation of components, even if human presence will not be allowed, during plasma pulses.

Second, its role is essential as a further confinement barrier (even containment in this case), forming two concrete leak tight vaults around the neutral beam injectors and the water cooling system, or even as a third confinement barrier in the tritium building (the metallic equipment inside glove boxes provide the first and second barriers in this case).

Third (Figure 4.5-1), there is a differential pressure maintained in the different zones around the tokamak, according to the risk of being contaminated by an accidental release of tritium or activated material during operation or maintenance. In this way, the atmosphere should leak only from smaller to larger contamination levels. These differential pressures are maintained by the air conditioning system. When a release is detected in an area, this area will be isolated from the others and its atmosphere will be filtered and detritiated before being sent to the environment. This policy explains the design arrangement of a separate cell around each vessel port access, especially justified during the maintenance procedure of removing components from the vessel. In case of an accidental release, the normal air conditioning system is shut down, and a large detritiation process becomes active.

In addition, the concrete walls provide appropriate shielding against emission from activated components, during their automatic transport via cask from one vessel port to the hot cell (and back) through the galleries.

The very robust structure of the tokamak and tritium buildings groups the interdependent components within these buildings on a common basemat to better react seismic conditions. If the actual site has much harsher conditions than the generic site used in the design, the common basemat will be put on isolators and the acceleration amplification suffered by the components above will be maintained at the accepted design level, as indicated previously.

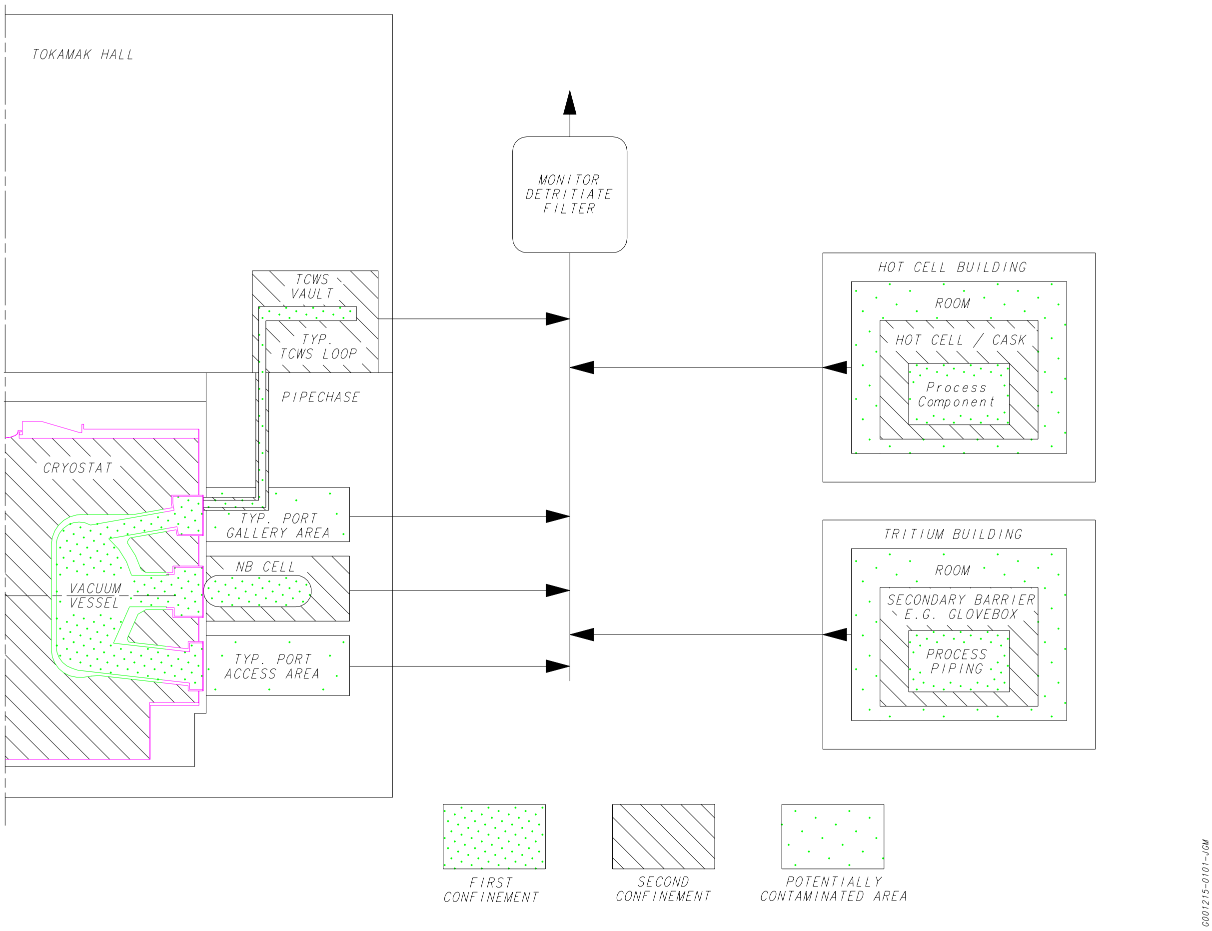


Figure 4.5-1 Schematic of Confinement Approach illustrating Successive Confinement Barriers that are Available

4.6 ITER Plant Operation and Control

Compared to today's experiments, the need for ITER to operate with a burning plasma under stationary conditions for more than 400s, while handling about 0.5 GJ of plasma thermal and magnetic energy, poses quite challenging and new constraints in the design of hardware that affects the control of plasma operation. The typical waveforms of a standard driven burn plasma pulse are shown in Figure 4.6-3.

Therefore, one of the most important objectives of plasma operation and control in ITER is the protection of tokamak systems against the normal and off-normal operating conditions, from safety and protection of investment viewpoints. Reliable control of the fusion burn conditions and provision for its rapid termination under off-normal conditions are crucial.

To support plasma operation, plant systems must be efficiently and reliably operated. In particular, the fuel cycle needs special attention in order to manage and control the tritium inventory within the system. In the water cooling system, the control of activated corrosion products and tritium content is very important. Although the basic mode of tokamak operation is pulsed, there are certain resulting requirements and constraints for operation of the tokamak supporting plant systems, which are intrinsically more steady state.

The ITER plasma control system comprises four major elements: control of scenario sequencing, plasma magnetic control, kinetic and divertor control, and fast plasma termination by a large amount of light impurity injections.

Control of plasma parameters can be characterised by three basic attributes of closed loop control systems – diagnostics, control algorithms and actuators as shown in Figure 4.6-1. The control algorithm is typically a proportional/integral/derivative (PIDS) feedback scheme. There are, however, alternate algorithms designed with more sophisticated optimisation procedures.

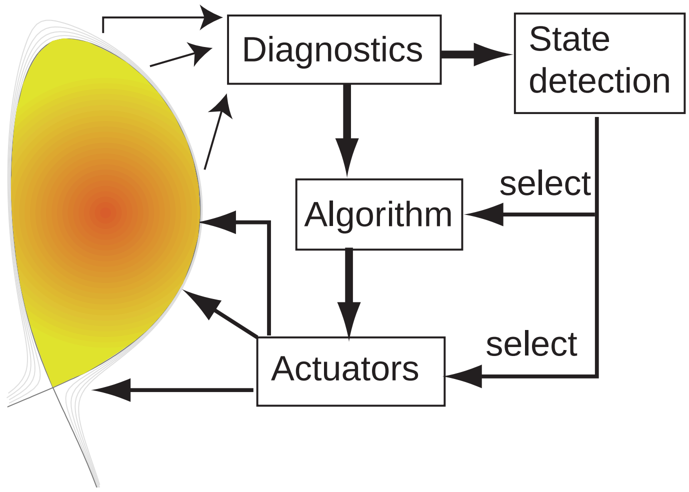


Figure 4.6-1 Plasma Feedback Control

Figure 4.6-1 introduces the concept of plasma state cognisance, and state-dependent control actions. Here the change of the plasma state can dynamically modify the control algorithms and choice of control actuators so as to more optimally control the overall plasma response. The implementation of state-cognisant control gives the control system a certain degree of autonomy. It will ultimately lead to a highly dynamic and state- and scenario-phase dependent 'expert system'.

A plasma control matrix for ITER to relate control actions or actuators and controllable parameters is shown in Figure 4.6-2. The vertical organization of the matrix reflects the division of the plasma control system into the four hierarchical categories mentioned above, namely scenario, magnetics, kinetics and fast shutdown.

