

REPORT NO.

ITR-IEBH-104 v1.0

TITLE

# ITER Engineering Basis Handbook

## Vol. 1: Genesis, Design and Evolution

### Chapter 4 - The role and distinctive features of ITER

CHIEF EDITOR

Dr. Gianfranco Federici

CHIEF EDITOR'S EMAIL

[gianfranco.federici@euro-fusion.org](mailto:gianfranco.federici@euro-fusion.org)

January 15th, 2026

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 – EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.



The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

[www.iter.org](http://www.iter.org)

## About the ITER Engineering Basis Handbook

This handbook consists of two volumes which describe the ITER design from its inception up to the design, construction and assembly in 2025.

The handbook is not designed to be read as a continuous sequence of chapters. Instead, it is composed of focused, self-contained sections that address specific topics. Each chapter can be read and understood independently, allowing readers to engage with the material most relevant to their needs without requiring familiarity with preceding chapters. As a result, the reader will find certain overlapping content in chapters.

It is to be noted that at the time of writing, the design for some systems is still on-going. Therefore, the reader should consider that whilst there is significant value of this important point-in-time study, an update would be required as the Project progresses.

A broad Project overview is given in the first volume, to provide the reader with background information necessary to understand the context in the subsequent more-detailed chapters of the second volume, dedicated to the individual systems composing ITER.

For the overall table of contents of the Handbook and to access each one of the chapters, please refer to <https://www.iter.org/scientists/iter-technical-reports/iter-engineering-basis-handbook>.

## Authors and Contributors of this Chapter

This chapter is authored by Gianfranco Federici and co-authored by the Handbook Editors Federico Casci, Stefano Chiochio, Richard Hawryluk, John How, Akko Maas, Masanori Onozuka, René Raffray, Gabriella Saibene, Bill Spears and Eisuke Tada.



## Volume 1

### GENESIS, DESIGN AND EVOLUTION

## Chapter 4

# THE ROLE AND DISTINCTIVE FEATURES OF ITER

### Table of Contents

|       |  |    |
|-------|--|----|
| 4.1   | Introduction .....   | 2  |
| 4.2   | Features of Burning Plasmas .....  | 5  |
| 4.3   | The Evolution of ITER Objectives and Parameters .....                          | 8  |
| 4.4   | A concise description of the ITER design .....                                 | 19 |
| 4.5   | Distinctive features of ITER and main differences with existing devices.....   | 27 |
| 4.6   | Main ITER design challenges and the role of fusion enabling technologies ..... | 37 |
| 4.6.1 | Magnets.....   | 37 |
| 4.6.2 | Vacuum Vessel .....  | 42 |
| 4.6.3 | Plasma Facing Components .....   | 45 |
| 4.6.4 | Plasma heating and current drive .....   | 49 |
| 4.6.5 | Plasma Diagnostics.....  | 52 |
| 4.6.6 | Tritium Systems.....   | 56 |
| 4.6.7 | Remote Handling .....  | 60 |
| 4.6.8 | Test Blanket Modules .....   | 64 |
| 4.6.9 | Integrated System Plant Design and Safety.....                                 | 70 |
|       | Summary .....  | 75 |
|       | Glossary .....   | 76 |
|       | References .....   | 78 |

# Chapter 4

## THE ROLE AND DISTINCTIVE FEATURES OF ITER

### 4.1 Introduction

Since the early eighties the fusion scientific community has focused its Research and Development activities on providing the base of data to ensure the successful design, construction and operation of an experimental fusion reactor to achieve reactor-relevant burning plasma conditions, which is recognised to be the primary challenge faced by fusion research today and the necessary step towards the demonstration of fusion as a source of energy. During this period the knowledge on reactor relevant plasma physics performance improved dramatically thanks mainly to advances achieved with experiments, modelling and technology developments. These advances have led to the design of ITER, conceived to be the first large tokamak to produce plasmas with the alpha-heating dominating over the external heating power.

ITER's mission- to establish the scientific and technological basis for the peaceful exploitation of fusion energy and to demonstrate the safety and environmental potential of fusion energy production and thereby provide a precedent for the safe operation of future fusion power plants - provides a fundamental principle for the design and construction of all elements of the ITER tokamak facility. It undoubtedly represents the critical step from today's magnetic confinement fusion (MCF) experiments to future fusion energy demonstration power plants.

The performance and operating requirements of ITER in terms of power, pulse lengths, duty cycles, and availabilities represent a very large extrapolation from current experience and place a

huge constraint on design integration, fabrication, assembly, component reliability, and maintenance procedures. However, the data derived from ITER on these issues are essential to realising practical fusion power plants.

From the beginning, these ambitious objectives affected the design and construction of the ITER tokamak facility. Although there is a strong physics and technology basis, there are still uncertainties and important issues for burning-plasma research, giving rise to significant risks and requiring robust design and technology solutions.

The main physics and engineering challenges impacting the design and fabrication of the ITER systems include:

- a) the unprecedented size and performance requirements for some of the core systems, i.e. the ITER superconducting magnet systems, the vacuum vessel etc;
- b) the huge electromagnetic forces on the mechanical structures of the tokamak;
- c) the high-power fluxes for long durations to the divertor targets and to the first wall. These must be actively cooled to remove the plasma heat fluxes as well as the neutron heating, the latter also deposited deep in the shielding blanket and the vacuum vessel surrounding the plasma. Further challenges arise from the additional demands imposed by very high transient heat pulses on the plasma facing components (PFCs) generated by plasma instabilities;
- d) the high neutron flux and the radiation shielding requirements complicated by the constraints of a complex tokamak geometry that challenges the disposition of fully effective shielding (neutron streaming across penetrations on the outboard also represents a serious design issue while a biological shield is necessary to keep the radiation biological dose outside the reactor within the maximum permissible dose for occupationally exposed individuals;
- e) the development and deployment of efficient and reliable plasma heating and current drive, fuelling, diagnostic and instability control/mitigation systems capable of routine operation in the burning plasma environment;
- f) the need to handle and reprocess tritium at an unprecedented scale to satisfy the fuel requirements of the burning plasma programme, to limit in-vessel fuel retention and to assure the required degree of tritium containment;
- g) the provision of remote maintenance tools and facilities to allow repair, maintenance and upgrades in an activated environment;
- h) the millimetric and in some cases sub-millimetric alignment and/or very demanding fabrication tolerances on components of complex shape and large dimensions;
- i) the need to develop fully qualified fabrication methods and to overcome the limited industrial manufacturing experience in some of the utilised technologies. This often occurs for first-of-a-kind (FOAK) supplies in some areas, and there is the need to address uncertainties with extensive testing and qualification;

- j) the integration of prototypical tritium breeding blanket modules and their associated subsystems into the tokamak environment for the first time;
- k) the need to comply with tight nuclear regulations in terms of safety set by the French regulators.

These challenges explain the thoroughness and the length of the design activities and of the supporting R&D which has been carried out. Designing and constructing the tokamak, auxiliary and plant systems which make up the ITER facility have necessitated extensive R&D programmes characterised by wide-ranging, long-term international collaborations and major innovations. The pooling of resources from all the major fusion programmes in the world alleviated the difficulties that a single programme would have in constructing such a large-scale fusion experiment. The cross fertilisation of ideas and expertise proved to be very beneficial in technical terms, and this, together with the shared risk and pooled resources, more than offset the greater complexity in organisation and management and the many interface problems to be solved.

However, ITER remains an experimental facility with operational risks arising from the uncertainties of the plasma physics and many FOAK components and systems, implying that there are still significant risks and uncertainties in the design, fabrication and assembly of the core components and their operation as single items and as part of the whole reactor, to ensure that the device can reach the required performance in terms of parameters such as the fusion power, fusion gain and pulse length. A built-in flexibility to accommodate these uncertainties, allowing adaptation in response to issues which emerge during operation, is essential, albeit not easy to implement.

The scientific and technological challenges associated with ITER's performance objectives and requirements are formidable, and this chapter attempts to highlight the main engineering challenges together with its major technical differences with existing devices. Specific manufacturing and assembly challenges are mentioned in Chapter 8 and extensively described in Vol. 2. Section 2 in this chapter concisely describes the main features of a burning plasma. Section 3 summarises the rationale behind the choice of the main objectives and the technical parameters of ITER and its evolution during the time. Section 4 provides a brief description of the ITER design. Section 5 addresses some of the most distinctive features of ITER and the main differences with existing devices. Finally, Section 6 describes the features of essential ITER enabling technologies and the associated challenges encountered in their design and realisation. A more exhaustive description of all the systems that composed the ITER facilities can be found in the second volume of this handbook.

ITER is a FOAK fusion facility based on its scale, technical specifications, and integration of systems, but is also a FOAK for licensing. As a matter of fact, regulatory challenges exist as the fusion process in the ITER machine involves specific nuclear hazards due to its radioactive inventory, namely tritium, activated products and wastes such as tritiated and activated wastes. In France, this means that the ITER machine is a nuclear facility which is classified as a Basic Nuclear

Installation (Installation Nucléaire de Base, INB) by French Regulations. Nuclear safety is a top priority issue for the ITER project, its staff and workers on site, and local population and the environment.

Due to nuclear safety characteristic of the machine and according to article 14 of the ITER Agreement, the IO has been asked to get a licence from the French Nuclear Safety Authority that has nuclear safety regulations that the IO must comply with for construction, operation and decommissioning of the ITER machine. The so called INB Order of February 2012 that forms the basis for licensing and authorisation for construction and operation of nuclear facilities in the French territory is of special relevance. The licencing process of ITER is described in Chapter 8 and the Quality Assurance (QA) and Quality Control (QC) Programme are also described in Chapter 8. The contribution of ITER to fusion science and engineering is immense. The next generation of fusion devices will be able to learn from ITER's return of experience across topics including the details of the design, manufacture, assembly and operation that could not have been learned otherwise.

## 4.2 Features of Burning Plasmas

In a burning plasma, ions undergo thermonuclear fusion reactions, which supply self-heating to the plasma [1]. The plasma requirements for achieving sufficient thermonuclear reactions in a fusion device are lowest using the hydrogen isotopes of deuterium and tritium, due to the high reaction rate at the same temperature (see Fig. 4.1). In fact, the total energy released per DT fusion event is 17.6 MeV, of which 20% is carried by the resultant alpha particle and 80% by a neutron.

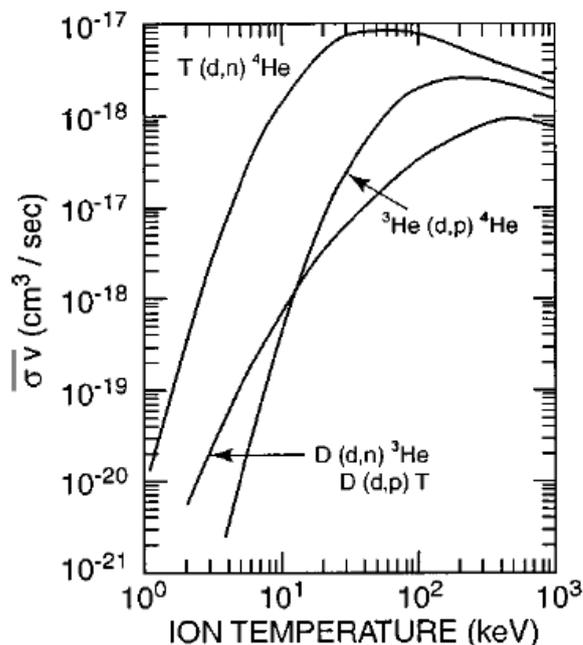


Fig. 4.1. Rate coefficients for fusion reactions of importance in magnetic fusion, plotted vs ion temperature [1].

A burning plasma is a plasma that is dominantly self-heated by fusion products from thermonuclear reactions in the plasma (i.e. alpha particles for DT fusion). This definition can be made more quantitative by considering the fusion energy gain  $Q$  defined as the ratio of the fusion power  $P_{fusion}$  to the externally supplied heating power  $P_{heat}$ :

$$Q = P_{fusion}/P_{heat} = 5P_{\alpha}/P_{heat}$$

The factor of 5 applies for DT fusion, since the power carried by alpha particles,  $P_{\alpha}$ , is 1/5 of the fusion power. The alpha particle heating fraction is defined as  $f_{\alpha} = P_{\alpha}/(P_{\alpha} + P_{heat})$ , which can also then be expressed as  $f_{\alpha} = Q/(Q + 5)$ . “Breakeven” is defined as  $Q = 1$ , which corresponds to an alpha heating fraction of  $f_{\alpha} = 17\%$ .

The “burning plasma” regime corresponds to  $Q \geq 5$ , for which self-heating by alpha particles supplies at least half of the plasma heating (i.e.  $f_{\alpha} \geq 50\%$ ). ITER is designed to achieve  $Q = 10$  (or  $f_{\alpha} = 66\%$ ). When the alpha particles supply all the heating ( $f_{\alpha} = 100\%$ ), external heating is unnecessary, and  $Q$  becomes infinite; this is defined as the “ignition” regime.

A review of the scientific issues for burning plasmas in ITER may be found in the documents ITER Physics Basis [2], Progress in the ITER Physics Basis [3] and the International Tokamak Physics Activity (ITPA) 2025 special issue on burning plasmas [4]. The ITER device will address key scientific issues in confinement, magnetohydrodynamic (MHD) stability, divertor physics, plasma–wall interactions, alpha-particle physics, and non-inductive steady-state operation at the reactor scale. It should, therefore, provide the additional understanding in physics and the key validation of technology required to allow the magnetic fusion program to proceed to the design of a

demonstration tokamak power plant. The scientific issues relevant for ITER are described in the second volume in Chapter 4 [5].

For MCF, which is the focus here, essentially, two paths have been followed in the approach to burning plasmas physics. One is the high-temperature path, which has been used by TFTR [6], JET [7], and other facilities like JT-60U [8] and which will also be used by ITER. The other is the high-density path, for which compact high magnetic-field tokamaks have been proposed and will be very briefly discussed in Section 3.

Important experiments using ITER-relevant fuel mixtures were conducted in the Tokamak Fusion Test Reactor (TFTR) in the US (1993–1997) [5] and in the Joint European Torus (JET) in Europe, first in 1991, then in the Deuterium Tritium Experiment 1 (DTE1) in 1997 [6] and more recently (DTE2) in 2021 [9] and (DTE3) 2023. In summary, by exploiting the unique capabilities of these devices, a few important physics and engineering issues of relevance for ITER design and operation were explored. These include:

- a) demonstration of high fusion power, sustained for a few seconds (this duration was limited by the thermal inertia of uncooled tokamak's wall and divertor and copper magnetic coils);
- b) demonstration of significant  $\alpha$ -particle heating effects;
- c) addressing of key plasma-wall interaction issues; the use of carbon as plasma facing material led to unacceptably high T fuel retention both at TFTR and JET, and in the latter, it was decided to refurbish the device (2009–2011) with a material mix that was considered to be more attractive for ITER, consisting of a combination of Be tiles in the main chamber and W in the divertor);
- d) demonstration at JET of radio-frequency heating schemes relevant to ITER DT operation;
- e) neutronics measurements and validation of neutron streaming modelling and of predictions of the ITER shutdown dose rates [10]. For reference, the neutron flux levels measured at the JET first wall were of the same order of magnitude of those expected at the rear ITER blanket/diagnostic first wall for ITER  $Q = 10$  operation and the in-vessel  $\gamma$ -radiation field in JET due to neutron activation after DTE2 was as intense as that expected in some maintenance locations of ITER.

In conclusion, in terms of fusion performance, TFTR operating with 50/50 DT plasmas achieved  $P_{DT} = 10.7$  MW and  $Q = 0.27$ , and JET achieved  $P_{DT}$  of 16 MW and  $Q = 0.6$  in 1996, and a fusion energy record output of 59 MJ in a five-second pulse in December 2021. Despite these records, neither TFTR nor JET have reached breakeven conditions, primarily due to their small plasma size (see Section 3). ITER will have a plasma volume 10 times that of JET.

## 4.3 The Evolution of ITER Objectives and Parameters

The size of ITER or any burning-plasma experiments is determined by several factors. One is the requirement to achieve sufficient confinement to reach the burning state, which is determined by scaling predictions for the energy confinement time or the fusion triple product (density  $\times$  temperature  $\times$  confinement time). This is determined from size-scaling relationships [11], [12] derived from experiments on tokamaks with plasma volumes spanning two orders of magnitude; the largest of these tokamaks already operate at plasma densities and temperatures like those expected in ITER though at reduced confinement times, given their smaller size. Another factor is the requirement that the power handling of the plasma-facing components, especially the divertor, must be able to handle the heat loads under burning plasma conditions. A third factor is the need to provide adequate radiation shielding of the critical equipment, i.e., the superconducting magnets.

