1 Overview & Summary

1.1 Introduction

1.1.1 Preface

This document presents the technical basis for the ITER Final Design Report foreseen during the current, Engineering Design Activities (EDA), phase of the ITER project. The report presents the results of collaborative design and supporting technical work undertaken by the ITER Joint Central Team (JCT) and the Home Teams (HT) of the Parties to the Agreement on Co-operation in the Engineering Design Activities for ITER[1] (the ITER EDA Agreement).

The overall programmatic objective of ITER, as defined in the ITER EDA Agreement, is "*to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes*".

The work presented in this report covers the full scope of activities foreseen in Article 2 (a) - (d) of the ITER EDA Agreement, i.e.:

"2 a) to establish the engineering design of ITER including

(i) a complete description of the device and its auxiliary systems and facilities,

(ii) detailed designs with specification, calculations and drawings of the components of ITER with specific regard to their interfaces,

(iii) a planning schedule for the various stages of supply, construction, assembly, tests and commissioning of ITER together with a corresponding plan for human and financial resources requirements, and

(iv) specifications allowing immediate calls for tender for the supply of items needed for the start-up of the construction of ITER if and when so decided,

(b) to establish the site requirements for ITER, and perform the necessary safety, environmental and economic analyses,

(c) to establish both the proposed program and the cost, manpower and schedule estimates for the operation, exploitation and decommissioning of ITER,

(d) to carry out validating research and development work required for performing the activities described above, including development, manufacturing and testing of scalable models to ensure engineering feasibility".

The report is based on detailed supporting technical documentation in all the above areas. In accordance with the terms of the ITER EDA Agreement, this documentation and other information generated in the EDA is available to each of the Parties to use either as part of an international collaborative programme or in its own domestic programme.

[1] ITER EDA Agreement and Protocol 1, ITER EDA Documentation Series No. 1, IAEA, Vienna 1992

The ITER Final Design Report documentation is organised hierarchically as shown in Figure 1.1.1-1. The notion behind this structure is that this documentation arrangement should remain valid also for construction and operation.

The main constituents of the ITER documentation are:

- A top level *Plant Design Specification (PDS)* document, where externally imposed essentially design-independent requirements at the highest level are defined, including safety principles and criteria.

- Design Requirements and Guidelines Level 1 (DRG1) deals with the requirements and specifications above the system level. This includes not only plant-wide requirements but also interfaces or specifications affecting the design of more than one single system. DRG1 identifies the functional and physical interfaces between two systems and refers to any document and drawings defining the interface in more details It effectively includes overall “configuration drawings”. More detailed "Design Background" documents are annexed. These annexes, for example, address in detail Loads Specifications, Quality Assurance, Design Criteria, Design Manuals and Guidelines, Safety Requirements, etc.

- Design Requirements and Guidelines Level 2 (DRG2) defines in one document the boundaries of each system and deals in more detail with the requirements and specifications at the system level. The system division is identical to that of the DDDs.

- Design Description Documents (DDDs) are one per system. They follow a normalized format for the detailed description of the system design and its performance.

- The Plant Description Document (PDD), that is this very document, is the global plant description. It summarises the design based on the details in the DDDs, gives an overview of major plant processes that usually involve more than one system, summarises plant level assessments, and overall planning. The latter items are described in more detail in "Plant Assessment" document annexes. These annexes describe and assess more in detail the entire Plant and processes involving more that one single system. For example: Plant Control, Plasma Performance, Safety Assessment, Assembly Process, Seismic Analysis, Material Assessments, Nuclear Analysis, etc.

- The complete set of Task Reports on detailed design and R&D technology compiled by the Home Teams.

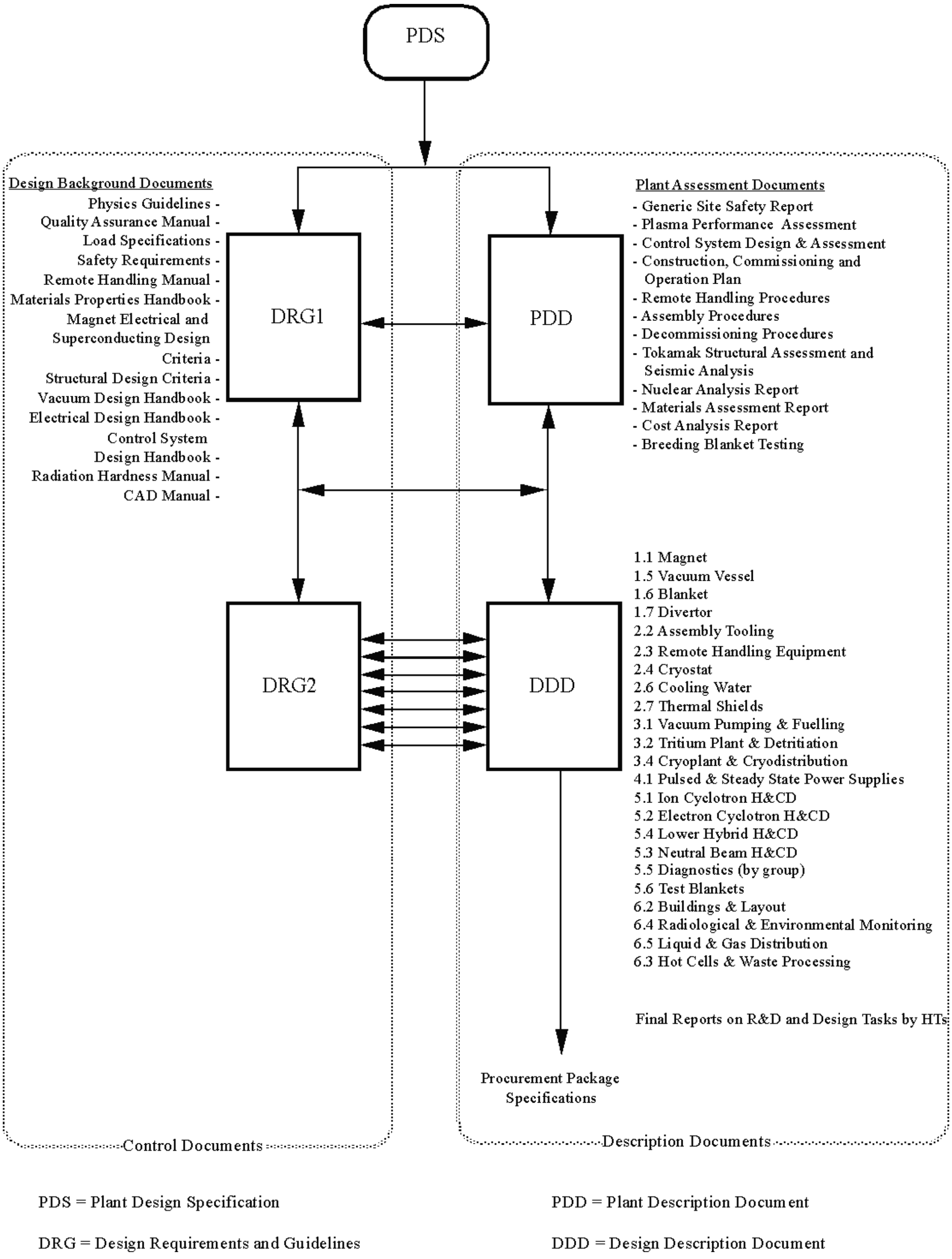

Figure 1.1.1-1 Overall hierarchy of ITER documentation

1.1.2 Evolution of the ITER Design

In 1998, at the end of the six years of joint work originally foreseen under the ITER EDA Agreement, a design for ITER had been developed[1] which fulfilled the overall programmatic objectives and complied with the detailed technical objectives, technical approaches, and the cost target adopted by the ITER Parties in 1992 at the start of the EDA.

When they accepted the 1998 report, the ITER Parties, anticipating the Agreement to extend the period of the EDA by three years[2] and recognising the possibility that they might be unable, for financial reasons, to proceed to the construction of the then foreseen device, established a Special Working Group (SWG)[3], and charged it:

- to *propose technical guidelines for possible changes to the detailed technical objectives and overall technical margins, with a view to establishing option(s) of minimum cost still satisfying the overall programmatic objective of the ITER EDA Agreement,* and
- to *provide information on broader concepts as a basis for its rationale for proposed guidelines, and articulate likely impacts on the development path towards fusion energy*.

In reporting on the first task, the SWG[4] proposed revised guidelines for Performance and Testing Requirements, Design Requirements, and Operation Requirements, noting that *"preliminary studies ... suggest that the direct capital costs of ITER can be reduced significantly by targeting the less demanding performance objectives recommended..."* and expressing the view that *"these less demanding performance objectives will satisfy the overall programmatic objectives of the ITER Agreement even though these performance objectives are necessarily less than those that could be achieved with the present* [1998] *design."*

With regard to their second charge, which essentially comes down to a choice between two strategies:

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

the SWG[5] found that "*the full non-linear interplay 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 step*" like 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. Furthermore, the SWG expressed the unanimous opinion that the world programme is *"scientifically and technically ready to take*

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[1]ITER Final Design Report, Cost Review and Safety Analysis, "ITER Council Proceedings: 1998", ITER Documentation Series No 15, IAEA, Vienna, p39

[2] Text of the Agreement extending the EDA Agreement, ibid, p102

[3] SWG Charter, ibid, p108

[4] ITER Special Working Group Report to the ITER Council on Task #1 Results, ibid, p148

[5] ITER Special Working Group Report to the ITER Council on Task #2 Results, "ITER Council Proceedings: 1999", ITER Documentation Series No 17, IAEA, Vienna, p33

the important ITER step." The Parties through the ITER Council subsequently endorsed this viewpoint.[1]

1.1.3 Guidelines and Objectives

The revised performance specifications adopted by the ITER Council in June 1998[2] are set out in full in Table 1.1.3-1; in summary they require ITER:

- to achieve extended burn in inductively-driven deuterium-tritium plasma operation with $Q \geq 10$ (Q is the ratio of fusion power to auxiliary power injected into the plasma), not precluding ignition, with an inductive burn duration between 300 and 500 s;
- to aim at demonstrating steady state operation using non-inductive current drive with $Q \geq 5$;

In terms of engineering performance and testing, the design should

- demonstrate availability and integration of essential fusion technologies,
- test components for a future reactor, and
- test tritium breeding module concepts; with a 14 MeV-neutron power load on the first wall ≥ 0.5 MW/m^2 and fluence ≥ 0.3 MWa/m^2.

In addition, the device should:

- use as far as possible technical solutions and concepts developed and qualified during the previous period of the EDA, and
- cost about 50% of the direct capital cost of the 1998 ITER Design.

[1] Record of the ITER Meeting, "ITER Council Proceedings: 1999", ITER Documentation Series No 17, IAEA, Vienna, p11

[2] ITER Final Design Report, Cost Review and Safety Analysis, "ITER Council Proceedings: 1998", ITER Documentation Series No 15, IAEA, Vienna, p148

Table 1.1.3-1 ITER Detailed Technical Objectives and Performance Specifications

Plasma Performance

The device should:

- *achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power of at least 10 for a range of operating scenarios and with a duration sufficient to achieve stationary conditions on the timescales characteristic of plasma processes.*
- *aim at demonstrating steady-state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5.*

In addition, the possibility of controlled ignition should not be precluded.

Engineering Performance and Testing

The device should:

- *demonstrate the availability and integration of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance);*
- *test components for a future reactor (such as systems to exhaust power and particles from the plasma);*
- *Test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high grade heat, and electricity production.*

Design Requirements

- *Engineering choices and design solutions should be adopted which implement the above performance requirements and make maximum appropriate use of existing R&D database (technology and physics) developed for ITER.*
- *The choice of machine parameters should be consistent with margins that give confidence in achieving the required plasma and engineering performance in accordance with physics design rules documented and agreed upon by the ITER Physics Expert Groups.*
- *The design should be capable of supporting advanced modes of plasma operation under investigation in existing experiments, and should permit a wide operating parameter space to allow for optimising plasma performance.*
- *The design should be confirmed by the scientific and technological database available at the end of the EDA.*
- *In order to satisfy the above plasma performance requirements an inductive flat top capability during burn of 300 to 500s, under nominal operating conditions, should be provided.*
- *In order to limit the fatigue of components, operation should be limited to a few 10s of thousands of pulses*
- *In view of the goal of demonstrating steady-state operation using non-inductive current drive in reactor-relevant regimes, the machine design should be able to support equilibria with high bootstrap current fraction and plasma heating dominated by alpha particles.*
- *To carry out nuclear and high heat flux component testing relevant to a future fusion reactor, the engineering requirements are*

  Average neutron flux ≥ 0.5 *MW/m*^{2}

  Average fluence ≥ 0.3 *MWa/m*^{2}
- *The option for later installation of a tritium breeding blanket on the outboard of the device should not be precluded.*
- *The engineering design choices should be made with the objective of achieving the minimum cost device that meets all the stated requirements.*

Operation Requirements

The operation should address the issues of burning plasma, steady state operation and improved modes of confinement, and testing of blanket modules.

- *Burning plasma experiments will address confinement, stability, exhaust of helium ash, and impurity control in plasmas dominated by alpha particle heating.*
- *Steady state experiments will address issues of non-inductive current drive and other means for profile and burn control and for achieving improved modes of confinement and stability.*
- *Operating modes should be determined having sufficient reliability for nuclear testing. Provision should be made for low-fluence functional tests of blanket modules to be conducted early in the experimental programme. Higher fluence nuclear tests will be mainly dedicated to DEMO-relevant blanket modules in the above flux and fluence conditions.*
- *In order to execute this program, the device is anticipated to operate over an approximately 20 year period. Planning for operation must provide for an adequate tritium supply. It is assumed that there will be an adequate supply from external sources throughout the operational life.*

1.1.4 Modeling of Design Alternatives

To find a set of consistent overall parameters of a tokamak device, a set of non-linear equations are solved, which describe different aspects of the machine performance, both in engineering and in physics. These "system" equations often represent simplifications of much more complex phenomena. The equations that define the physics performance and power balance are often zero dimensional, including the scaling law for energy confinement predictions. Engineering equations, both for plasma and structures, can be very detailed but, with some generally applicable exceptions, must be extracted and qualified by a specific design solution already studied in depth. A costing algorithm completes the suite of procedures, giving the capability to investigate cost trends as a function of dependent variables.

For any given finite Q, four parameters, i.e., the plasma aspect ratio, maximum toroidal field (TF), plasma elongation, and poloidal magnetic flux consumed during the plasma burn phase, are not mutually independent. Allowable elongation, with a given set of plasma vertical position and shape control constraints, is in fact also a function of the aspect ratio. Moreover, for any given burn flux and aspect ratio, the peak field in the TF magnet is automatically determined. There is a limit on plasma triangularity which is strongly interconnected with the divertor geometry, shape control, and issues related to the single null divertor operation, such as the distance separating active and inactive separatrixes (see Figure 1.2.1-5).

On this basis, the system studies indicated a domain of feasible design space, with aspect ratios in the range 2.5 to 3.5 and a major radius around 6 m, able to meet the technical guidelines ad objectives, with a shallow cost minimum across the aspect ratio range. The shallowness of the cost curve and the inevitable approximate nature of the system studies made it clear that no particular choice can be made on the optimal aspect ratio based on estimated costs alone. In addition, there are other important aspects (e.g. plasma access and in-vessel maintenance) for which the cost or performance impact may not be easily factored into a systems optimisation.

1.1.5 Convergence to an Outline Design

To provide a basis for rigorous exploration and quantification of the issues and costing, representative design options that span an appropriate range of aspect ratio and magnetic field were selected for further elaboration and more comprehensive consideration. A task force involving the JCT and the HTs met during 1998 and 1999 to analyse and compare them.

The development of specific representative options provided a more tangible appreciation of the key issues, and a practical framework for the process of convergence was explored and clarified in the joint Task Force. The Task Force recommendations were instrumental in developing consensus on the criteria and rationale for the selection of major parameters and concepts as the precursor to converging and integrating the various considerations into a single coherent outline design which is described in the rest of this report.

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 taking into account their recommendations. The governing body of ITER subsequently approved the design in June 2000 (Moscow ITER Meeting), recognising it as a single mature design for ITER consistent with its revised objectives.

The proposed design is based on:

1. physics understanding: the ITER Physics Basis[1] (IPB) plus new results of “voluntary” physics R&D from Parties;
2. R&D results in technology development since 1992[2], 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 design criteria for limiting the consequences of ITER operation for 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 $Q \geq 10$, in high confinement modes of operation (H mode);
3. 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 required to develop into 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 will be available with an adequate amount of power.

1.1.6 Conclusion

This report marks the achievement of the full technical scope of activities indicated in the ITER EDA Agreement, with a final design which meets the programmatic objective defined in the Agreement and satisfies detailed scientific, technical and costing objectives set by the ITER Council in 1998. With the accompanying body of supporting documentation, the Parties now have at their disposal, in accordance with the purpose of the ITER EDA Agreement, a well founded and robust ITER design that confers a high degree of confidence that it will meet its objectives. While there is still technical work that can be done to finalise the details of procurements and to optimise costs, all technical data necessary for future

[1] Nuclear Fusion 39 (1999) 2137-2664

[2] Y. Shimomura for the ITER Central Team and Home Teams, ITER Technology R&D, Fusion Engineering and Design 55 (2001), 97 - 358.

decisions on the construction of ITER is now available. Following the completion of Explorations, the next step is for the Parties negotiators to agree on a preferred site to allow specific site adaptation, and a text for the construction agreement ready to sign.

1.2 Design Overview

1.2.1 Design

ITER is a long pulse tokamak with elongated plasma and single null poloidal divertor (Figure 1.2.1-1 to Figure 1.2.1-5 and Table 1.2.1-1 to Table 1.2.1-3). The nominal inductive operation produces a DT fusion power of 500 MW for a burn length of 400 s, with the injection of 50 MW of auxiliary power.

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 (TF) coils, a central solenoid (CS), external poloidal field (PF) coils, and correction coils (CC). The centring force acting on the D-shaped toroidal magnets is reacted by these coils by wedging in the vault formed by their straight sections. The TF coil windings are enclosed in strong cases used also to support the external PF coils. The vacuum vessel is a double-walled structure also supported on the toroidal field coils. The magnet system together with the vacuum vessel and internals are supported by gravity supports, one beneath each TF coil.

Inside the vacuum vessel, the internal, replaceable components, including blanket modules, divertor cassettes, and port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics modules, absorb the radiated heat as well as most of the neutrons from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The shielding blanket design does not preclude its later replacement on the outboard side by a tritium-breeding blanket constrained to the same temperature cooling water as the shielding blanket.

The heat deposited in the internal components and in the vessel is rejected to the environment by means of the tokamak cooling water system (comprising individual heat transfer systems) designed to exclude releases of tritium and activated corrosion products to the environment. Some elements of these heat transfer systems are also employed to bake and consequently clean the plasma-facing surfaces inside the vessel by releasing trapped impurities. The entire tokamak is enclosed in a cryostat, with thermal shields between the hot components and the cryogenically cooled magnets.