PLASMA CONTROL MATRIX

● = Major direct effect

◎ = Appreciable secondary effect

○ = Possible secondary effect

blank = No appreciable effect or not applicable

	Control Action (Controllable Parameter or System) →	Scenario and Magnetics						Fueling and Exhaust						Auxiliary Heating and Current Drive (options)						Shutdown
	Measurable Quantity or Attribute to be Controlled	TF field (static, 0-5.3 T)	PF currents	PF voltages	Prefill pressure (D_2)	Startup EC (ω, P, t, Δt)	Error field compensation current	DT fueling (gas, into SOL)	DT fueling (shallow pellet, ~1 km/s)	Impurity fueling (He,Ne,Ar... to SOL) (gas)	DT divertor fueling	Impurity divertor fueling (Ne, Ar, ...)	Pumping speed (0-300 m^3/s)	NBI power (0-max)	ICH power (0-max, ω, mode)	ECH power (0-max, launch angle)	FWCD (power, radial location)	ECCD (power, radial location)	LHCD (power, $n_\parallel$)	Shutdown Pellet
1: Scenario 2: Magnetics	Plasma current, q_{edge}	●		●																
	Plasma shape (R, a,			●																
	Plasma shape (FW gaps)			●																
	IC coupling impedance			●				◎		◎	○								○	
	Plasma current initiation		●	●	●	●														
	Locked mode susceptibility		◎				●							●						
3(a): Core Kinetics	Plasma density							●	●	●	●	○	○	○						
	Fusion power							●	●	◎	●	◎	◎	○	○	○	○			
	He fraction									◎		○	●	◎	◎	◎	◎	◎	◎	
	Core D/T ratio							●	●	●										
	Core impurity fraction									●		◎								
	Core radiation fraction									◎		◎	●	◎	◎	◎	◎	◎	◎	
	Core plasma rotation (f_{rot})													●						
	W_{th} or $_N$ (at given P_{fus})	●						○	○					◎	◎	◎	◎	◎	○	
	Axial safety factor q(0)													◎			○	○	○	
	Current profile j(r)	●												◎			●	●	●	
	Sawtooth period	◎												◎			◎	◎	○	
3(b): Edge Kinetics	ELM period, magnitude			○				●	◎	●										
	n_{edge}							●	◎	◎			◎						◎	
	SOL flow							●	◎	◎			●							
	SOL radiation fraction									●		◎								
3(c): Divertor	Divertor power input							◎	◎	●	◎	●	◎	◎	◎	◎	◎	◎	◎	
	In-divertor radiation (x,y)									◎		●	◎							
	Target plasma (n,T)							●	◎	●	◎	◎	◎							
	Target power or temp.			○				●	◎	●	◎	●	●	◎	◎	◎	◎	◎	◎	
	Divertor neutral pressure			○				◎	◎	●		●	●							
	Divertor He fraction							◎	◎	●	◎	●	◎							
4: Shutdown	Fast P_{fus} and I_p shutdown																			●

Figure 4.6-2 ITER Plasma Control Matrix

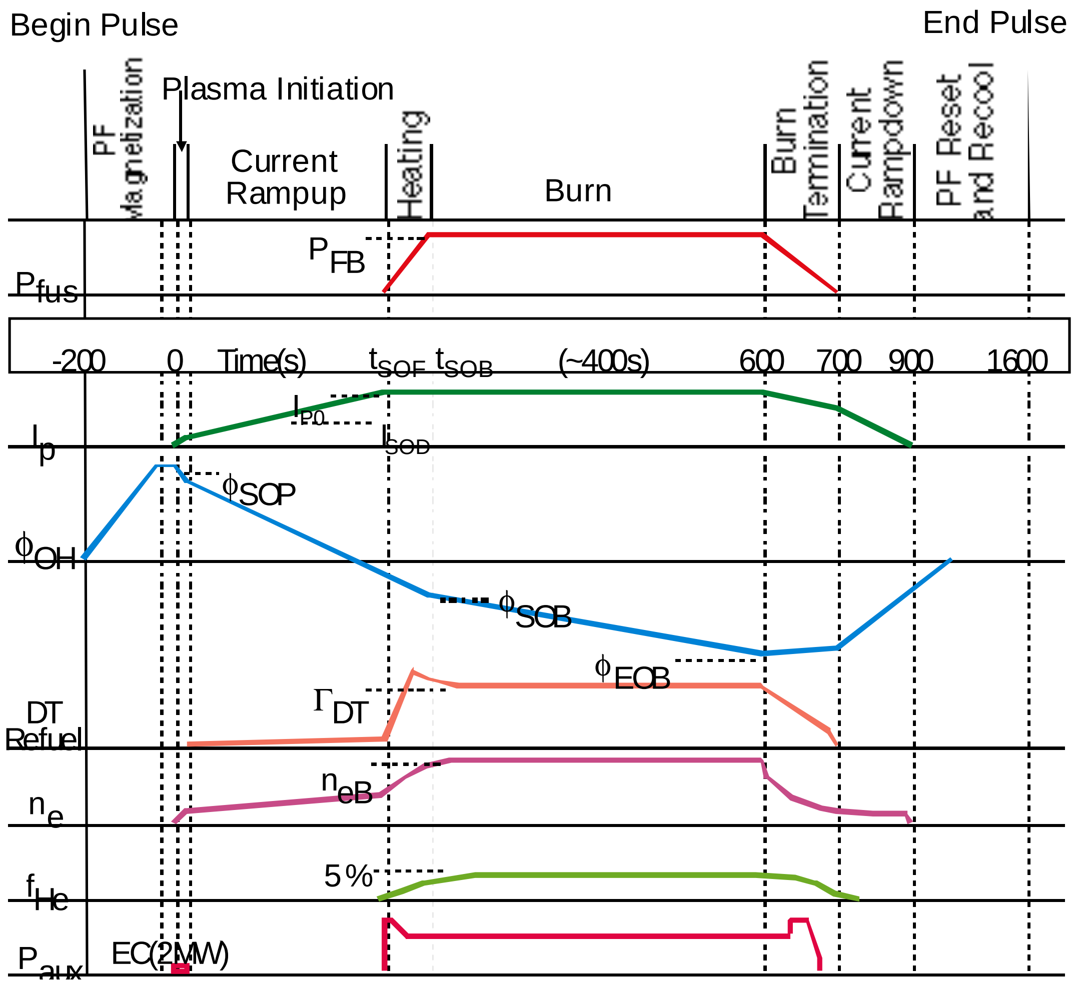


Figure 4.6-3 Waveforms for Standard Driven-burn Operation Scenario

4.6.1 ITER Plant Control System

The ITER plant operation is controlled and monitored by the "Command Control and Data Acquisition and Communication" (CODAC) system. The CODAC system consists of a centrally positioned supervisory control system (SCS) and sub-control systems dedicated to each plant subsystem under the supervision of the SCS.

The SCS provides high level commands to plant subsystems, and monitors their operation, in order to achieve integrated control of the entire plant. An interlock system, sometimes in parallel with the CODAC system, ensures plant-wide machine protection, as well as personnel protection, in case of off-normal events. It monitors operational events of the plant, and performs preventative and protective actions to maintain the system components in a safe operating condition. The interlock system is also hierarchically structured and has individual interlock subsystems which are dedicated to each plant subsystem under the central supervisory interlock system.

The integrated control of the entire ITER plant will be achieved by the CODAC, and the interlock system. A concept of the ITER control system is schematically shown in Figure 4.6-4.

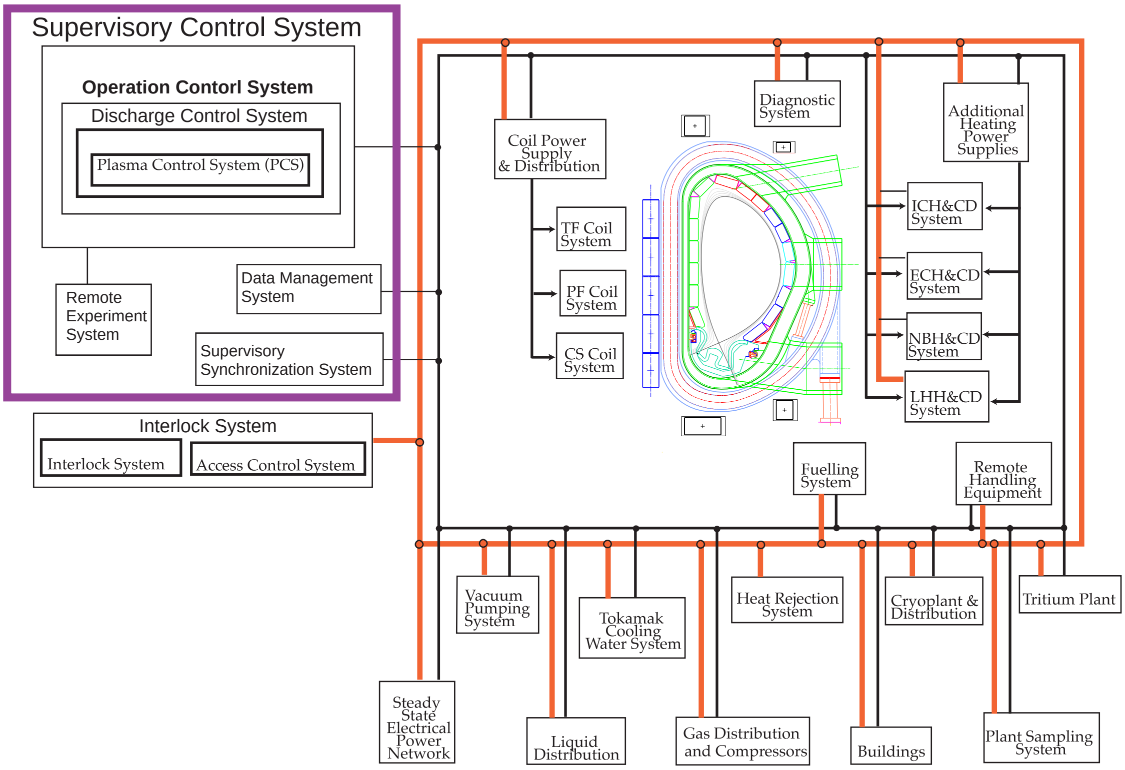


Figure 4.6-4 ITER Plant Control System

5 Construction, Commissioning and Decommissioning Plans

5.1 Introduction

The planning schedule for supply, construction/assembly, commissioning and decommissioning set out below depends on a number of assumptions detailed in the following. As the design progresses, decisions reached by the Parties may confirm or alter the assumptions that have led to its present status. The actual plan will therefore depend on the licensing procedure, as well as the organization and arrangements that will be put in place for the procurement/construction commissioning.