Clarification, on the basis on simple intuitive considerations, is first needed as to why ITER must be larger than current devices to reach its mission objective.

- 1) First, it must achieve sufficient plasma confinement.

The so-called triple product,  $n T \tau_e$  (where  $n$  is the plasma density,  $T$  is the plasma temperature and  $\tau_e$  is the plasma confinement time) required to sustain a fusion plasma with a  $Q=10$  in ITER for long pulses (of the order of several minutes) must be typically:

$$n T \tau_e > 3 \times 10^{21} \text{ keV s m}^{-3} \quad (1)$$

The optimal plasma temperature is typically around  $10^{-29}$  keV. The density is not a free parameter as one needs a magnetic field to confine the plasma and the magnetic pressure scales with the square of the magnetic field ( $B^2$ ). The ratio between the plasma kinetic pressure,  $p$  (the product  $nT$ ) and the magnetic pressure is called beta ( $\beta$ ) and its achievable value is unfortunately relatively low (few %) to avoid or minimise plasma MHD instabilities.

$$\beta \sim nT/B^2 \quad (1a)$$

Therefore, the plasma pressure scales with  $B^2$  and for a typical value of the density of  $10^{20} \text{ m}^{-3}$ ,  $\tau_e$  in ITER must be around 3s to satisfy equation 1.

The energy confinement time  $\tau_e$  depends on many factors and is determined from empirical scaling laws derived from experiments in tokamaks. For a machine-like ITER, it can be shown that, according to one of the most common empirical scaling (IPB98):

$$\tau_E \sim R^{2.6} B^{1.55} \quad (1b)$$

Thus combining (1a) and (1b) one derives that

$$nT\tau_E \sim R^{2.6} B^{3.55} \quad (1c)$$

to reach a value of  $3 \times 10^{21}$  keV s m<sup>-3</sup> (from equation (1)) the machine radius must be of the order of 6.2 m. These qualitative arguments are developed in more detailed calculations which consider other physics effects including density limits and more recent empirical scaling laws and should not be used for design purposes. This is discussed further in Volume 2, Chapter 4.

2) A second important design consideration is that the plasma-facing components, especially the divertor, must be able to handle the heat loads under burning plasma conditions.

One of the crucial points in the dimensioning of ITER and any power producing fusion plant, remains the amount of power that can be reliably produced and controlled within it. This heavily depends, amongst other things, on the heat load that can be tolerated by the divertor under normal and off-normal operation [13].

In fission power plants the heat produced by the fission nuclear reactions in the fuel rods is absorbed and transported by the coolant (typically water) where they are immersed. In fusion devices instead, the heat by the fusion reactions is produced in the plasma through the production of fusion products, alpha-particles and neutrons. The energetic alpha particles (~20%) transfer their energy to plasma electrons and ions, which deposit the energy on the surfaces of the surrounding internal components. The neutron energy (~80%) is deposited deep in the bulk of surrounding components. The heat from the plasma charged particles is not uniformly distributed, because their trajectories are dictated by the magnetic field, being the divertor region the most loaded area. In ITER the divertor is located at the bottom of the vacuum vessel in a toroidally continuous ring or target plate, where heat loads substantially in excess of 20 MW m<sup>-2</sup> can be reached if the plasma radiation in the divertor is not considered (note that at the surface of the sun the heat flux is estimated to be of the order of 50 MW m<sup>-2</sup>).

Due to the limits of existing state-of-the-art divertor technologies (see Section 6.3), the ITER design is assumed to operate with at least a partially detached divertor. Specific high-density plasma conditions are therefore created in the divertor region, with the effect that a significant fraction of the power that would be conducted to the divertor target ( $q_{tar}$ ) is instead dissipated by plasma radiation in the divertor scrape-off layer (SOL) before reaching the target plates. The necessary high dissipation is planned to be obtained with the use of seeded, radiative impurities, such as Neon (Ne), Argon (Ar) or Krypton (Kr), which re-distribute the necessary fraction of the exhaust power onto the first wall in form of photons.

The use of these impurities is however not without consequences for the machine operation, as a certain fraction of the seeded atoms, in fact, is expected to penetrate the plasma core, where, depending on the edge profile characteristics can cause either a reduction of the fusion power via fuel dilution or trigger some radiative instability. It is therefore necessary to find an adequate balance between the radiation level in the SOL and the impurity content in the core, and adequate plasma configurations and scenarios will have to be established to achieve fusion power production and at the same time, radiative protection of the divertor.

The criteria to be employed in the preliminary phases of a tokamak fusion reactor dimensioning to ensure the integrity of the divertor for sufficiently long operating times is discussed in another source [14]. This translates into limits in terms of major radius  $R$  and in terms of toroidal magnetic field  $B$ . The heat flux on the divertor targets ( $q_{div}$ ) under attachment conditions can be given as [14]:

$$q_{div} \propto B_{t,0}^{2.52} R_0^{0.16} \quad (2)$$

where  $B_{t,0}$  is the toroidal field strength at the centre of the plasma and  $R_0$  is the major radius (see Fig. 4.2). This relation indicates that, by increasing the magnetic field device, the heat flux on the divertor targets increases, further exacerbating the critical problem of the power exhaust. Clearly, ITER and future fusion power reactors are expected to operate under detached or partially detached divertor conditions, and the design provides for the injection of suitable impurities as well as sufficient particle recycling for the achievement of detachment or other suitable divertor conditions. Still, it is thought that the divertor heat flux under attachment is a representative figure of merit to define design parameters and compare different design solutions.

ITER must work with long pulses, and the possibility of accidental divertor re-attachment must be considered from the earliest design phases. If the divertor is damaged by a reattachment event, the consequence might be an in-vessel loss of coolant accident (LOCA), breaking the barrier between coolant and vacuum chamber and thus potentially allowing some amount of radioactive material to be released from the vessel in the primary circuit. Due to the implications on downtime and cost, arising from in-vessel LOCA events, the design of the in-vessel components must be as robust and reliable as possible to minimise the occurrence of such events.

3) A third factor that affect the machine size is the need to provide adequate radiation shielding of the critical equipment, i.e., the superconducting magnets.

Figure 4.2 shows a schematic vertical cross section of ITER and the physical interfaces between the plasma, the blanket and the other systems like vacuum vessel, superconducting Toroidal Field and Central Solenoid coils .

The utilisation of the space on the inner side of the torus represents a crucial aspect in tokamak design and providing sufficient shielding in the restricted space of the inboard region is a challenge.

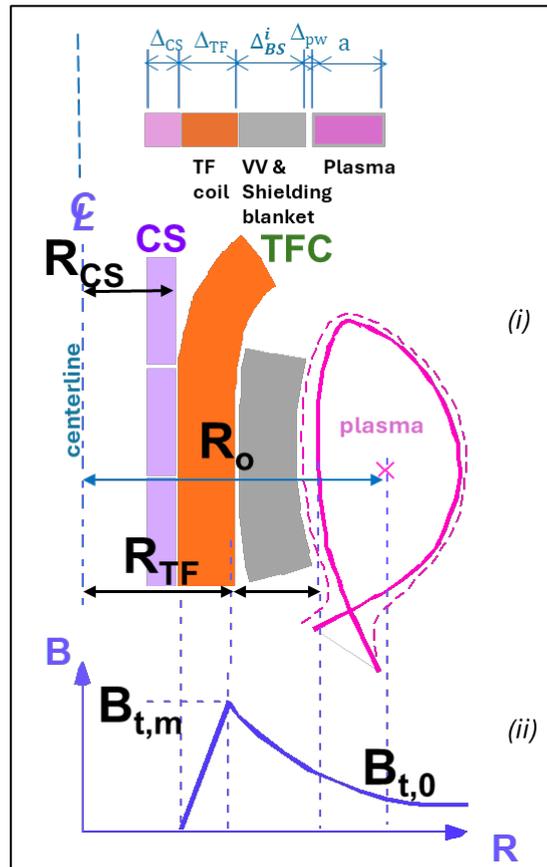


Fig. 4.2. (i) Schematic radial build of a tokamak showing the key fusion components at the inboard: central solenoid, toroidal field coils, vacuum vessel and shielding blanket; (ii) radial profile of the magnetic toroidal field that peaks at the coil. (source figure [15])

The fusion power in a tokamak is a function of:

$$P_F \propto \beta^2 B_{t,o}^4 V_p \quad (3)$$

where  $\beta$  has been defined above,  $B_{t,o}$  is the toroidal field strength at the centre of the plasma and  $V_p$  is the plasma volume. By increasing  $B_{t,o}$  and/or beta one clearly obtains a significant increase in power output. However, beta is limited by plasma stability and  $B_{t,o}$  by the maximum practical magnetic field,  $B_{t,m}$ , at the magnet windings that is limited by technological constraints, e.g.  $\sim 13$  T for  $Nb_3Sn$ .

The relationship between  $B_{t,m}$  and  $B_{t,o}$  (see Fig. 3.1)[16],[17] is given by:

$$B_{t,o} = \left(1 - \frac{1}{A} - \frac{\Delta_{pw} + \Delta_{BS}^i}{R}\right) B_{t,m} \quad (4)$$

where  $A$  is the aspect ratio ( $R/a$ ) ( $A = 3.1$  for ITER),  $R$  is the major radius of the plasma, and  $\Delta_{p,w}$  is the clearance between the plasma and the first wall (roughly  $\sim 0.2$  m). The parameter  $\Delta_{BS}^i$  is the thickness of the region occupied at the inboard by the blanket and the vacuum vessel, and includes also maintenance clearance and the thermal shield (i.e. the distance in mid-plane from the first wall to the toroidal field (TF) coil windings). The cost of the TF coils, typically, increases as  $B_{t,m}^2$ .

Eq. (4) clearly shows that by reducing  $\Delta_{BS}^i$ , for a given  $B_{t,m}$ , one can increase the value of the toroidal field strength at the centre of the plasma, and, thus, the reactor power, or for a given reactor power can reduce the machine size (i.e.  $R$ ).

Similarly, the flux core radius,  $R_{CS}$ , for the central solenoid (CS) coils is given by (see Fig. 4.2):

$$R_{CS} = R_o - (a + \Delta_{pw}) - \Delta_{BS}^i - \Delta_{TF}^i \quad (5)$$

where  $\Delta_{TF}^i$  is the thickness of the inner TF coil leg and its support structures and  $\Delta_{CS}$  is the thickness of the CS. For a given  $R$ ,  $a$ ,  $\Delta_{p,w}$ ,  $B_{t,o}$ , and  $P_F$ , reducing  $\Delta_{BS}^i$  reduces also  $B_{t,m}$  and  $\Delta_m$  and  $R_{CS}$  increases. Increasing  $R_{CS}$  reduces the ohmic heating field,  $B_{CS}$  ( $B_{CS} \sim 1/R_{CS}^2$ ). Besides the technological constraints on  $B_{CS}$ , the cost of the CS coils, and more importantly the cost of the power supply increases rapidly with  $B_{CS}$ .

All these factors, provide a strong incentive to reduce  $\Delta_{BS}^i$ . However, providing the radiation attenuation in the blanket/shield necessary for magnet protection favours a relatively large  $\Delta_{BS}^i$ . In a future device meant to produce electricity and its own tritium, further space would be required for the breeding blanket.

The previous discussion clearly shows that the dimensioning of ITER and any other fusion device is not arbitrary but strongly linked to mission objectives and predefined design and operating requirements. During the last 40 years, the design of ITER, its objectives, and design parameters have evolved because the experimental plasma physics knowledge base has on the one hand grown with the results stemming from the operation of several tokamak devices, the improved understanding and predictive capabilities of sophisticated simulation codes, and the commissioning of several new devices, and on the other hand the progress in understanding has meant that some early, more optimistic, predictions of plasma performance had to be changed. A brief review of the underlying assumptions behind these studies clarifies the definition of, and the rationale for, the choice of ITER objectives and parameters.

The choice of the plasma current is one of the parameters that, for a given magnet technology, has the strongest impact on device size and capital cost. The original values set for plasma currents were based on the empirical scalings for energy confinement which enjoyed the widest support at that time. In particular, a major driver towards moving to a higher current, as shown in Table 4.2, was the fact that in the 1980s an unfavourable dependence of the confinement time on the heating power was found in both low (L) confinement- and high [18] (H) confinement modes [19]. However, the factor of 2 increase in confinement observed in H-modes [20] re-opened the door for a viable fusion reactor design.

For a given plasma current, the size and cost of the device depend mainly on the plasma shape (e.g. elongation, triangularity, aspect ratio), on engineering allowables, design concepts of the assembly, as well as the plant layout.

Table 4.1 shows the evolution of objectives of ITER while the evolution of its design parameters and performance, and its precursors is shown in Table 4.2. One can easily see that the correlation between performance objectives and main parameters of the devices evolved towards increasing the size whilst maintaining similar objectives, as is shown below where the main objectives and parameters of some of these design studies are summarised.

Tab. 4.1. Historical evolution of ITER Objectives.

|         | Concept Design Activities (CDA) (1988)   | Engineering Design Activities (EDA) (1992)  | Reduced Technical Objectives/Reduced Cost (RTO/RC)[21] (2001)                       |
|---------|--|---|---|
| General | Demonstrate technologies essential to a reactor in an integrated system and perform integrated testing of the high-heat-flux and nuclear components required to utilise fusion power for practical purposes. | Aim to keep device costs within limits comparable to those of the CDA.  | 50% of device cost of EDA design.   |
| Plasma  | Demonstrate controlled ignition and extended burn of DT plasmas, with steady state as an ultimate objective.   | Demonstrate controlled ignition and extended burn of DT plasmas. Aim at demonstrating steady state operation. | Inductively driven $Q \geq 10$ .<br>Aim at steady state operation with $Q \geq 5$ . |

|        | Concept Design Activities (CDA) (1988)  | Engineering Design Activities (EDA) (1992)  | Reduced Technical Objectives/Reduced Cost (RTO/RC)[21] (2001)   |
|--------|---|---|---|
| Engin  | <p>Validate design concepts and qualify engineering components for a fusion power reactor. Demonstrate reliability and maintainability.</p> <p>Demonstrate potential for safe and environmentally acceptable operation of a power-producing fusion reactor.</p> | <p>Demonstrate essential fusion reactor technologies (e.g. superconducting magnets and remote maintenance).</p> <p>Operate safely and demonstrate safety and environmental potential of fusion power.</p>       | <p>Demonstrate availability and integration of essential fusion reactor technologies (e.g. superconducting magnets and remote maintenance).</p> <p>Assure safe and reliable operation.</p>                      |
| Testin | <p>Serve as a test facility for neutronics, blanket modules, tritium production, and advanced plasma technologies. Extract high-grade heat from reactor-relevant blanket modules appropriate for the generation of electricity.</p>                             | <p>Test reactor components (e.g. exhaust systems). Test reactor-relevant tritium-breeding blankets, demonstrating capability leading to TBR&gt;1 in a reactor, high grade heat, and electricity generation.</p> | <p>Test reactor components (e.g. exhaust systems). Test reactor-relevant tritium-breeding blankets, demonstrating capability leading to TBR&gt;1 in a reactor, high grade heat, and electricity generation.</p> |

|       | Concept Design Activities (CDA) (1988)  | Engineering Design Activities (EDA) (1992)  | Reduced Technical Objectives/Reduced Cost (RTO/RC)[21] (2001)   |
|-------|---|---|---|
| Opera | Two phases: a physics phase devoted mainly to achieving the plasma physics objectives, and a technology phase devoted to engineering objectives and the testing programme. Machine modifications within and between phases. | <p>Basic Performance Phase (BPP) lasting 10 years with a few thousand hours of DT operation, to address controlled ignition, extended burn and steady state operation (using initial heating systems), blanket testing.</p> <p>Enhanced performance Phase (EPP) lasting 10 years improving overall performance (with potential H&amp;CD upgrade) and conducting higher fluence component and materials testing. May demonstrate reactor-relevant blanket.</p> <p>Implementation of EPP depends on BPP review.</p> | <p>~20 years operation. Operation modes should be determined having sufficient reliability for nuclear testing. Low-fluence functional tests of blankets should be conducted early on. DEMO-relevant tests can then be conducted later in higher fluence/flux conditions.</p> |

A further detailed description of the evolution of the ITER design parameters and technical characteristics of the various points design is presented in Section 7.1.