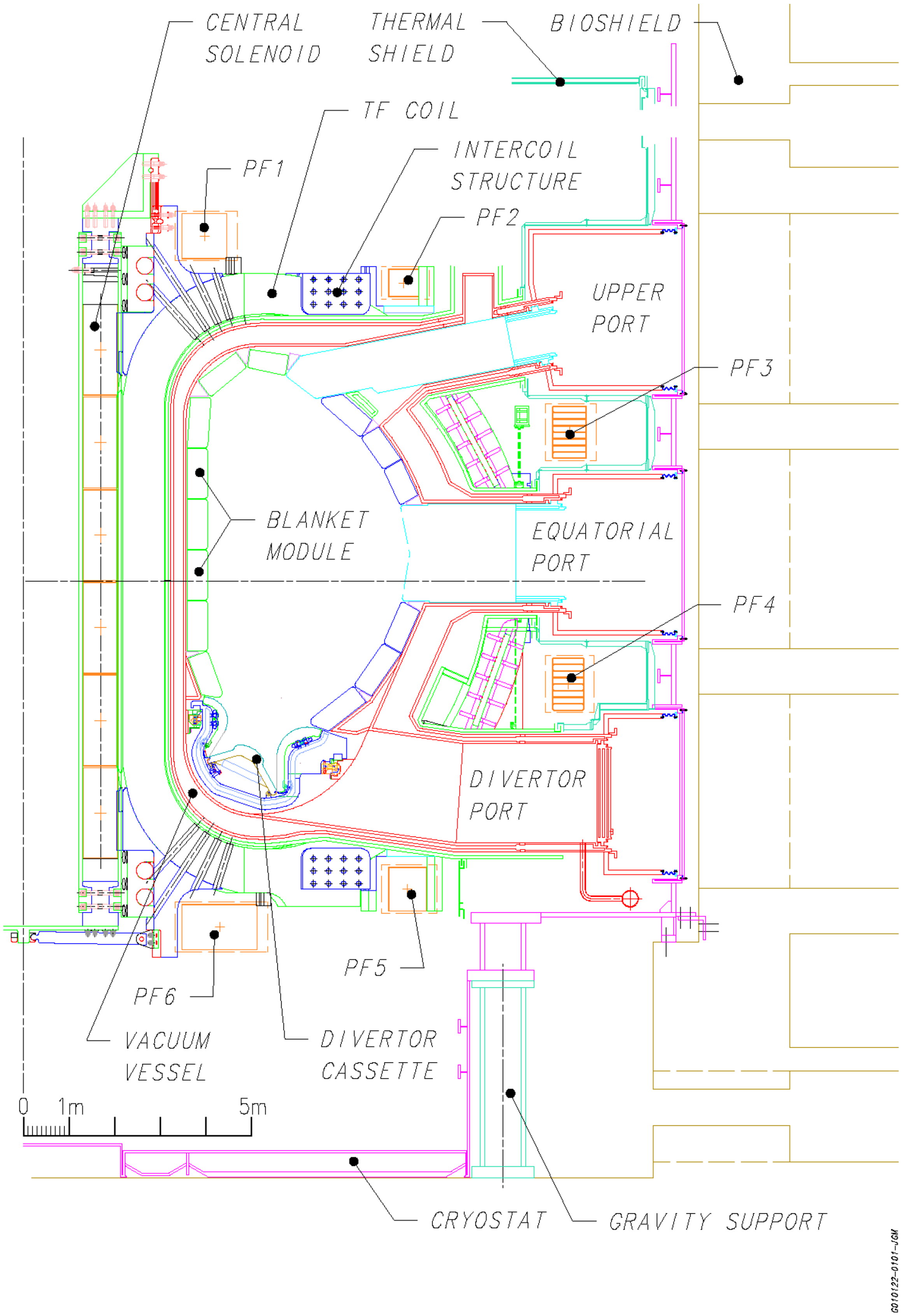


Figure 1.2.1-1 ITER Tokamak Cross-section

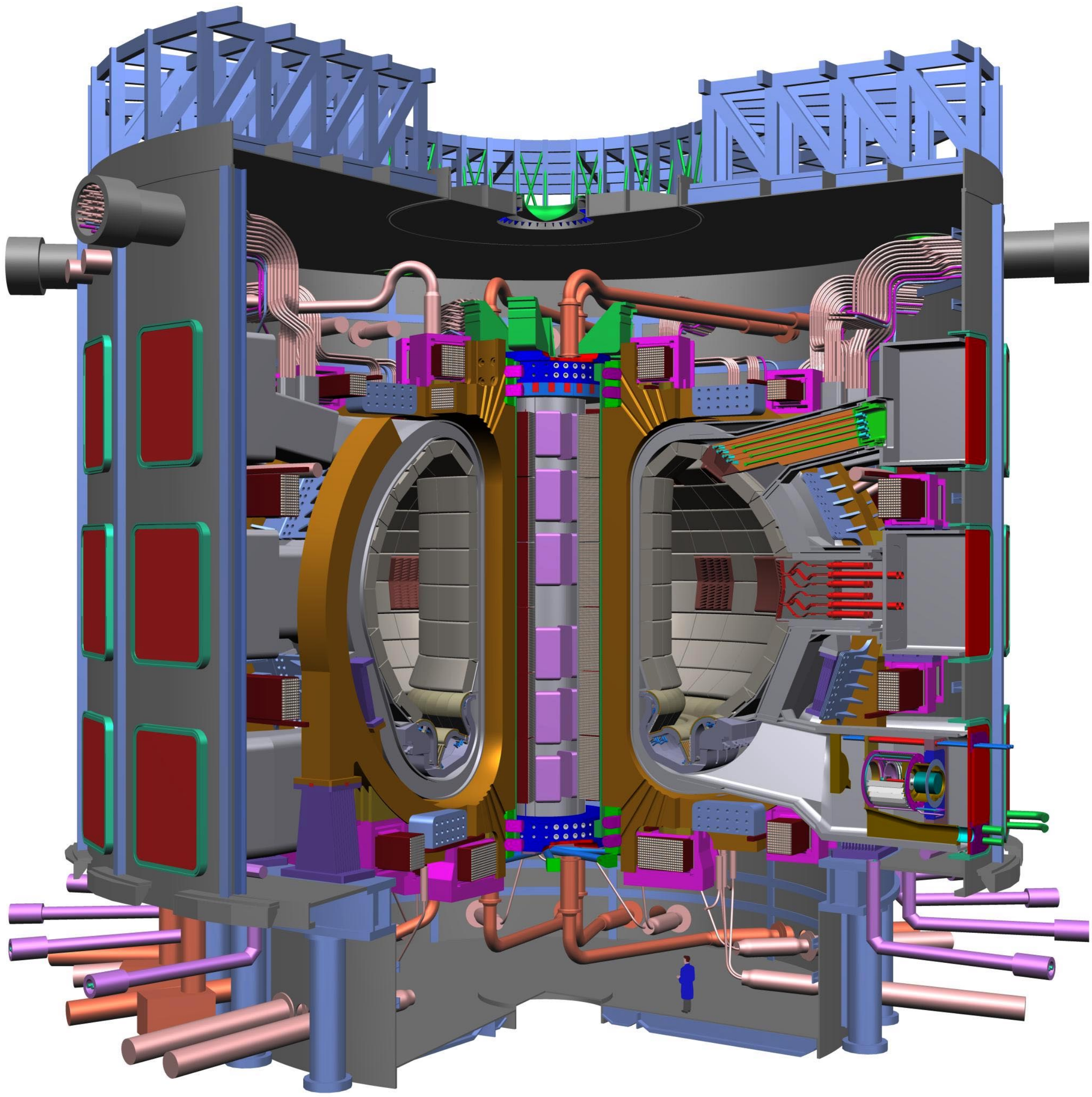


Figure 1.2.1-2 ITER Tokamak Cutaway

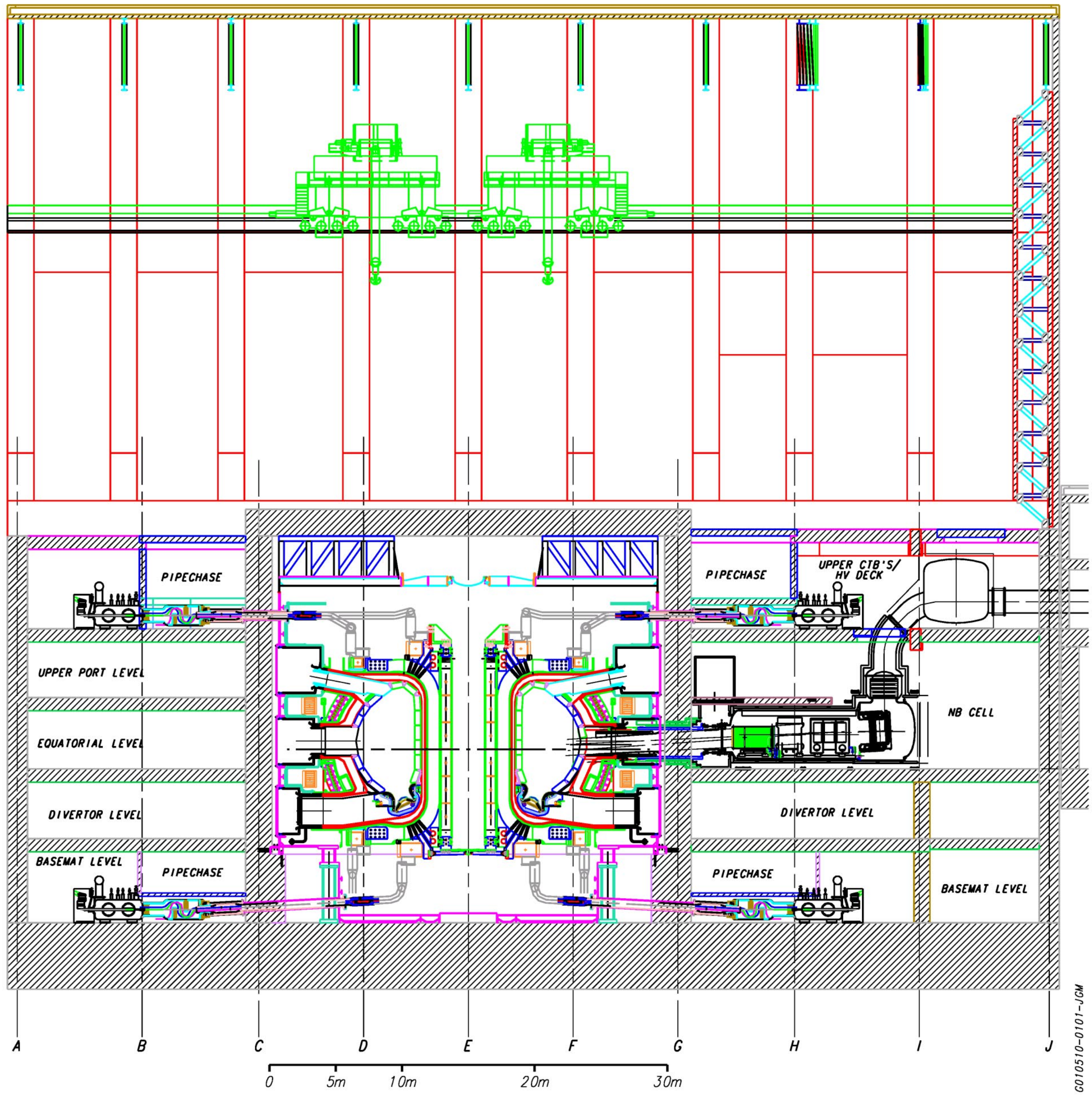


Figure 1.2.1-3 Cross-section NS Through the Tokamak Building

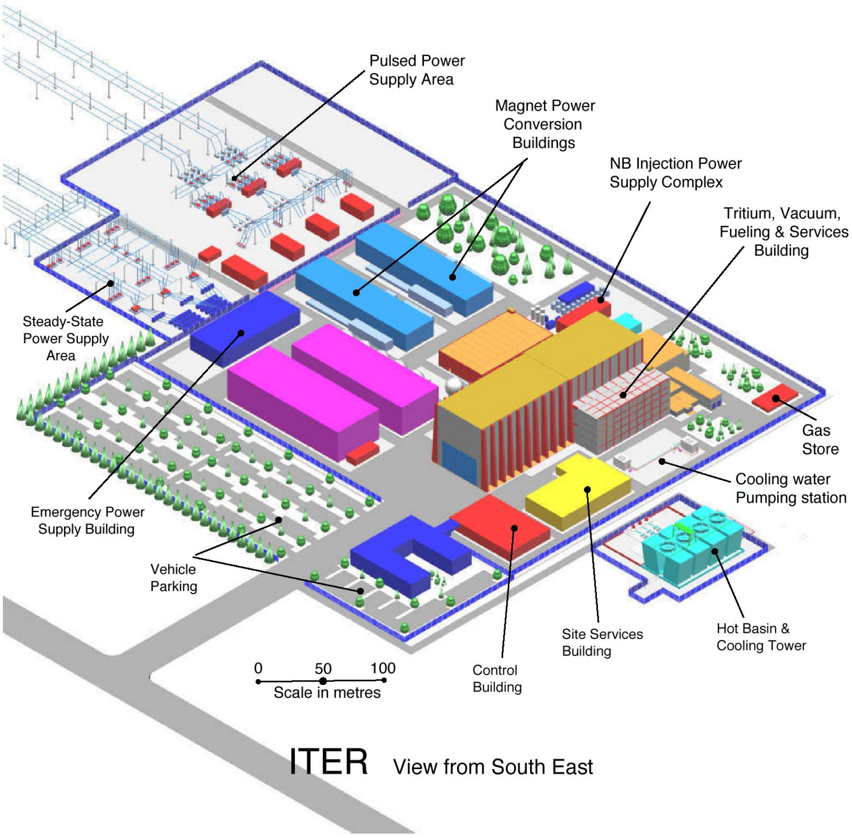
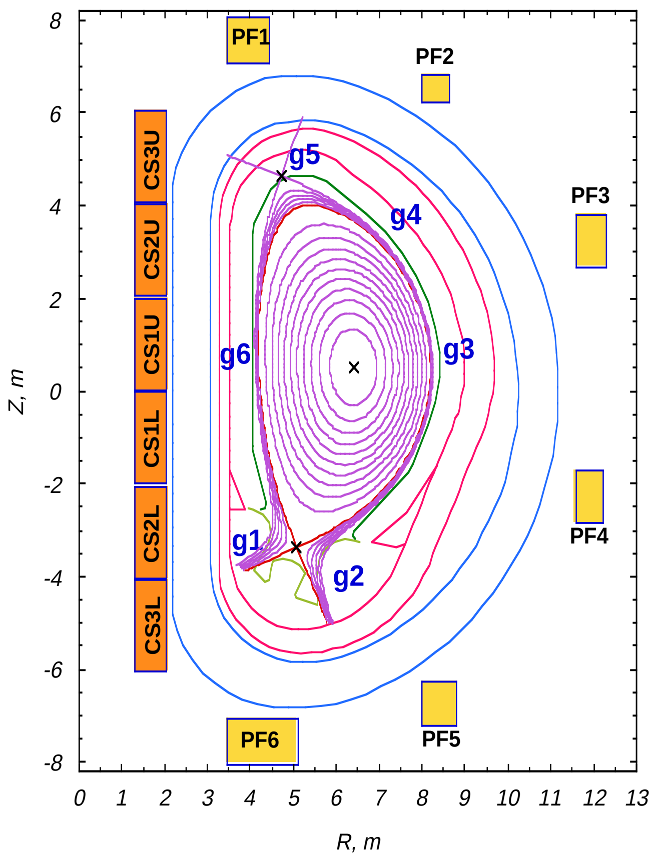


Figure 1.2.1-4 Generic ITER Site View

Figure 1.2.1-5 ITER Nominal Plasma Configuration

(The upper and lower X-points are on different magnetic surfaces, defining two separatrixes, the active one within the inactive one.)

(The g1, … g6 refer to gaps whose sizes are control variables for plasma stability.)



The tokamak fuelling system is designed to inject gas and solid hydrogen pellets. During plasma start-up, low-density gaseous fuel will be introduced into the vacuum vessel chamber by the gas injection system. The plasma will progress from electron-cyclotron-heating-assisted initiation, in a circular configuration touching the outboard limiter, to an elongated divertor configuration as the plasma current is ramped up. Once the current flat top value (nominally 15 MA for inductive operation) is reached, subsequent plasma fuelling (gas or pellets) together with additional heating for ~ 100 s leads to a high Q DT burn with a fusion power of about 500 MW. With non-inductive current drive from the heating systems, the burn duration is envisaged to be extended to 1 hour. In inductive scenarios, before the inductive flux available has been fully used, reducing the fuelling rate so as to slowly ramp-down the fusion power terminates the burn. This phase is followed by plasma current ramp-down and finally by plasma termination. The inductively driven pulse has a nominal burn duration of 400 s, with a pulse repetition period as short as 1800 s. The integrated plasma control is provided by the PF system, and the pumping, fuelling (D, T and impurities such as N_2, Ar) and heating systems all based on feedback from diagnostic sensors.

With regard to safety and licensing issues, the current design focuses on confinement as the overriding safety function, other functions being recognised as being required to protect confinement. Design requirements have been derived from safety principles and release guidelines adopted by the project by identifying the systems, structures, components and procedural measures that can prevent or mitigate releases, and by allocating performance targets (both capability and reliability) to these. 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. 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 1.2.1-1 Main Plasma Parameters and Dimensions

Total Fusion Power	500 MW *(700 MW)*
Q — fusion power/additional heating power	≥ 10
Average 14MeV neutron wall loading	0.57 MW/m^2 *(0.8 MW/m^2)*
Plasma inductive burn time	≥ 400 s
Plasma major radius (R)	6.2 m
Plasma minor radius (a)	2.0 m
Plasma current (I_p)	15 MA *(17 MA [(1)])*
Vertical elongation @95% flux surface/separatrix (κ_{95})	1.70/1.85
Triangularity @95% flux surface/separatrix (δ_{95})	0.33/0.49
Safety factor @95% flux surface (q_{95})	3.0
Toroidal field @6.2 m radius (B_T)	5.3 T
Plasma volume	837 m^3
Plasma surface	678 m^2
Installed auxiliary heating/current drive power	73 MW [(2)]

(1) The machine is capable of a plasma current up to 17MA, with the parameters shown in parentheses) within some limitations over some other parameters (e.g., pulse length).

(2) A total plasma heating power up to 110MW may be installed in subsequent operation phases.

Table 1.2.1-2 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	Six modules of 5 hexa- and 1 quad-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 for TF ripple reduction
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, modular cassettes with separable high heat flux components
Materials	W alloy and C plasma facing components Copper alloy heat sink, SS 316 LN structure
Cryostat	
Structure	Reinforced 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, up to 110 MW maximum
Candidate systems	Electron Cyclotron, Ion Cyclotron, Lower Hybrid, Negative Ion Neutral Beam
Electrical Power Supply	
Total pulsed active/reactive power from grid	500 MW / 400 Mvar
Total steady state active/reactive power	110 MW/ 78 Mvar

Table 1.2.1-3 Heating and Current Drive Systems

	NB (1MeV)	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 used either i) in up to 3 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.				

1.2.2 Operation Scenarios and Phases

As an experimental device, ITER is required to be able to cope with various operation scenarios and configurations. 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.

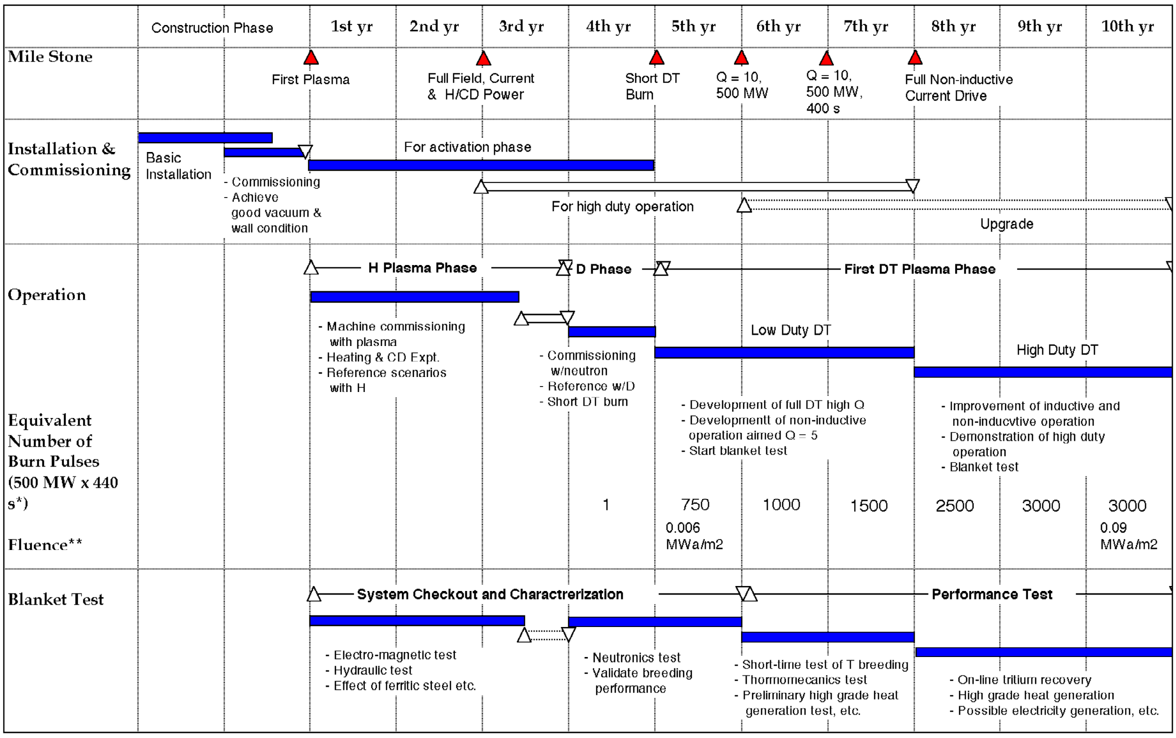
Four reference scenarios are identified for design purposes. Three alternative scenarios are specified for assessment purposes to investigate how plasma operations will be possible within the envelope of the machine operational capability assuming a reduction of other concurrent requirements (e.g. pulse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

<u>Design scenarios</u>

1. Inductive operation I: P_{fus}=500 MW, Q=10, I_p=15 MA, with heating during current ramp-up
2. Inductive operation II: P_{fus}=400 MW, Q=10, I_p=15 MA, without heating during current ramp-up
3. Hybrid operation (i.e., plasma current driven simultaneously by inductive and non-inductive means)
4. Non-inductive operation type I: weak negative shear (WNS) operation

<u>Assessed scenarios</u>

5. Inductive operation III: P_{fus}=700 MW, I_p=17 MA, with heating during current ramp-up
6. Non-inductive operation type II: strong negative shear (SNS) operation
7. Non-inductive operation type III: weak positive shear (WPS) operation



* The burn time of 440 s includes 400 s flat top plus 40 s of full power neutron flux to allow for contributions during ramp-up and ramp-down
** Average fluence at first wall (neutron wall load is 0.56 MW/m^2 on average and 0.77 MW/m^2 at outboard equator)

Figure 1.2.2-1 Initial Operation Plan

During its lifetime, ITER will be operated in successive phases.