The construction agreement is expected to be signed at the end of 2002 or the beginning of 2003 following formal negotiations. The ITER legal entity (ILE) will be established after ratification of the agreement within each Party. This organisation will start the formal regulatory procedure and procurement process for the long lead-time items. The regulatory approval process, however, will remain speculative until a site is formally selected. As the site proposals are received before or at a sufficiently early stage of negotiations, it will be possible to assess the time needed for licensing in the various possible host Parties and the effects on the overall schedule. Since the start of the actual construction on the site depends upon when the licence to construct is issued by the regulatory authority, dates in the construction and commissioning plan are, therefore, measured in months from a start date ("T = 0") defined as the date at which the actual construction work of excavation for the tokamak buildings is started.

Furthermore, the following assumptions pertain at T = 0.

- Informal dialogue with regulatory authorities should be established and should orientate the technical preparation toward a licence application with a view to solving the major technical issues prior to establishment of the ILE. Documents required for the formal regulatory process are assumed to be prepared before the ILE exists, so as to allow the ILE to begin the formal regulatory process immediately after the establishment of the ILE.
- Procurement specification of equipment/material for the longest lead-time items and critical buildings are assumed to be finalized during the co-ordinated technical activities (CTA).
- Procurement sharing is assumed to be agreed among the Parties during the CTA so as to permit the placing of all contracts at the appropriate time.
- The construction site work starts immediately at T = 0. It is assumed that site preparation has been started sufficiently early by the host Party so as not to place constraints on the start of construction.

5.2 Overall and Summary Schedule

The overall schedule that leads up to the first hydrogen plasma operation is shown in Figure 5.2-1. It represents a reference scenario, which is a success-oriented schedule of procurement, construction, assembly and commissioning of ITER, based on the assumptions. The detailed construction schedule is developed to correspond to each procurement package specified for the cost estimate (see later). The schedule for each package includes procurement specification preparation, bid process, vendor's design (if appropriate), manufacturing (if appropriate), transport to site (if appropriate), installation and commissioning.

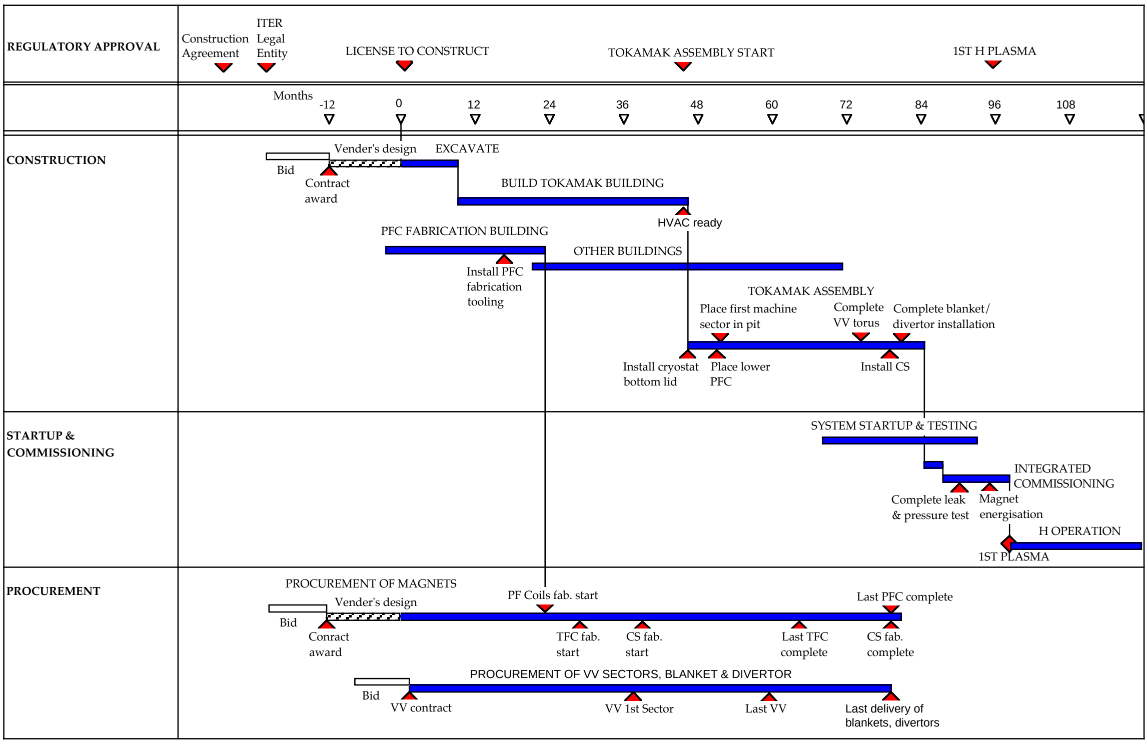


Figure 5.2-1 Overall Schedule up to First Plasma

5.3 Construction and Procurements

5.3.1 Procurement Assumptions

The lead-times for the different components of ITER vary widely. Also some items – including the buildings, parts of the cryostat, magnets, and vacuum vessel - are logically on the critical path irrespective of their time schedule, whereas others can be delayed until an ultimate date when they will become part of the critical path. In addition, cash flow may pose constraints in conflict with the need to make procurements at a date compatible with smooth planning of implementation. For the purposes of evolving this schedule, procurement is assumed to occur such that systems/components are delivered just in time, i.e. at the latest time, on the critical path, for assembly and installation/construction, in accordance with the construction logic. In a second evolution of this plan, some items could be moved earlier in the schedule to gain same margins, which would remove some items from the critical path line. However if cashflow peaks caused by critical items are too high, only an extension of the overall schedule is possible as a solution.

Another important assumption is that the placing of purchase orders is allowed on the establishment of the ILE, even before the licence to construct is awarded (T = 0), in order to allow vendor’s design and tooling preparation for the critical lead-time components and buildings. In reality, for non-safety-related items (e.g. magnets), manufacturing can even be started before the granting of a construction licence, provided it is clear one will eventually be granted. For safety-related items, however, construction can only start after the license to construct is issued, if necessary through a second contract. Documentation required for purchasing the various items will be completed in time for the scheduled procurement. The procurement bid process is assumed to take typically six to twelve months from the release of tender documents to industries to the awarding of contracts.

5.3.2 Buildings and License to Construct

The critical path of the plan is the regulatory licensing procedure and the construction of the tokamak buildings. In order to start excavation immediately at T = 0, the design of the complex must be complete by then. The contract for the vendor's preparation of the complex, thus has to be awarded at least twelve months before. Considering a period for the procurement bid process, the tender documents have to be released 18 to 24 months before T = 0. If the establishment of the ILE occurs 24 months before T = 0 and the license process can be completed in this period, the construction period defined from the establishment of the ILE to first hydrogen plasma discharge is ten years. If the regulatory process takes more than two years, the construction period becomes longer.

Excavation is to be completed within 9 months. Installation of the large pipes of the heat rejection system that are below grade will follow. The tokamak buildings must be functional, including cranes and HVAC system, by the end of 45 months from T = 0 in order to allow the timely start of tokamak assembly.

Two cryoplant buildings are to be built and serve dual purposes, as the PF coils fabrication buildings in the early stage, and cryoplant cold box and compressor buildings later. To maximise the time available for PF coil manufacture and to allow the cryoplant to be used for NB injector stand-alone commissioning as soon as possible, the construction of these buildings is also started at T = 0, or sooner for buildings not related to safety/licensing.

5.3.3 Procurement of Long Lead-Time Items

The tokamak building is ready for machine assembly at month 46. In order to start the pre-assembly of TF coils and vacuum vessel sectors in the assembly hall, the first two TF coils at least must be delivered by month 45 and the last two by month 62. It is essential to purchase an initial quantity of Nb_3Sn conductor to train the lines for the forthcoming serial strand production and to award the contract for the design of the TF coils manufacturing prior to T = 0, as soon as the ILE is established. This should be possible if (1) a sufficient number of contractors in the world have been qualified and trained, (2) the detailed specifications for manufacturing TF coils are fully available by the time of the signing of the construction agreement, and (3) procurement sharing will be agreed among the Parties before the signing of the construction agreement. This allows the call for tender to be issued immediately after establishment of the ILE.

Most of the PF coils are too large to consider their transfer from the factory to the site (unless both factory and ITER site have deep water access). Thus, fabrication on site is likely to be unavoidable. To save cost, the cryoplant buildings are used. However, the two smallest coils, PF1 and PF6, may be fabricated at an off-site factory. The lower PF coils, PF5 and PF6 must be ready to be placed at the bottom of the pit at the beginning of the tokamak assembly starting in month 46. The other coils are stored until the time of installation.

Nine (40º) sectors are shipped to the construction site and pre-assembled with the TF coils and vacuum vessel thermal shield (VVTS) in the assembly hall before being installed into the pit. Three VV sectors are simultaneously at different stages of manufacture. The manufacturing time for one sector including the welding of a port stub extension is 21 months. Another 3 months are taken into account for shipping and inspection for acceptance at the site. The first sector should be available by month 45 to start machine assembly. The manufacturing time for the 9 sectors is 42 months in total. The last sector should be accepted at month 62.