From Table 4.2 one can see that the weight of materials substantially exceeds that of the CDA device, and the overall size of ITER has increased by almost 30% from 1990 to 1994. This increase in size arose partly from the need to increase volt-seconds due to the change in objectives in 1992 to provide a 1,000-second inductive pulse, and partly from maintaining confinement capability (probability of ignition) while adopting a lower, more conservative value for plasma elongation. In addition, a large volume single-null lower divertor was adopted in 1992, to provide maximum space for assembly and improve the maintenance of the divertor which would have been more difficult in the double null configuration of the CDA ITER.

Although the details of the mode of divertor operation were not yet provided, the concept was for a so-called “detached divertor”, i.e., a divertor designed to favour the formation of high density/neutral density “cloud” in front of the divertor plates, where the heat flux carried by the plasma along field lines can be dissipated by radiation, charge-exchange other atomic processes, figuratively “quenching the plasma”. These processes distribute the power over the entire surface area of the divertor channels, achieving a reduction of the heat flux density to the material surfaces to values that can be handled by the divertor.

This design evolution was substantiated by more detailed knowledge of the plasma physics and benefitted from advances of technologies. The ability and experience of the designers play a major role in identifying reliable and cost-effective design point solutions, striking the right balance between innovation and industrial practice. For example, a layer-wound cable-in-conduit conductor (CICC) was adopted for the toroidal field and poloidal field coils, to allow all connections to be external to the core of the tokamak. A TF Coil design bucked on the CS was adopted where the TF coils and CS are mutually supported against a bucking cylinder. This approach was intended to provide greater compactness, possibly at the cost of more difficult manufacturability and added complexity.

Tab. 4.2. Evolution of objectives and design parameters of ITER and precursors.

| Device   | INTOR<br>1982 | NET<br>1986 | ITER<br>1990 | ITER<br>1994 | ITER<br>1998 | ITER<br>2001 |
|--|---------------|-------------|--------------|--------------|--------------|--------------|
| Fusion power   | 585           | 650         | 1000         | 1500         | 1500         | 500          |
| Installed auxiliary power (MW)                           | 50            | 50          | 145          | 50           | 100          | 73           |
| Q-fusion power/additional heating power                  | (ignited)     | (ignited)   | (ignited)    | (ignited)    | (ignited)    | ≥10          |
| Average 14 MeV n-wall loading (MW/m <sup>2</sup> )       | 1.22          | 1           | 1            | 1            | 1            | 0.57         |
| Average n-fluence (MWa/m <sup>2</sup> ) operation/design | 3             | 0.8/2       | 1/3          | 1/3          | 1/3          | 0.3–0.5      |
| Pulse duration (inductive)(s)                            | 150           | >200        | 400          | 1000         | 1000         | 300–500      |
| Plasma current (I <sub>p</sub> )                         | 8             | 10.8        | 22           | 24           | 21           | 15           |
| Toroidal field at plasma major radius (T)                | 5.5           | 5           | 4.85         | 6            | 5.7          | 5.3          |

|  |         |          |          |         |         |         |
|--|---------|----------|----------|---------|---------|---------|
| Plasma elongation                            | 1.6     | 2.05     | 1.98     | 1.6     | 1.6     | 1.7     |
| Major/ minor radius (R/a)                    | 5.2/1.2 | 5.2/1.35 | 6.0/2.15 | 7.7/3.0 | 8.1/2.8 | 6.2/2.0 |
| Divertor (SN=single null;<br>DN=double null) | SN      | DN       | DN       | SN      | SN      | SN      |

Concerning the neutron fluence and the pulse length, it was initially considered that reactor relevant testing of a breeding blanket would be best achieved by operating ITER for a significant fraction of the target fluence of a reactor, i.e. about 3 MWa/m<sup>2</sup> and a pulse length of thousands of seconds to reach stationary conditions in the thicker regions of the blanket under test. The cost versus benefit of such ambitious objectives in physics and technology was challenged at the end of 1998, when both the objectives and the cost effectiveness of the proposed device came into question. This continues to be challenged as cheaper means to obtain the necessary data become apparent (see Section 6.8).

The need to save in capital costs while maintaining the major programmatic goals led to a redefinition of the ITER design parameters and machine configuration (with a capital cost 45–50% of that of ITER 1998). A study on the Reduced Technical Objectives/Reduced Cost ITER (RTO/RC ITER) (see Section 7.1.6) showed that reducing the technical objectives without changing the nature of the design was not enough on its own to achieve the 50% reduction in cost target. Thus, further design changes, compared to the 1998 design of ITER, would be necessary.

The main changes proposed include less demanding performance objectives (setting as a goal operation at Q = 10 instead of ignition), shorter plasma inductive burn duration, higher elongation and triangularity to increase the plasma current for the same R, improve the density limit and the beta limit, and lower fluence. This led to the need to design a suitably robust plasma vertical stability to take advantage of elongation and triangularity, ranging from the use of a close-fitting passive shell to copper cladding on the vessel, and passive internal saddle coils, resulting with the need to use active stabilisation coils inside the vessel.

Additional measures were implemented to reduce the machine size, including:

- the use of wedged toroidal field (TF) coils with a segmented central solenoid and poloidal field (PF) coils linked above and below the equatorial plane to handle a higher plasma elongation and triangularity within power supply limits;
- the need to maintain the largest plasma volume at the expense of a reduced divertor volume, increased neutron heating of the TF and CS winding due to reduced shield thickness, and the elimination of the blanket backplate (which meant connecting

mechanically and hydraulically the blanket modules and their coolant channels directly to the vessel);

- the need to maintain the largest possible port access at the expense of a reduced intercoil mechanical structure.

It was recognised that the ITER design variant (ITER 2001 in Table 4.2), now in construction, could more reliably achieve the power density and wall loading objectives, and thus was a more prudent development path to a reactor.

As mentioned previously, a major alternative route to studying burning plasma behaviour envisages using compact, higher field experiments, which may be less expensive.

High field tokamaks, based on copper coils, have been used in the United States, Italy and Japan [22],[23] [24]. Concepts were also proposed, but never built, for devices like BPX (CIT) [25], the Fusion Ignition Research Experiment (FIRE) [26], and IGNITOR [27], [28] to explore high-Q operation and possibly ignition. The main motivation behind the choice of high field has been to explore the physics of regimes with a high  $nT\tau_e$  parameter in limited size and cost experiments. High magnetic fields in a reduced size support designs with higher plasma current and density although at lower confinement time.

The potential downsides of this approach are the higher heat loads expected on the divertor. New divertor solutions with performances well beyond today's concepts would be needed and complex mechanical and electrical engineering provisions must be adopted. For example, higher field windings generate higher forces in the mechanical structures in and around the plasma, in particular in the toroidal field (TF) coils, which constitute the structural core of the machine. The stresses on the inboard leg of the TF coil casing tend to quickly reach the maximum allowable stress for a given geometry, effectively limiting the field. This stress roughly scales with  $B^2/S$  (where  $S$  is cross section of the structural elements).

Copper coils, even cooled at liquid nitrogen conditions to lower copper resistivity and increase strength, have a limited pulse duration of a few seconds or tens of seconds, due to the high electric power required to sustain the extremely high currents and current densities in the windings.

Recently, a new interesting route has emerged with the advent of high temperature superconductors (HTSs) that offers the promise of operating at higher magnetic field and temperature for longer pulses. This allows the use of higher field magnets and has then been promoted by several groups around the world to substantially reduce the size of a fusion power reactor system and as a breakthrough innovation that could dramatically accelerate fusion power deployment. There are some major challenges though, including making sure that the higher electromagnetic forces resulting from the increased field can be accommodated by the magnet structure and that the reduced reactor size still provides sufficient space for the required shielding

of the magnets and other components, which, even for a reactor the size of ITER, had proven to be very problematic.

Main concerns in the case of a thin-film type HTS, which is currently used, are uncertainties in terms of the performance required for tokamak reactors, such as whether it can operate stably enough against magnetic loads, alternating current losses and disturbances in a complex magnetic field and nuclear irradiation environments, whether it can provide the precise magnetic field profile necessary for plasma shaping and operation, and whether it can accurately and quickly detect abnormalities (quench detection) in the superconducting state for its protection. The discoveries and innovations in HTS that have brought them into such consideration have occurred after the consolidation of the ITER magnet design. In the future, such a technical basis may be available. Nevertheless, HTSs are used to reduce heat losses occurring when conventional power supply lines are connected to the superconducting terminals of ITER's magnets.

## 4.4 A concise description of the ITER design

At present, ITER is the largest tokamak being built to produce fusion energy by confining a deuterium-tritium plasma in which the  $\alpha$ -particle heating dominates all other forms of plasma heating. Its primary mission is to regularly sustain a DT plasma producing  $\sim 500$  MW of fusion power for durations of 300–500s with a ratio of fusion output power to input heating power,  $Q$ , of at least 10. ITER is also designed to explore the physics basis for the continuous operation of fusion power plants by investigating 'steady state' plasma operation by means of non-inductive current drive for periods of up to several thousand seconds while maintaining a fusion gain,  $Q$ , of  $\sim 5$ . The project's technical goals encompass significant technological demonstrations to prepare the design basis for a tokamak fusion power plant. One of the most important awaited demonstrations is that of breeding blanket modules that are inserted in a limited number of properly equipped equatorial ports.

These tests are considered essential in evaluating the performance of the various options to be tested and confirming the feasibility of tritium self-sufficiency and the extraction of tritium and high-grade heat. Achieving tritium self-sufficiency will be an inescapable requirement for any next-step fusion facility beyond ITER. However, no fusion breeding blanket has ever been built or tested. Hence, to ensure its crucial integrated functions and reliability in DEMO and future power plants, ITER presents a first and unique opportunity to test the response of representative component mock-ups, specifically called Test Blanket Modules (TBMs) at relevant operating conditions, in an actual fusion environment, albeit at very low neutron fluences (see Section 6.8).

ITER is a long-pulse tokamak with elongated plasma shape and single null poloidal divertor located at the bottom of the vacuum chamber. The ITER design is based on scientific knowledge and extrapolations derived from the operation of tokamaks over the past decades and on the technical know-how flowing from the fusion technology R&D programmes around the world.

The design has been validated by wide-ranging physics and engineering work, including detailed analyses, specific experiments in existing fusion research facilities and dedicated technology developments and tests.

The major components of the tokamak are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside a toroidal vacuum vessel. The magnet system comprises toroidal field coils, a central solenoid, external poloidal field coils, and correction coils. The superconducting magnets are cooled with helium at 4.5 K. The toroidal field coil windings are enclosed in strong mechanical containment structures. The vacuum vessel is a double-walled structure with water filled between the walls to provide cooling and radiation shielding capabilities.

The magnet system and the vacuum vessel, together with internals, are supported separately by the so-called gravity supports. Inside the vacuum vessel, the internal, replaceable components, including blanket modules, divertor cassettes, and port components such as the heating antennas, test blanket modules, diagnostic modules, and/or port plugs, absorb the radiated heat as well as most of the neutrons from the plasma, protecting the vessel and magnet coils from excessive heat and damage by nuclear radiation. The heat deposited in the internal components and in the vessel is removed from the vessel via the tokamak cooling water system. The entire tokamak is enclosed in a cryostat, with thermal shields between the hot components and the cryogenically cooled magnets.

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 a circular configuration to an elongated divertor configuration as the plasma current is ramped up. As the current develops (nominally up to 15 MA), subsequent plasma fuelling (gas or pellets) and additional heating leads to a high energy gain burn and finally to a controlled burn with a fusion power of about 500 MW. With non-inductive current drive from the heating systems, it is envisaged that the burn duration will ultimately be extended towards 3000 s. In inductive scenarios, before the inductive flux available has been fully used, reducing the fuelling rate 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 300 to 500 s, with a pulse repetition period as short as 1800 s.

Fig 4.3 shows a cutaway of the ITER Tokamak while some key ITER figures, the main ITER systems, and the main operation phases are summarised in Tables 4.3, 4.4 and 4.5, respectively.

Table 4.3: Some key figures about ITER.

- The ITER Tokamak is the largest ever built, with a plasma volume of 840 m<sup>3</sup>. The maximum plasma volume in tokamaks operating today is 160 m<sup>3</sup> reached in Japan's JT-60SA;
- The ITER tokamak weighs 23,000 t, the equivalent of three Eiffel Towers;
- The ITER vacuum vessel measures 19.4m across (outer diameter), 11.4 m high, and weighs approximately 5,200 t of which 1,611 t is the vacuum vessel, 1733 t for the shielding plates, 1781 t for the ports and 111 t for the supports. With the installation of the blanket and the divertor, the vacuum vessel will weigh 8,500 t. Its double wall thickness varies from 0.34 m at the inboard to 0.75 m at the outboard. The internal surface of the vacuum vessel is 850 m<sup>2</sup> and it has a volume of 1400 m<sup>3</sup>. The double-walled structure is filled with water to provide cooling and radiation shielding capabilities;
- The structure of the ITER central solenoid, the large, 1,000 t electromagnet in the centre of the machine, must be strong enough to contain a force equivalent to twice the thrust of a space shuttle at take-off: 60 MN, or over 6,000 t of force. The CS coils produce inductive flux to ramp up and maintain the plasma current and contribute to shaping the plasma. In ITER the CS is made of six independent Nb<sub>3</sub>Sn modules, with a total height of 13 m, a diameter of 4 m and a total weight of about 1000 t. The maximum operating current in the CS modules is 45 kA, leading to a maximum field of 13T reached in the centre of the stacked modules and a magnetic energy stored in CS coils of 7 GJ. This generates sufficient magnetic flux to allow initiating and sustaining a plasma current of 15 MA for about 400 s under burning plasma conditions;
- Every one of the ITER's 18 D-shaped Nb<sub>3</sub>Sn [toroidal field](#) coils is 17 m high and 9 m. wide and weighs 360 t (the approximate weight of a fully loaded Boeing 747-300 airplane), distributed among its components: 110t for the winding pack, 190 t for the casing and 60t for pre-compression systems, shear keys and bolts. The total magnetic energy in the TF coils is around 40 GJ. The current flowing in the TF conductors is 68 kA, which produces a maximum magnetic field of 11.8 T on the coil;
- The PF coils control the radial position of the plasma, contributing to its stability by maintaining the clearance between the plasma and the first wall. PF magnets are also essential elements of the slow vertical position control and of the plasma shaping capability. In ITER the six annular NbTi PF coils are located outside of the TF magnet structure. They are designed to operate at a maximum current of 45 kA, which generates a maximum magnetic field of 6 T on the coils and a total magnetic energy of 4 GJ. The largest magnet has a diameter of 24 m and the heaviest is 400 t, with a total combined weight of 2,163 t. All of the coils mentioned above are toroidally symmetric;
- Some 400,000 t rests on the lower basemat of the Tokamak Complex including the buildings, the 23,000 t machine and equipment. This is more than the weight of New York's Empire State Building.
- The Tokamak Building is slightly taller than the Arc de Triomphe, Paris. Measuring 73 m (60 m above ground and 13 m below), it is the tallest structure on the ITER site.

- The main feature of the 180 ha ITER site in Saint Paul-lez-Durance, southern France, is a man-made level platform that was completed in 2009. This 42 ha platform measures 1 km long by 400 m wide, and compares in size to 60 football fields.

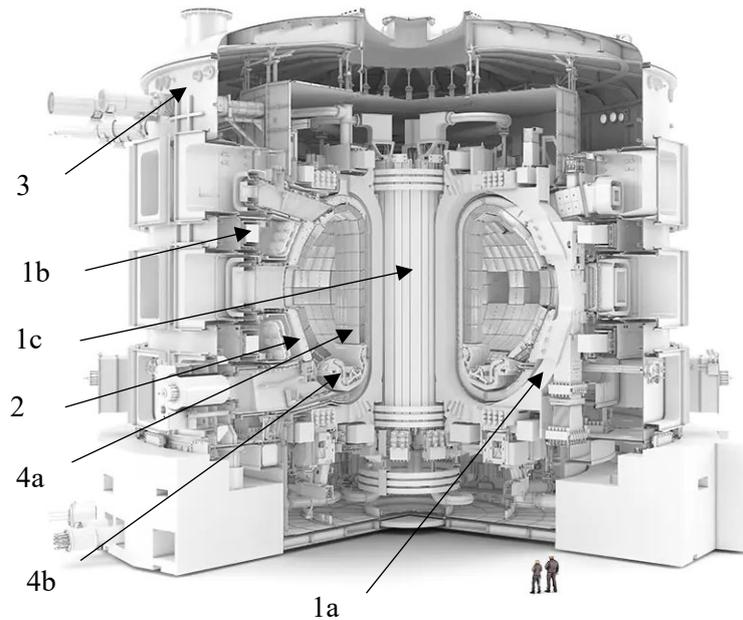


Fig. 4.3. Cutaway view of the ITER tokamak (numbers shown refer to some key components/systems, as listed in Tab. 4.4).