H Phase

This is a non-nuclear phase using only hydrogen or helium plasmas, planned mainly for complete commissioning of the tokamak system in a non-nuclear environment where remote handling maintenance 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.

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 a suitably high plasma density.

D Phase

The characteristics of deuterium plasma 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 further. Since some tritium will be generated in the plasma, fusion power production for short periods of time without fully implementing the cooling and tritium-recycle systems could therefore also be demonstrated. By using limited amounts of tritium in a deuterium plasma, the integrated nuclear commissioning of the device will be possible. In particular, the shielding performance will be tested.

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 reactor relevant test blanket modules will also be tested whenever significant neutron fluxes will be 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 a higher neutron fluence. This phase will address the issues of higher availability and further improved modes of plasma operation. The implementation and the programme for this phase will be decided following a review of the results from the preceding three operational phases and an assessment of the merits and priorities of programmatic proposals.

A decision on incorporating in the vessel a tritium breeding blanket during the course of the second DT phase will be taken on the basis of the availability of this fuel from external sources, its relative cost, the results of breeder blanket module testing, and acquired experience with plasma and machine performance.

1.3 Plasma Performance

According to the conclusions of the ITER Physics Basis, obtained from broadly based experimental and modelling activities within the fusion programmes of the ITER Parties, the regime assumed for nominal inductive operation of ITER is the ELMy H-mode confinement regime in the presence of edge localized modes (ELMs).

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 ELMs. The interest in ELMy H-modes follows from experimental observations that show that this mode reduces transport throughout the whole plasma. The standard working hypothesis, supported by many observations, is that H-mode occurs when the power transported across the separatrix (P_{loss}), which must be compensated for by internal and external heating, exceeds a threshold value (P_{L-H}).

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

$$\tau_{E,th}^{IPB98(y,2)} = 0.0562 H_H I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_x^{0.78} \qquad \text{(rms err. 0.13)}$$

Where the units are s, MA, T, $10^{19}m^{-3}$, MW, m and amu, where ε=a/R and P is the total (from internal and external sources) power crossing the separatrix as P_{loss}, 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 $P_{loss} > P_{L-H}$, with

$$P_{L\text{-}H} = 2.84 M^{-1} B_T^{0.82} \bar{n}_e^{0.58} R^{1.00} a^{0.81} \qquad \text{(Rms err. 0.27)}$$

Where the units are MW, amu, T, $10^{20}m^{-3}$, m. No statistical uncertainty factor is included in the second equation, so all uncertainties can be handled by assumed variations in H_H.

1.3.1 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_{add}, which can be formulated approximately as:

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

With X ~ 50-60, a slowly varying function of parameters. This relation provides the basis for I_P = 15MA, R/a = 3.1 if Q = 10, X = 55 and H_H = 1.

Expressing now $I_p \cdot R/a = 5 \cdot B_T \cdot a/q \cdot f$, 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; additionally 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 of B_T at the expense of a smaller plasma volume, a poorer access to the plasma for heating and maintenance as well as a more difficult plasma shape control. A compromise amongst these and other more detailed considerations leads the choice to the present value.

1.3.2 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, shown for example in Figure 1.3.2-1, are the consequence of transport properties which are governed mainly 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.

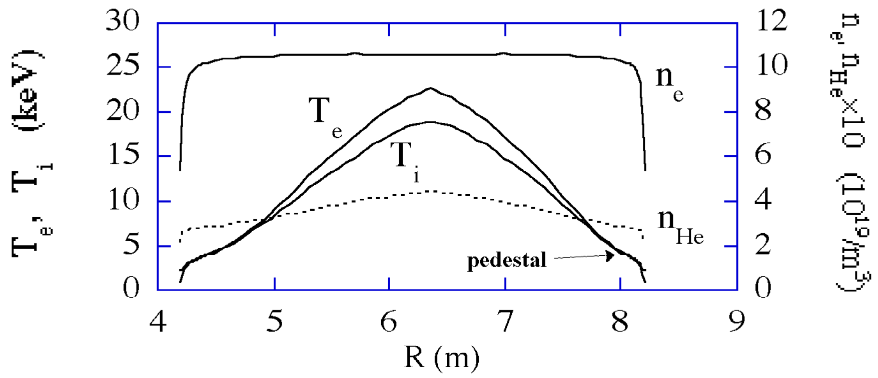


Figure 1.3.2-1 **Profiles of Electron Temperature (T_e), Ion Temperature (T_i), Electron Density (n_e), Helium Density (n_{He})**

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, simulation codes have been written to model plasma evolution. These programs 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.

1.3.3 H-mode Pedestal and ELMs

The transport barrier that occurs just inside the magnetic separatrix in H-mode provides a thin layer where the pressure increases sharply (a large radial gradient is established). 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 remaining 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 in the plasma boundary region, due to an instability that 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' frequency becomes smaller, their amplitude increases and the energy removed from the pedestal by each ELM becomes larger and, as it is deposited onto the divertor targets, leads to rather strong erosion. The physical phenomena involved are not understood in quantitative terms at present.

1.3.4 Internal Confinement Barrier

In some conditions (not completely understood or controlled until now), a confinement barrier might occur inside the core, and limit even more the turbulent heat conduction across the plasma. For its existence this barrier again appears to require a threshold in the power crossing it. The barrier 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 minor radius. In addition, because a toroidal current is driven by the pressure gradient (the so called "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.

1.3.5 Non-axisymmetric Perturbations, Islands, and β limits

Because fusion power production scales as β^2B^4, there is motivation 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 $< 4\ l_i \simeq 3.5$ (where l_i is the plasma internal inductance).

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 magnetic "islands" which grow from a small seed width to a much larger dimension as β_N reaches values considerably lower than 3.5. The observed limit in present experiments is around 2.5 but it may decrease with ρ^* (which decreases with increase in the size of the device) by a factor ~ 3. Nevertheless, the stabilisation of these "neoclassical tearing modes" (NTMs) has been achieved by a localised plasma current addition, driven by electron cyclotron waves on the specific magnetic surfaces, a method that will be employed in ITER.

Moreover, the existence of small-amplitude non-axisymmetric "error" fields produced by residual asymmetries in the magnetic coil positions or in ferromagnetic material distributions, can lead to the development of large magnetic islands, again on low order rational magnetic surfaces, and subsequently to disruptions. These error fields should be eliminated by appropriate currents in the correction coils system producing 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 chaotic ergodisation of the magnetic field lines, subsequently a large heat flow occurs along field lines to the boundary walls, cooling the plasma and leading to an influx of impurities. This is followed by a rapid decrease of the plasma current. 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 significant fast electron "runaway" currents, if the confinement of energetic electrons is not limited by the magnetic fluctuations which may remain from the previous field ergodisation phase. These effects, if repeated often at the same location, can lead to sufficient damage that a refit of the components will become necessary, and they are therefore to be avoided whenever possible.

1.3.6 Divertor and Power Exhaust

The magnetic field configuration in Figure 1.2.1-5 shows closed nested magnetic surfaces with increasing internal volumes from the plasma magnetic axis until a separatrix occurs, outside of which magnetic surfaces are open. The particles travel along largely toroidal field lines which slowly rotate poloidally around these surfaces, but they can diffuse outwards due to collisions. The particles diffusing out of the plasma through the separatrix flow along the field lines until they hit a “target”. Thus the plasma contact with the wall is located at 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 to induce power losses, the power density on the target (even taking into account its inclined position and the flux expansion due to a smaller poloidal field B_p) would be too large for the capability of heat removal and the surface material temperature. This power should remain 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.

Alternatively, if the plasma density is large 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 more favourable condition, 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 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 highlights one of the main functions of the divertor: to protect the main plasma from impurities originating from plasma-wall interactions.

Another important 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 as small as possible (a few % of the electron density) in order not to dilute the reacting ions D and T.

These helium particles are born in DT fusion reactions with an energy of 3.5 MeV. They quickly 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 therefore 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 appears not to be detrimental in ITER according to the present understanding (with the expected values of alpha pressure and its gradient).

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

Together with He, also the outward flow of D and T is pumped at the edge. The plasma density is the result of this outward flow against the gas fuelling near the separatrix and/or solid (D or T) pellet periodic injection (a few tens of mm^3 at a few Hz). The objective with pellets is to push them as far as possible toward the plasma core, well inside the H-mode barrier. With this type of deep fuelling a density gradient will be present leading to an even more favourable plasma performance.

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

1.3.7 Plasma performance

The rules and methodologies for projection of plasma performance to the ITER scale are those established in the ITER Physics Basis, which has been developed from broadly-based experimental and modelling activities within the magnetic fusion programmes of the ITER Parties.

Key limiting factors for inductive operation are normalised beta (β_N), density in relation to the Greenwald density (n_G), and the L-H mode transition power threshold (P_{LH}). A view can be formed of the range of possible plasma parameters at which Q = 10 by analysing, by means of transport codes, with flat density profile across the plasma, possible operational domains in relation to the above limiting factors, for given values of plasma current and confinement enhancement factor as illustrated in Figure 1.3.7-1.

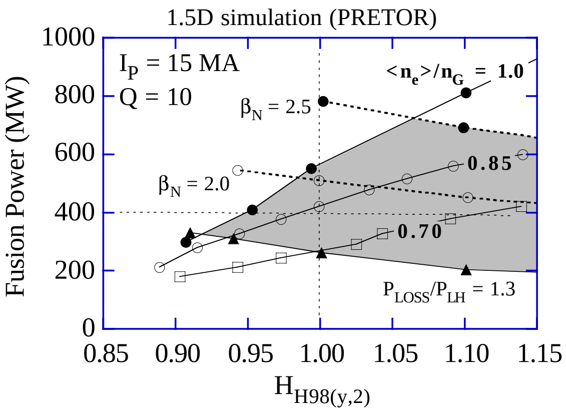


Figure 1.3.7-1 Q = 10 domain (shaded) for I_p = 15 MA (q_{95} = 3.0).

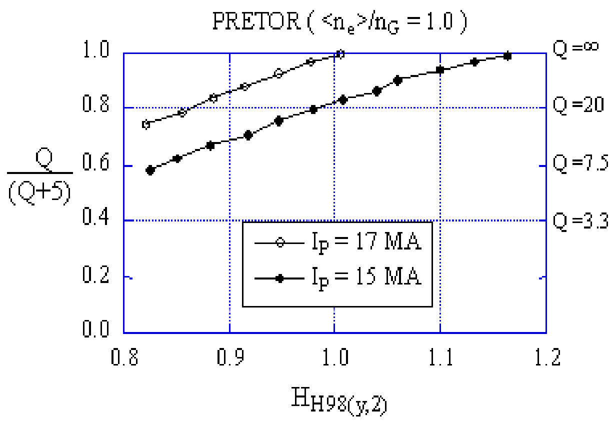


Figure 1.3.7-2 Self-heating Rate versus H_H for I_P=15 and 17 MA.

It is evident from Figure 1.3.7-1 , that:

- for operation at a safety factor at the 95% flux surface, q_{95} = 3 the fusion output power from the new ITER design is in the region of 200-600 MW (at $H_{H98(y,2)}$ = 1), corresponding to a 14MeV neutron flux at the outboard first wall of 0.31-0.92 MWm^{-2}, so that the device retains a significant capability for technology studies, such as tests of tritium breeding blanket modules;

- the margin in H-mode threshold power (at $H_{H98(y,2)} = 1$) is significantly greater than the predicted uncertainty derived from the scaling;
- the device has a capability for Q = 10 operation at $n/n_{GW} \sim 0.7$ and $\beta_N \sim 1.5$ (when $H_{H98(y,2)} = 1$).

The flexibility of the design gives a wide range of operation points not only at Q = 10. For example, different operating points can be obtained by fixing the plasma current and density assumptions and changing the confinement enhancement or the auxiliary power level. These operating points are shown in Figure 1.3.7-2 for I_P=15 and 17MA and for $n=n_G$ and in Figure 1.3.7-3 with n/n_G = 0.7-1.0 and H_H = 1. For all these points, power reaching the edge pedestal exceeds by 1.3 times the L-H transition power. A large fusion gain (Q>20) can be obtained at 15MA with H_H=1. The same conditions would imply ignition (Q=∞) at I_P = 17MA while ignition at I_P = 15MA would be also possible at $H_{H98(y,2)} \sim 1.15$.

The inductive burn pulse length at 15MA and Q=10 is 400s-500s being limited by the available magnetic flux. For this reason the burn time can be increased as the plasma current is reduced and the non-inductive drive power is increased. Figure 1.3.7-4 shows how the burn time and Q change versus plasma current. A pulse length of more than 1000s, for blanket testing, may be easily achieved with Q=5.

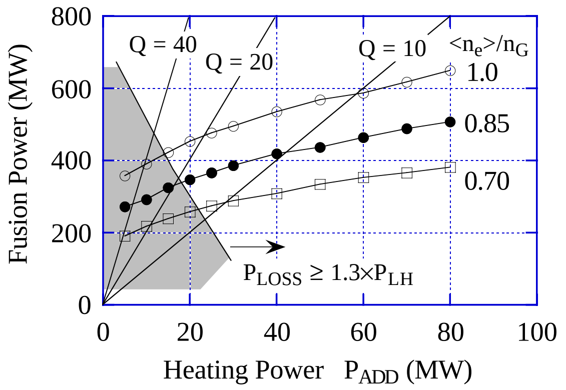


Figure 1.3.7-3

Fusion gain Q at various additional heating power and plasma density
I_P = 15 MA, $H_{H98(y,2)} = 1.0$, $\tau_{He}^*/\tau_E = 5$

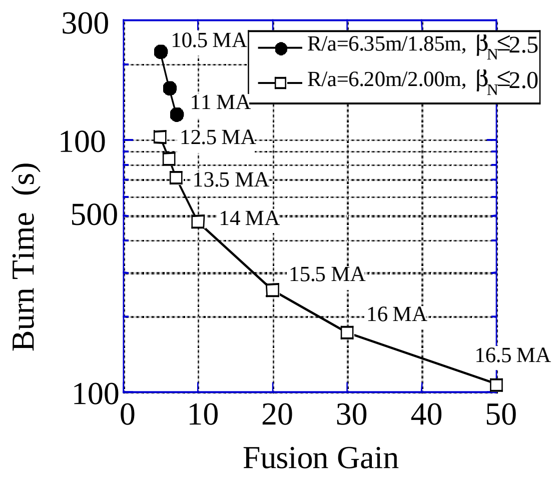


Figure 1.3.7-4

Pulse length versus fusion gain Q.
Each plasma current given in the figure is the minimum value for the corresponding Q value with H_H=1.0 n_e/n_G = 0.85, $\beta_N \leq 2.0$ or 2.5, $P_{loss}/P_{LH} \geq 1.3$ and τ^*_{He}/τ_E = 5. The fusion power is in the range 400- 700 MW.

1.4 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. Figure 1.4.1-1 shows a functional diagram with the basic plant system configuration introducing all systems.

1.4.1 Magnets

The plasma is confined and shaped by a combination of magnetic fields from three main origins: toroidal field coils, poloidal field coils and plasma currents. The nested magnetic surfaces are able to confine a plasma pressure equivalent to a few atmospheres, with a density 10^6 times smaller than in the atmosphere ($n = 10^{20}/m^3$, $T \approx 10$ keV). 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.

1.4.1.1 Toroidal Field Coils

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, the confinement of which leads to significant forces on each coil restrained by a thick steel case to resist circumferential tension (≈ 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 connected together (Figure 1.4.1-2) by bolted structures, and by two compression rings made of unidirectional glass fibres, that provide an initial inward radial force on each coil (2 x 30 MN).

This very robust assembly is provided mainly to resist the toroidal forces induced by interaction of the TF coil current with the transverse poloidal field from plasma and poloidal field coils. These forces produce a distribution of torque around the TF coil proportional to the magnetic flux crossing unit length (the net torque is thus 0). These local forces are pulsed, and therefore mechanical 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 (in particular at their lower curved region) where they are resisted by the friction between coil sides (under high compression) and by specific keys.

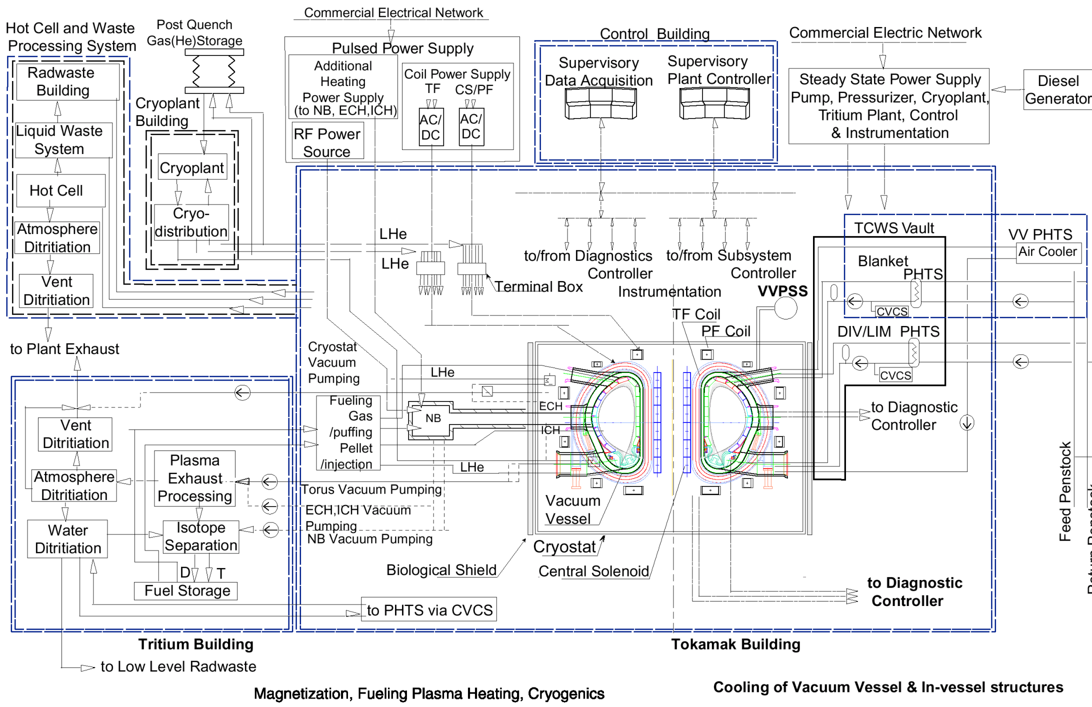


Figure 1.4.1-1 Basic Plant System Configuration

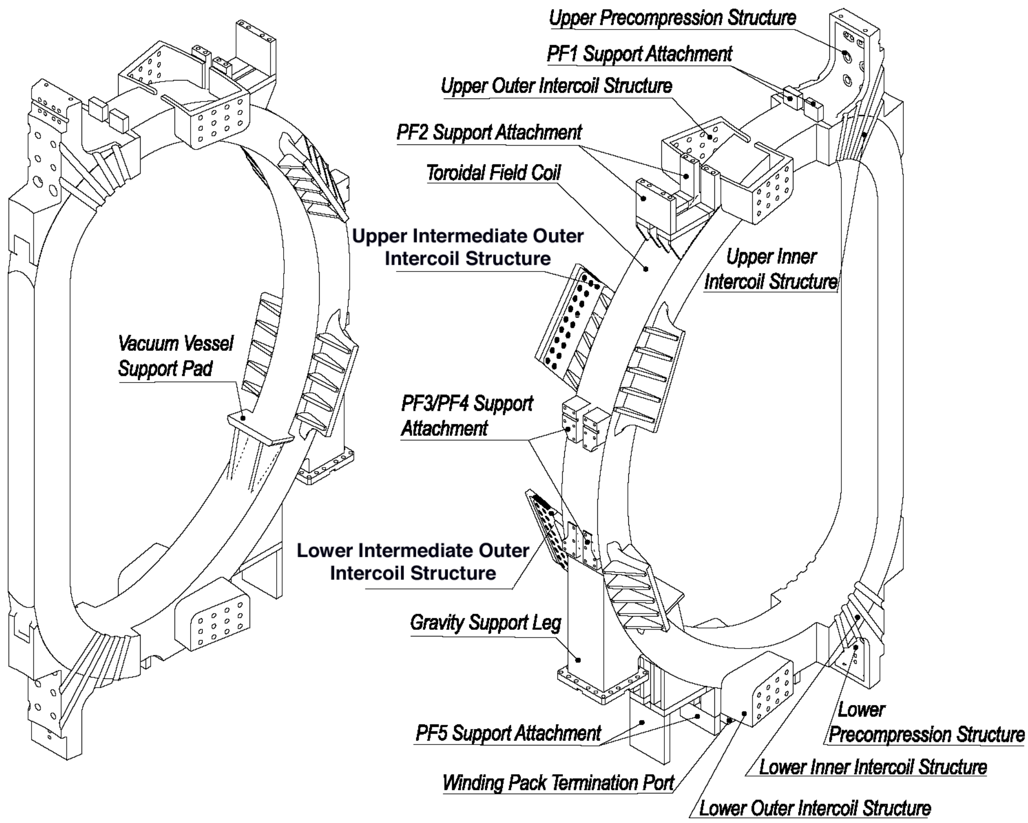


Figure 1.4.1-2 TF Coil Structure

1.4.1.2 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 PF coils placed outside the TF coils. All these axis-symmetric 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 whereas NbTi can be used in the PF coils since the maximum field value is lower than 6T. Redundant turns are built into the trapped coils to allow for failures.