5.4 Tokamak Assembly

The assembly procedure is grouped in five activities, lower cryostat assembly, TF/VV/VVTS (machine 40º sector) sub-assembly in the assembly hall, integrated TF/VV/VVTS assembly in the pit, in-vessel component assembly after completion of the torus and establishment of the magnetic axis, and ex-vessel component assembly. It takes two and a half years to complete the assembly of the vacuum vessel torus with TF coils and VV thermal shields. About a year is needed to assemble the in-vessel components and ex-vessel components simultaneously after completion of the torus and establishment of the magnetic axis.

The tokamak assembly starts with the installation of the bottom lid of the cryostat, placed on the basemat of the pit at month 45. The lower cryostat cylinder is then installed. For manufacturing of the bottom lid 18 months are needed at the factory and a further 6 months for construction work on the site before installation. Early procurement is required, namely within nine to twelve months after T = 0. The upper cylinder and top lid of the cryostat will be fabricated to match the tokamak assembly process. The completion of the cryostat with the installation of the top lid is at month 84.

Drain tanks and the main pipes for the primary heat transfer loops and heat rejection system are placed on the basemat level of the pit during the construction of the tokamak building and in the surrounding service tunnels. Thus these components have to be procured early.

5.5 Commissioning Plan

5.5.1 Individual Sub-System Test

Testing of each individual plant subsystem has to start immediately when permitted by their delivery and by the corresponding assembly work. This testing is the last phase of procurement for each individual subsystem. Individual plant subsystem tests will be done by simulating interfaces with other systems or by using dummy loads or bypasses. Links with the CODAC supervisory control system are compulsory at this time. Each individual subsystem must be ready before the next phase of integrated testing with other subsystems. Subsystems not needed in an early phase will be commissioned in parallel with operation.

A more complex example is that of the remote handling (RH) equipment. Many RH tools, especially transporters, will be tested on mock-ups during the construction phase. Some complete RH techniques will be used as part of the initial installation either because it is more efficient or for demonstration purposes. Therefore, the major RH equipment will be installed and commissioned before first plasma.

5.5.2 Integrated Commissioning up to the First Plasma Discharge

There will be the need for adequate testing of controls and interfaces between subsystems. CODAC is designed to permit testing of the complete system in the absence of one or more of the sub-subsystems. This one year of integrated testing requires that all the key subsystems have been successful in their individual tests.

All systems are tested to the extent possible without plasma. This includes the following major items.

- Vacuum leak and pressure test of vessel and cryostat
- Hydraulic test and baking
- Magnet cooldown (40 days)
- TF coil energisation
- Pulsing without plasma and wall conditioning

At the end of this phase the following will be achieved:

- readiness of operation of the tokamak machine (e.g., vacuum, baking, sufficiently high toroidal magnetic field (4 T) to match ECH&CD for start-up and discharge cleaning, and more than 50% of coil current in all PF coils);
- readiness of all subsystems, including the additional heating and current drive system, start-up diagnostics set, and fuelling (except the tritium system), which are needed for H plasma operation; initial test blanket modules (or blanking plugs) have been installed and commissioned; some subsystems may be completed later (as indicated above).

5.5.3 Commissioning after First Plasma

After the first plasma there will be further integrated commissioning over about four years leading to full operation in DT. The first 2.5 years of operation without DT is defined as a "pre-nuclear commissioning phase" and "nuclear commissioning" (about one year) will be done by using DD discharges with limited amounts of tritium.

5.6 Decommissioning Plan

It is assumed that the ITER organization at the end of operation will be responsible for starting the machine decommissioning through a de-activation period after which the facility will be handed over to a new organization inside the ITER host country. It is therefore necessary to provide a feasible and flexible plan for the decommissioning of the ITER machine and associated active components. The plan is based on the rationale of resources and equipment usage optimization, and takes into account the statutory occupational radiological exposure (ORE) limits. The plan provides a framework for the organisation to decide when and how to implement the ITER facility dismantling, depending on priorities applicable at the time. Flexibility is provided by the use of two separate phases.

During the first phase, the machine will, immediately after shutdown, be de-activated and cleaned by removing tritium from the in-vessel components and any removable dust. Also, any liquid used in the ITER machine systems will be removed (no component cooling will be further required) and processed to remove activation products prior to their disposal. De-activation will include the removal and safe disposal of all the in-vessel components and, possibly, the ex-vessel components. The main vacuum vessel may be prepared for dismantling by the cutting of the inner vessel wall. The ITER de-activation will also provide corrosion protection, for components which are vulnerable to corrosion during the storage and dismantling period, if such corrosion would lead to a spread of contamination, or present unacceptable hazards to the public or workers. These activities will be carried out by the ITER organization using the remote handling facilities and staff existing at the end of operation. At the end of phase 1, the ITER facility will be handed over to the organization inside the host country that will be responsible for the subsequent phase of decommissioning after a dormant period for radioactive decay.

The plan does not include the dismantling of the buildings and of the non-active components (except, when applicable, for the ex-vessel components), or the disposal of wastes from decommissioning. Outside of the pit, the re-use or scrap value of components is higher than the cost of dismantling them.

Table 5.6-1 Summary of the ITER Decommissioning Plan

<table>
<tr><th>PHASE</th><th>ACTIVITY</th><th>DESCRIPTION</th><th>DURATION</th></tr>
<tr><td rowspan="2">1</td><td>De-activation</td><td>1 Removal of mobilizable tritium and dust from the machine using available techniques and equipment. Removal and de-activation of coolants.
2 Classification and packaging of active, contaminated and toxic material.
3 Removal of all the in-vessel components.

OPTION 1: removal of ex-vessel components (if not done in phase 2).</td><td>~ 5 years</td></tr>
<tr><td colspan="3">The ITER facility is handed over to an organization inside the host country</td></tr>
<tr><td colspan="2">Radioactivity decay period</td><td>1 The vacuum vessel radioactivity is left to decay to a level which allows the extraction of vessel sectors into the tokamak building (during phase 2) for size reduction and disposal.
2 No site activities are required except security and monitoring.</td><td>As required</td></tr>
<tr><td>2</td><td>Final Dismantling and Disposal</td><td>1 Removal of vacuum vessel sectors and their size reduction by remote/semi-remote operations.

OPTION 2: removal of ex-vessel components (if not done in phase 1)

2 Classification and packaging of active, contaminated and toxic material</td><td>~ 6 years</td></tr>
</table>

6 Cost Estimates

6.1 Resources Required for ITER Construction

6.1.1 Cost Estimating Approach for ITER Construction

The approach to cost estimating for the construction of ITER is based on the presumption that ITER will be constructed as an international joint project in which the participants' (Parties') contributions will mainly be specific systems or components contributed directly to the project ("in kind").

The main objective of ITER cost estimates is to provide a realistic and sufficient basis for ITER Parties to make their decisions on the scope of their involvement and to select the desirable systems for them to manufacture. The estimates have been developed from the engineering designs following a "bottom-up" approach which emphasises physical estimates (such as labour hours, material quantities, physical processes, etc.) so as to ensure that the data are comprehensive and coherent and provide a basis for evaluating results from different Parties.

In considering the ITER cost estimate, it is important to recognize that:

- economic conditions in the Parties vary widely over time and these changes are not necessarily or adequately reflected in relative monetary exchange rates;
- domestic industrial practices, contracting policies and labour costs for manufacturing prototypes depend on Parties and are not reflected in relative currency exchanges (only valid for goods for which a worldwide market exists);
- the overall ITER management approach and specific procurement and contracting practices have not been determined and the host party has not been selected;
- the aggregate costs that will be incurred in constructing ITER will depend greatly on how the responsibilities for specific components are distributed between the different participants, and on the procurement policies pursued by each.

6.1.2 Basic Data — Procurement Packages for Cost Estimation

In order to elicit the basic data for ITER cost estimates, about 85 "procurement packages" have been developed for the elements of the project work break-down structure (WBS), each defined at a level consistent with a plausible procurement contract. Each package comprises comprehensive information, including the functional requirements, detailed designs, specifications, interfaces and other relevant data that would be needed by potential suppliers in order to prepare for contract quotations, e.g. the proposed split of responsibilities between supplier and customer and the necessary QA arrangements.

In some areas the packages provide for the possibility of splitting contracts between several suppliers in case more than one Party might wish to participate in a specific area. In some others, this splitting between suppliers in more than one Party is necessary in order to produce the volume required because of the limited capacity of each supplier (e.g. superconductor strand, shield/blanket modules, divertor targets).

Industrial companies or large laboratories with relevant experience were invited, through the Home Teams, to generate, from the procurement packages, their best estimates of the likely current costs of supply, assuming that all data necessary to support procurement would be available on schedule. To allow review and comparative evaluation of the estimates, the participants were requested to provide detailed supporting data and detailed descriptions of the potential deliverables and processes, in the format of standard ITER Cost Estimation Workbooks.

The information thus generated offers a comprehensive database for cost analysis, comparison and evaluation.

However, the cost estimates provided by each Party are not intended to be the lowest values which could be obtained in this Party, keeping the same technical specifications, since they were not the result of competitive estimation and tendering.

6.1.3 Evaluation of Cost Estimates

The JCT has analysed the results of the procurement package studies in consultation with Home Teams concerned, with the objective of deriving "evaluated estimates" which distil the database to a consistent set of project cost estimates.

The final result is a complete set of "evaluated cost estimates" for building ITER, expressed in IUA, which is robust to currency fluctuations and domestic escalation rates and which can be used by the Parties jointly and individually in reviewing their options and the possible budgetary effects of participating in ITER construction [1 IUA = $ 1000 (Jan 1989 value)].