Tab. 4.4 Main ITER systems.

|   |  |   |
|---|--|---|
| <p>1. Magnets</p> <ul style="list-style-type: none"> <li>a. Toroidal Fields Coils (18)</li> <li>b. Poloidal Fields Coils (6)</li> <li>c. Central Solenoid (6)</li> <li>d. Pre-Compression Rings (6)</li> <li>e. Correction coils (18)</li> <li>f. In-vessel coils for VS and ELM control.</li> </ul> <p>2. Vacuum Vessel (9 sectors)</p> <p>3. Cryostat</p> | <p>4. In-Vessel</p> <ul style="list-style-type: none"> <li>a. Shielding Blanket</li> <li>b. Divertor</li> </ul> <p>5. Auxiliary Heating Systems</p> <ul style="list-style-type: none"> <li>a. Neutral Beam</li> <li>b. EC launchers</li> <li>c. RF Antennas</li> </ul> <p>6. Diagnostics</p> | <p>7. Fuel Cycle</p> <p>8. Remote Handling</p> <p>9. Test Blanket Modules</p> |
|---|--|---|

The ITER plan (baseline 2024) foresees three main operation phases: 1) Start of Research Operation (SRO); 2) First Deuterium-Tritium phase (DT-1); and 3) Second Deuterium-Tritium phase (DT-2). Each operational phase is preceded by integrated commissioning phases aimed at ensuring that each of the components and plant systems installed during the tokamak assembly phase or upgrade has been installed correctly. It is also an opportunity to transfer knowledge to the operations team, test all the procedures, satisfy any licensing requirements, and get ready to start the first experiments.

With the advent of the ITER baseline 2024 [29][163], the operation strategy is based on the progressive acquisition of knowledge, combining a stepwise approach to demonstrating nuclear safety with the stepwise approach to operation, based on several phases: two first operational phases (SRO & DT-1) with limited nuclear safety stakes, constituting learning phases for the acquisition of data on systems and components, and a third operational phase (DT-2) with the target of achieving all the goals of the Project Specification, which will then be fully established on the basis of the results of the first two phases.

*Tab. 4.5. Main ITER Operation phases.*

**Start of Research Operation (SRO):** This phase starts with the demonstration of the first tokamak plasma and concludes with the demonstration of tokamak operation up to the nominal design parameters of 15 MA / 5.3 T in a diverted plasma configuration, including the use of the electron cyclotron (EC) and ion cyclotron (IC) heating and current drive (H&CD). Within the SRO phase an experimental campaign with deuterium plasmas at reduced current up to 7.5 MA and toroidal field (2.65 T) will take place to explore the operational space and control of H-mode plasmas in ITER. This deuterium campaign will have a limited neutron budget ( $\sim 1.5 \cdot 10^{20}$ ) to allow entry of personnel into the vacuum vessel for installation of components in the post-SRO shutdown. Risks of damage to the water-cooled blanket in-vessel components are removed by installing in their stead a temporary inertially cooled first wall. This will allow the full characterisation of disruption loads, the validation of safety-related assumptions, and the commissioning of the disruption mitigations system up to full magnetic energy (15 MA / 5.3 T). Since the first wall is inertially cooled (the divertor is water cooled) the duration of the flat top with high heating power levels will be limited to about 30–50 s. In essence, in this phase, the engineering evaluation of the ITER tokamak as an integrated system with as-built and as-assembled components/systems will be performed. The SRO phase is foreseen to last 27 months. The post-SRO phase for assembly and commissioning towards DT-1 is envisaged to last less than 36 months.

**First Deuterium-Tritium operation phase (DT-1):** In this phase, DT plasma scenarios will be developed to demonstrate the attainment of the key project goal of 500 MW of fusion power with a multiplication factor  $Q \geq 10$  for lengths of at least 300 s and to demonstrate high-duty operation (i.e. one pulse every 30 minutes) with fusion power levels of 250 MW for at least 300 s. Research in this phase will address, amongst others, a wide range of burning plasma

physics and scenario integration issues, and will provide demonstrations of key technologies required for demonstration fusion reactors, including the installation and testing of four test blanket systems as planned by the ITER TBM Program. The neutron fluence in this phase will be limited to  $\sim 1\%$  of the ultimate project fluence goal ( $\sim 3.5 \cdot 10^{25}$  neutrons) to enable the performance of ex-vessel maintenance activities, while respecting the specified shutdown dose rate requirements for workers. This phase will provide key reference data to confirm safety-related evaluations for DT Operation in ITER (e.g. radiation maps, T retention and removal, dust production etc.), which will be used to define the details of the licensing requirements in the DT-2 operation phase. We note that, while the neutron production in this phase is limited, it is high enough to require the use of remote handling tools to perform in-vessel shutdown activities in DT-1. The DT-1 operation phase is foreseen to last up to 10 years to include five experimental campaigns.

**Second Deuterium-Tritium operation phase (DT-2):** In this phase, DT plasma scenarios will be developed to demonstrate all the project's fusion power production goals. These goals are specifically: the demonstration of 500 MW of fusion power with fusion power multiplication factor  $Q \geq 10$  for lengths of 300-500 s in high duty operation and of long pulse and steady-state scenarios with  $Q \geq 5$  and burn lengths of 1000 s and 3000 s respectively, as defined in the Project Specifications and Project Requirements [5]. In addition, research will be carried out to support the ITER Members' demonstration fusion reactor programmes including both scenario development issues (e.g. heat flux exhaust), design/operational issues (e.g. optimum H&CD mix) and their TBM programs, in principle, up to neutron fluences of at least 0.3 MWa/m<sup>2</sup>.

Before the start of DT-2 specific integrated commissioning of the newly available or upgraded components or systems and recommissioning of the existing ones will take place. The DT-2 phase is foreseen to last up to 10 years to include five experimental campaigns. Table 4.4 summarises key tokamak components and systems that will be available to execute the programme in each of the phases. The successful implementation and execution of the ITER Research Plan rely on close collaborations among the ITER Organization and the ITER Member's fusion research institutes in both the experimental and theory/modelling areas.

Table 4.4 shows key tokamak components/systems available to execute the programme in each of the phases described in table 4.5.

Tab. 4.5. Main Tokamak components and systems in each of the phases.

| System/<br>Ancillary<br>Available   | SRO           | DT-1          | DT-2          | System/<br>Ancillary<br>Available | SRO  | DT-1   | DT-2   |
|---|---------------|---------------|---------------|-----------------------------------|--|--|--|
| Vacuum Vessel, Thermal Shield and Cryostat  | Final Config. | Final Config. | Final Config. | Disruption Mitigation System      | Final Config.  | Final Config.  | Final Config.  |
| Toroidal Field, Poloidal Field, Central Solenoid and Error Field Correction Super-conducting Magnets and Power Supplies | Final Config. | Final Config. | Final Config. | Electron Cyclotron Heating        | Upper Launchers installed, 3 operational<br>1 Equatorial Launcher<br>(40 MW) | 3 or 4 Upper Launchers operational<br>2 Equatorial Launchers<br>(60–67 MW) | 3 or 4 Upper Launchers operational<br>2 Equatorial Launchers<br>(60–67 MW) |
| In-vessel Vertical Stability and ELM Control Coils and Power Supplies   | Final Config. | Final Config. | Final Config. | Ion Cyclotron Heating             | 1 Antenna<br>(10 MW)   | 1 Antenna<br>(10–20 MW)  | 1 Antenna<br>(10–20 MW)  |
| Cryostat and Torus Cryopumps  | Final Config. | Final Config. | Final Config. | Neutral Beam Heating              |  | 2 Injectors<br>(33 MW)   | 2 or 3 Injectors<br>(33–49.5 MW)   |

| System/<br>Ancillary<br>Available                 | SRO                                      | DT-1             | DT-2             | System/<br>Ancillary<br>Available | SRO  | DT-1   | DT-2  |
|---|--|------------------|------------------|-----------------------------------|--|--|---|
| Blanket<br>Shield<br>Modules<br>and First<br>Wall | Temp.<br>Config.<br>inertially<br>cooled | Final<br>Config. | Final<br>Config. | Diagnostic<br>Neutral<br>Beam     |  | Final<br>Config.   | Final<br>Config.  |
| Divertor  | Final<br>Config.                         | Final<br>Config. | Final<br>Config. | Diagnostics                       | Basic set for<br>SRO phase<br>(incl. for the<br>safety-<br>related<br>knowledge<br>acquisition<br>programme) | Near<br>complete<br>set,<br>including<br>DT fusion<br>products | Complete<br>set   |
| Glow<br>Discharge<br>System                       | Partial<br>Config.                       | Final<br>Config. | Final<br>Config. | Hot Cell<br>Facility              | Partial<br>config. for<br>operation<br>on TFA<br>liquid<br>radwaste,<br>independent<br>ly of HC<br>building  | operational<br>for DT-1  | operational<br>with<br>expanded<br>capabilities<br>for DT-2 |
| Boronisation<br>Gas<br>Distribution<br>(B2D6)     | Final<br>Config.                         | Final<br>Config. | Final<br>Config. | Test<br>Blanket<br>Modules        |  | DT-1 TBMs  | DT-2 TBMs   |
| Gas and<br>Pellet<br>Injection<br>Systems         | Partial<br>Config.                       | Final<br>Config. | Final<br>Config. | Tritium<br>Plant                  |  | operational  | operational<br>with<br>expanded<br>capabilities<br>for DT-2 |

The complexity of a fusion power plant like ITER, with regard to integration of plant systems, is much higher than that in a fission power plant due to the larger number of plant systems.

The general plant architecture is arranged around the tokamak, with a cylindrical bioshield of 2 m thickness around the cryostat, and floor levels corresponding to the cryostat penetrations. Additional levels are used for the integration of auxiliary equipment for the various plant systems and for accident mitigation systems. Currently, the tokamak building also represents a final nuclear

confinement barrier for the radioactive material towards the environment and the public. This safety function requires the plant and safety systems to limit the release of radioactive substances during normal operation and in accidental conditions well below the safety limits.

Furthermore, fusion is a nuclear technology and as such will receive scrutiny from regulators under relevant principles such as ALARA (risks As Low As Reasonably Achievable). This means that licensing considerations related to shielding, safety, and remote handling (RH) must be considered from the outset and can play a significant role in the design. Final assessments in this area can only be conducted once engineering design details have been developed and thus require a certain maturity of design. When the decision to site ITER in France was taken in 2005, a commitment was taken to observe all French regulations. By virtue of its tritium inventory, ITER falls in scope of nuclear licensing as an INB along with fission reactors and other facilities with a radioactive material inventory. It is therefore regulated by the Nuclear Safety Authority and Radioprotection (Autorité de Sûreté Nucléaire et de Radioprotection, ASN) and subject to the provisions of the nuclear transparency law (TSN law).

More recently, discussions in some countries and at the IAEA have considered the approach to the licensing of fusion power plants in the future. In some cases, notably the USA and the UK, it is foreseen that fusion plant will not be licensed under existing nuclear regulations intended for fission reactors, but under new regulations that will be proportionate to the reduced risks presented by fusion. In the UK, this position has been enshrined in law. However, these developments have come too late to influence the licensing of ITER, which remains under existing French nuclear regulations. This subject is further discussed in Chapters 8 and 10.

## 4.5 Distinctive features of ITER and main differences with existing devices

The ITER goals, especially for the pulse lengths, duty cycles, and availabilities, represent a very large extrapolation from current experience and place a large demand on design integration, power handling, neutron shielding, component reliability, and maintenance procedures.

The essential differences between the attainments of today's tokamak research facilities and what should be achieved in ITER are summarised in Table 4.6 together with the related primary issues. The increase in fusion power and in plasma energy content together with the increase in pulse duration and cumulative run time represent the largest changes in operation conditions in ITER. These will have by far the greatest consequences, giving rise to important effects, which are only partially observed and accessible in present day experiments, and will provide new physics insight, and open new design, operation and safety issues.

A new class of important medium- and large-size devices that rely on the use of superconducting magnets is now in operation within ITER Member countries. They include for example the Large Helical Device [30] in Japan, the stellarator Wendelstein 7-X [31] in Germany, and several tokamaks, including WEST [32] in France, KSTAR [33], [34] in Korea, EAST [35], [36] in China and JT-60SA [37] in Japan. They routinely operate with long pulses and require the use of actively cooled components. However, many present-day machines still operate with copper-coils and in short pulse mode, with plasmas maintained for periods of the order of seconds, typical pause times of 5–30 minutes and power loads sufficiently low to be handled by inertial cooling.

In present-day tokamak devices, erosion of the main chamber wall and divertor strike plates act as a source of impurities in the discharge but does not affect the component lifetime, mainly because of the very low duty cycle. Erosion effects are on the scale of microns for a typical run campaign (1,000–5,000 s/operation year) [38]. Similarly, fuel economy has never been an issue in deuterium fuelled experiments, and only in the late 1990s have the limitations associated with the use of tritium and its incomplete recovery in experiments in TFTR [39] and JET [40] brought the issue of fuel retention under closer scrutiny. In contrast, long plasma duration and high duty cycle operation in a next step device such as ITER will lead to levels of erosion/redeposition and tritium retention that could affect operational availability, due to the necessity of tritium removal and/or divertor plate replacement. Also, divertor heat loading caused by disruptions and Type-I ELMs would imply melting/ablation for any material in the divertor wetted by the plasma.

Two of the main challenges pertaining to the large amount of tritium and need to control the in-vessel tritium inventory, and the problem of plasma transients and their mitigations, are briefly described below.

Tab. 4.6. Distinctive features of ITER and main differences with existing devices.

- |   |
|---|
| <p>1) <b>High energy content</b> (<i>several hundreds of MJ vs. few MJ in current devices</i>) and power flow (<i>several hundred MW in the scrape-off-layer (SOL) vs. tens of MW in current devices</i>):</p> <ul style="list-style-type: none"><li>• more intense disruptions and disruption-related damage effects (e.g., material ablation and melting) resulting in local damage and in a substantial influx of impurities into the plasma;</li><li>• vertical displacement events (VDEs) will produce much larger forces on vessel components in ITER than in present day tokamaks. As the plasma column moves during a VDE, a substantial fraction of the plasma current is transferred to a “halo” region around the main plasma. Since the halo intersects the vessel wall, the return path for this “halo current” passes through conducting components of the vessel structures. The flow of this return current will be perpendicular to the main magnetic field, thus exerting a large mechanical force on these structures;</li><li>• high electric fields produced by disruptions can also give rise to production of run-away electrons with multi-MeV energy. There are still substantial uncertainties related to the</li></ul> |
|---|

runaway current that is predicted to be much larger in ITER due to the avalanche effect, and to the quantification of material damage effects; the database from tokamaks is somewhat limited in this respect.

- plasma power densities for long durations in the divertor will be larger than in present-day experiments and require an effective and reliable way to disperse the power on the divertor surfaces.
- type-I ELMs will, if unmitigated, cause melting/ablation of any material in the divertor wetted by the plasma.

2) **Long pulse duration** (*few hundred seconds vs. few seconds in most of current devices*):

- requires active cooling of PFCs and vessel structures, and techniques to pump helium ash during each pulse;
- requires effective control of plasma purity and plasma-wall-interactions to achieve high plasma performance.

3) **High duty factor or pulse repetition rate** (*ultimately needed for meaningful breeding blanket testing in the second phase of operation*).

- high performance plasma operation with limited shutdown times for wall conditioning and tritium recovery;
- effective and fast remote maintenance to minimise shutdown times.