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 double walled vacuum vessel, which passively reacts to slow down the plasma motion. 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 of the expected plasma energy changes, so as to minimise the sources of instabilities.

Second, using an active feedback position control system, the currents in the largest 4 PF coils will be changed through a special power supply feeding them in an anti-symmetric way, across the plasma equatorial plane. These changes provide an additional radial magnetic field leading to the required vertical 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 1.2.1-5) 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 disturbance (e.g. loss of or change in current distribution/internal inductance or loss of plasma thermal energy).

In the inductive scenario, the plasma current is generated by the change in magnetic flux linked with the plasma torus (this is 277 Vs in the nominal 15 MA inductive scenario). 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 external PF coils contribute to make sure that as little poloidal field as possible is present in the plasma region during initiation. Providing a few MW of plasma heating through electromagnetic waves will minimize the flux consumption during plasma initiation and current increase. This method will help securing a robust start-up as well as a sufficient flux variation available (37 Vs) to sustain the current flat top during at least 400s at a plasma current of 15 MA.

Additionally, after the plasma current is set up inductively, a non-inductive scenario may follow. In this type of scenario the plasma current flat top is extended towards steady state by driving the current non-inductively by means of a set of high energy (1 MeV) beams of neutral D tangentially injected at a small angle to field lines. Current drive may 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).

1.4.1.3 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 (Figure 1.4.1-3) are composed of 3 sets of six saddle coils, around the torus located between PF and 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 that are not stabilised by the conductive walls, on the longer time scale associated with the wall resistance.

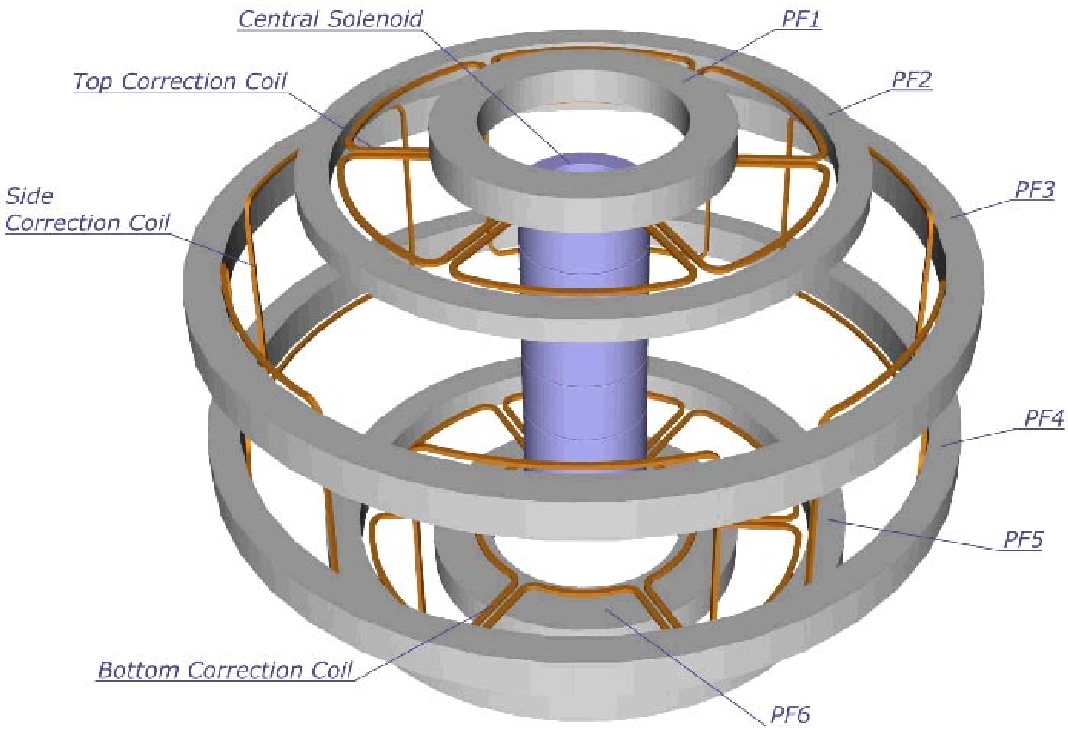


Figure 1.4.1-3 ITER Poloidal Field Coils and Error Field Correction Coils

1.4.1.4 Superconducting Coil Protection

The superconductor of all coils is 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 is small enough to minimise the energy dissipated into the coil and to limit its local temperature increase. However 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. A compromise value of 11s has therefore been chosen for the time constant.

In addition, all these coils must be protected against the heat coming from their surroundings. Therefore, a large cryostat vessel places all the coils in a vacuum good enough to limit convective heat transfers. Additionally a thermal shield (VVTS), cooled at about 80 K by a flow of helium, is provided between the coils and hot parts to shield against radiative heat transfer. The geometry of this thermal shield is evidently rather complex, but the avoidance of radiation hot spots is necessary to limit the already significant amount of power to be removed from the coils at 4.5K. This permanent heat load (~15 kW) due to nuclear radiation, and conduction through supports, adds to the non-ideal efficiency of the circulation pumps feeding the supercritical helium in each coil.

1.4.1.5 Superconducting Coil Cryogenic Cooling

On top of the steady state cryogenic heat load there is a significant pulsed heat load on the coils from two separate sources: 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 (or even more during a plasma disruption). Being the cryogenic plant essentially a steady state system, between the coils and the cryogenic plant, an energy storage is present to cushion the pulsed loads.

In effect, this 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 that 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 about 55 kW.

1.4.2 Vessel and In-vessel Systems

1.4.2.1 Neutron Shielding

The 14 MeV neutrons, i.e. 80% of the fusion energy produced, transfer energy to the water coolant, and subsequently to the environment, by colliding with the materials present around the plasma (mostly steel and water) in the blanket modules and in the vacuum vessel. The small neutron energy, not absorbed in these two shields, is released in the cold TF coil structure, and should be absolutely minimised. Typical maximum nuclear heating in the design is ~ 15 kW.

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 so as to allow human access, in case of any need for repair. As a result, 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 while minimizing the shielding envelope requirement.

The radial thickness distribution between blanket and vessel mainly derives from the requirement of reweldability of the vessel inner shell until the ITER end of life. This involves 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.

1.4.2.2 Blanket Modules

The shielding blanket is divided into two parts: a front part that may be separated from a back one (Figure 1.4.2-1). The back part with a radial thickness of around 30 cm is a pure shield made of steel and water. The front part, the “first wall”, includes diverse materials: l cm thick beryllium armour protection, l cm thick copper to diffuse the heat load as much as possible, and around 10 cm of steel structure. This component will become the most activated and tritium-contaminated in the entire ITER device. 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 possibly changed.

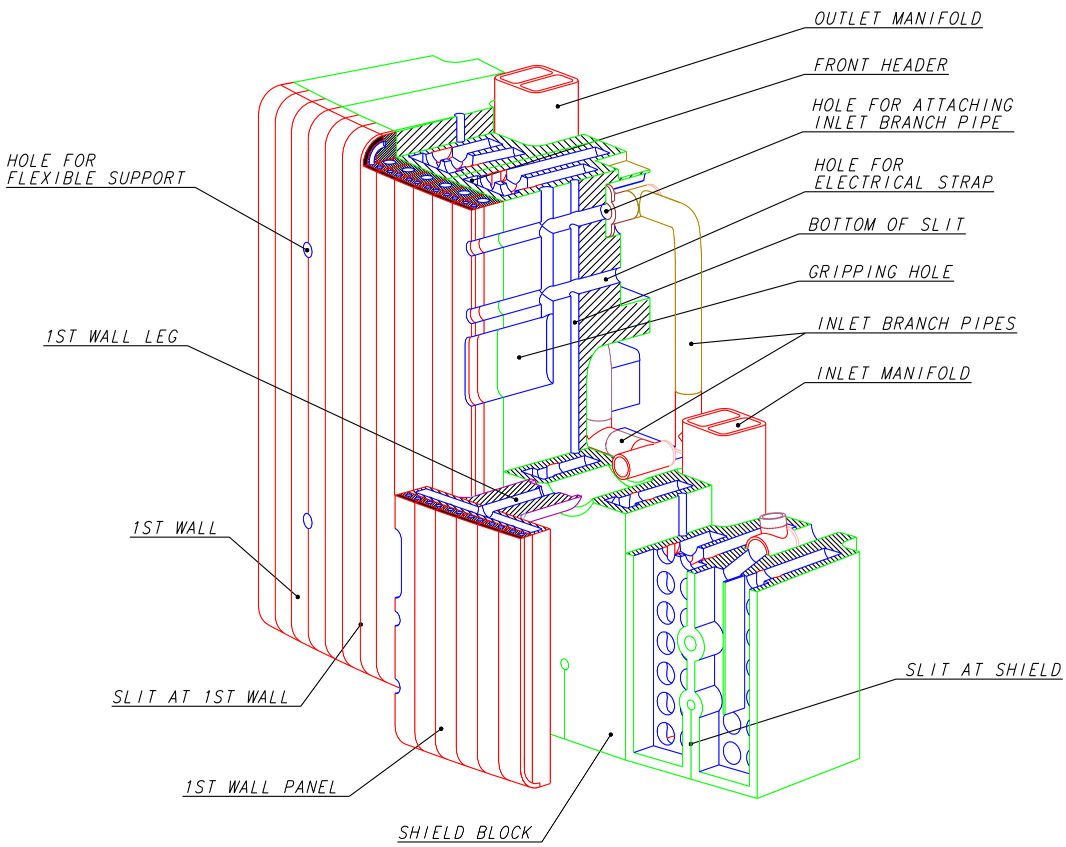


Figure 1.4.2-1 Blanket Module

In order to allow a practical method of maintenance, the blanket wall is modular (~ 420 in total) with a maximum weight of 4.5 t (and about 1.5 m^2 facing the plasma) and moreover the front part of each module is divided in 4-6 first wall panels. Each module is attached to the vessel by 4 flexible links, radially stiff but pliant against toroidal or poloidal motions. This flexibility is required because, across the blanket thickness, the absorbed power density decreases sharply and, whilst 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 thermal expansion together with a "bowing" effect during each plasma pulse. The toroidal and poloidal external forces (i.e. during a disruption) acting on the module are therefore reacted by additional mechanical keys provided with sufficiently large compliance clearances.

1.4.2.3 Blanket maintenance

The maintenance and repair of a blanket module is performed by first removing it from the vessel. For this purpose, a vehicle, equipped with an end gripper, is positioned along a toroidal rail deployed along the vessel torus centreline. The end gripper is engineered to cut the connection to the water pipe feeders and to unbolt the module, and to bring it to an equatorial maintenance door. At this location it will be transferred into a cask, and subsequently to the hot cell for repair or replacement. The cask operates by docking and undocking to the ports of the vessel and of the hot cell, avoiding contamination to the environment. Similar casks are 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.

1.4.2.4 Divertor

The divertor shares with the blanket a similar modular philosophy and maintenance procedure. The cassettes (54 in total) are removed from the vessel at three lower access ports, to which they are beforehand conveyed by a toroidal mover mounted on annular rails attached to the vessel floor. These rails also act as the mounting point of the cassettes during operation.

Besides providing shielding of the vessel, the modular cassettes (Figure 1.4.2-2) support the divertor target plates, a set of particularly high heat flux components, built with high conductivity armour of carbon fibre composite (CFC) and tungsten.

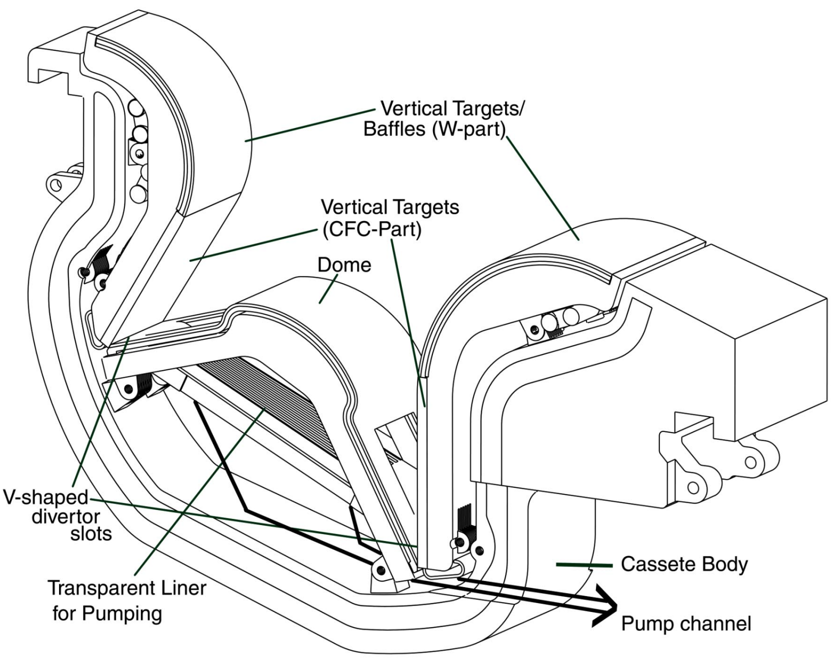
These materials can be eroded by the plasma particles, mostly during short pulses of high heat loads, associated with ELMs or plasma disruptions. This erosion process not only will call for replacement from time to time of the worn out divertor targets, but also may create dust, and in particular tritiated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

1.4.2.5 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. Groups of two or 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 (e.g. by "spiking" specific coolant channels with tracer elements) allows the identification of possible modules or cassettes leaking water, from tests outside the cryostat, a crucial procedure to be able to rapidly localise the leaks in vacuum.

The pressurised coolant water input is continuously maintained around 100°C whereas the output coolant temperature during a pulse at nominal fusion power will be around 150°C. At the end of a pulse, control valves allow the large heat exchanger to the heat rejection system to be short circuited so as not to cool these in-vessel components below 100°C. During standby, the coolant flow is 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. Subsequent to maintenance periods, the pressurisation will be increased, and the water coolant used to heat and to bake the in-vessel components to 240°C.

Figure 1.4.2-2
Divertor Cassette



1.4.2.6 Cryogenic Pumps

Well recessed and shielded from neutrons but inside the divertor port are the torus cryogenic pumps operating at 4.5 K. These have the capacity to pump hydrogenic atoms as well as helium by adsorption and condensation. The pumping performance can be varied and the condensed gases can be removed by heating the pumping panels to 80 K and pumping away the gas released using a roughing pump after a shutter towards the vacuum chamber has been closed. For long plasma pulses, this procedure may be carried out on-line sequentially through all the installed cryogenic pumps in order to limit the amount of hydrogen in each pump below its deflagration level in case of an accidental ingress of oxygen. This limiting amount of hydrogen corresponds to pumping 200 Pam^3s^{-1} of DT for 450 s, with six pumps.

1.4.2.7 Vacuum Vessel

The vacuum vessel is a component with multiple functions, namely it:

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

- supports the in-vessel components and their resultant mechanical loads;
- participates in shielding against neutrons, and in removing the corresponding power during a pulse, and moreover in removing the decay heat of all in-vessel components in case of there being no other coolant available;
- provides a continuous conductive shell for plasma MHD stabilisation with a toroidal one turn resistance of ~8μΩ;
- provides all access to the plasma through ports, for diagnostics, heating systems, pumping, water piping, etc.;
- provides the first confinement barrier for tritium and activated dust with a very high reliability.

All these functions are central to the operation of ITER and thus require a very robust mechanical design analysed for stresses in all possible normal and off-normal conditions. The vessel is built with two shells linked by ribs and fitted with nuclear radiation shielding material, and ferromagnetic inserts in the shadow of the TF coils to reduce the TF ripple value.

To ensure reliable water cooling, two independent loops are used. These 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.

1.4.2.8 Vacuum Vessel Pressure Suppression System

In the case of a water pipe rupture inside the vessel, the subsequent chamber pressure will be limited below 0.2 MPa by the opening of rupture disks and communication with a large container located above the tokamak vacuum 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 of flowing into the vessel will be driven into drain tanks located at the bottom of the tokamak building.

1.4.3 Mechanical Loads and Machine Supports/Attachments

Parts of the technical challenge of the ITER design is due to the large mechanical loads that are applied to the various components.

The mechanical loads acting on ITER fall into four categories.

1. inertial loads due to gravitational and seismic accelerations,
2. kinetic loads due to coolant and atmospheric pressures,
3. thermal loads,
4. electromagnetic loads, 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 Vertical Displacement Events (VDEs).

The chosen ITER support hierarchy is schematically drawn in Figure 1.4.3-1 where all core components of the machine are attached to the TF coil cases. Generally speaking, 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.

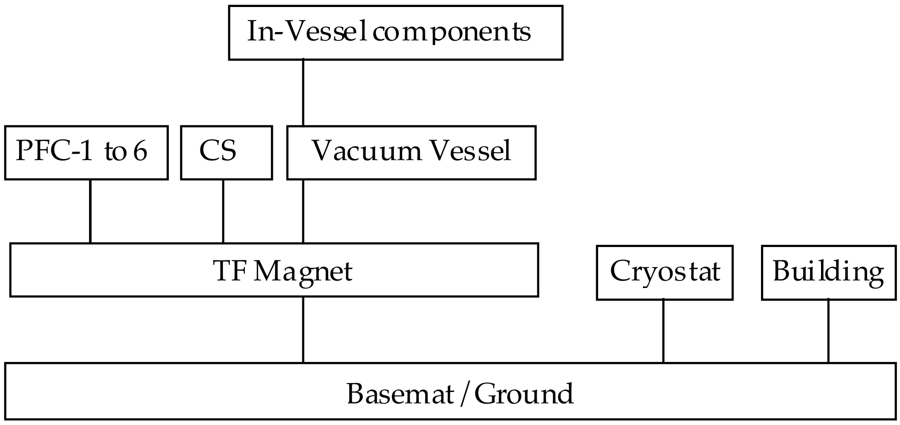


Figure 1.4.3-1 Schematic of Supports Hierarchy

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

1.4.3.1 Seismic Loads

Earthquakes simultaneously produce vertical and horizontal random ground motions that are typically statistically independent, the horizontal ones having the most important impact on the design. Even if the ground peak horizontal acceleration is a fraction of gravity, a seismic event is, in many cases, one of the most demanding loading conditions, 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 that involve horizontal shearing as well as rocking motions

As an additional design constraint, the relative distance between components must be maintained limited in particular in the inboard radial build region where the indirect design “cost” of clearances is large (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 for the design of all components and their support leading to a configuration with some margin. In case of the selection of a site with significantly larger seismic loads, the use of horizontal seismic isolators below the building basemat has been shown to be effective at lowering the peak acceleration to acceptable values.