Data expressed in terms of physical quantities can be readily compared directly, after confirming the definition of the terms. Data expressed in money values needs to be converted into common terms for the purposes of comparison and analysis.

Financial data have first been converted from current costs to the established reference date for ITER cost estimates of Jan 1989, using standard inflation factors for each Party (Table 6.1-1).

This methodology has already been applied for the ITER 1998 design cost estimates.

It is not assumed that the 1989 exchange rates between currencies is better than those at any other date. Even, exchange rates between two currencies and inflation rates in their two countries do not sometimes vary in a coherent way (examples are Canadian and US Dollars, British Pound and Euro).

Nevertheless, looking backwards in time, using this approach the Parties have their own appreciation of these old rates and can apply at once the correction factor they feel appropriate (because of labour cost, industrial practices or any other reason) to quantify directly their potential contribution in their current money.

Moreover, a constant reference unit is the most appropriate tool to cost a project which extends in time over many decades in an international framework.

Table 6.1-1 Exchange and Escalation Assumptions

Currency	Conv. Rates to US$ 2000 (for comparison only)	Conversion Rates to Euro (Fixed)	Conversion Rates to US$, (January 1989)	Escalation Factors for Reference Years					Conversion Factor 2000 to IUA (x 10^{-3})
				1989	1997	1998	1999*	2000*	
US $	1.000		1.000	1.00	1.32	1.35	1.37	1.39	1.392
Can $	1.507		1.207	1.00	1.21	1.22	1.24	1.25	1.509
Euro	1.036	1.0000	0.876	1.00	1.37	1.41	1.44	1.46	1.279
DM		1.9558							
FF		6.5596							
Lira		1936.3							
UK Pound	0.634		0.585	1.00	1.42	1.47	1.51	1.55	0.906
Yen	107		128	1.00	1.13	1.14	1.15	1.16	0.148
Rouble	28.34		39.45	1.00	1.00	1.00	1.00	1.00	39.45

Note: * Extrapolated data
References [1] : International Financial Statistics Yearbook, International Monetary Fund, 1999, [2] : Eurostat Yearbook, Edition 98/99.

In order to emphasise the relative costs of the different systems/components, the following method has been used to derive from the basic data a JCT cost estimate, consistent across the whole ITER plant:

- world market prices have been used where they exist, for instance for standard materials or items of equipment;
- unified standard labour costs have been established which reflect (see Table 7.1):
  - rates in IUA/khour established for the main categories of labour, averaged across skill levels and Parties;
  - the different levels of manufacturing support costs, including facility, equipment, overhead and profit, for different types of industrial works and working conditions (e.g. staff in their existing home facilities, on site, or in a facility charged as an itemised cost).

After analysis of the proposals from the different Parties for amounts of tooling, material quantities and labour hours to manufacture each identified item, the JCT has established its own assessed numbers for these cost driving elements and, using the above-mentioned lists of standard materials prices and labour costs, has derived an “evaluated cost estimate” for each identified item.

Summing all items per procurement package, the JCT provides through this methodology a normalised cost for each package, the most credible given the present uncertainties on ITER construction management and procurement. For reasons described above, when applying to each package the conversion factor from IUA to the present value of currency in one Party, the result should not be expected to match the cost incurred by this Party to provide the given component.

Table 6.1-2 ITER Labour Rates

The cost estimates prepared for the EDA Final Design Report include a number of parameters which need to be normalised. A large table was established to provide a set of labour rates for both field construction and shop manufacturing operations. These rates were used for all estimates for both home and field manufacturing operations as well as site installations. Established in IUA /khours, these rates are assumed constant in time (which means that, converted in any Party's currency, they will follow the inflation rate of that Party).

Labour costs are composed of labour wage rates and support costs:

- *The Labour Wage rates are the Hourly Wages each worker earns plus fringe benefit packages including employer's contribution to taxes, health benefits, vacation benefits if any, and related employer-paid labour cost items.*

- *Support costs, also presented as an hourly rate, include the cost of use of the manufacturing facility and equipments, ship supervision, management, consumables, other overheads, and profit.*

For some commodities, such as for the TF or PF coils, in which Facilities, Special Tooling and Equipment are estimated separately, the support costs have been decreased accordingly. For those items to be fabricated on site (in the same Country as Home), it is assumed that the support costs will be the same as at Home, but that a premium (0.005 IUA) is added to the hourly wage.

The Wage Rates were determined by using an average of wage rate cost data from the U.S. ("R.S. Means, 30 City Average, January 1997"), Japan ("Chingin Jijyo", Factory labour direct cost and "Seskisan Shiryo", construction labour rate), and EU (Eurostat). All rates include 5% casual overtime. Manufacturing labour cost is based on a composite of Highly Skilled, Skilled, and Semi-Skilled Labour. Site Installation Labour is based on a composite of Craft Labour.

From these values of 1997 using conversion factors to IUA (according to inflation rate in each Party), one can note that in average:

1. *the normalised labour wages are 20% above the European values and 20% below the Japanese ones;*
2. *the normalised support cost per hour is equal to the European value and 10% below the Japanese one; it amounts to a value between about 2 (for welding) and 3 (for machining) times the normalised labour wage, and only about 1.25 (for welding) and 1.80 (for machining) if the special equipment is estimated separately.*

A few global values for Labour rates are mostly used in the ITER "estimated cost" (in IUA/khour):

1 *for Engineering, a mix of professional designers (90), professional engineers (67), CAD and procurement technicians (38), in average 67-73,*
2 *for machining including full support (67) or limited support (48),*
3 *for welding including full support (55) or limited support (40),*
4 *for QA and testing, (45),*
5 *for assembly/installation, (35 to 45, depending on the amount of professional support).*
6

For civil work, the costing is done through the use of a table of commodity rates for all activities referred to the "unit measurement" of the quantity of the commodity used (m^3, m^2, t,etc..). They include material and labour amounts as well as specific support per unit.

6.1.4 Conclusions on the Approach to the ITER Construction Cost

The approach to develop a JCT evaluated cost estimate for ITER construction, expressed in IUA tries, to the extent possible, to remove variations in costs that are due to differences in estimating practices by the different Parties, and to exchange rate fluctuations. This means that although ITER costs for each item reflect the same "value" in two Parties, they will not, when expressed in these Parties' currencies, necessarily correspond to each other through exchange rates at any date.

The approach provides fair and consistent relative costs for the different ITER systems and components. The Parties can, jointly, appreciate in advance the relative contributions (in percentage) that each might make to building ITER and, individually, estimate from the underlying physical data the absolute costs (in their own currency) that each might expect to incur in providing specific components, inside their contribution "in kind", by applying their own appropriate conversion factor to IUA.

This detailed "evaluated cost estimate" aids the "design to cost" and, later, the "manufacture to cost" approaches by which design/process changes are made to maintain costs within the budgeted amount because itemized quantities and manufacturing man-hours are clearly defined.

6.1.5 Cost Estimates Summary

6.1.5.1 Component/system "Evaluated Cost Estimate"

The evaluated cost estimates for ITER construction in kIUA, summarised by procurement packages, are presented in the Table "ITER Cost Estimate Summary". A much larger "Business Confidential" document provides for each procurement package the JCT detailed evaluation. The table also indicates, for each package, which of the Parties provided data, what is its direct capital cost, its percentage of the total capital investment, the cost of spares and of deferred items to be supported by operation funds, and their comparison with the relevant numbers for the ITER 1998 design.

As recognised by Experts from the four Parties, who have reviewed the details of the component/systems costing which support these estimates, they are not intended to be the lowest values which could be obtained, keeping the same technical specifications, since they were not the result of competitive estimation and tendering within each Party.

Globally for ITER construction they represent the most credible cost estimates, given the present uncertainties on ITER Construction Management, Siting and Cost sharing: a global value inside which one can be confident to be able to build ITER.

In addition, there is room to achieve substantial savings in some areas already identified, and more might be found through the needed and expected industry feedback on design to optimise manufacturing processes.

6.1.5.2 Siting and Construction Costs

The present ITER design, and its "evaluated cost estimate", follows the "ITER Site Requirements and Site Design Assumptions", which have been approved by the ITER Council[4]. Site specific adaptations of the design may induce changes in the cost of some systems; they will be analysed during the Coordinated Technical Activities (CTA), when potential sites are characterised.

Similarly, the present design is consistent with codes, and standards which have been defined inside the project. These rules are coherent but are not identical to those of any specific Party, even if they do not contradict them. Regulatory bodies from potential host country may request application of different and specific design rules or quality assurance measures. This can induce cost variations to be analysed again during the CTA.

The "ITER Site Requirements and Site Design Assumptions" describe a list of Host responsibilities, for which the project bears no cost, and which include in summary:

- infrastructure for industrial support of ITER and socio-economic provisions,
- land for the ITER site,
- high quality (potable), and raw water, treatment for sanitary and industrial sewage,
- supply to steady state electrical power network and a tie line capable of large pulse power for magnet and plasma heating,
- off site fire protection equipment and personnel,
- receipt of waste of all types, generated from ITER operation and decommissionning

6.1.5.3 Procurement Scheme

Each procurement package relies on a single dominant technology, and therefore mainly leads to a single industrial contract and avoids the need for a large amount of subcontracts. On the other hand, the possibility of sharing the work to be done for a package between more than one industry was always considered, even if this splitting leads to an (acceptable ?) increase in cost. This method provides for the ITER Parties the possibility, if they wish, to contribute in all fusion specific technologies. For example, the following list offers a possible splitting of procurement orders:

[4] PDS Chapter 4

- the Nb_3Sn and NbTi conductors (355 kIUA in total) should be provided by contributions from all Parties; even more, industries in the Parties should be encouraged to prepare for a larger production capability, in particular in Russia for niobium of the required quality;

- two suppliers are proposed to share equally the manufacturing of the TF coils windings (117 kIUA in total);

- one or two equal suppliers for the procurement of TF coil mechanical structures (168 kIUA in total).