4) **Long cumulative run time** (*several thousands of hours of operation over several calendar years*):

- the erosion lifetime of PFCs may be sufficiently short that several replacements will be required during the lifetime of the device;
- large-scale erosion also raises tritium retention and dust (safety) issues that may determine the feasibility of reactor designs
- neutron damage effects in the bulk of surrounding materials and structures;
- plasma-facing surfaces will be modified in situ by the plasma, which will mix the plasma-facing materials. The composition, physical structure, and properties of deposited materials will be different to manufactured materials.

5) **Routine operation with large amount of tritium** (*10–100 g/pulse vs. fractions of g in current devices*):

- this requires a closed fuel cycle with efficient reprocessing of the exhaust stream (e.g., limited but successful operation experience of TFTR and JET);
- requires adequate tritium supply, minimisation of the inventory of tritium retained on in-vessel components and efficient methods of recovery;
- address safety concerns of any vulnerable tritium inventory.

**6) Superconducting magnet technology**

- a thick neutron shield is required to protect the toroidal and poloidal coils, thereby increasing the major radius and the overall size of the device; neutron and gamma irradiation destroy organic insulators and superconductors, and also increase the resistivity of the stabilising Cu (typical limit is  $<10^9$  rad);
- since the superconducting magnets will remain energised during the interval between pulses, certain techniques used in tokamaks for between-pulse wall preparation/conditioning will not be feasible.

**7) Fast remote maintenance for repair/refurbishment of the in-vessel and some of the ex-vessel components:**

- viability of remote methods of maintenance, repair and upgrading of fusion devices, is a key technology for development of fusion reactors (e.g., limited but very successful experience at JET);
- requires a specific system, component and sub-component design philosophy to minimise shut-down times (effective remote maintenance and repair), and to meet stringent constraints imposed by a radioactive environment.

**8) Safety**

- requires stringent and detailed safety assessments and safety-related procedures and controls, particularly once components have become activated and changes (in components or their operation) need to be made;
- stringent worker and public radiation protection measures;
- public outreach to address community acceptance and understand safety and risk in context.

**Control of the in-vessel tritium inventory** - Safe management and accounting of tritium is recognised to be crucial for the acceptance of fusion as an environmentally benign power source. Tritium retention in co-deposited materials from PFCs and dust formation and accumulation have emerged as a primary concern for ITER, with strong implications for in-vessel component design, material selection, operational schedule and safety. Special controls imposed on the handling of tritium require that the quantity of tritium retained in the torus be accounted for, and the inventory limited in order to permit continued operation within the licensed site inventory limit.

An excessive tritium inventory in the torus would present a safety hazard in the form of a potential tritium release to the atmosphere in case of a loss-of-vacuum event. Besides tritium inventory control, tritium removal from the wall is required to control plasma fuelling by tritium implanted in the wall and to reduce the tritium outgassing. Independently of safety limits, control of the in-vessel tritium inventory is also necessary to manage the available tritium supply. Tritium retention

mechanisms are reviewed elsewhere (see for example [41]). Table 4.7 shows a list of key quantities related to tritium in existing tokamaks and ITER [40].

Tritium retention and the control of the tritium inventory in ITER have been key drivers for the choice of plasma-facing materials and their operational conditions. Experimental results in the 90 s [40] showed very large retention of tritium co-deposition with carbon, and the need to mitigate/control tritium accumulation via co-deposition in the torus. Retention by other mechanisms such as implantation and surface adsorption, which may be significant for small, short pulse machines, is expected to rapidly reach saturation in ITER.

The experimental results with tritium fuel for the plasma led to the decision to abandon carbon and choose a beryllium (wall) and tungsten (divertor) combination as alternative solutions for the PFCs of ITER. Consideration on beryllium toxicity, its low melting point and risk of severe damage has now led to another change in the PFC materials for ITER, that is to use tungsten both in the divertor and in the first wall (See Vol.2 [42]). The recent DT JET-ILW experiments have indeed demonstrated a ten-fold reduction of fuel retention rates in deuterium compared to reference pulses in JET-C [43]. Furthermore, first results of ASDEX-Upgrade with a full tungsten wall and divertor have demonstrated reduced D fuel retention compared to C-dominated conditions in 2007-2008 44[.

Tab. 4.7. Tritium in previous fusion facilities and ITER.

| Parameters                          | TFTR      |      |      | JET   |           |           |           | ITER             |
|-------------------------------------|-----------|------|------|-------|-----------|-----------|-----------|------------------|
|                                     | 1993-1995 | 1996 | 1997 | 1991  | DTE1 1997 | DTE2 2021 | DTE3 2023 |                  |
| Peak fusion power (MW)              | 10.7      | 8.5  | 7.8  | 1.8   | ~16       | 13        | 14        | 500              |
| Total discharge duration (s)        | 8         | 4.2  | 6.5  | 10–30 | 20–30     | 12–34     | 12–34     | 300–500          |
| Total number of discharges          | 14724     | 5324 | 3619 | 2     | ~593      | ~922      | ~257      | >10 <sup>4</sup> |
| Typical # of tritium pulses per day | 0–5       | 0–5  | 0–5  | –     | 10        | 9         | 11        | 10-20            |
| T processed by facility (g)         | 73        | 17   | 15   | 0.1   | 100       | 1000      | 117       | –                |
| T introduced in the NBIs            | 3.3       | 0.84 | 1.1  | 0.005 | 65        | 763       | 0         |                  |
| T introduced in the torus           |           |      |      |       | 35        | 240       | 117       |                  |

|   |                    |       |        |                                   |                     |                      |       |                       |       |                |
|---|--------------------|-------|--------|-----------------------------------|---------------------|----------------------|-------|-----------------------|-------|----------------|
| T introduced per pulse (g/pulse)          | <0.048             | <0.01 | <0.014 |                                   | <0.25               | 0.2–1.5              | 0.1-1 | 50                    |       |                |
| T inventory in the torus and NBIs (g)     | 1.7                | 1.6   | 1.8    | 0.004                             | 11.5 <sup>(b)</sup> | N/A                  | N/A   | —                     |       |                |
| Avg. retention (excluding clean-up)       | 51% <sup>(a)</sup> |       |        | —                                 | 40%                 | <1% <sup>(e,f)</sup> | N/A   | <1% (predicted)       |       |                |
| Increment of tritium inventory (g)        | 1.7                | 0.81  | 0.76   | 0.004                             | 11.5 <sup>(c)</sup> | N/A                  | N/A   | <20 g/pulse           |       |                |
| T removed during clean-up period (g)      | 0.96               | 0.49  | 0.98   | 0.0045                            | 5.5 <sup>(c)</sup>  | 2 <sup>(l)</sup>     | N/A   | ~99% <sup>(g)</sup>   |       |                |
| T remaining at end of clean-up period (g) | 0.74               | 1.06  | 0.85   | —                                 | ~6 <sup>(c)</sup>   | N/A                  | N/A   |                       |       |                |
| T permitted in the vessel (g)             | 2                  |       |        | 20 (first wall) 11 <sup>(d)</sup> |                     |                      | 11    | <1000 <sup>(h)</sup>  |       |                |
| On site T inventory (g)                   | 5                  |       |        | N/A                               | 21                  | 69                   | 58    | <~4000 <sup>(h)</sup> |       |                |
| Fuel cycle                                | closed             |       |        | closed                            | batch               | batch                |       | closed                |       |                |
| Exhaust processing                        | batch              |       |        |                                   |                     |                      | batch | batch                 | batch | semicontinuous |
| Breeding blanket                          | none               |       |        |                                   |                     |                      |       |                       |       | Test modules   |

- a) This is an average value over the period of 1993–97, excluding dedicated tritium removal campaigns.
- b) This was the tritium inventory in all systems outside the active gas handling system (AGHS) (i.e. neutral beam injectors (NBI), torus), but individual analysis of batches of gas from the different subsystems indicate that the torus contributes >90% of the inventory.
- c) Some cleanup was also done in the middle of DTE1, to repair a small water leak in the fast shutter of the neutral beam. At that time 11.4 g of T2 had been introduced into the torus and about 4.6 g of tritium was retained on the walls. The wall load was reduced to 2.3 g in a four day period with ~120 pulses.
- d) During T and DTE2 campaigns, JET operations were restricted by a daily limit of 11 g (44 bar-l) of tritium on the inventory allowed in torus and cryo-panels (divertor and neutral beams), as set by the JET D-T safety case.
- e) In total, ~2 g / T were removed from JET PFCs during the clean-up sequence.

- f) The in-vessel tritium inventory after the DTE2 and T campaigns at this date remains unquantified. In the absence of post-mortem analysis of JET PFCs one can only provide an upper bound estimate of T retention in JET with Be/W wall, based on results from gas balance analysis of D fuel retention from earlier campaigns, which yielded a 2% fuel retention fraction. These results need to be compared with the large T retention measured in JET with C-wall after DTE1: initial T inventory of 40% of the injected amount of T of 35 g, decreasing to ~17% (~6g of T) after days of tokamak plasma operation in deuterium.
- g) Design requirement.

**Mitigation of disruptions and other off-normal transients** - The main characteristics and the resulting consequences of off-normal transients in ITER are summarised in Table 4.8.

*Tab. 4.8. Type, range of parameters chosen to design ITER components and major consequences of plasma transient events in the ITER design.*

| Event                            | Key Characteristics  | Major Consequence(s) and/or comments(s)  |
|----------------------------------|--|--|
| Major disruptions                | frequency: ~10%;<br>thermal energy $\approx 0.35$ GJ<br>thermal quench time: 1–10 ms;<br>magnetic energy $\approx 0.6$ GJ<br>current quench time: 50–1000 ms;<br>max. current decay rate: 500 MA / s | vaporisation (and also melting for metals) of divertor targets, and nearby surfaces;<br>during the thermal quench, 80–100% of the thermal energy ( $W_{th}$ ) is transported by conduction to divertor; $\leq 30\%$ by radiation to first wall or baffle;<br>during the ensuing current quench phase; most of the remaining magnetic energy, about 600 MJ (only a fraction of the poloidal field energy) is conducted to the wall and can lead to melting. initiates VDE and runaway conversion. |
| VDE <sup>(*)</sup>               | duration $\approx$ plasma quench;<br>direction = up or down<br>(depending on changes of plasma current and inductance);<br>halo current $\leq 40\%$ .  | follows each disruption;<br>slow current quench--> worst VDE;<br>part of the magnetic energy ( $W_{mag}$ ) is lost to the first wall;<br>produces in-vessel halo currents and forces on in-vessel components.  |
| Loss of equilibrium control VDEs | frequency: 1% of pulses<br>drift $\approx 1$ –5 s to wall contact<br>onset: first wall contact initiates H mode loss, $W_{th}$   | initiated by poloidal field control failure;<br>electromagnetic effects (halo currents/vertical forces);   |

| Event                          | Key Characteristics   | Major Consequence(s) and/or comments(s)  |
|--------------------------------|---|--|
|                                | loss, melting, rapid current quench or disruption   | major thermal effects on first wall;<br>affected first wall region: upper/inside or divertor entrance.   |
| Runaway current conversion     | knock-on avalanche<br>$E \approx 10\text{--}15 \text{ MeV}$ ,<br>$W_{th} \approx 60 \text{ MJ}$ (uncertain) | many uncertainties;<br>toroidal localisation depends on first wall alignment;<br>shutdown species will influence runaway electron current.   |
| Fast shutdown (active control) | $\tau_{th}, \tau_{mag} \leq 1 \text{ s}$ ;<br>by impurities and H/D injection.                              | fast plasma power and current shutdown means for thermal protection of the first wall and divertor targets;<br>by impurity and D injection, D favoured to minimise runaway conversion.   |
| Type I ELMs                    | frequency: 2–0.5 Hz,<br>$\Delta W_{th} / W_{th} \sim 2\text{--}6 \%$<br>deposition time: 0.1–1 ms;          | ~50–80 % of the energy lost from the core plasma is observed to strike the divertor target (e.g. in DIII-D, ASDEX-Upgrade);<br>JET, DIII-D, ASDEX-Upgrade reported inboard energies;<br>2–4 times that on outer divertor (see text for further details). |

(\*) During a VDE, the vertical plasma movement precedes the thermal quench; during a major disruption, the thermal quench precedes the vertical movement.

To offset the effects of thermal transients, the ITER design incorporates several specific technologies to control, mitigate or suppress the principal instabilities, i.e. neoclassical tearing modes (NTMs), edge-localised modes (ELMs), resistive wall modes (RWMs), vertical instabilities and major disruptions. It has, for example, been established for many years that NTMs can be stabilised by using localised Electron Cyclotron Current Drive (ECCD) to suppress the growth of the tearing mode island (e.g. [45], [46], [47]). The ITER Electron Cyclotron Heating (ECH) system includes four EC launchers located in the upper port plugs specifically for this application.

The need to enhance ITER’s control capability for other forms of MHD instability and to improve the reliability of operation has motivated significant R&D studies which have provided the physics basis for several design changes incorporated during the past decade. To ensure more robust control of the plasma vertical position, a pair of in-vessel coils, one located near the top of the

vacuum vessel and the other located on the lower half of the outboard vessel wall, is included in the design. These coils, coupled in a “saddle-coil” arrangement, enhances the vertical stabilisation capability by adding a second fast-response circuit to that provided by the external superconducting PF coils. This will allow a significantly more rapid response to fast changes in plasma vertical position and will ensure that vertical excursions of the plasma having a larger initial displacement, for example in response to fast changes in plasma  $\beta$  or internal inductance, can be controlled reliably [48], [49].

A second set of in-vessel coils has been incorporated to address the challenge of ELM control: the transient heat and particle fluxes associated with large amplitude ELMs in the ITER reference plasma scenario can lead to localised melting or significant erosion of the PFCs, in particular, of the divertor high heat flux components, reducing their operational lifetime (see for example [50]). It is, therefore, necessary to reduce the heat pulse caused by ELMs by more than an order of magnitude, or to fully suppress the ELMs, while retaining the high plasma performance characteristics of the H-mode. On the other hand, ELMs can also play a beneficial role in controlling the influx of tungsten sputtered from the divertor, helping to limit the tungsten concentration in the plasma core. Controlling the ELM instability-driven transport and intrinsic transport processes in the plasma edge pedestal to achieve the required benign conditions, presents a challenging problem.

Following initial experimental demonstrations of ELM suppression by magnetic perturbations in DIII-D [51], [52], a set of in-vessel ELM control coils was incorporated in the ITER design from 2007 to complement the pellet pace making capability previously foreseen, which exploits high frequency injection of cryogenic deuterium pellets to generate high frequency/low amplitude ELMs (see for example [11], [69]). This combination of techniques provides greater flexibility in addressing ELM control/suppression in ITER burning plasma operation. The ELM control coil array consists of 27 coils mounted on the outboard wall of the vacuum vessel in a toroidal/poloidal array of 9 – 9 – 3 coils. It is also foreseen that the power supplies for this coil array can be upgraded during the ITER Operations Phase to provide a capability for RWM control.

Disruption mitigation on the ITER scale must be highly reliable and efficient to ensure that the design lifetime of in-vessel components can be met. Even in the first phases of operation, the energies stored in the plasma are sufficient to cause the melting of plasma facing components.

Thermal energies above 25 MJ are expected to cause localised melting of divertor tiles during a disruption thermal quench and magnetic energies associated with currents above about 6 MA can produce localised melting of first wall components during unmitigated current quenches. In addition, the probability of generating runaway electrons increases with plasma current and at 15 MA a seed runaway population at the end of the thermal quench of the order of  $10^3$  electrons may already be critical. Indeed, on the basis of present knowledge, runaway electrons generated in ITER during a 15 MA disruption have the potential to deposit a total energy of up to hundreds of MJ on the first wall, the divertor dome and/or the outer divertor baffle.

Failure to suppress runaway electron generation, or achieving insufficient mitigation, will result in significant damage to plasma-facing components, including a risk of water leaks. The common concept of disruption mitigation is to dissipate the energy through line radiation on the timescale of the disruption. While there is confidence that this concept is in principle able to reduce conductive heat fluxes and electromagnetic loads, there are large uncertainties for other aspects of disruption mitigation. Most important will be to ensure that runaway electron formation is excluded when mitigating thermal and electro-magnetic loads.

Designing a disruption mitigation system (DMS) that fulfils this essential requirement requires much better understanding of the generation of runaway electron seed populations during the MHD driven thermal quench than exists at present (see Section. 6.6). It is expected that increasing the density over the entire plasma cross-section will be essential. This can only be achieved with sufficient understanding of the material penetration process and how this process can be optimised for all disruption scenarios once ITER operates.