1.4.3.2 Electromagnetic loads

Beyond the TF coil loads, either static in-plane from the toroidal field itself, or out-of-plane cyclic due to their interaction with the poloidal field, as well as the consequence of a emergency TF energy fast discharge on the VV stress level, other important electromagnetic loads are associated with transient phenomena that are consequences of changes in plasma current, internal energy or position. They act on the PF coils and all conductive structures close to the plasma (blanket modules, divertor cassettes, vacuum vessel).

For slow transients (time scales longer than those which induce significant currents in these structures), there is 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.

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

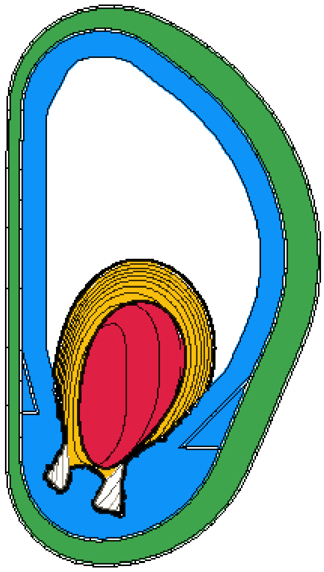


Figure 1.4.3-2 Plasma Current Density during a Downward VDE

In the case of a disruption, the load severity is larger the shorter is the 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 stiffest 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: as a result, VDEs should occur only during a major disruption or a failure of the control system;
- during a major disruption, the plasma will move inward and upward as a consequence of the direct inductive coupling between plasma and passive structure, but vertical forces will be much smaller than for a full downward VDE (Figure 1.4.3-2), where loads up to 80MN may develop and which could occur only in the absence of control; to react to this latter event a “killer pellet” may be used to trigger the plasma quench early during its downward motion, and thus limit the arising vertical loads.

For a conservative design, in ITER these types of events have been often combined with, for example, earthquakes and/or TF coil fast discharge.

1.4.4 Fuel Cycle

The tritium used in ITER will be supplied by external sources. During plasma operation, in order to generate 500 MW of total fusion power, about 0.1 g of tritium will be burnt every

100 s. However, considering the divertor/plasma-purity operational conditions that call for maximum pumping speed and un-burnt fuel recalculation, more than 25 g of tritium will be injected into and pumped from the vessel during the same 100 s.

The need is obvious to process the pumped gases on line, to remove impurities and separate the tritium, and to store it for recycling. A schematic of this fuel cycle is shown in Figure 1.4.4-1. It includes first a permeator to separate impurities from hydrogen in line with the pumping exhaust. After that, the impurity flow is processed before final exhaust, with an ALARA (as low as reasonably achievable) content of tritium. The hydrogen flow is processed to separate the different isotopic 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.

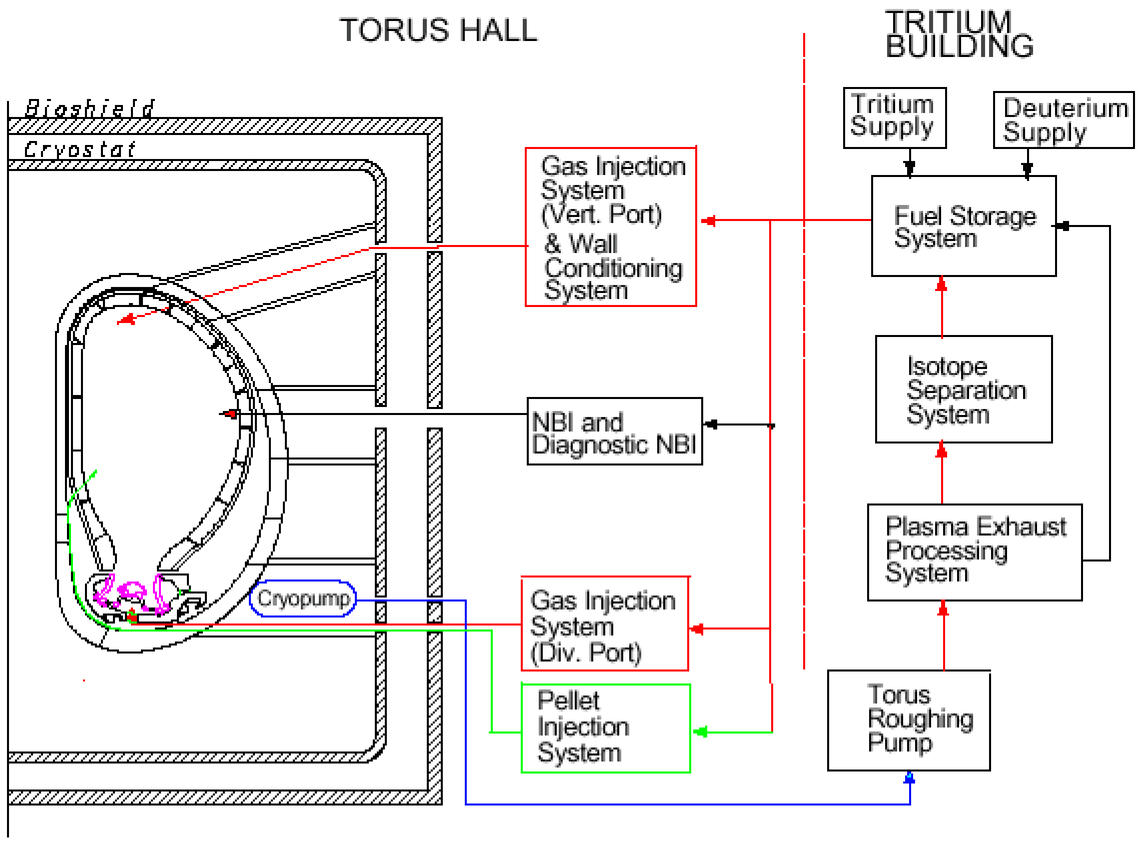


Figure 1.4.4-1 Block Diagram of Fuel Cycle

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 entire fuel cycle is an important issue in particular because a part of the tritium injected in the plasma may remain in the vessel trapped by co-deposition with carbon dust, and it is important to know how much since, if it can be mobilised, there may be a limit placed on operation.

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 re-inject the tritium into the fuel cycle.

1.4.5 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.

- 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.

- Part of the building are 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 case of the tritium building (the metallic equipment inside glove boxes provide the first and second barriers in this case).

- A differential pressure (Figure 1.4.5-1) is 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 will move only from lower to higher contamination levels. These differential pressures are maintained by the air conditioning system. The design arrangement of a separate cell around each vessel port access allows the atmosphere of each cell to be maintained through a venting system capable of detritiation and filtering. This is especially justified during the maintenance procedure when removing components from the vessel occurs.

- 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 is based on the existence of a common stiff basemat designed to react seismic conditions. Should the actual site have much more severe 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 below the accepted design level.

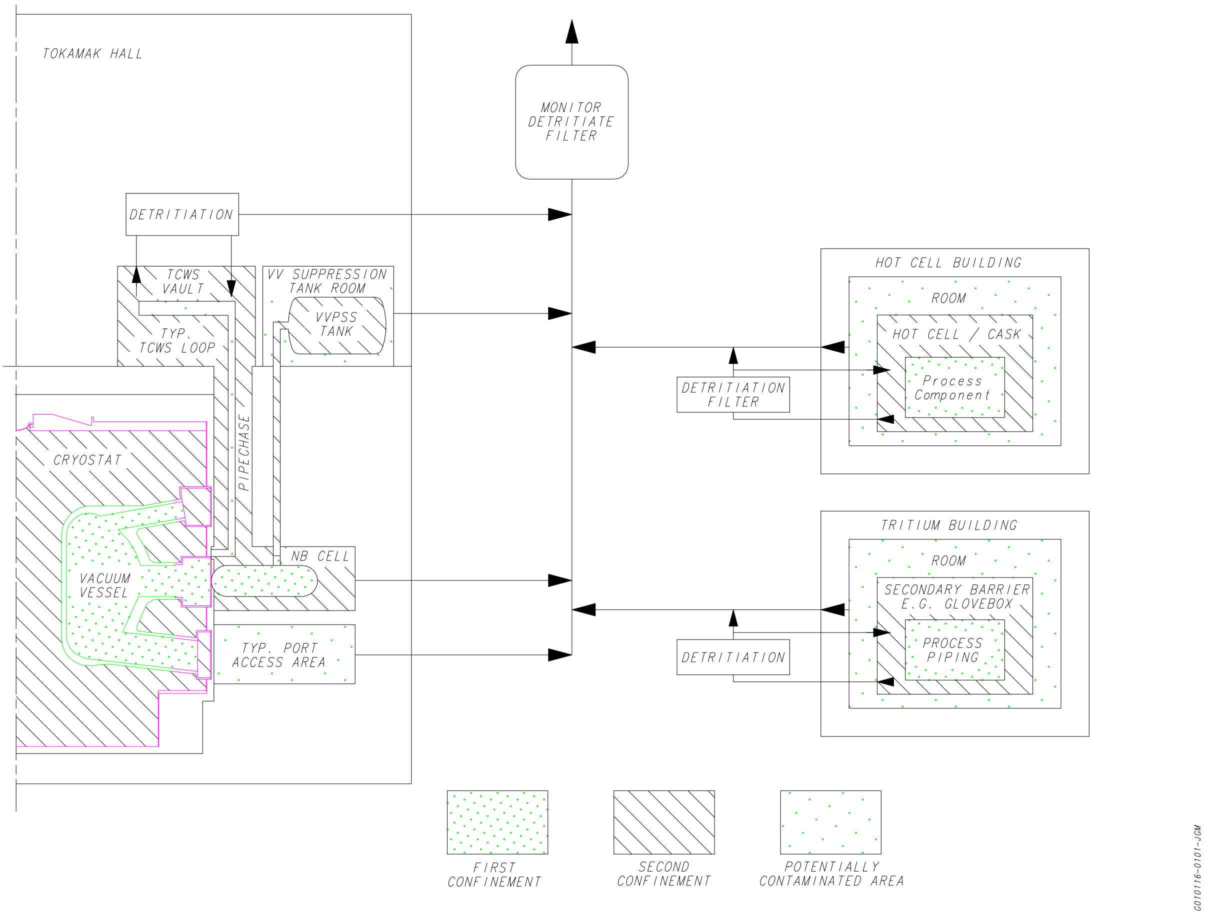


Figure 1.4.5-1 Schematic of Confinement Approach illustrating Successive Confinement Barriers that are Available

1.4.6 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 controls plasma operation. The typical waveforms of a standard driven burn plasma pulse are shown in Figure 1.4.6-1.

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. Reliable control of the fusion burn conditions and provision for its rapid termination under off-normal conditions are thus crucial.

To support plasma operation, all plant systems must be efficiently and reliably controlled. 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.

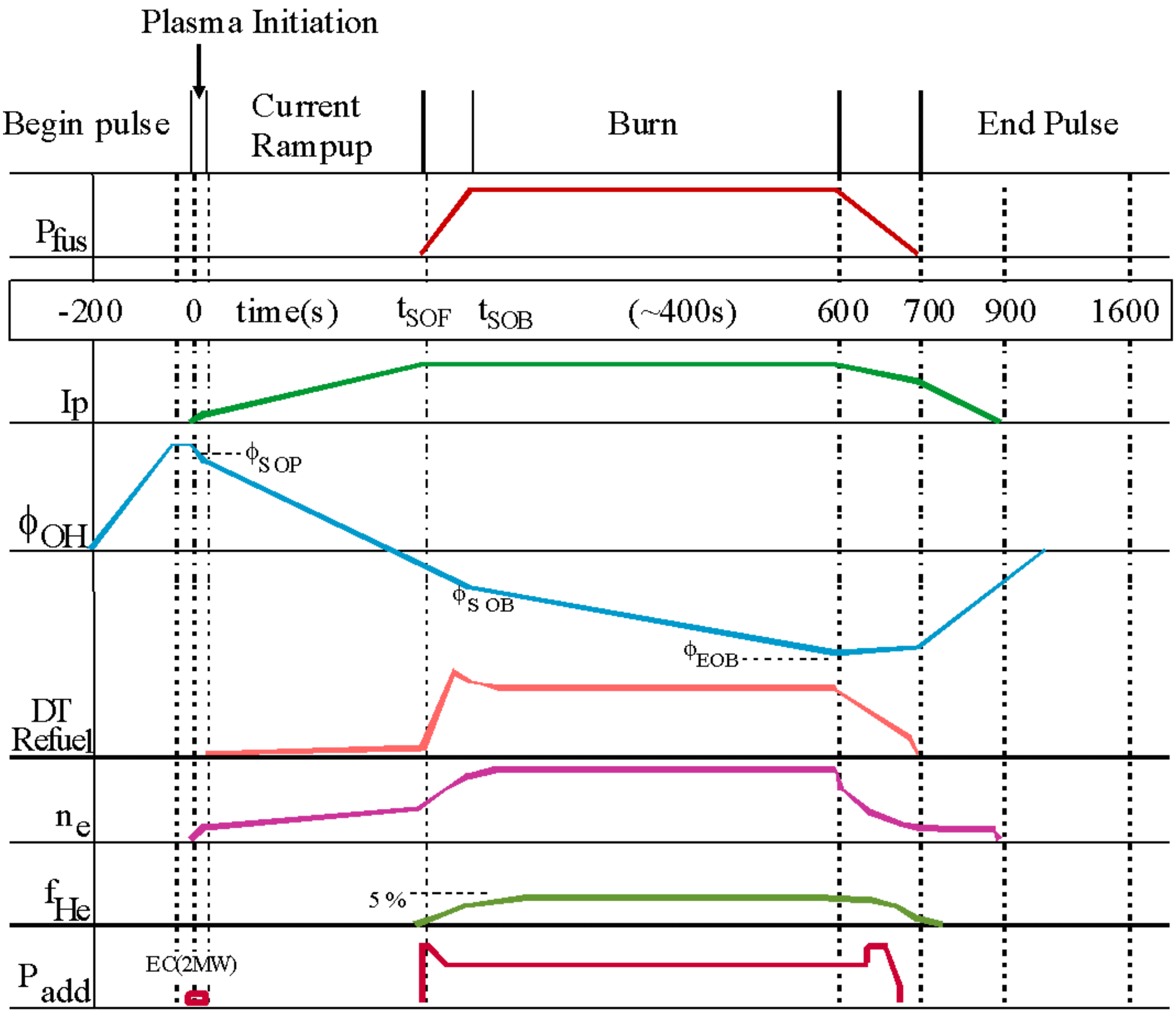


Figure 1.4.6-1 Waveforms for Standard Driven-burn Operation Scenario

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 impurity injection.

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

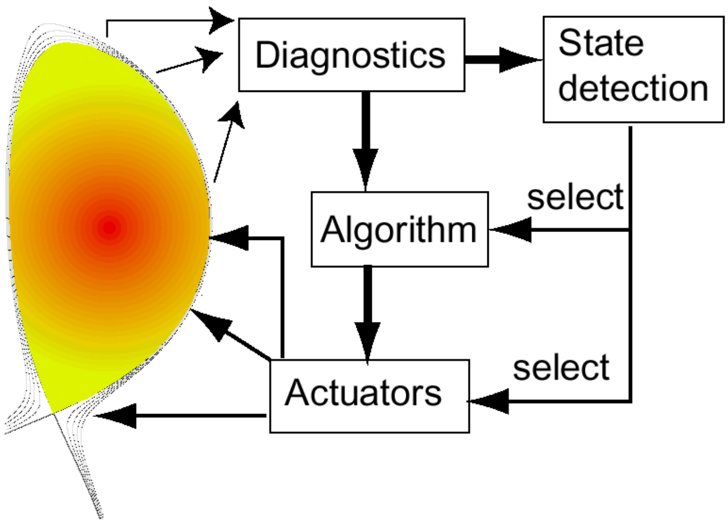


Figure 1.4.6-2 Plasma Feedback Control

Figure 1.4.6-2 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 1.4.6-3.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.

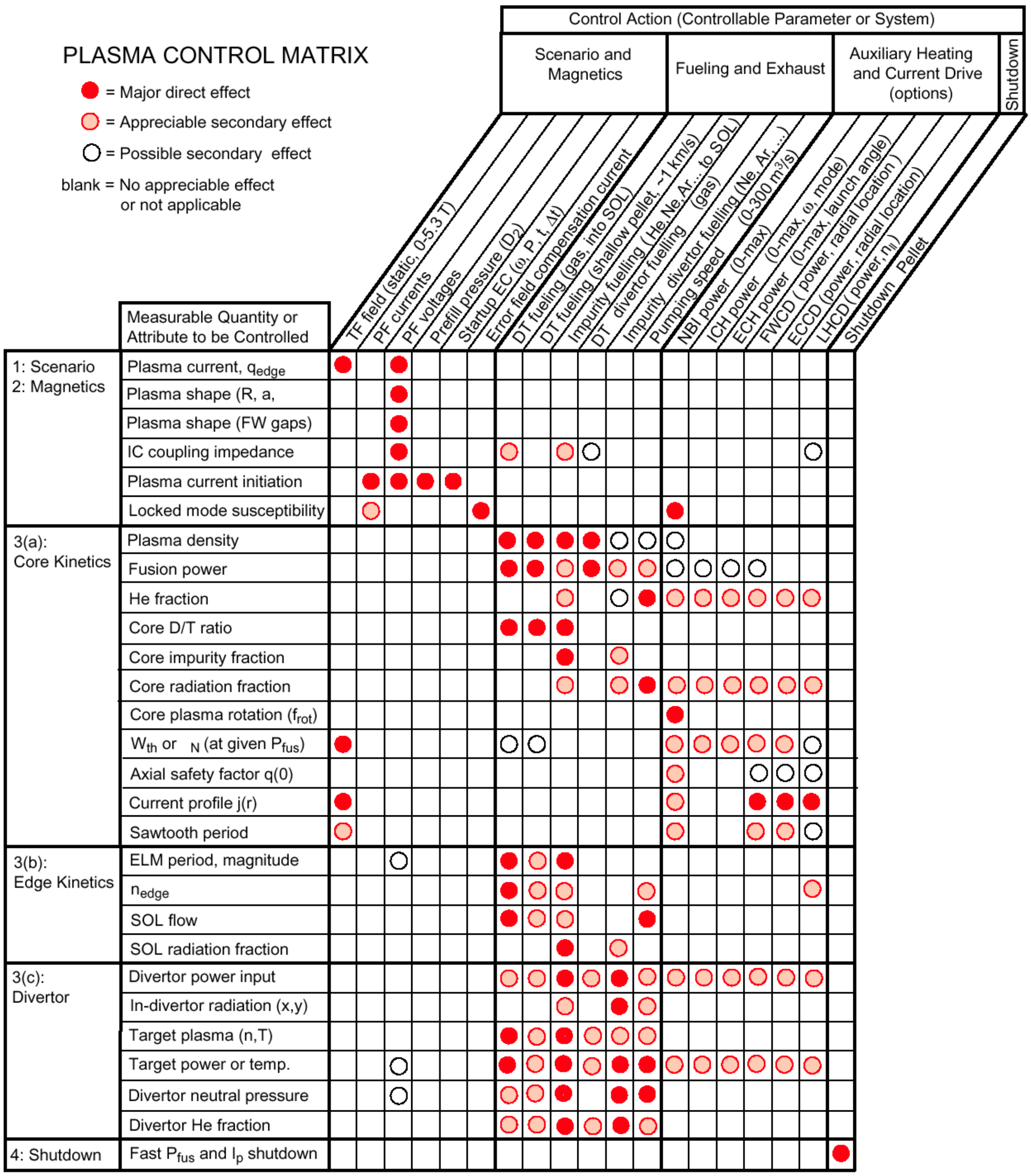


Figure 1.4.6-3 ITER Plasma Control Matrix

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. A conceptual schematic of the ITER plant control system is schematically shown in Figure 1.4.6-4.