In these cases, it is thought appropriate that one contractor will have the responsibility to develop the design for the special toolings, which will be duplicated for use by the second contractor. The two sets of tools are anyway necessary to meet an acceptable manufacturing schedule. Therefore, this sharing possibility between two Parties' industries leads to only a small increase of cost due to a small loss of labour efficiency.

- one manufacturer for the central solenoid (31 kIUA);

- one manufacturer for the PF coils, (50 kIUA), assumed to work on the ITER site and thus from the Host;

- one manufacturer for coil Feeders (41 kIUA): a complex area of superconductor and cryogenic technologies and instrumentation;

- at most, three equal industrial partners for the vacuum vessel manufacture (155 kIUA in total), and one or two more to produce the port structures (75 kIUA in total), if they agree on cooperation in the engineering of the specific process toolings;

- no difficulty should be met to share between different industries the delivery of more than 420 blanket modules (143 kIUA in total);

- the high flux components for the divertor are assumed to be provided by three, or more industries (65 kIUA in total);

- a few of the packages require only one responsible supplier per package (joined by industrial subcontractors): assembly of the machine, (50 kIUA), reinforced concrete buildings (312 kIUA), the steel frame buildings (69 kIUA) – all activities on ITER site and thus for the Host industries – the cryostat manufacturing (76 kIUA), the cooling water piping (60 kIUA), and the steady state electrical power network (40 kIUA) which all require also a large amount of work on site;

- all other packages should be procured through more than one supplier; particular ones provide for heating and current drive systems and diagnostics, and are discussed below.

In summary, the procurement of the machine core (1465 kIUA) could be possibly split into about 40 different contracts of 40 kIUA in average. The procurement of Auxiliaries and Heating and CD systems, except the Concrete Building (800 kIUA) could be split into 40 different contracts of 20 kIUA on average.

6.1.5.4 Heating and Current Drive Systems

The ITER requirements for plasma heating and current drive cannot be satisfied by a single method. Four have been envisaged (NB, IC, EC and LH), which could provide ITER with flexibility of operation. The machine design makes possible the installation of three NB lines, each assumed to provide 16.5 MW of neutral deuterium atoms at 1 MeV. All RF heating systems have been designed to provide 20 MW to the plasma per port. All these systems costs have been estimated per port of injection.

Provisions in the layout of equipments are such that heating and current drive systems procurement can be staged, to reach a maximum of 50 MW of NB (three lines), and 40 MW for any of the three RF methods: in all possible scenario, installed power cannot provide more than $\simeq$ 130 MW and more than 110 MW available at the same time.

The present status of R&D results in any of those heating methods has not achieved the level required to be confident in their assumed performance and therefore in their availability at the start up of ITER operation. An R&D effort in all Parties, more extended and more efficient, is absolutely needed with the highest priority.

For the time being, it is assumed that the start up scenario will use two NB lines (33 MW, 96 kIUA), one equatorial port equiped with an IC launcher (20 MW, 32 kIUA)) and 20 MW of EC power (77.5 kIUA) which can be delivered either through one equatorial port or through 3 upper ports (for NTM stabilisation).

6.1.5.5 Diagnostic System

ITER requires a comprehensive diagnostic system for monitoring in real time the conditions of the different machine components and for measuring the value of the key plasma parameters, in order either to control ITER operations or to increase the understanding in physical phenomena.

The diagnostic system comprises about forty individual measurement systems. The responsibility for design and procurement of these specific systems should be shared by the Laboratories of the Parties which aim at participating in ITER operations through their physicists. The Central ITER Team cannot bear alone this responsibility, even if it should specify all interfaces with the machine and in particular the responsibility of all "generic" packages.

The cost of the diagnostic system has been estimated through:

- seven diagnostic specific procurement packages: A: Magnetic diagnostics – B: Neutron systems – C: Optical systems – D: Bolometry – E: Spectroscopic systems – F: Microwave systems – G: Operational systems.
- and six "generic" packages: N01: In vessel services - N03: Diagnostic port plugs and first closure flanges - N04: Port interspace structures and second closure flanges - N05: Divertor diagnostic components - N06: Ex vessel services - N07: Windows assemblies.

In addition, the cost of the Diagnostic Neutral Beam has been obtained together with the Heating Neutral Beam cost. Other related costs are for Blanket Shield Modules for Diagnostics Ports and the Test Stand in the Hot Cell, which are included in other packages.

Costs of providing the diagnostic system are divided into several components. The cost of diagnostics required for initial operation (startup) is determined separately from the cost of those that are not required until the DT operation (deferred). Included in the startup costs is the cost of the in-vessel and inter-space equipment, and interfaces, for the deferred systems which would otherwise be expensive and time consuming to install later. Further, the diagnostic costing identifies items which would have to be provided during construction under control of the central ITER Team (in-machine items) and those which could be provided by the ITER Parties directly (ex-machine items). Examples of the former could be in-vessel services and wiring, while examples of the latter could be specialised lasers and spectrometers. As in previous costing exercises, the cost of the specialist diagnostic effort to support the design, procurement and implementation of the diagnostic systems is costed separately in PPY: part of this effort will be required early to prepare for consistent plans and interfaces with the central ITER Team.

The procurement of diagnostics for the startup requires 121 PPY from laboratories and 118 kIUA of which 72.2 kIUA are under direct design control of the central ITER team; deferred diagnostics will require in addition 106 PPY and 42.3 kIUA.

6.1.6 ITER Direct Capital Cost

Taking account of the previous assumptions, the total “JCT estimated capital investment” for ITER amounts to 2,755 kIUA; in addition the cost of spares and items needed only a few years after start of operation (full DT operation) amounts to 258 kIUA and is deferred to be supported by operating funds. The present investment cost is only 49.2 % of the previous estimate for the ITER 1998 design, which amounted to 5603 kIUA and 302 kIUA deferred.

6.2 Construction Management and Engineering Support

An estimate of the cost of construction management and support cannot be done without assumptions on the future organisation to execute the construction and the manner of contracting and managing contracts for procurement.

6.2.1 Assumptions

For this purpose, it is assumed that the ITER Legal Entity (ILE), which will be responsible for the management of ITER during its whole life time, will provide a direct and effective line of accountability by incorporating all actors in a single management entity, including:

- an International Team at the ITER site which will have the overall responsibility to meet the project objectives and to ensure the design continuity and coherence.
  For this goal, it will :
    - define the technical specifications for procurement packages, the general QA rules and contracting rules,
    - control, analyse and decide upon design changes or deviations,
    - maintain databases on R&D and manufacturing results,
    - integrate all aspects of ITER: technology and physics - safety and licensing - assembly - CODAC and diagnostics - cost and schedule.

- a National Team, as part of the ILE in each Party, which will manage and follow up the technical content of the procurement contributed by the Party, when the financial and legal contents of the relevant contracts are being taken care of by a Domestic Agency

  For this goal, each National Team will :
  - adapt technical specifications to its national usages and ensure engineering at detailed level,
  - implement QA,
  - assure technical control of each domestic supplier contract by a permanent presence in suppliers' premises, and a schedule control by accepting contractual payments according only to work progress.

With these assumptions it is clear that the size of the International Team can be deduced approximately from its functions, but the size of each Party National Team will depend on the level of the Party's contribution to ITER construction, on these specific packages in its contribution, and on the specific national practices in contract management.

Presently, when the respective choices of the ITER Partners in matters of procurement packages are unknown, a global approximation can only be given of the manpower necessary to follow up all the procurement packages, expressed in professional and technical manyears, integrating all National Teams, and taking no account of the possible splitting of procurement contracts between Parties and its relevant increase in manpower for their follow-up.

6.2.2 Estimated Personnel Cost During Construction

To exercise its responsibilities, the International Team will probably include a core management group and a few technical groups, in charge of physics, safety, engineering, assembly, etc.. These groups should be able to ensure technical continuity with the EDA and CTA, and, as construction approaches its end, these groups, suitably increased by personnel from the National Teams who have followed procurements, will eventually be involved in the integrated commissioning and start up of operation of the facilities. The number of professionals of this International Team can thus range from about 80 at the beginning of construction, to about 200 towards the end.

The support personnel for the core management (mostly administrative) and for the technical groups (mostly CAD) is expected to be in number equal to the number of professionals.

Following the assumed schedule (seven years of construction beginning two years after establishment of the ILE – the one year of integrated commissioning is assumed to be the first year of operation) the global manyears for the International Team during construction amounts to 840 PPY and a similar number for support personnel.

The global estimate of professionals and support personnel (clerical, technicians and CAD) for the different National Teams to follow up all procurement contracts in all Parties amounts to about 960 PPY and twice this number for support personnel. The number of professionals is assumed to be about 120-140 in average for six years and decreasing during the last three years of construction, for probably being transferred to the site to participate in the components installation and commissioning.

Assuming the annual cost of one professional and one support staff to be 150 IUA and 75 IUA respectively, the cost estimate for the International Team during construction until the start of ITER operation (integrated commissioning of the whole machine) amounts to 189 kIUA and the global estimate for all the National Teams during the same period amounts about 288 kIUA. Again, these costs are normalised and global, including all Parties; they might not be representative for the specific conditions of each Party. Excluded from the previous numbers are those relevant to diagnostic procurement referred to above.