Extensive theory and modelling R&D related to disruption mitigation is in progress, aiming at injecting enough material in the plasma to cool it, promote radiation and avoid formation of runaway electrons. Material injection could be done either by high-pressure gas valves (Massive Gas Injection (MGI)) or by Shattered Pellet Injection (SPI) [53]. After extensive simulation, R&D, and tests on existing tokamaks [54], [55] the SPI technique has been chosen for ITER, and relies on forming cryogenic pellets, accelerating them in a single-shot gas gun so they shatter on a surface close to the plasma. Ablation of the fragments in the plasma supplies large amount of material in a shorter time and can penetrate deeper in the plasma than in the case of gas injection.

Although SPI technology has been demonstrated on several fusion devices, the whole setup with ITER-size pellets has not been tested yet. It has proven to be challenging to produce hydrogen pellets of the required size and to launch them intact [56]. Also, reliability and ensuring exact arrival time to the plasma is of utmost importance to ensure a toroidally symmetric radiation pattern. ITER's DMS task force is charged with defining the design requirements and proposing the necessary developments [57]. At the time of writing, H<sub>2</sub> ITER size 28 mm diameter pellets were successfully manufactured at the Oak Ridge National Laboratory (ORNL) in the US, and fired with H<sub>2</sub> gas in the ITER SPI barrel design geometry. Other laboratories have similar results with a slightly different geometry.

## 4.6 Main ITER design challenges and the role of fusion enabling technologies

Considerable advances have been made during the last four decades in magnetic plasma confinement properties and fusion performance parameters, as well as in fusion enabling technologies developments. These include the technologies to confine the plasma (magnet coil sets, plasma facing components) and those which are used to manipulate the plasma parameters and their spatial and temporal profiles (plasma heating and current drive, and plasma fuelling systems). These technological advances led to important milestones in existing devices, including:

- record plasma temperatures (40 keV) and fusion power (>10 MW) through additional heating systems (i.e. neutral beam injection and radiofrequency heating) and tritium processing systems;
- high performance plasmas using pellet injection (plasma fuelling);
- H-mode as a result of plasma shaping, wall conditioning techniques and understanding of plasma wall interactions;
- production of low impurity concentrations containing plasmas through plasma-facing component (PFC) development and plasma wall conditioning techniques;
- demonstration of non-inductive current drive by RF heating and neutral beam injection;
- stabilisation of MHD modes via EC wave injection;
- sustained operation above the empirical density limit with pellet injection;
- disruption mitigation using fuelling technologies for rapid plasma quench;
- expansion of the  $\beta$  limit via the control of NTMs and RWMs;
- mitigation of energy losses due to ELMs;
- techniques for minimising the in-vessel tritium inventory.

Such advances are going to play a fundamental role in ITER.

The rest of this chapter will describe the main engineering and technology challenges associated with the underlying design features and technology choices adopted for some of ITER's systems and components, emphasising those where major gaps with existing devices remain due to foreseen harsher operating conditions not fully testable in present devices (see Section 3). The main plasma physics challenges related to burning plasma physics in ITER including energy confinement, MHD stability and control, plasma exhaust and impurity control, and energetic particle behaviour are described elsewhere (see [5], [58], [59]).

### 4.6.1 Magnets

ITER's magnets utilise superconductors (SC) which, compared to traditional technology, allow the power consumption to be reduced, even if SC coils must be cooled with supercritical helium (S-He) in the range of 4 K ( $-269\text{ }^{\circ}\text{C}$ ). Superconducting coils are generally manufactured using low-temperature superconductors, such as niobium-titanium (NbTi) for magnetic fields up to about 8 T or niobium-tin ( $\text{Nb}_3\text{Sn}$ ) for fields between about 8 and 16 T. In the manufacturing process of  $\text{Nb}_3\text{Sn}$  magnets a specific heat treatment (HT) is required to allow the formation of the SC material. Since  $\text{Nb}_3\text{Sn}$  becomes brittle after the HT and experiences degradation of its performances if subject to tensile strain, the heat treatment generally follows the winding phase of the manufacturing procedure. This is the so-called Wind & React (W&R) approach, which is used for ITER coils.

The ITER superconducting magnet system is composed of:

- 18 toroidal field coils using  $\text{Nb}_3\text{Sn}$  superconductors, operating at a maximum (conductor) current of 68 kA and providing a toroidal magnetic field on the plasma axis of up to 5.3 T, with a maximum field in the inboard length of 11.8 T;
- a central solenoid assembled from 6 modules also fabricated from  $\text{Nb}_3\text{Sn}$  superconductor, operating at up to 45 kA and producing a maximum field of 13 T;
- 6 poloidal field coils fabricated from NbTi superconductor, having diameters of up to 24 m, and producing maximum fields (at the conductor) of 6 T at operating currents of up to 45 kA;
- 18 error field correction coils, also fabricated from NbTi superconductor.

A full description of the design and fabrication challenge of these ITER magnets is given in Section 9.4.1.

The overall operational requirements of the magnets define the basic functionality of the tokamak, but leave open many design choices which also affect the design of the rest of the machine. The 18 TF coils determine the basic toroidal segmentation of the machine and were chosen to meet the requirements of access ports (both number and size) and the toroidal magnetic ripple at the plasma edge.

This low temperature superconducting (LTS) magnet system [60] forms a mission-critical core component of the ITER tokamak, allowing very long plasma burn durations ranging from several 100 s to several 1000 s [61], and represents the largest superconducting magnet system ever built, with 51 GJ of stored energy. These magnets are technologically highly advanced components, using composite  $\text{Nb}_3\text{Sn}$  with 4–6 K forced flow cooled conductors that stretch current manufacturing technology to its limits in order to maximise plasma performance and minimise cost. They work at the highest possible electrical (20–30 kV), mechanical (primary stresses up to 600 MPa) and superconducting performance consistent with safe and reliable operation over the life of the machine.

ITER selected Nb<sub>3</sub>Sn strands for the TF and CS coils in the first conceptual design phase, 1988–1991, setting performance parameters that acted as a driver for industrial development over the following 25 years. The ITER conductor used in the superconducting coils of the magnet system was ultimately chosen in 1993 as a rope-type cable-in-conduit conductor (CICC) made of Nb<sub>3</sub>Sn for the two high field systems (TF coils, CS coils) and of NbTi for the three lower field systems (poloidal field coil, correction coils (CC), feeder busbars) cooled by a forced flow of supercritical helium.

The cable is formed by multi-stage twisting of superconducting strands, with the final stage (except for the CC and busbar) consisting of 6 bundles twisted around a central cooling channel. The cable is enclosed in a stainless-steel pipe or jacket that is formed from an assembly of butt-welded seamless pipes. The role of the stainless steel jacket around the cable is therefore limited to supporting the local conductor forces and provide a containment for the forced flow helium. There are also asymmetric coils, namely the superconducting correction coils, with the role of correcting magnetic field errors [62], [63] potentially caused by deviations from target dimensions during manufacture and tolerances in assembly positioning.

Following qualification by model coil tests in 2001–2002 (see Section 7.3), conductor production for the TF and CS coils started in 2007 [64] and was completed in 2015 with an overall production of about 700 t of strand material [65]. For the PF, the total weight of the superconductor was 240 t and it was produced in about the same time. The difference is that while for Nb<sub>3</sub>Sn ITER needed ~10 times the normal worldwide production, for NbTi only 240 t was used - a small amount compared to the total production for the medical industry [66]. LTS magnet technology was developed at industrial level through ITER R&D and in support of large-scale projects, such as the Large Hadron Collider (LHC) at CERN. In addition, recent fusion devices have been built and are in operation using superconducting coils (see Section 5).

A new compact tokamak, named SPARC, is also under construction in the US using a novel magnet technology that relies on the use of 16 T high-temperature superconductors (HTS). HTS are presently used in the design of ITER current leads, which is a component of the feeders that transfer the large currents from room-temperature power supplies to very low-temperature superconducting coils at a minimal heat load to the cryogenic system. In ITER, the leads consist of a section of a bismuth strontium calcium copper oxide (BSCCO 2223) HTS and a conventional copper section connected through an electrical joint [67]. The joint and copper are cooled by helium gas at 50 K, the cold end of the BSCCO 2223 at 4.5 K.

The major design drivers for the ITER conductor design and the selection of Nb<sub>3</sub>Sn as superconducting material for TF and CS magnets were the high operating current of the magnet systems and correspondingly high magnetic field on the windings of magnets, to provide the magnetic flux to drive the plasma current inductively and a high field at the plasma axis to maximise confinement. Also, as ITER will produce several 100 s of MW of fusion power, reducing nuclear heating from neutrons deposited in the TF inboard leg to acceptable levels was an

important design consideration of the vacuum vessel and shield blanket. A central cooling channel made of open stainless-steel spiral provides improved cooling of the superconductor and the use of Nb<sub>3</sub>Sn provides a large temperature range capability. Both factors contribute to a heat extraction capability at 4 K of several 10 s of kW from the TF coils, allowing sufficient cooling in ITER during nuclear plasma operations as well as to compensate for inadequacies found in the nuclear shielding as designs matured to manufacturing reality.

A cross-section of a TF coil (left side), and the layout of the ITER TF conductor with details of a SC strand are shown in Fig. 4.4 (source: Ref. [68]).

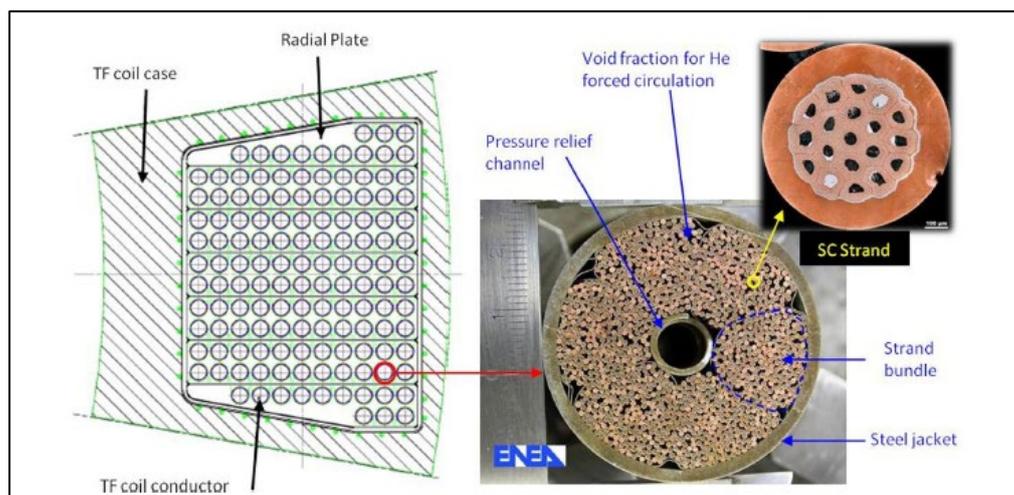


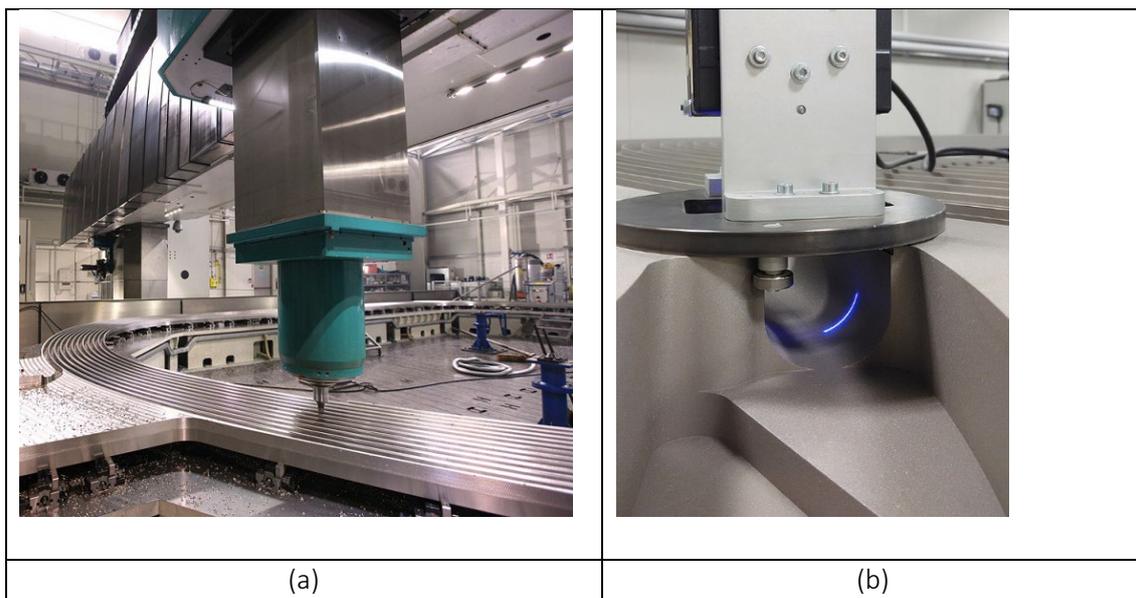
Fig. 4.4. Cross section of a TF coil (left side); layout of the ITER TF conductor with details of a SC strand (right side) [69].

A specific aspect of the magnet system for fusion reactors is the need for mechanical structures to sustain the enormous forces acting on the coils. For example, for ITER the centering force per TF coil is 403 MN and the vertical force on one half of a TF coil is 202 MN. The maximum vertical force per coil is 327 MN and 160 MN for the CS and PF magnets, respectively. For supporting the out-of-plane forces, in the inboard region the coil straight legs are wedged along their side walls, and the curved parts are linked by six pre-compression rings. To counteract the twisting moment of the TF coils, due to the out-of-plane forces from the controlling poloidal field coils, the outer legs are connected by the outer intercoil structures, which form toroidal shear belts around the magnets. The overall force balance in the TF set is maintained by the TF cases, including the central cylinder formed by the wedging, the inner pre-compression rings and the outer intercoil structures. The toroidal field coil assembly rests, via 18 gravity supports, on a tokamak pedestal ring supported by a concrete crown on the floor slab of the Tokamak Building.

Magnets are permanent components and must be designed to achieve maximum reliability and a fatigue life compatible with 30,000 full power tokamak pulses. Many design and manufacturing quality aspects have a bearing on magnet reliability, but experience with magnet operation indicates that the most common causes of magnet faults are insulation failures. A specific feature

of the ITER TF coil design is the use of thick austenitic SS radial plates to embed the insulated conductors in machined grooves [12]. ITER is the only fusion device to date that has adopted this solution because of the large size, and forces, with the aim to transfer the Lorentz forces acting on each conductor directly to the case. The radial plates prevent the accumulation of forces on the conductor and its turn insulation, reducing the probability of insulation degradation due to mechanical fatigue.

Figure 4.5 shows the fabrication of the radial plates that hold the conductor of the toroidal field coils. The radial plates are 8.5 m x 15 m D-shaped stainless-steel structures with grooves machined on both sides along a spiral trajectory. As shown in Fig. 4.4 each toroidal field coil is composed of seven double pancake modules which are composed of a radial plate within whose grooves the insulated conductor is embedded. Each coil contains five regular radial plates (with 12 grooves per side) and two side radial plates (with respectively nine and three grooves on the two sides) that were manufactured utilising seven stainless steel forged plates made of 316 LN and butt welded with local vacuum electron beam technology. Each section is fully machined to the final tolerances except the welded areas.



*Fig. 4.5. (a) Tool machining the grooves of a radial plate; (b) Measuring with extreme precision the grooves of the radial plate [159].*

A key driver of the ITER magnet design has been the need to generate and control plasmas with a relatively high elongation  $k_{95}$  of about 1.7 and a relatively high triangularity,  $\delta_{95}$  of 0.35 69. These plasma-shaping specifications are more demanding than those of the 1998 ITER design 70 and have made it necessary to adopt a design where the CS is vertically segmented into several electrically independent modules to provide non-uniform current distributions along the vertical axis. The six CS modules require electrical connections which can be located only in an annular gap between the TF coils and the CS. The consequences of the need for independent CS modules

and the gap for connections are that the CS must become a pancake-wound, free-standing coil subject to high cyclic tensile stresses.