In order to achieve integrated control of the entire plant, the SCS provides high level commands to plant subsystems while monitoring their operation. An interlock system, largely independent of the CODAC system, ensures plant-wide machine protection, as well as personnel protection. In so doing, 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 elements dedicated to each plant subsystem under the central supervisory interlock system.

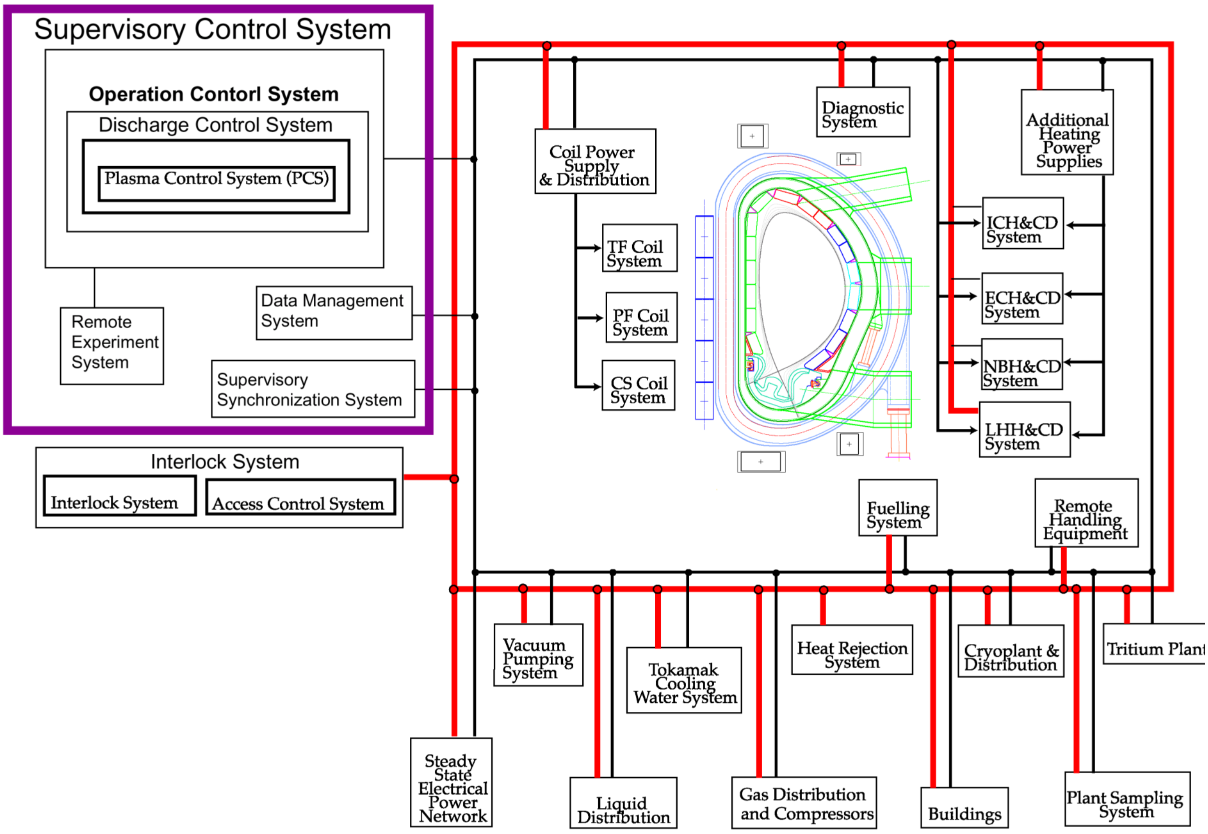


Figure 1.4.6-4ITER Plant Control System

1.5 R&D Overview

1.5.1 Introduction

The overall philosophy for the ITER design has been to use established approaches through detailed analysis and to validate their application to ITER through technology R&D, including fabrication and testing of full scale or scalable models of key components. Seven large projects were established to confirm the industrial fabrication processes and quality assurance for major key components of the basic machine and their maintenance scheme, namely,

- central solenoid model coil (CS MC) and toroidal field model coil (TF MC) projects,
- vacuum vessel sector, blanket module, and divertor cassette projects,
- blanket and divertor remote handling projects.

Other R&D concerning safety related issues, auxiliary systems including heating and current drive systems, fuelling and pumping system, tritium process system, power supplies and diagnostics are also critical areas. All the key technical issues required for ITER construction have been identified and Home Teams have carried out the associated R&D with a total resource of about 660 kIUA (1 kIUA = 1 M$ in 1989) during the nine years of collaboration (the US contributed up to July 1999). R&D resources were distributed to the various technical areas as shown in Table 1.5.1-1 with three-fourths of the resources devoted to the seven large R&D projects.

Table 1.5.1-1 Percentage of Resources Devoted to the Different R&D Areas

R&D Area	%
Magnets (incl. L-1 & L-2 Projects)	27.9
Vacuum Vessel (incl. L-3 Project)	5.3
Blanket and First Wall including Materials (Incl. L-4 Project)	16.3
Divertor & PFC including Materials (incl. L-5 Projects)	15.1
In-vessel Remote Handling (incl. L-6 & L-7 Project)	11.3
Subtotal	75.8%
Fuelling & Pumping	1.9
Tritium System	3.4
Power Supply	1.8
IC H&CD	1.1
EC H&CD	3.7
NB H&CD	3.1
Diagnostics	2.5
Safety Related R&D	3.4
Miscellaneous (incl. Standard Component Development)	3.3
Total	100.0

The technical output from the R&D confirms the manufacturing techniques and quality assurance incorporated in the ITER design, and supports the manufacturing cost estimates for important key cost drivers. The testing of models is continuing to demonstrate their performance margins and/or to optimise their operational performance. During the EDA, the successful progress in R&D has provided valuable and relevant experience in the management of industrial scale, cross-party ventures, offering important insights and experience for a possible future collaborative construction activity in a joint implementation of ITER.

1.5.2 CS Model Coil and TF Model Coil

These two projects are working towards developing the superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. The model coil projects are intended to drive the development of the ITER full-scale conductor, including the manufacturing of strand, cable, conduit and terminations, and the conductor R&D in relation to AC losses, stability and joint performance. These model coil projects also integrate the supporting R&D programmes on coil manufacturing technologies, including electrical insulation, winding processes (wind, react, and transfer) and quality assurance. 29 t of Nb_3Sn strand, from seven different suppliers throughout the four Parties, have been produced and qualified. This reliable production expanded and demonstrated the industrial manufacturing capability for the eventual production of the 480 t of high performance Nb_3Sn strand as required for ITER-FEAT.

The CS model coil is the largest, high field, pulse super conducting magnet in the world. By using approximately 25 t of the strand, the inner module (US), the outer module (JA), and the insert coil (JA) were fabricated (Figure 1.5.2-1) and assembled in the ITER dedicated test facility at JAERI. In April 2000, the maximum field of 13 T with a cable current of 46 kA has been successfully achieved. The stored energy of 640 MJ at 13 T has been safely dumped with a time constant as short as 6 s (11 s in the ITER CS). This model coil is similar in size and characteristics to one of the six modules of the ITER Central Solenoid. The SC insert coil was successfully tested with 10,000 cycles by August 2000 to simulate ITER operation (0 to 40 kA cycles in a steady 13 T background field of the CS model coil.

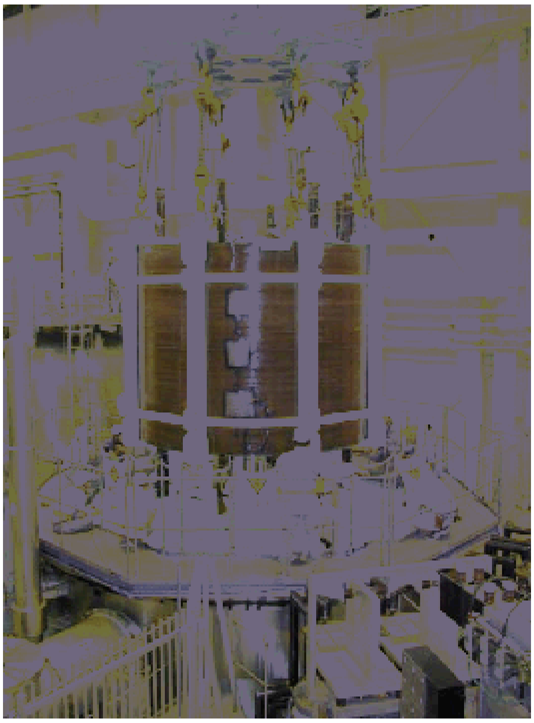


Figure 1.5.2-1 CS Model coil. The Outer Module being placed outside the Inner Module which has already been installed in the vacuum chamber.

The TF model coil is fabricated (Figure 1.5.2-2) and fully assembled in the EU. It uses a cable similar to the one used in the full-size TF coil. The diameter of the TF model coil is smaller but the cross section is comparable in size to that of the ITER TF coil.

The coil will be tested in summer 2001 in the TOSKA facility at FZK Karlsruhe with a field of 9.7 T at 80 kA (instead of 11.8 T and 68 kA in ITER). In addition, at the same time, a TF insert coil with a single layer will be tested inside the bore of the CS model coil test facility at JAERI at a field up to 13 T.

For the development of the manufacture of the TF coil case, large forged and cast pieces (about 30 t and 20 t respectively) have been produced in the EU. Investigation of the material properties at 4.5K has revealed values adequate for their use at different locations in the coil. Different welding processes have been qualified.

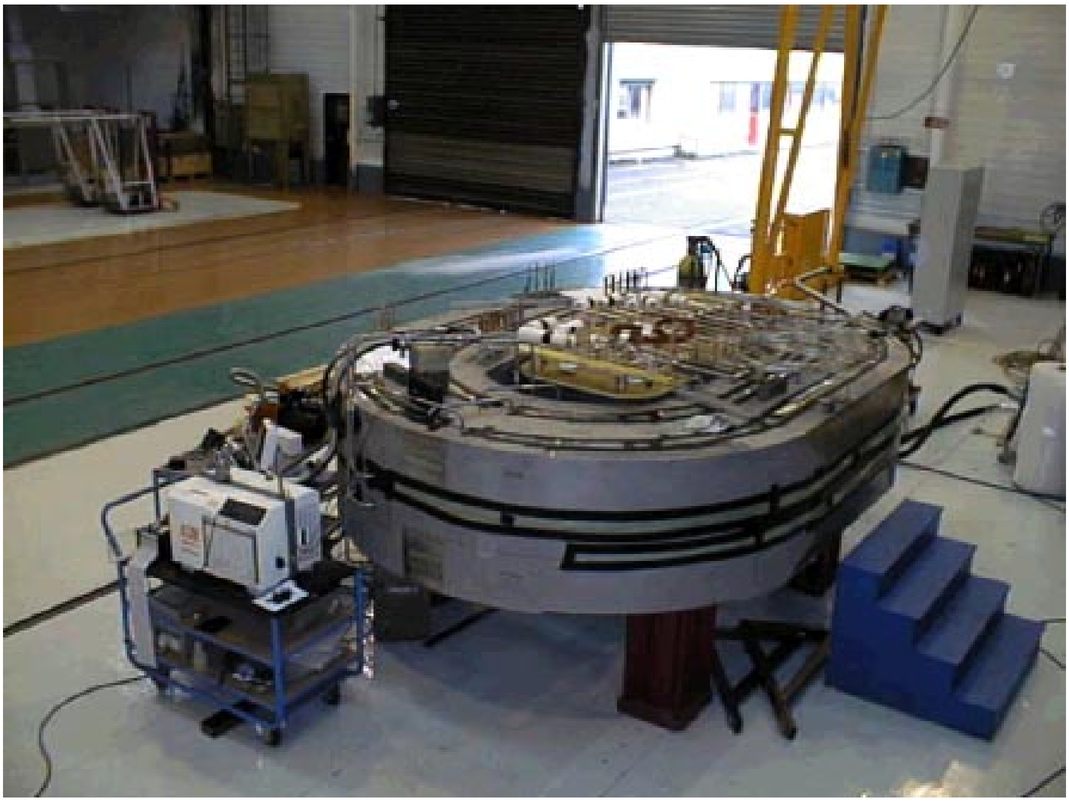


Figure 1.5.2-2 TF Model COIL undergoing vacuum tests

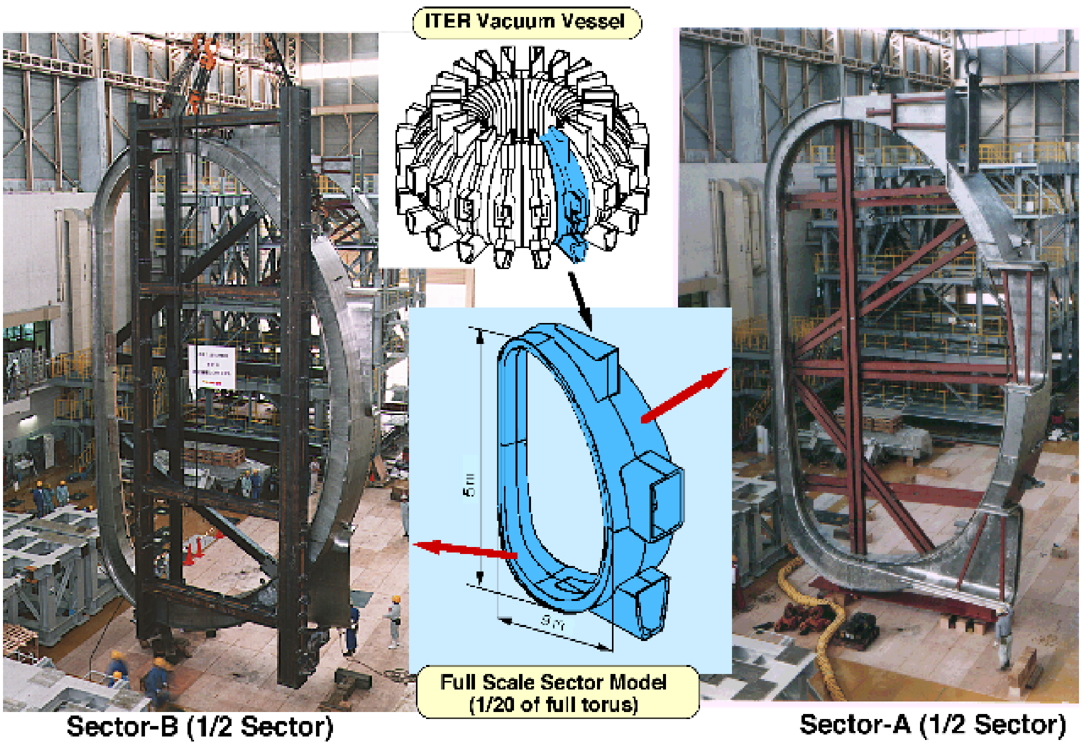


Figure 5.3-1 Full Scale Vacuum Vessel Sector R&D

1.5.3 Vacuum Vessel Sector

Two full-scale vacuum vessel segments (half-sectors) have been completed by JA industrial firms, using a range of welding techniques, within the required tolerances. At JAERI, they were welded to each other and the equatorial port fabricated by the RF was attached to the vessel to simulate the field joint (Figure 5.3-1) which will be done at the ITER site during the assembly of the machine. Remote welding and cutting systems prepared by the US were also tested and applied.

1.5.4 Blanket Module

The blanket module project aims at producing and testing full-scale modules of the first wall elements and full-scale, partial prototypes of mechanical and hydraulic attachments.

Material interfaces such as Be-Cu and Cu-stainless steel have been successfully bonded by using hot isostatic pressing (HIP) and other advanced techniques. Full-size shield block has been completed by using powder HIP (EU) and other techniques (JA) like forging, drilling, and welding. The mechanical attachments made of titanium alloy have been developed and tested in the RF (Figure 1.5.4-1).

Figure 1.5.4-1 Flexible Mechanical Attachments made of Ti-Alloy (RF)

In parallel with these fabrications, heat cycle and irradiation tests have been performed for the base materials and the bonded structures, and have demonstrated that the performance is well within the required level.

1.5.5 Divertor Cassette

The divertor cassette project aims at demonstrating that a divertor can be built within tolerances and withstand the high thermal and mechanical loads.

A full-scale prototype of a half cassette has been built by the four Parties. Plasma-facing components (PFC) shipped from the JA and the RF were installed on the inner divertor cassette body fabricated in the US, and hydraulic flux and mechanical tests were performed at Sandia National Laboratory. Other PFC mock-ups fabricated by the EU and the RF were also installed on the outer divertor cassette body fabricated by the EU (Figure 1.5.5-1).

Various high heat flux components were fabricated and tested in the four Parties. High heat cycle tests show that carbon fibre composite (CFC) monoblock survives 20 MW/m^2 x 2000 cycles (EU) and tungsten (W) armours survive 15 MW/m^2 x 1000 cycles (EU / RF). A large divertor target mockup with CFC attached to dispersion-strengthened copper (DSCu) through oxygen-free copper (OFCu) has been successfully tested with 20 MW/m^2 x 1000 cycles from a large hydrogen ion beam with a diameter of 40 cm (JA).

Figure 1.5.5-1 Integrated Outer Divertor Cassette (EU)

Irradiation tests have been also performed. For example, CFC brazed on Cu survived 20 MW/m^2 x 1000 cycles after 0.3 dpa irradiation at 320°C. Tests with pulse heat deposition simulating the thermal load due to disruptions have demonstrated erosion but no disruptive failure of CFC armours even with 0.4 dpa irradiation. (The average neutron fluence of 0.3 MWa/m^2 at the first wall gives 0.38 – 0.59 dpa on the CFC divertor target.)

1.5.6 Blanket and Divertor Remote Handling Systems

The last two of the large projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions on reasonable timescales, in particular to replace damaged blanket or divertor modules remotely. In this area, full-scale tools and facilities have been developed. Their testing will be extended over a long period of time, including the ITER operation phase, to optimise their use in detail, minimise any intervention time, and allow training of operators.

The blanket module remote handling project makes use of a transport vehicle on a monorail inside the vacuum vessel for the installation and removal of blanket modules, brought to or from a vacuum vessel port by a docked transfer cask. After successful operation of a 1/4 scale model, the fabrication of a blanket test platform with full-scale equipment and tools, such as a 180° rail, a vehicle with telescopic type manipulator, and a welding/cutting/inspection tool, have been completed in JA (Figure 1.5.6-1). The simulation of installation and removal of a simplified, dummy shield blanket module of 4 tons has been successfully performed by using a teach-and-repeat procedure with only 0.25 mm of clearance in positioning.