6.2.3 Possible R&D Cost During Construction

In addition to personnel costs, a certain amount of R&D during construction should always be considered. As already mentioned, R&D for all heating and current drive methods is required with high priority.

The EDA has provided the principle qualification of design solutions to be implemented in ITER. Nevertheless, during the manufacturing of components, proposed process improvements and design changes or unexpected difficulties could require new tests.

Moreover, to achieve in industrial production reliable results and good efficiency in the manufacturing of a large amount of high technology components (e.g. superconductor strands, high heat flux components, etc…), it is probably more efficient to launch, at the chosen industrial firm, a manufacturing R&D before contracting the global procurement. Along a similar line, it is conceivable that some Nb_3Sn coils (TF and CS) should be tested at their operational cryogenic temperature to confirm quality, even if presently the cost/ benefit ratio of technical results to be expected from these tests does not appear high enough.

It is therefore prudent to expect a spending in R&D of 60-80 kIUA during ITER construction.

6.3 Resources for ITER Operation

6.3.1 Project Manpower Costs

Manpower costs of permanent staff on site and cost of extra manpower brought in from time to time to aid in maintenance of the device are costed assuming an average level of 200 professionals and 400 support staff (clerical, technicians and CAD), at 150 IUA and 75 IUA respectively per year; thus the annual personnel cost is about 60 kIUA.

The permanent professional and support staff above are expected to operate and maintain the facility, and support the experimental programme, for example diagnostics, or installation and testing of the blanket modules. However, visitors to the site to conduct experiments (experimentalists or theoreticians) are not included in the manpower cost: their costs are assumed borne by the Parties. This is consistent with including the cost of diagnostics in the construction or operating costs, but not the Laboratories staff having the responsibility of their procurement.

6.3.2 Energy Costs

Electric power costs, which not only include the power required for pulse operation, but also must cover energy consumption during various levels of standby/maintenance of the machine, depend on aspects of the load time profile and on the characteristics of the national electricity network of the host site. Whereas electric peak power to be delivered, and the average power to be made available over a certain time period, place a premium on the cost of energy consumed, the consumption dominates the cost in any well-provisioned site. Thus the electricity costs can be estimated knowing the steady state power levels required in the various operating states, and the fraction of time spent in those states, superimposing the integral of the pulsed power demand when in the plasma operating state.

A typical scenario is shown in Table 6.3-1. The steady state power loads are representative values over the whole life of the plant, but are somewhat uncertain. Similarly the time devoted to maintenance (or conversely to burn/dwell) as also difficult to specify at this stage. For this reason ranges are given, with the implications noted. A unit cost of 0.05 IUA/MWh is used.

Table 6.3-1 Key Features of Electricity Cost Calculation

	Low	Nominal	High
Steady state power loads			
Plasma operating state (POS)	100	100	100
Short-term standby (STS)	70	80	90
Short-term maintenance (STM)	50	60	70
Long-term maintenance (LTM)	30	35	40
Time usage (lifetime average)			
In LTM	0.25	0.375	0.5
In STM	0.5	0.375	0.25
Average annual cost (kIUA)	24.7	29.1	34.0

The most sensitive parameters of this calculation are shown above. The results are not sensitive to the average burn time, or to the details of the operation stages over the years, since the integral burn time is constrained to deliver the nominal fluence. Typical yearly cost variations from the average, at various operation stages, are below 5 kIUA.

6.3.3 Fuel Costs

The ITER plant must be operated, taking into account the available tritium externally supplied. The net tritium consumption is 0.4 g/plasma pulse at 500 MW burn with a flat top of 400 s. During the first 10 years of ITER operation, the total burn duration at 500 MW is planned to be about 0.15 years. The total consumption during the first 10 years is 4.7 kg. The typical tritium receipt, consumption, and site inventory during the first ten years are given in For commissioning the tritium plant, several tens of grams of tritium will be needed. This will be carried out in parallel with hydrogen plasma operation. A small amount of tritium (< 0.01 g) will be burned during the deuterium plasma operation.

The maximum total tritium transportation per year will be about 1.2 kg/year in the first 10 years. Assuming a 50 g transport tritium container, there will be two shipments every month.

Table 6.3-2. During ITER operation, all tritium will be supplied by external sources.

For commissioning the tritium plant, several tens of grams of tritium will be needed. This will be carried out in parallel with hydrogen plasma operation. A small amount of tritium (< 0.01 g) will be burned during the deuterium plasma operation.

The maximum total tritium transportation per year will be about 1.2 kg/year in the first 10 years. Assuming a 50 g transport tritium container, there will be two shipments every month.

Table 6.3-2 Receipt, Consumption and Site Inventory of Tritium

Year	Receipt (kg)	Consumed (kg)	Site Inventory (kg)
1	0.0	0.0	0.0
2	0.0	0.0	0.0
3	0.1	0.0	0.1
4	0.8	< 0.01	~ 0.9
5	0.8	0.3	~ 1.4
6	0.8	0.4	~ 1.8
7	0.8	0.6	~ 2.0
8	1.0	1.0	~ 2.0
9	1.2	1.2	~ 2.0
10	1.2	1.2	~ 2.0

Fuel costs include deuterium and tritium burnt during operation, plus that lost by decay of the inventory (taken as 2 kg) during plant operation. The deuterium cost is negligible at 2 IUA/kg. There is no market for tritium for the quantities required, and thus tritium may have little or no monetary value. Nevertheless a largely hypothetical 10 kIUA/kg for tritium purchase is used. The total tritium received on site during the first 10 years of operation, according to Table 6.2.5.1., amounts to 6.7 kg. The total consumption of tritium during the plant life time may be 16 kg to provide a fluence of 0.3 MWa/m^2 in average on the first wall; this corresponds, due to tritium decay, to a purchase of about 17.5 kg of tritium. This is well within, for instance, the available Canadian reserves.

Therefore the fuel costs are in average 6.7 kIUA per year during the first ten years, and probably 11.5 kIUA per year after.

6.3.4 Capital Improvement, Spare Parts, Maintenance Costs

Analysis of the expected capital costs to keep the ITER facility in the required effective state has shown that very different ratios (annual cost against the initial investment) should be considered for the different systems, going from almost 0% (e.g. magnets) to 10% (for RF power generators and diagnostics), up to 15% (e.g. computers).

Considering all systems, the required maintenance cost amounts to 2.5% of the initial investment, about 70kIUA per year. To this cost should be added investments deferred initially to operation costs and the cost of replacement of the divertor high heat flux components (possibly five times during the plant life time). This leads to a total of 90kIUA per year in average during operation.

6.3.5 Conclusion: Average Annual Operation Costs

In summary, the ITER average annual operation costs amount to about 60 kIUA for the personnel permanently on site, 30 kIUA for the energy consumption, 8 kIUA for the tritium purchase and 90 kIUA for spare parts, maintenance and improvements, i.e. a total average per year of 188 kIUA. Again this value will depend on the ITER site, mostly through the electricity cost (assumed to be 0.05 IUA/MWh), and on the specific arrangement between the Parties on how to support the personnel cost.

6.4 Decommissioning Costs

The technical implementation and schedule have been detailed according to a credible option. Even if the Host Party may consider other options, the global cost to be borne should not change drastically as long as a full dismantling of the machine is not required before the vacuum vessel activity has decayed substantially.

The ITER facility, because of the remote maintenance implemented during operation, offers initially most of the tools, procedures, and even trained staff, to accomplish the decommissioning operations. This capacity is an essential element in keeping their cost down.

The manpower estimate is based on the requirements for the dismantling of the main active parts of the ITER facility only. The non-active parts are not considered, because their residual values are probably higher than their dismantling costs.

For the technical operations and their schedule in two active phases described in Chapter 6.3, the estimated integrated work force over 11 years amounts to about 2,800 manyears. The average cost per manyear is rated at 90 IUA as a mix of different staff categories. In addition, it is assumed that new hardware may be required to replace a few aging tools, or to enhance the hot cell, and radwaste processing efficiency. For this purpose, one third of the manpower cost is put as an AFI, as observed in previous experience.

Other costs (dependent on the Host country) are not included in the present estimate:

- radwaste disposal,
- components and facilities salvage value after dismantling where applicable (e.g. materials below “clearance”),
- non-active parts dismantling and salvage value,
- site restoration,
- financing-related costs, if spending is made at a later stage.

Under the assumptions and limitations listed above, the estimated cost for decommissioning amounts to 250 kIUA for manpower costs and 85 kIUA for possible hardware costs.

6.5 Summary

For reference, a summary of the cost estimates for all phases of ITER plant lifetime is set out in Table 6.5-1 and compared with estimates for the 1998 design. This summary is subject to all the qualifications and considerations outlined above.

In particular the construction cost estimates:

- assume the agreed Site Requirements and Site Design Assumptions, and are thus valid for a generic site,
- include the policy for additional heating and diagnostics procurement,
- exclude costs associated with site hosting, notably, the provision of land, off-site facilities and all service supplies up to the boundary fence,
- exclude items deferred beyond the start of operation.

The average yearly operation cost estimates include permanent staff costs on site to operate and maintain the facility, but exclude the cost of visitors (physicists) to conduct the experimental programme of ITER.