The CS assembly includes the stack of six modules and a preload structure that compresses the CS magnet vertically during the assembly and the cool-down of the machine. The preload structure is made by a set of tie-plates located at the inner and outer diameters of the CS stack. The CS conductor has a very thick square jacket which is self-supporting against the radial and vertical magnetic loads. Sliding supports allowing radial displacements are attached to the TF coil cases to support the PF coils. The structural support of each PF magnet is guaranteed by the thick square jacket of the PF conductor

As a result of a Design Review undertaken in 2007–2008, two additional sets of in-vessel coil systems were added in the ITER design baseline [59], to provide fast and reliable vertical instability control for the growth rates predicted for the high plasma current, high elongation plasma scenarios. These coils are constructed from water-cooled, steel-jacketed copper conductors.

At the time of writing, the magnets and associated feeders, are entirely manufactured. All 19 ITER TF coils, including one spare, have been manufactured by Japan and Europe, and delivered to the ITER site. The last to be assembled Poloidal Field Coil, PF1 coil, procured by Russia, has also been delivered.

The slippage of the ITER machine assembly schedule has led the project management team to take the opportunity to build a Magnet Cold Test Bench (MCTB) on site to test some of the TF coils and the PF1 coil at their operational temperature (4 K) and at full current. In addition to the tests to be performed, the test bench integration and operation is an opportunity to gain experience with regards to ITER upcoming commissioning and operation phases.

The main objectives of the tests are to check the high voltage ground insulation at different temperatures, the different quench protection systems foreseen for operation and the coil performance up to nominal current (68 kA for TF and 48 kA for PF1). This will be an opportunity to test them in operating conditions. Tests of the instrumentation chains and control logics foreseen for ITER operation will also be performed. An ITER Cold Terminal Box will be connected to the coil for the helium and electrical supplies. The cryogenic system will provide enough refrigeration capacity with one out of the three ITER refrigeration cold boxes and will supply supercritical helium to the TF winding pack, casing and busbars at 4.75 K, 0.6 MPa with a nominal mass flow of 250 g/s. One important system to be tested will be the magnet protection system, including demonstration of quench detection capabilities.

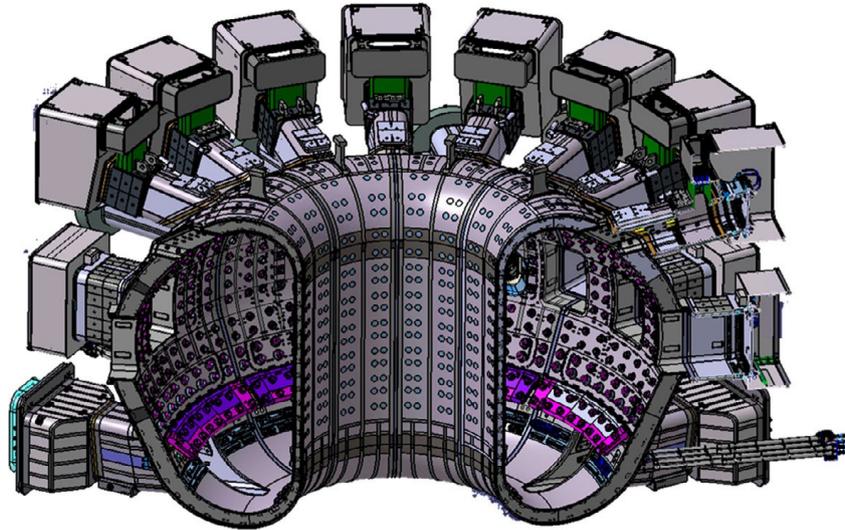
#### 4.6.2 Vacuum Vessel

In a tokamak, the vacuum vessel (VV) is a toroidal pressure-retaining vessel with a D-shaped cross section providing the primary plasma vacuum conditions, with ports allowing access to the

plasma. The VV contributes to the shielding of the surrounding coils. It is enclosed by the TF and PF coils, which, together with the intercoil structures, form a cage-like structure with interspaces used to route the VV ports. ASDEX Upgrade has a single-walled VV [71]. In JET the interspace of the double-walled VV is not filled with water, a neutron moderator, but is used as a secondary vacuum barrier and for gas baking [72]. In fusion devices with superconducting magnets, however, the VV must contribute to the shielding of neutrons to limit the neutron heating of the magnets operated at cryogenic temperatures.

To reduce the distance between the plasma and the TF conductor, and hence maximise the high field region accessible to the plasma, and at the same time still provide sufficient neutron shielding to the TF coils, the space occupied by the VV (and dedicated to reducing the neutron flux on the inboard side) must be minimised [73]. In ITER, the neutron flux must be further reduced, in particular on the outboard side, to avoid excessive activation of the components inside the cryostat in order to allow personnel access in exceptional situations requiring unplanned maintenance. Special attention had to be given to minimise radiation streaming through major penetrations.

For the VV to shield neutrons efficiently, a double-wall steel structure filled with liquid cooling water acting as neutron moderator, has proven to be most suitable (see Fig. 4.6). Additional plates of borated steel are stacked in the interspace to reach a favourable mixture of steel and water of about 60/40% by volume [74].



*Fig. 4.6. View of four sectors of the assembled ITER VV with upper, equatorial and lower ports and in-vessel component support structures (pink) on the inner shell. Support structures below the lower ports and port plugs not shown. The splice plates used to assemble the sectors and some of the ports are highlighted in dark grey. [160].*

The vacuum vessel in ITER acts as a primary safety barrier to prevent the release of radioactivity into the environment. Because of its safety important classification, a strict control of the manufacturing and inspections is needed, which has shown to be complex than expected [29]

To avoid high thermal stresses due to differential thermal expansion, high stress interactions and thermal conduction from the VV to the cryogenic coils [75], the VV is supported independently of the magnets. Moreover, in nuclear devices such as ITER, the safety-classified VV (see Section 6.9 in this Chapter) cannot be supported by the non-safety-classified magnet system. The VV supports the in-vessel components (IVCs) and hence the plasma-facing components, mainly the blanket/first wall (FW) and the divertor targets. The VV has stringent manufacturing and assembly requirements. Leading edges of the PFCs will be subject to very high heat loads due to charged particles escaping from the plasma confinement [76].

To avoid leading-edge overheating of PFCs, it is therefore necessary to align the PFCs to the static toroidal field with a precision that, for components of this size, is uncommonly tight. The ITER FW has a prescribed alignment tolerance of around 5–10 mm [77] and the ITER divertor targets of around 2mm [78]. The customisation of the individual IVC support structures can compensate for misalignments in the VV inner surface to some degree [78] Nonetheless, for the successful installation and alignment of the PFCs, small VV tolerances are required.

To enable plasma operation, the currents in the PF and CS coils are varied for three different purposes: (i) to provide toroidal magnetic flux to break down the hydrogen gas into a plasma and initiate the discharge, (ii) to drive the plasma toroidal current, and (iii) to actively control the plasma shape, radial and vertical positions. The VV is subject to the resultant magnetic field

variations as it is a toroidally continuous structure. To reduce its EM shielding effect during plasma breakdown, several existing tokamaks, including TFTR, JET, ASDEX Upgrade or JFT-2 M, integrate bellows or electrical breaks between the VV sectors, increasing the toroidal resistivity. Shunts can prevent arcing at the bellows and at electrical breaks between the VV sectors. But in the case of plasma vertical motion, the toroidal electrical conductivity of the VV causes currents to be induced that counteract the plasma movement, providing passive vertical stability [79]. Due to their flexibility, bellows between VV sectors prevent the VV acting as a ring structure that would be well suited to bear vertical and out-of-plane loads.

Operational experience with tokamaks having vertically elongated plasmas showed that the VV's role in the passive stabilisation of vertical plasma motion was an important factor in limiting the loss of vertical position control. In addition, VDEs associated with such loss of control produced significant forces on the VV and in-vessel structures. For example, the occurrence of VDEs required the JET VV to be reinforced [80]. With the increase of the plasma currents, e.g. toward ITER, the VV design approach incorporating bellows or electrical breaks was abandoned. Indeed, the use of bellows was unprecedented in nuclear containment vessels and the expected difficulties in obtaining a nuclear licence were also important considerations. Both shells of the ITER VV sectors are therefore fully welded at the sector field joints using splice plates [81]. Due to progress in operational and plasma control techniques for establishing plasma breakdown, the use of a toroidally continuous VV is no longer considered a major issue [82], [83].

Although substantial design and R&D efforts have been directed in the past to develop methods of manufacturing the vacuum vessel, including the realisation of full-scale mock ups (Chapter 7) and to refine techniques for welding and inspection, several critical issues have emerged with the fabrication of the vacuum vessel sectors in ITER. The limited prediction and control of the magnitude of the welding distortions, the dimensional accuracy and the achievable tolerances have led to several non-conformities that are now being addressed through intensive repair work. This is described in Ref. [29] and in Vol. 2.

### 4.6.3 Plasma Facing Components

Because of the long duration of each plasma pulse (several hundred seconds) ITER requires actively cooled plasma-facing components to cover almost all the internal surface of the vacuum vessel. Designing this interface between the thermonuclear plasma and the rest of the tokamak has been arguably one of the greatest technical challenges of ITER and will continue to be a challenge for the development of future fusion power reactors. The interaction between the plasma edge and the adjacent surfaces profoundly influences conditions in the core plasma and can damage the plasma-facing material and lead to long machine downtimes for repair.

The production of neutrons in the burning plasma will also lead to deposition of heat and radiation damage effects in structures relatively far away from the plasma e.g. the shielding blanket, the vacuum vessel, and the front ends of additional heating systems and diagnostics. Robust solutions

for issues of plasma power handling, plasma-wall-interactions, and shielding are required for the realisation of a commercially attractive fusion reactor.

The PFCs in ITER consist primarily of two main systems: the FW as the front component of the shielding blanket, and the divertor (see Fig. 4.6), with approximate plasma-facing areas of 610 and 140 m<sup>2</sup>, respectively. They are both mechanically attached to the VV, divertor and the FW panels through the blanket shield blocks, and they must remove up to 850 MW of thermal power [84].

The main drivers for the design and technology solution adopted for the divertor and the first wall are the very high surface heat fluxes, the cyclic nature of the heat fluxes, and the electromagnetic loads associated with disruptions and vertical displacement events (VDEs). The typical thermal loads to which the PFCs are subjected, especially in the divertor regions, have required an unprecedented R&D and engineering effort to develop suitable high-heat flux technologies, as well as to design efficient means of heat removal. The solution adopted for ITER relies on the use of copper-alloys as heat sink, and water as coolant. This requires active cooling via sub-cooled pressurised water at inlet pressure and temperature of 4 MPa and 70 °C, respectively, and the inclusion of turbulence promoters (helical tapes) or special configurations (hypervapotrons) in the coolant channels of the higher heat flux regions.

In the 90 s, the joining of carbon fibre composites (CFC) and tungsten to copper heat sinks largely relied on silver based brazing techniques. In a tokamak fusion environment, the silver would transmute to cadmium which is considered unacceptable in an ultra-high vacuum due to its low vapour pressure. A vigorous R&D programme was instigated that considered flat tile, saddle-block and monoblock armour geometries for the divertor (see Fig. 4.7 from Ref. [85]) and a range of brazing, diffusion bonding, HIPing and metal casting techniques for joining the armour to the heat sink. Hypervapotrons were used in JET as beam stopping elements in both neutral beam injectors and in the neutral beam test bed. The monoblock type design was initially developed for the CFC plasma facing material of the NET and ITER reactor designs [86], [87] and it was later adapted to tungsten armours. In this design, the armour surrounds the heat sink, yielding a particularly robust design, for a high number of high heat flux cycles [88], [89].

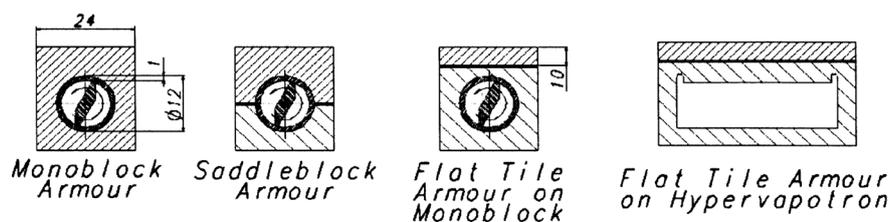


Fig. 4.7. Heat sink geometry divertor tiles [86]

In parallel to joining development, evaluation of the heat sink material concentrated on precipitation hardened CuCrZr and aluminium oxide dispersion strengthened Cu alloy, DS–Cu. However, because CuCrZr has slightly better tensile properties and significantly higher fracture toughness than the DS–Cu (ten times higher at 350 °C), it was selected as this latter property is important for a component that needs to withstand large impulsive electro-magnetic loads. To offset the risks that the manufacturing thermal cycles can degrade the strength and thermal conductivity of CuCrZr, a number of promising joining processes were developed keeping strict control of the manufacturing cycle and maintaining the mechanical properties of the CuCrZr. All the most successful methods rely on using a pure Cu interlayer between armour and heat sink. For tungsten, a hot radial pressing (HRP) manufacturing technique, based on the radial diffusion bonding principle performed between the cooling tube and the armour tile, was proposed [90] and proved to be very successful. The bonding is achieved by pressurising the cooling tube while the joining interface is kept at the vacuum and temperature conditions. This technique has been used for the manufacturing of relevant mock-ups of the ITER divertor vertical target.

To address the most critical issue of the reliability of the armour to heat sink joints, reliance was placed on testing the various design solutions in high heat flux (HHF) test beds. A vigorous qualification programme was launched involving HHF tests in a few relevant electron beam facilities equipped with water loops, an advanced beam scanning system and relevant diagnostics (IR camera, X-ray detector, calorimetry). Thanks to fast two-directional scanning of the narrow beam, steady-state surface heat fluxes typically up to 20 MW/m<sup>2</sup> were generated, allowing ITER divertor relevant surface heat fluxes to be simulated for thermal fatigue studies. These included for example: 1) ITER Divertor Test Facility (IDTF) [91] in the Efremov Institute, near St. Petersburg, 2) the JAERI Electron Beam Irradiation Stand (JEBIS) facility in Japan [92], and 3) the FE200 facility in France [93].

A key aspect for PFCs is the choice of plasma facing materials. For the first divertor, ITER originally considered CFCs for plasma facing materials while beryllium was, until recently, the primary choice for the first wall. Now tungsten will be used as armour material for both the divertor and the first wall. This topic is covered briefly below and will be discussed in more detail in Section. 9.1.

The major factor that has led to the progressive reduction of the use of carbon in ITER has been its affinity with hydrogen and the retention of large amount of tritium redeposited material (bulk retention also exists but is much smaller). This was very clearly confirmed by operational experience in existing tokamaks. In 2010, ITER decided to move to a full-W divertor. Tungsten has favourable characteristics for the divertor targets, which will be subject to the highest heat and particle fluxes: it has a high energy threshold for sputtering by plasma particles, the highest melting point of any metal, and an acceptably low affinity for hydrogen, implying a low rate of fuel retention. However, as noted above, it will be necessary to implement specific operation scenarios to avoid excessive production of W and limit the concentration of tungsten in the plasma core.

While the first wall is typically subject to much lower heat fluxes than the divertor, it is exposed to higher energy particles escaping from the core plasma.

During plasma transients associated with disruptions, the plasma current decays with a timescale of several tens of milliseconds. This generates eddy currents in the PFCs, which produce high electromagnetic loads by coupling to the toroidal field and generate forces which are several times higher than the dead weight of the components. This is an important factor affecting the design and material selection, and is one of the reasons why the first wall material has recently been changed from Be to W. The Be tiles (“fingers”) were separated by thin gaps, designed to electrically isolate the tiles when eddy currents occur. Tile gap bridging of beryllium-clad first wall panels induced by melt motion of Be at the top of the main chamber is likely to occur during an unmitigated upward-going disruption current quench at high plasma current. Besides the obvious lifetime implications for PFCs, the occurrence of just one of such a large-scale melting event has the potential to significantly increase the eddy currents induced by disruptions in the first wall panels in subsequent VDEs, even if these were well mitigated. This would be caused by the flow of molten Be filling the gaps between first wall fingers.

This is likely to happen since the typical molten erosion thickness with Be is in the few mm scale for 15 MA unmitigated VDEs or minor disruptions. This behaviour has been recently reproduced in experiments at JET. As a result, the larger eddy currents would then significantly increase the torque applied by electromagnetic forces on the FW panel fingers during follow-up disruptions (mitigated or unmitigated) with consequences for the lifetime of the affected FW panels. Recent modelling suggests that during high-current downward VDEs, the Be armour can also melt. In contrast, the W armour would not melt even up to the full plasma current of 15 MA.