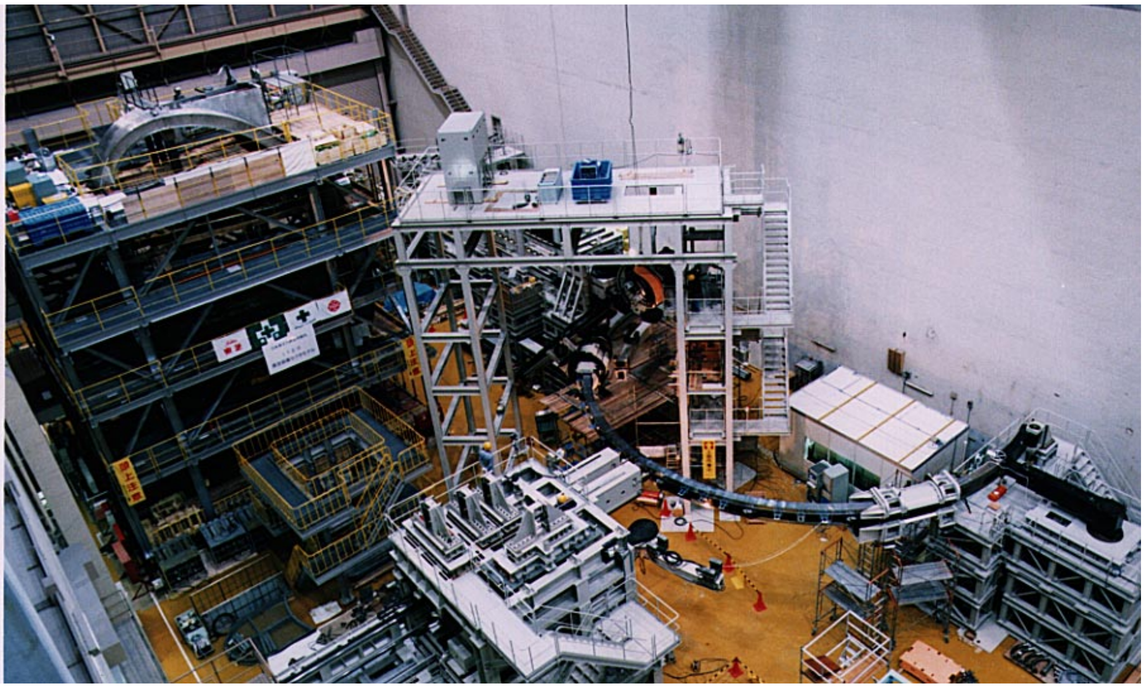


Figure 1.5.6-1 Blanket maintenance Test Platform at JAERI (JA)

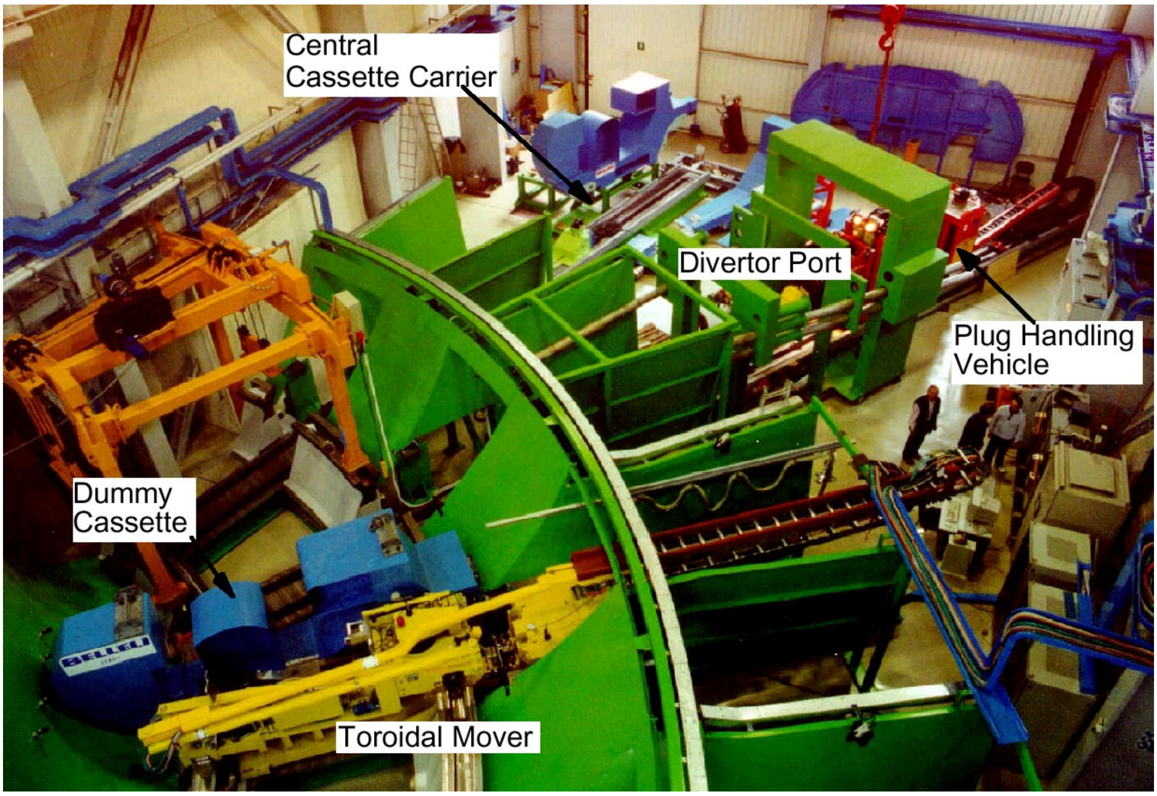


Figure 1.5.6-2 Divertor Maintenance Test Platform at Brasimone (EU)

The divertor remote handling project has involved the design and manufacture of full-scale prototype remote handling equipment and tools, and their successful testing in a divertor test platform (Figure 1.5.6-2) to simulate a portion of the divertor area of the tokamak and in a divertor refurbishment platform to simulate the refurbishment facility. The system is based on a toroidal transporter that moves a cassette in front of a remote handling port from where the cassette is extracted with a radial mover that is deployed from a transfer cask docked to the port. The real in-vessel operation will be done in a gamma field of 10^4 Gy / h. Key elements such as motor, position sensor, wire/cable, glass lens, electrical insulator, periscope and strain gauge have shown to survive tests at 10^6 –10^7 Gy.

1.5.7 Other R&D

A 50%-scale cryogenic pump for DT, He and impurities has been completed and is under testing in the EU (Figure 1.5.7-1).

A tritium pellet injector (DT and T_2) has been tested in the US ejecting a large pellet (10 mm) from an 80 cm radius curved guide tube at 285 m/s. Further tritium pellet injector development is continued in the RF.

Figure 1.5.7-1 50% Scale Cryogenic Pump (EU)

Gyrotrons at 170 GHz are being developed in all Parties aiming at longer pulses towards steady state (0.5 MW x 8 s in JA, and at 1 MW x 1 s in RF).

Key components for the ICRF (ion cyclotron range of frequencies) plasma heating antenna and the transmission line have been developed and tested at a higher voltage than the expected operational voltage, waiting for the test of a high power density antenna in a present day machine (JET, EU).

Almost full-size negative ion sources and high voltage technology (1 MeV) have been developed for the neutron beam plasma heating in JA (Figure 1.5.7-2) and in EU, but present performance has to be increased to meet the requirements.

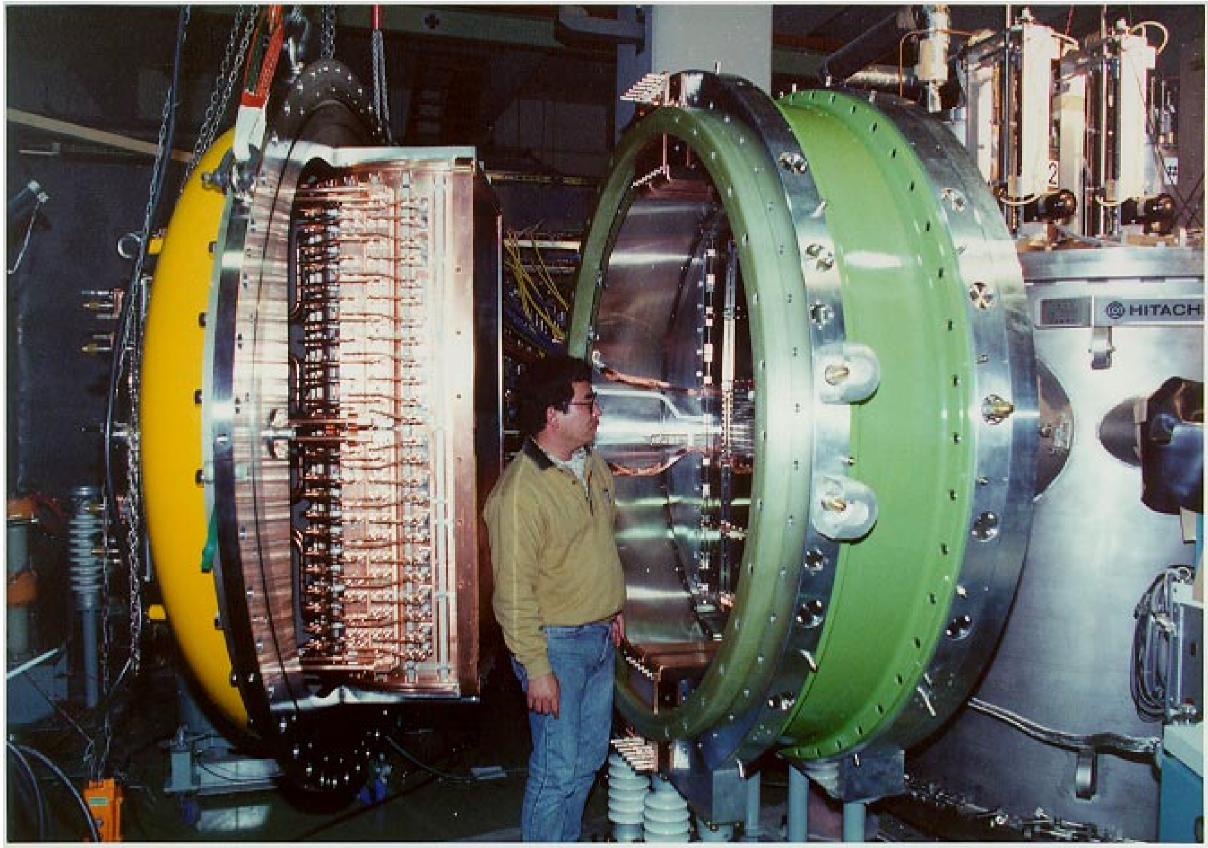


Figure 1.5.7-2 Prototype of negative ion source at JAERI (JA)

Mechanical bypass switches and fast-make switches have been developed and successfully tested at 66 kA as well as explosively actuated circuit breakers at 66 kA at Efremov Institute (RF) for the magnet system power supplies.

Irradiation tests of key components of plasma diagnostics have provided the values required for shielding of components. Lifetime of mirrors positioned near the plasma will be limited by deposition/sputtering and are under investigation

A tritium processing system using about 100 g tritium was successfully operated for 12 weeks in the US.

Safety-related R&D, such as the characterization of dust in tokamaks, tritium co-deposited with carbon, and experiments on steam-material reactions, has provided inputs for key phenomena and data for safety assessments, and the current R&D emphasis is on verification and validation of data, models and computer codes. Neutron shielding tests using 14 MeV neutron sources in Japan and the EU demonstrate that the accuracy of shielding calculations is within 10 %.

1.6 Safety and Environmental Assessment

1.6.1 Objectives and Approach

It is one of the foremost goals of ITER to demonstrate the safety and environmental potential of fusion power. Extensive design and assessment activities are undertaken to ensure the safety and environmental acceptability of ITER and to ensure that ITER can be sited in the territory of any of the Parties with only minor changes to accommodate or take advantage of site-specific features.

A consensus across the Parties on safety principles and criteria for limiting the consequences to the public and the environment from ITER operation has been reached based on internationally recognised safety criteria and radiological limits following ICRP and IAEA recommendations, and in particular on the concept of defence-in-depth and the As-Low-As-Reasonably-Achievable (ALARA) principle. Safety-related design requirements have been established, and assessments have been or are being made to evaluate the success in meeting these requirements in the facility, system and component designs.

Comprehensive safety and environmental assessments of the ITER design and operation have been completed for a generic site with the involvement of Home Team safety experts. These assessments include:

1. estimation of the environmental impact of ITER during normal operation including waste management and decommissioning,
2. evaluation of operating personnel safety,
3. in the case of postulated off-normal and even hypothetical events, estimates of environmental consequences and public safety.

1.6.2 Environmental Impact

The ITER design incorporates many features to ensure that the environmental impact during normal operation will be insignificant, including confinement barriers to prevent releases as well as air/water detritiation and filtration systems to treat releases. The facility can be operated to satisfy the restrictive safety and environmental guidelines that have been established by the project for the design. In addition to meeting prescribed host country limits, the ALARA principle is recommended by international nuclear safety experts and incorporated in national regulations of many countries. The International Basic Safety Standards explain ALARA as the objective to keep all exposures to values such that further expenditures for design, construction, and operation would not be warranted by the corresponding reduction in radiation exposure. For ITER this process involves systematically reviewing systems, activities and pathways with a release potential, estimating releases, and examining ways to reduce the main contributors.

In a comprehensive analysis, sources of potential releases have been identified, release pathways determined, and design features and release control systems assessed. Conservative assumptions have been also made so as not to underestimate potential releases.

Potential doses to members of the public (i.e. the most exposed individual) during normal operation, for a 'generic' or 'average' site, are less than 1% of the natural background level.

The continued application of the process to implement the ALARA principle may further reduce the estimated normal releases.

1.6.3 Waste and Decommissioning

In the frame of ITER's safety objectives, the control of radioactive materials, decommissioning and waste is carefully considered. All activities associated with radioactive materials and waste will comply with the host country regulations and practices. The management will be co-ordinated between the ITER project and the host country. In the absence of an actual host country for ITER, the ultimate waste amounts have been estimated provisionally on the basis of 'clearance'. According to this IAEA concept, by this process, activated materials of known composition can be released from regulatory control if their activation levels are less than the specified clearance level, and used unrestrictedly in the future.

The radioactive materials arising during operation and remaining after final shutdown include activated materials (due to fusion neutrons) and contaminated materials (due to activated tokamak dust, activated corrosion products and tritium) and mixtures thereof. Decay and decontamination will reduce the radioactivity with time after final shutdown. Therefore, a significant fraction of activated material has the potential to be cleared. Since this fraction increases with time at present the ITER project provisionally assumes that radioactive material not allowing for clearance after a decay time of up to 100 years is 'waste' requiring disposal in a repository. Estimated material masses are shown in Table 1.6.3-1.

The experimental nature of the ITER project often calls for frequent replacements of plasma facing components. For this reason the design approach has included practices to reduce the quantities and hazards of radioactive materials, such as modular components, choice of materials, control of impurities, shielding, and re-useable components. About 750 t of such waste is expected during operation which can be stored on-site in the hot cell while low-level contaminated materials (e.g. filters) would be shipped to an off-site storage facility. Moreover, to ensure that ITER can be safely dismantled at the end of its useful operating life, decommissioning plans have been developed, as described in 1.8.5.

Table 1.6.3-1 Masses of Radioactive Materials

Total radioactive material at shutdown	~30,000 [t]
Material remaining as waste after a decay time up to 100 years	~6,000 [t]

1.6.4 Worker Safety

ITER has established a program for personnel protection against hazards anticipated during construction, operation and maintenance activities. The program addresses radiation protection and conventional hazards. The objective of this work is to ensure that occupational safety is considered in the design of systems and components and, mainly through maintenance procedures, thereby to gain confidence that a high level of worker safety will be achievable.

Assessment of all major system maintenance demonstrates that ITER will maintain occupational exposures below the project guidelines. The project's commitment to ALARA ensures the continuing review, analysis, and improvement of design and maintenance procedures to ensure that exposures and radiological risks are not only below the guidelines, but also maintained as low as reasonably achievable.

1.6.5 Safety Analysis

During full DT operation ITER will contain radioactive materials that require careful management. Table 1.6.5-1 shows values of tritium assumed for assessment purposes in ITER (the "assessment value"). For the neutrons from the fusion reactions activating surrounding materials, the majority of activation products will be bound in solid metal structures of the in-vessel components. The most relevant source of mobilizable activation products is activated dust originating from plasma facing materials. Of all plasma facing materials envisaged for use in ITER it is tungsten that has by far the largest radiological hazard potential. The assessment value for tungsten dust in the vacuum vessel is 350 kg. Compared to the assessment values, smaller project guidelines are set for tritium and dust to cover uncertainties.

Table 1.6.5-1 Tritium Inventories for ITER: Assessment Values

Type of inventory	[g]
In-vessel mobilizable (in plasma facing components, dust, co-deposited etc.)	1000
Fuel cycle circulating inventory	700
Total site inventory	**< 3000**

A comprehensive analysis of off-normal events has been performed to assess the effectiveness of the implementation of the safety requirements and functions in the ITER design. Failures and combinations of failures have been postulated to critically verify that the design is tolerant to such failures and to ensure a robust safe operating envelope for the experimental programme.

The analyses included conservative assumptions of initial facility operating and off-normal conditions and thoroughly examined possible ways for tritium, activated corrosion products in coolants, and neutron-activated tokamak dust, to be released to the environment. Results show that radioactive releases for all of these reference events are well below the project release guidelines that would lead to doses (to the most exposed individual) comparable to the average annual natural background exposure for a generic site. The assessments provide confidence that the operation of ITER will result in no significant risk to the general public from postulated accidents.

In the course of the ITER design, the systematic identification of possible accidents has always been part of the safety assessment. In this process, different approaches were chosen to treat the problem of completeness methodically. One method investigated all possible failures on a system-by-system basis. A complementary method started with postulating a large environmental release of radioactivity and identifying the associated failures in the plant that could lead to it. Another approach systematically identified the energy sources in the plant that may drive accidents. All these methods have been employed during the design process of ITER and have led to similar lists of important accidents. These are discussed below together with the design features in ITER to prevent and mitigate environmental releases of radioactivity.

A fundamental safety principle of ITER is that uncertainties in plasma physics will not have an effect on public safety. The maximum possible fusion power transient is limited by the beta- and density limit to a level below 2 GW. The energy content and fuel in the plasma is limited such that only local melting of plasma facing components could occur. No safety credit is given to experimental in-vessel components. As a consequence of this approach ITER is designed to safely accommodate multiple first wall or divertor cooling channel failures. These postulated failures bound all possible damage to the in-vessel components due to unexpected plasma behaviour.

The first safety-relevant barrier for the plasma and in-vessel components is the vacuum vessel and its extensions (such as equipment in ports, and particularly the neutral beam injection lines). A second safety-relevant barrier is the cryostat and its extensions (such as vaults around the neutral beam injectors and cooling circuits of the heat transfer systems). For the tritium systems the process equipment forms the first barrier and secondary enclosures such as glove boxes or secondary piping are provided to deal with leaks in the primary confinement. Rooms that may receive leakage of radioactivity are equipped with HVAC systems that keep the pressure sub-atmospheric and provide a capability to filter dust and tritium in the exhaust if needed. Any uncontrolled flow of radioactivity is “inwards”, because the internal confinement areas are either evacuated (vacuum vessel and cryostat) or kept at sub-atmospheric pressure. Thus small leaks are excluded from leading to uncontrolled releases of radioactivity.

One very favourable safety feature of ITER is that the decay heat is limited to 11 MW at shutdown and very quickly decreases to only 0.6 MW just one day after shutdown. In case of loss of in-vessel cooling the decay heat will be passively transported by thermal radiation to the vacuum vessel. Even if active vessel cooling pumps fail, the vacuum vessel cooling system will remove the decay heat passively by natural circulation of the water coolant to an air-heat exchanger.

Figure 1.6.5-1 shows the resulting temperature history of the in-vessel components. A maximum temperature of 360ºC is reached 9 hours after plasma shutdown.

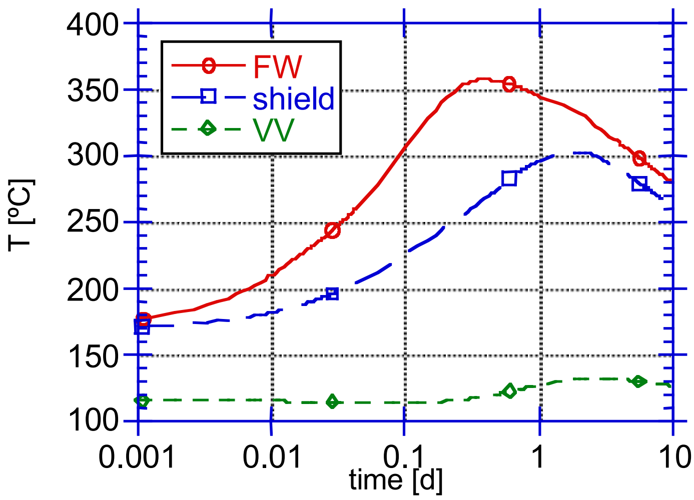


Figure 1.6.5-1

Decay heat driven temperature transient in ITER in case of loss of in-vessel cooling and failure of the active vessel cooling. The heat sink is the VV cooling system that removes the decay heat passively by natural circulation to an air heat exchanger.

The vacuum vessel is equipped with a pressure suppression system which consists of a large container of cold water to condense steam which will be formed inside the vacuum vessel during coolant leakage. For small leaks (equivalent to a break of 1 to 10 cooling channels) the maximum steam pressure inside the vacuum vessel would be less than one atmosphere and the pressure suppression system will not need to be connected. Postulating a multiple FW failure of two toroidal rings (one inboard, one outboard) as a bounding event leads to fast pressurization of the vacuum vessel and rupture of disks to allow connection to the pressure suppression system. The moderate coolant temperatures (100ºC inlet, 150ºC outlet) combined with large connecting ducts between the vacuum vessel and the pressure suppression system limit the maximum pressure in the vacuum vessel to 1.8 bar throughout this event, a value which gives rise to extremely small stress levels in the thick vessel shell.