Table 6.5-1 Summary of ITER Cost Estimates

	ITER-FEAT (1) kIUA	Ratio 1/2 %	ITER/1998 Design (2) kIUA
Construction costs			
A) Direct capital cost	**2755**	49.2	**5603**
B) Management and support	**477**	61.2	**780**
C) R&D during construction	**60-80**	$\simeq$ 50	**150**
Operation costs (average/year)			
A) Permanent personnel	60	66	90
B) Energy	$\simeq$ 30	50	$\simeq$ 60
C) Fuel	$\simeq$ 8	40	$\simeq$ 20
D) Maintenance/improvements	~ 90	50	$\simeq$ 180
Total	**188**	54	**350**
Decommissioning cost (total)	**335**	110	**$\simeq$ 300**

WBS No.	Procurement Package	No.		HT Cost Estimate EU	JA	RF	ITER-FEAT Direct Capital Cost (1) (kIUA)	Percentage / Total	Spares deferred Investment (2) (kIUA)	Ratio of (1) / (3) %	ITER 1998 Design Direct capital cost (3) (kIUA)	Deferred Investment (4) (kIUA)
				O : full estimate P : partial estimate X : no estimate								
1.1	**Magnet Systems**	1	Toroidal Field Coils Windings	O	O	O	117.0		4.0			
		2	Magnet Structures	O	O	O	168.3		5.4			
		3	Poloidal Field Coil & Correction Coils	O	O	O	49.7		0.0			
		4	Central Solenoid Coil	O	O	O	31.1		3.4			
		5	Feeders	X	O	X	41.0		3.2			
		6	Conductor	O	O	O	355.0		24.2			
			Subtotal				**762.1**	27.7	**40.2**	42.3	1800.0	
1.5	**Vacuum Vessel**	1	Main Vessel	O	O	X	155.0		0.0			
		2	Port Assemblies	O	O	O	75.0		0.0			
			Subtotal				**230.0**	8.4	**0.0**	113.0	203.0	
1.6	**Blanket System**	1	Blanket Manifolds, Filler Shields	O	O	O	6.0		0.0			
		2-1	FW/Blanket Shield Modules	O	O	P	142.6		7.3			
		2-2	Port Limiters	O	O	O	6.1		1.3			
		3	Blanket Module Connections	P	O	O	10.5		0.0			
			Subtotal				**165.2**	6.0	**8.6**	38.6	428.5	
1.7	**Divertor**	1	Cassette Integration & Testing	O	P	X	10.7		0.5			
		2	Plasma Facing Components	O	O	O	65.3		6.4			
			Subtotal				**76.0**	2.8	**6.9**	34.5	220.6	
2.2	**Machine Assembly**	1	Assembly Operations	O	O	O	50.3		0.0			
		2	Assembly Tooling	O	O	O	42.4		0.0			
			Subtotal				**92.7**	3.4	**0.0**	71.0	130.5	
2.4	**Cryostat**	1	Cryostat & VV Pressure Suppr. System	O	O	X	**75.8**	2.8	**0.0**	70.6	107.4	
2.7	**Thermal Shields**	1	Thermal Shield System	O	O	O	**28.8**	1.0	**0.0**	75.2	38.3	
3.1	**Vacuum Pumping & Fueling System**	1	Non-standard Cryopumps & Related Equipment	O	O	X	8.9		2.3			
		2	Roughing Pump Sets & Change-over Boxes	O	O	X	6.7		0.0			
		3	Leak Detection Stations	X	X	X	5.0		0.0			
		4	Standard Components	X	X	X	5.3		0.0			
		5	Pellet Injector	X	X	X	2.5		2.5			
		6	Gas Injector Valve Boxes	X	X	X	4.3		1.3			
		7	GDC Conditioning System	O	O	X	1.5		0.6			
			Subtotal				**34.2**	**1.2**	**6.8**	36.7	93.3	
	Machine Core		*Subtotal*				*1464.8*	*53.2*	*62.5*	*48.5*	*3021.6*	*0*

2.3	**R/H Equipment**		Blanket RH Equipment	O	O	X	27.1		0.8			
			In-Vessel Div. Maintenance Equipment	O	O	X	11.7		0.3			
			Transfer Cask System	O	O	X	10.0		6.4			
			Viewing/Metrology Systems	O	P	X	6.0		0.8			
			Neutral Beam RH Equipment	O	O	X	0.6		5.4			
			Hot Cell Repair/Maintenance Equipment	O	O	X	5.7		38.6			
			Subtotal				**61.1**	**2.2**	**52.3**	34.0	180.0	87.4
2.6	Cooling water	1	PHTS and CVCS	O	O	X	71.0		2.6			
		2	Heat Rejection System	O	O	X	37.4		10.8			
		3	Component Cooling & CW Systems	O	O	X	23.1		3.4			
			Subtotal				**131.5**	**4.8**	**16.8**	38.7	339.4	10
3.2	**Tritium Plant**	1	Tokamak Exhaust Processing System	O	O	X	6.1		6.9			
		2	Storage & Delivery, Long Term Storage Syste	O	O	X	11.9		2.6			
		3	HTD Isotope Separation	O	O	X	0.0		6.2			
		4	Atmosphere Detritiation	O	O	X	12.8		17.4			
		5	Water Detritiation	O	O	X	3.7		10.8			
		6	Tritium Plant Analytical System	O	O	X	1.0		0.6			
		7	Tritium Plant Automatic Control System	O	O	X	1.1		0.8			
			Subtotal				**36.6**	**1.3**	**45.2**	47.2	77.6	98
3.4	**Cryodistribution**		Cryoplant	O	O	X	60.5		2.5			
			Cryolines	O	O	X	12.2		4.0			
			Cryodistribution Components	O	O	X	16.2		1.4			
			Subtotal				**88.9**	**3.2**	**7.9**	61.7	144.2	24.2
4.1	**Power Supplies & Distribution**	1	High Voltage Substation and AC Distribution System	O	O	X	26.6		0.4			
		2	AC/DC Converters, Reactive Power Compensators and Harmonic Filters	O	O	X	81.2		1.0			
		3	Switching Networks, Discharge Circuits, DC Distribution and Instrumentation	O	O	O	67.2		1.8			
			Subtotal				**175.0**	**6.4**	**3.2**	54.2	323.0	0
4.1		8	Steady State Electrical Power Network	O	O	O						
			Subtotal				**39.7**	**1.4**	**0.3**	53.1	74.7	0
6.2	**Buildings**	1	Reinforced Concrete and Site Infrastructure Elements	O	O	X	311.5		0.0			
		2	Steel Frame Buildings	O	O	X	68.8		0.0			
			Subtotal				**380.3**	**13.8**	**12.0**	46.5	818.1	11.4
6.3	**Waste Treatment and Storage**	1	Waste Treatment and Storage	O	X	X	**2.1**		**7.0**		87.4	0
6.4	**Radiological Protection**	1	Radiological Protection	O	O	X	**1.0**		**3.2**		7.4	0
	Auxiliaries		*Subtotal*				*916.2*	*33.3*	*148.0*	44.7	*2051.8*	*231*

5.1	**IC H&CD**	1	Antenna Arrays and Vacuum Transm. Lines	O	X	X	4.5		0.0			
		2	Main Transm. Line and Matching System	O	X	X	4.8		0.0			
		3	RF Power Sources & RF Monitoring Control	O	X	X	16.0		2.0			
		4	Power Supply	O	X	X	6.9		0.0			
			Subtotal				**32.2**	**1.2**	**2.0**	0.0	0.0	0
5.2	**EC H&CD**	1A	Equatorial Launcher	X	O	O	7.3		0.0			
		1B	Upper Launcher	X	O	O	8.9		0.0			
		2	Transmission Line	O	O	O	17.9		0.0			
		3	RF Power Sources and Controls	O	O	O	29.5		3.0			
		4	Power Supply	O	O	X	13.9		0.0			
			Subtotal				**77.5**	**2.8**	**3.0**	45.8	169.2	0
5.3	**NB H&CD**	1	Assembly and Testing	O	O	X	3.8		0.0			
		2	Beam Source and High voltage Bushing	O	O	O	9.5		0.0			
		3	Beamline Components	O	O	O	3.9		0.0			
		4	Pressure/Vacuum Vessels, Drift Duct and Passive Magnetic Shielding	O	O	X	11.9		0.0			
		5a	Active Corr./Compensation Coils	O	O	X	4.4		0.0			
		6	Power Supply	O	O	X	62.5		0.0			
			Subtotal				**96.0**	**3.5**	**0.2**	60.0	160.0	0
	H and CD		*73MW (100MW in 1998)*				*205.7*	*7.4*	*5.2*	*62.5*	*329.2*	*0*
5.5	**Diagnostics**	A	Diagnostics. Magnetics	X	X	P	3.3		0.0			
		B	Neutron Systems	X	P	P	3.9		6.2			
		C	Optical Systems	X	X	P	16.9		8.8			
		D	Bolometry	O	X	X	6.7		0.0			
		E	Spectroscopic Systems	X	P	P	8.9		13.6			
		F	Microwave Systems	X	X	X	8.3		9.4			
		G	Operational Systems	X	X	X	11.0		0.0			
		N	Standard Diagnostics	O	X	X	36.3		4.2			
			Diagnostic Neutral Beam:Line	O	O	X	13.0		0.1			
			Diagnostic Neutral Beam:Power Supply	O	O	X	9.7		0.0			
			Subtotal				**118.0**	**4.3**	**42.3**	78.2	150.8	71
	CODAC		Hardware and Software (rough estimate)				**50**	**1.8**	**0.0**		50.0	0
			Grand Total				**2754.7**	**99.9**	**258.0**	**49.2**	**5603.4**	**302**