Ex-vessel pipe failures are investigated as plausible incidents occurring in the heat transport system. The plasma burn is actively terminated on a cooling pressure drop signal to avoid large temperatures in the blanket due to a continued plasma burn without cooling. For a pipe rupture during plasma operation, the maximum absolute pressure inside the tokamak cooling water system (TCWS) vault is 1.5 bar which is below its design pressure of 2 bar. Should the divertor cooling loop be affected by an ex-vessel loss of coolant, failure of the plasma facing components is likely as a result of the high surface heat loads during plasma operation and the expected disruption loads at plasma termination. This would cause pressurization of the vacuum vessel, mobilization of tritium and dust and subsequent transport into the vault through the divertor cooling pipes. The subsequent environmental releases due to leakage from the vault are about one order of magnitude below the project release guidelines for accidents.

Another class of incidents with the potential for environmental releases are bypass events of the first two confinement barriers. In these events it is postulated that the leak tightness of both barriers in one of the many hundred penetrations into the vacuum vessel would fail. Such a penetration could be a heating system waveguide or a diagnostic line. Also in this case the ITER design feature to limit vacuum vessel pressurization to below two atmospheres has the advantage that such a confinement bypass event can be excluded as a consequential failure after a small in-vessel loss of cooling event. In any case, the amount of gas released from this failed penetration tightness will be confined in the separate cell around the relevant vacuum vessel port and the atmosphere there will be detritiated and filtered before exhaust to the environment.

Another postulated event analysed to bound structural failures is a full terminal short of a toroidal field (TF) coil, which is a highly unlikely event that requires 2 independent types of faults. The maximum current induced in a shorted TF coil during a fast discharge is limited by inductive coupling with the other coils during the fast discharge and by quench as the conductor is driven to critical conditions. Some local plastic deformation may be expected to occur in TF coil structures but gross structural failure is prevented.

As bounding events in the fuel cycle, failures in both the fuelling line and secondary confinement piping are investigated. The room will be isolated from normal ventilation and connected by valves to the standby vent detritiation system. The total expected tritium release is more than one order of magnitude below the project release guideline for accidents.

The extensive analysis of a comprehensive range of postulated failures using pessimistic or limiting conditions demonstrates that the modest safety requirements for ITER are adequate, and that releases, if any, would be a small fraction of the project release guidelines. This contributes to the demonstration that operation of ITER will result in no significant risk to the public.

1.6.6 No evacuation objective

Ultimate safety margins of the facility are examined by analysis of hypothetical events by arbitrarily assuming that more and more failures occur. ITER has the goal to demonstrate the achievement of the "no evacuation" objective according to the IAEA established guideline for evacuation which is 50 mSv of avertable dose. Even under the worst imaginable sequence of events from internal or identified external origins, the design and operation of the facility protects the public to such a degree so that there is no technical justification for dependence on public evacuation as a backup.

For example, one hypothetical event is the postulation of failure of all cooling systems in ITER (including vessel coolant natural circulation) when removing decay heat. Even for this extreme situation the first wall only heats up to a maximum temperature of about 650ºC. This temperature does not induce any significant structural failure and does not degrade the confinement functions.

Another extreme situation postulates common cause failure damage of both the vacuum vessel and cryostat boundaries by some unidentified magnet energy release. This event would not lead to large environmental releases because the cryogenic surfaces would effectively capture most of the mobilized source term and the atmosphere detritiation and filtering would control the building exhaust.

The analysis shows that the design is tolerant to failures, that there is no single component whose failure leads to very large consequences, that there is no single event that can simultaneously damage the multiple confinement barriers, and hence that the ITER design provides a high level of public protection.

1.6.7 Safety assessment conclusions

The following conclusions can be drawn from the comprehensive accident analysis: ITER is 'safe' with little dependence on engineered, dedicated 'safety systems' for public protection because of the inherent, fail-safe nature of the fusion energy reaction, limited mobilizable radioactive inventories, multiple layers of confinement, and passive means for decay heat removal. As a result of the safety characteristics of fusion, and engineered design features, it requires an almost inconceivable combination of failures to lead to a significant release of radioactive material.

In summary, the assessments documented in the GSSR shows that ITER can be constructed and operated without undue risk to health and safety and without significant environmental impact. The analyses and assessments completed with the involvement of the Home Teams experts offer a well-developed technical basis for regulatory applications in potential host countries.

1.7 Quality Assurance Program

The central objective of the ITER quality assurance (QA) program is to ensure that the level of quality necessary to achieve ITER objectives, in performance and safety, is specified and implemented. For this reason the ITER QA program covers all items or activities important to the safety and performance of ITER throughout its life. This has meant the adoption by ITER, in line with organisations qualified to ISO 9000, of:

- a quality policy,
- defined organisation structure, responsibilities, and interfaces,
- personnel qualifications and training,
- quality requirements for contracting organisations,
- documentation,
- defined processes and management systems to achieve these.

In particular, during the design/construction phase of the project the assurance of quality implies the following practices.

- Management. A proper organization of the project management is central in the successful implementation of the project itself. This includes the definition of responsibilities to ensure authority over all actions described below.
- Design. The design of components follows standard practices defined in many international standards (such as ANSI N45.2.11). These practices involve the use of interface control, definition of requirements, design verification, design reviews, configuration management, control of design documents.
- Procurement. This will be controlled to the extent necessary to ensure conformance of ITER procurement specifications with the ITER design. Items or services to be purchased will be specified in writing. Such documents will be reviewed, approved "for procurement" and controlled. Procurement contracts shall be placed with qualified suppliers whose previous performance has been assessed and recorded.
- Manufacture. This activity will be controlled to the extent necessary to ensure items conform with ITER procurement specifications. This implies that work will have to be carried out in well-controlled conditions using approved drawings, procedures, standards and other documents, according to approved pre-established checklists of operations. Approved procurement documents and records shall be maintained to reflect the actual specifications. Item or services status and relevant documentation shall be reviewed and formally accepted by ITER prior to delivery. Manufacturing documents and records shall be maintained to reflect the actual configuration of the item, and approved "as built" on item completion.
- Assembly. Installation is clearly to be controlled to the extent necessary to ensure that the installation of a particular item will not compromise the integrity of the item to be installed, or of the ITER plant.
- Inspection. Inspection and test equipment will also have to be controlled to ensure that equipment used for process monitoring, data collection, inspection and tests are of the proper range, type, accuracy and precision.

1.8 Construction, Commissioning and Decommissioning Plans

1.8.1 Introduction

The planning schedule for procurement, construction/assembly, commissioning and decommissioning set out below depends on a number of assumptions detailed in the following. As the negotiations toward the joint implementation of ITER progress, 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 ITER Joint Implementation 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.
Furthermore, the following assumptions pertain at T = 0.

- Informal dialogue with regulatory authorities should be established and should orient the technical preparation toward a licence application with a view to solving the major technical issues prior to 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 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.

1.8.2 Overall and Summary Schedule

The overall schedule that leads up to the first hydrogen plasma operation is shown in Figure 1.8.3-1. It represents a reference scenario for the schedule of procurement, construction, assembly and commissioning of ITER.

1.8.3 Construction and Procurements

1.8.3.1 Procurement Assumptions

The lead-times for the different components of ITER vary widely. For the purpose 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 should be moved earlier in the schedule to gain some margins, which would remove some items from the critical path line. However if cash flow peaks caused by critical items are too high, only an extension of the overall schedule is possible as a solution.

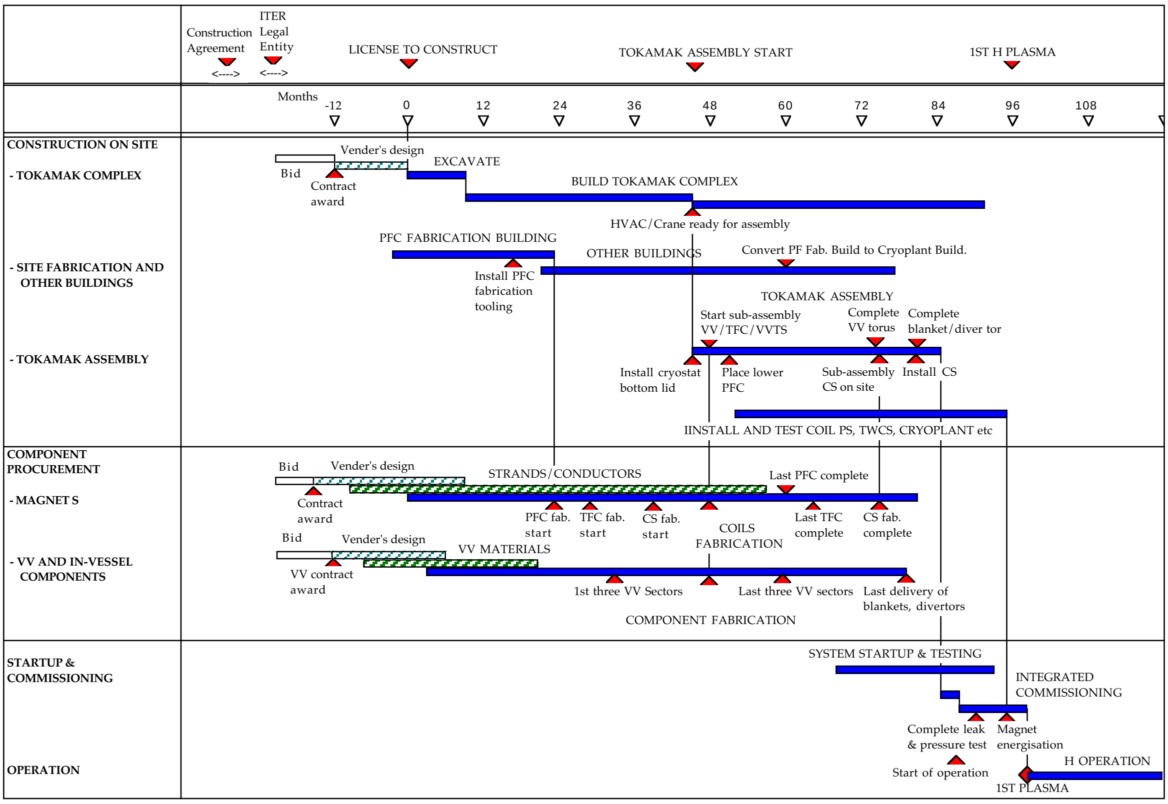


Figure 1.8.3-1 Overall Schedule up to First Plasma

Another important assumption is that the bidding process preceding placing of purchase orders will start only on the establishment of the ILE, 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.

1.8.3.2 Buildings and License to Construct

The critical path of the plan starts with the regulatory licensing procedure and the construction of the tokamak buildings. In order to start excavation immediately at T = 0, the manufacturing 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. After the establishment of the ILE, if the license process can be completed in less than 24 months, the construction period defined from the establishment of the ILE to first hydrogen plasma discharge is minimal, equal to ten years. If the regulatory process takes more than two years from establishment of the ILE, the construction period becomes correspondingly longer.

1.8.3.3 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. Two parallel manufacturing lines are necessary for the procurement of TF coils inside the planned schedule. 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. Nine (40º) vessel 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 parallel fabrication lines are necessary for the vacuum vessel procurement to fit into the schedule. The last sector should be accepted at month 62.

1.8.4 Commissioning Plan

Testing of each individual plant subsystem has to start immediately when permitted by their delivery and by the corresponding assembly work. Subsystems not needed in an early phase will be commissioned in parallel with operation.

There will be the need, for about one year, 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, and the complete Operations Team is assembled and trained on site. This is the first year of ITER operation. 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.

1.8.5 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 (1st phase) period after which the facility will be handed over to a new organization inside the ITER host country. 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. De-activation will include the removal and safe disposal of all the in-vessel components and, possibly, the ex-vessel components. At the end of this phase, 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.

1.9 Cost Estimates

1.9.1 Resources Required for ITER Construction

Although the details of arrangements for implementing ITER remain to be agreed, the items to be purchased are likely to be divided into three categories.

A. Those that can only sensibly be purchased by the host country.

B. Those which are of minor technical interest or size, and whose cost burden must therefore be shared by all Parties. For these items a centrally administered fund can be established.

C. Items of interest to all the Parties due primarily to their high technology content. To ensure each Party obtains its fair share of these items, the Parties must agree beforehand which ones each will contribute. To do this, all Parties must agree on their value to the project. This requires an agreed valuation of items, as described below. Each Party will contribute its agreed items "in kind", using the purchasing procedures and funding arrangements it prefers. Thus the actual costs to each Party (i.e., their share of A+B+C) may not correspond to the project valuation - it may differ due to competitive tendering as well as different unit costs.

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.

To arrive at an evaluated cost, or valuation, of ITER, about 85 “procurement packages” were developed for the elements of the project work breakdown structure (WBS). 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.

Industrial companies or large laboratories with relevant experience were invited, through the Home Teams, to generate, from the procurement packages, their quantification of the stages in the manufacturing process, and their best estimates of the likely current costs of supply. The information thus generated offers a comprehensive database for cost analysis, comparison and evaluation.

The Joint Central Team analysed the results of the procurement package studies in consultation with Home Teams concerned. Financial data were converted from current costs to the established reference date for ITER cost estimates of Jan 1989, using standard inflation factors for each Party. It is not assumed that the 1989 exchange rates between currencies is better than those at any other date. In particular, exchange rates between any two currencies and interest/inflation rates in any two countries most often do not vary in any coherent way.

Physical quantities (e.g. man-hours, material quantities) were then assessed starting from the industrial estimates, and multiplied by a single set of labour rates (depending on the speciality and type of labour) and material unit costs across all packages, independent of the Party. World market prices have been used where they.

The final result is an evaluated cost estimate for building ITER, expressed in IUA [1 IUA = $ 1000 (Jan 1989 value)], 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.

The evaluated cost estimate for ITER construction is presented in Table 1.9.1-1.

Taking account of the previous assumptions, the total value of the 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 (for full DT operation) amounts to 258 kIUA. The present investment cost is about 50 % of the previous estimate for the ITER 1998 design, which amounted to 5603 kIUA and 302 kIUA deferred.

The procurement of the machine core (about 1500 kIUA) could be possibly split into about 40 different contracts of 40 kIUA in average. The procurement of Auxiliaries and Heating and CD systems, excluding the Concrete Building (about 800 kIUA) could be split into about 40 different contracts of 20 kIUA on average.

Table 1.9.1-1 Summary ITER Direct Capital Cost

	Direct Capital Cost (kIUA)	Percentage of Total	Deferred Investment (kIUA)
Magnet Systems	762.1	28%	40.2
Vacuum Vessel	230.0	8%	0.0
Blanket System	165.2	6%	8.6
Divertor	76.0	3%	6.9
Machine Assembly	92.7	3%	0.0
Cryostat	75.8	3%	0.0
Thermal Shields	28.8	1%	0.0
Vacuum Pumping & Fueling System	34.2	1%	6.8
Machine Core, subtotal	**1464.8**	**53%**	**62.5**
R/H Equipment	61.1	2%	52.3
Cooling Water Systems	131.5	5%	16.8
Tritium Plant	36.6	1%	45.2
Cryoplant & Distribution	88.9	3%	7.9
Power Supplies & Distribution	214.7	8%	3.5
Buildings	380.3	14%	12.0
Waste Treatment and Storage	2.1	0%	7.0
Radiological Protection	1.0	0%	3.2
Auxiliaries, subtotal	**916.2**	**33%**	**147.9**
IC H&CD	32.2	1%	2.0
EC H&CD	77.5	3%	3.0
NB H&CD	96.0	3%	0.2
Heating and CD, subtotal	**205.7**	**7%**	**5.2**
Diagnostics	118.0	4%	42.3
CODAC	50	2%	0.0
Grand Total	**2754.7**	100%	**257.9**

The ITER construction plan requires a specific schedule for the procurement of each system and therefore appropriate profiles of commitments and payments, the total of which will amount to the JCT estimated capital investment. The most important to consider is the profiles of payments and therefore the necessary cash flow. This important issue is discussed in detail later in Chapter 7 where ,under some assumptions, the peak annual payment reaches about 350 kIUA in the second year after T=0 and continues at this level until the seventh year so that most of the direct capital cost is paid in nine years.

1.9.2 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.

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, 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 that will have the overall responsibility to meet the project objectives and to ensure the design continuity and coherence.
- National Teams, as part of the ILE in each Party, which will manage and follow up the technical content of the procurement contributed by each Party, when the financial and legal contents of the relevant contracts are being taken care of by a Domestic Agency.

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's 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.

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 global man-years 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. Assuming the annual cost of one professional and one support staff member 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. Excluded from the above numbers are those relevant to diagnostic procurement that should be the responsibility of the Parties Laboratories (estimated resources of 220 professional person years needed) which will afterwards participate in ITER operations.

In addition to personnel costs, a certain amount of R&D during construction will still be necessary. Although the EDA has provided the principle qualification of design solutions to be implemented in ITER, during the manufacturing of components, proposed process improvements and design changes or unexpected difficulties could require new tests. It is therefore prudent to expect a spending in R&D of 60-80 kIUA during ITER construction.

1.9.3 Resources for ITER Operation

Manpower costs of permanent staff on site 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.

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. With some typical scenario and a unit cost of 0.05 IUA/MWh the estimated cost is about 30kIUA/year.

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. Fuel costs include tritium burnt during operation, plus that lost by decay of the inventory (taken as 2 kg) during plant operation. 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, amounts to 6.7 kg. whereas the total consumption of tritium during the plant life time may be up to 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 will be 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.

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.

1.9.4 Decommissioning Costs

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 the cost down. The estimated cost for decommissioning amounts to 250 kIUA for manpower costs and 85 kIUA for possible hardware costs.

1.9.5 Summary

A summary of the cost estimates for all phases of ITER plant life is set out in Table 1.9.5-1.

Table 1.9.5-1 Summary of ITER Cost Estimates

	Cost (kIUA)
Construction costs	
Direct capital cost	2755
Management and support	477
R&D during construction	60-80
Operation costs (average/year)	
Permanent personnel	60
Energy	~30
Fuel	~ 8
Maintenance/improvements	~90
Total	188
Decommissioning cost	335

1.10 Conclusions

The ITER project has its origins in the common recognition by the world's leading fusion programmes world-wide of :

- the potential of fusion as a practical long-term energy source, with acceptable environmental characteristics
- the need for the next step on the path towards realising fusion energy to be the construction and operation of a burning plasma experiment allowing, in one device, full exploration of the physics issues as well as proof of principle, testing of key technological features of possible fusion power stations and demonstration of their safety and environmental characteristics, and
- the attractions of preparing to take such a step in an international collaborative framework which would allow participants to share costs and pool scientific and technological expertise towards a common goal.

After Conceptual Design Activities (CDA) between 1988-1990, the Engineering Design Activities (EDA) began in 1992 and are now completed with the ITER design as summarized in this report. Following the choice of site and the commitment by the ITER Parties of suitable funds, the construction phase (about 10 years) may start. This would be followed by an exploitation phase lasting roughly 20 years.

Nine years of intensive joint work by the ITER Joint Central Team and Home Teams of the four Parties (three after 1999) under the auspices of the IAEA have yielded a mature design supported by a body of validating physics and technology R&D, safety and environmental analyses and industrial costing studies. The ITER design meets all detailed objectives set by the ITER Parties, with margins in physics and technology to allow for uncertainties, whilst satisfying a cost target that makes possible for participants to benefit from the sharing of costs and the pooling of expertise that joint implementation allows.

The demanding technical challenges of the project and its international collaborative nature have led to the breaking of new technical ground in fusion science and engineering. In addition to the technical results, the project has demanded and enabled new modes of closer programmatic collaboration, among :

- physicists, in co-ordinating and collating experiments and results world-wide,
- technologists, in pursuing large, multi-Party projects on the key technological aspects of ITER design , and
- safety and environmental specialists, in pursuing a global approach to the specific safety and environmental characteristics of fusion as an energy source.

These advances provide assurance of the practicality of prospective joint implementation.

The ITER co-operation, in combination with the continuing general progress in fusion research, has brought its Parties and the world fusion development programme to the point at which they are technically ready and able to proceed to construction, thus bringing to successful fruition the Parties' efforts, investments and aspirations to date. By enabling, in a single device, full exploration of the physics issues as well as proof of principle and testing of key technological features of possible fusion power stations, ITER will provide the integration step necessary to establish scientific and technical feasibility of fusion as an energy source.