An additional reason that has led to reconsidering the use of Be includes the fact that Be is a toxic material, adding complexity and mobility constraints on the already demanding accessibility of the welds during the first wall panel installation and early leak repairs. Avoiding the use of Be eliminates associated health risks, reduces the source term in case of accidents, reduces the accumulation of tritium in the VV, eliminates the need for an onsite Health Physics Be laboratory and related personnel, and reduces related engineering to manage confinement and controls. Finally, Be is not a relevant plasma-facing material for a fusion power plant due to its high erosion rate. Thus, the use of W for the first wall implies that it will not be necessary to replace all the panels at a later date to act as a test bed for future reactors with a fully relevant first wall material, as it would have been the case if Be had been used.

However, one of the positive aspects of Be as first wall material is that it provides good vacuum conditions by reducing the oxygen levels through the formation of very stable beryllium oxides (oxygen gettering). This is not provided by W and there are some operational risks (e.g., increased plasma radiation) that have to be addressed by relying on the in-situ deposition of boron layers by means of boronisation systems (more details can be found in [30]).

Both the blanket/first wall [94] and the divertor [95], [96] are designed and manufactured according to the highest quality standards, and are suitable for replacement by RH during the operational life of the ITER tokamak. Since these components operate in an ultra-high vacuum environment, they are subject to severe design, manufacturing, material and non-destructive testing requirements to minimise the risk of leaks. A sub-millimetric alignment between the plasma and the PFC surfaces is also required, which is challenging, since the vacuum vessel assembly is necessarily carried out under less demanding tolerances in view of its height (11.4 m) and outer diameter (19.4 m). This implies that, before installation, the mechanical attachments of the PFCs must be custom-machined once the survey of the full vacuum vessel has been completed and the location of the plasma magnetic axis has been determined.

The shielding blanket system consists of 440 blanket modules, each module consisting of a FW panel attached to a shield block itself attached to the vacuum vessel. The blanket system is one of the most technically challenging components of the ITER machine, not only having to accommodate high heat fluxes from the plasma and most of the nuclear heating, but also large electromagnetic loads during off-normal events and demanding interfaces with many key systems (including the vacuum vessel, blanket manifolds and in-vessel coils and diagnostics) and the plasma.

These have impacted the design choices, including the use of a shaped first wall to accommodate the plasma heat fluxes under different plasma scenarios, a straight inboard profile with an increased radial thickness to provide sufficient shielding for the TF coil, a shield block design with numerous cut-outs to accommodate the blanket manifold and in-vessel coil space reservation, with optimised slits to maintain an acceptable load transfer to the vacuum vessel, and an attachment system providing the right support to accommodate the high EM loads while providing well characterised paths to guide halo and induced currents to the VV. In addition, special blanket designs are in place around heating and diagnostic ports, to allow for the discontinuity of the ports' entry in the vacuum vessel.

Details of design and fabrication of the complete blanket system can be found in Vol. 2.

#### 4.6.4 Plasma heating and current drive

Tokamaks are intrinsically ohmically heated by the toroidal plasma current, however this heating mechanism is not sufficient, and additional heating systems relying on the injection of energetic neutrals or radio frequency waves are needed [97], [98], [99].

One of the most studied high power heating schemes is neutral beam injection, i.e. the injection of neutral hydrogen particles, which penetrate through the confining magnetic field, collide effectively with the plasma particles, get ionised and confined, and transfer their energy to the plasma. The penetration length for a beam in the plasma – before most of its power is deposited due to ionisation – depends essentially on the injection beam energy and plasma line integrated

density. The typical particle energies of present-day fusion devices are in the range between 50 keV and 130 keV mainly relying on positive ion neutral beam injection. However, owing to its size, ITER demands a significantly higher particle energy of 1 MeV for its neutral beam injection.

Neutral beam injection based on positive ion technology is unattractive because of the low neutralisation efficiency for hydrogen or hydrogen isotopes at high particle energies. Negative ions sources must be used because negative ions can be very efficiently produced at the required energy. However, development of a high-power NBI at 1 MeV for use in ITER is a very complex task that requires extensive R&D, which is currently on-going. Owing to a variety of plasma resonances with high frequency electromagnetic radiation and several absorption mechanisms, high frequency wave heating of plasmas was always very attractive. One has to distinguish between Radio Frequency (RF) waves (in ITER  $\sim$  40–50 MHz, ion cyclotron frequency) and microwaves (in ITER 170 GHz, electron cyclotron frequency). Strong coupling of electromagnetic waves to the plasma particles can be achieved at the cyclotron resonances of ions and electrons or at a coupled resonance of both for RF plasma heating by ion cyclotron resonance heating (ICRH), and electron cyclotron resonance heating and current drive (EC H&CD). In recent years the launching of high-power microwaves at the electron cyclotron frequency into toroidally confined plasmas has gained increased interest because of several advantageous physical properties: the localised EC heating of the electrons, the possibility of tailoring the current profile, effective current drive, suppressing MHD instabilities and studying plasma transport by means of heat pulse propagation.

Additionally, a lower-hybrid resonance heating (LHRH) system was never considered to be a day one baseline H&CD system, but a possible upgrade after achieving the  $Q=10$  objective, specifically to support the 5000 s  $Q=5$  plasmas for the later phase of operation by method of current drive. However, important technical issues have emerged affecting the feasibility of the lower hybrid system and its specific application to ITER plasmas. One was the failure in successfully developing 5GHz klystrons, the frequency required for ITER. The second issue was related to LH wave injection from the low field side of the Tokamak. Experiments in C-Mod showed that this injection scheme leads to parasitic edge absorption at high density, absolutely not desired for ITER applications. While existing Tokamaks, DIII-D in particular, showed that this issue is avoided by high-field side injection, this solution was not practically implementable in ITER due to the limited access availability to high-field side.

Around 2017, modelling efforts intensified and it was concluded that the ITER goal of a fully non-inductive  $Q=5$  plasma could be achieved without LH (but increased ECH and NBI) with an expectation on plasma confinement ( $H_{98}\sim 1.5$ ) for 10 MA / 5.3 T plasmas. These results led to the decision to remove the LH from the baseline in 2019.

Due to the remaining physics uncertainties in some areas, ITER changed the heating mix as risk mitigation in 2024. The previous (2016) baseline envisioned a total injected power capability of 73 MW. This was provided through 20 MW of ECH (upgradable to 40 MW), 20 MW of ICH

(upgradable to 40 MW) and 33 MW of NBI (upgradable to 50 MW). The potential upgrades were part of the options which could be decided upon during the operation phase, based on project needs and heating system efficiency. It should be noted that these values correspond to powers injected into the vacuum vessel and are expected to couple to the plasma in the absence of plasma-related losses (e.g. inefficient ICH coupling, large NBI shine-through or partial absorption of ECH). As described in Ref. [29], the increased heating power levels are required to support plasma scenario development to high current with low fluence (either D or D with low T content plasmas) by supplementing the absent (or low) alpha heating (which requires high DT reactivity and thus neutron flux). The higher available heating power also allows for substantial radiative power fractions in the core plasma (up to 50%) which are frequently found in present experiments with a W wall when H-mode scenarios are first developed (e.g., before optimisation). An overview of the specifications of the ITER H&CD systems is the ITER 2024 Baseline is given in table 4.9.

Tab. 4.9. Specification and performance of the ITER H&CD systems.

| ITER design specifications |  |  |   |
|----------------------------|--|--|---|
|                            | Sources  | Objectives   | Achieved (experiments and test beds)  |
| EC                         | 40–67 MW / CW<br>170 GHz                                   | Heating & Current Drive<br>MHD control<br>Plasma breakdown and start-up (3 MW)             | On plasmas:<br>0.11 MW / 54 min<br>3 MW for 2 s in experiments<br>On test beds: 170GHz gyrotron: 1 MW, 800 s, 55% efficiency in Japan   |
| IC                         | 10–20 MW / CW<br>35–65 MHz                                 | Localised ion heating<br>Central CD<br>Sawtooth control<br>Ion cyclotron wall conditioning | Experimental results:<br>L-mode plasmas: 22 MW / 3 s,<br>4.3 MW / 1min, 0.38 MW / 54 min<br>H-mode plasmas: 16.5 MW / 3 s<br>1.6 GJ of IC energy (stellarator)  |
| H-NB                       | 2(3)×16.5 MW<br>1 MeV, 3600 s<br>200 A/m <sup>2</sup> in D | Heating, central and off-axis CD, rotation   | On test beds:<br>>300 A/m <sup>2</sup> in H-, >250 A/m <sup>2</sup> in D-.<br>>830 keV H-, >730 keV D-.<br>Ion source operated for 1 h<br>Experiments (one injector):<br>~6.2 MW 418 keV for 1.6 s<br>~6 MW, 187 keV H <sub>0</sub> for 2 s |

Concerning EC, good progress has been achieved on 1MW gyrotrons capable of very long pulse operation [100] and on ambitious designs which could perhaps reach both higher power and long pulses [101], [102]. There has also been progress on the design of quasi optical launchers [103], [104] capable of versatile functions ranging from central heating to NTMs instability suppression. Issues are in the domain of reliability and margins for long pulse full power operation of gyrotrons, as well as the control of stray radiation arising from quasi-optical transmission in the launchers.

On the IC side, coupling transients due to plasma edge perturbations such as ELMs, pellet injection or sawtooth can now be mitigated using passive circuits as demonstrated on several experiments [105]. However, as effectiveness of IC heating in a W machine has some residual uncertainties based on the RF generated impurities, it has been decided to not rely on the availability of 20 MW of power from this system for SRO. Instead, it is proposed to install an IC system of 10 MW power for SRO which provides ion cyclotron wall conditioning (ICWC),

With regard to H-NB, powerful negative ion beam systems are now used in major experiments [106]. In the 1990s there has been impressive progress in negative ion sources and accelerators with the construction of multi-megawatt negative-ion-based NBI systems at LHD (H0, 180 keV) and JT-60U (D0, 500 keV).

ITER, currently relies on two heating beams capable each of delivering 16.5 MW for 3600 s. The accelerator is specified [107] at 1 MeV for 40 A of D ion corresponding to a current density of 200 A/m<sup>2</sup>. Despite considerable progress in this field, such voltages and current density have never been realised simultaneously and major R&D is still required to demonstrate 1 MeV operation at the required ITER power level and pulse length. Recently, the specifications have been reduced to around 840 keV and the use of hydrogen instead of deuterium beams. A large neutral beam test facility has been built and is operating in Padua, Italy [108] to perform integrated developments on a prototype beam line and to check the performance and the reliability of all the key design choices.

It consists of two prototypes: SPIDER (an ITER-scale negative ion source designed to achieve all ion source requirements) and MITICA (a full-size 1 MeV heating neutral beam injector, capable of full acceleration voltage and power). In addition, two additional test facilities at IPP Garching have been supporting the development of the radio-frequency-driven ion source for ITER neutral beam injection for many years as part of the European roadmap: BATMAN Upgrade, a prototype ion source, and ELISE, an ion source one half the size of ITER's [109], [110].

#### 4.6.5 Plasma Diagnostics

Measurements in fusion plasmas utilise a wide variety of diagnostic techniques. Some methods are passive and use the detection of particles or radiation emitted spontaneously from the plasma, while other methods use active probing of the plasma with beams of particles or electromagnetic radiation from an external source. Diagnostics based on electromagnetic radiation are particularly important; fusion plasmas emit electromagnetic radiation over a very wide spectral range, from high energy gamma rays through hard and soft X-rays, and ultra-violet, visible, infra-red through to microwave radiation. Each wavelength range conveys information about specific aspects of the plasma, and coverage of the whole range is necessary.

While the physics of the operation of the diagnostics, as established on past and present machines is, in many cases, directly applicable to ITER, the technology and engineering of the

implementation of the systems differ substantially and involve many difficult challenges. The challenges arise from several different aspects, especially the unavoidable location of many of the diagnostic components inside the vacuum vessel and in the ports where they are subject to high levels of radiation and heating. This means that many phenomena new to diagnostic design can occur and must be considered.

The main challenges of diagnostics arise from operation aspects of ITER [111], [112]. The nuclear environment sets stringent demands on the engineering of the diagnostic systems, for example, the diagnostic systems must meet the strict ITER requirements for the containment of tritium and vacuum integrity; they must be able to withstand high pressures that can potentially occur under accident scenarios, activation must be minimised, and it must be possible to maintain and replace, with remote handling equipment, those components that are activated. The combination of the new physical phenomena, the demanding measurement requirements, especially for long pulse operation, and the engineering requirements for systems installed in and supporting the operation of a nuclear facility, lead to many challenges in diagnostic design.

The diagnostics can be broadly broken up into a number of categories as shown for example in Table 4.11, magnetic, optical, bolometric, neutron and particle, spectroscopic and edge systems. The plasma control system (PCS) will utilise measurements provided by a significant fraction of these systems. A typical list of diagnostics and their purpose is shown in table 4.10.

Tab. 4.10. A typical list of diagnostics and their purpose.

| Type     | Parameter  | Diagnostic  |
|----------|--|---|
| Magnetic | <ul style="list-style-type: none"> <li>- main plasma position</li> <li>- divertor plasma position</li> <li>- component (halo) currents</li> <li>- error fields</li> </ul>  | <ul style="list-style-type: none"> <li>- vessel wall inductive sensors</li> <li>- divertor magnetic coils</li> <li>- Rogowski coils</li> <li>- diamagnetic loops</li> </ul>   |
| Neutron  | <ul style="list-style-type: none"> <li>- spatial fusion power profile</li> <li>- temporal and positional flux</li> <li>- shielding effectiveness</li> <li>- reaction rates</li> <li>- cumulative results</li> <li>- plasma energy leakage by fast ions</li> <li>- confined alpha particle density profile</li> </ul> | <ul style="list-style-type: none"> <li>- neutron cameras</li> <li>- micro-fission chambers (in-vessel)</li> <li>- neutron flux monitors (ex-vessel)</li> <li>- gamma-ray spectrometer</li> <li>- activation system</li> <li>- lost alpha detectors</li> <li>- knock-on tail neutron spectrometer</li> </ul> |

|                                    |  |  |
|------------------------------------|--|--|
| Optical and infrared               | <ul style="list-style-type: none"> <li>- electron temperature and density profiles</li> <li>- electron density</li> <li>- ion temperature and density profiles</li> </ul>  | <ul style="list-style-type: none"> <li>- Thomson scattering</li> <li>- toroidal interferometer/polarimeter</li> <li>- collective scattering system</li> </ul>  |
| Thermal Radiation                  | <ul style="list-style-type: none"> <li>- plasma radiation spatial and temporal profiles</li> </ul>   | <ul style="list-style-type: none"> <li>- bolometer arrays</li> </ul>   |
| Spectroscopy and neutral particles | <ul style="list-style-type: none"> <li>- velocity and spatial distribution of fast ions</li> <li>- impurity profile (<math>Z &lt; 5</math>)</li> <li>- integral impurities</li> <li>- plasma rotation, ion temperature</li> <li>- ion temperature, plasma rotation</li> <li>- current density and electric field profiles</li> <li>- impurity concentration</li> <li>- ion temperature profile</li> <li>- impurity spatial profiles</li> </ul> | <ul style="list-style-type: none"> <li>- H-alpha spectroscopy</li> <li>- visual continuum array</li> <li>- main plasma and divertor impurity monitors</li> <li>- x-ray crystal spectrometers</li> <li>- charge exchange recombination spectroscopy</li> <li>- motional Stark effect</li> <li>- soft x-ray array</li> <li>- neutral particle analysers</li> <li>- laser-induced fluorescence</li> </ul> |
| Microwaves                         | <ul style="list-style-type: none"> <li>- MHD mode detection</li> <li>- electron density profile</li> <li>- plasma position control</li> <li>- divertor erosion</li> <li>- electron density temperature product</li> <li>- plasma density profile</li> <li>- ion density</li> </ul>   | <ul style="list-style-type: none"> <li>- electron cyclotron emissions</li> <li>- main plasma reflectometer</li> <li>- plasma position reflectometer</li> <li>- divertor reflectometer</li> <li>- divertor electron cyclotron absorption</li> <li>- main plasma microwave scattering</li> <li>- fast wave reflectometry</li> </ul>  |
| Plasma Facing Components           | <ul style="list-style-type: none"> <li>- fast ion energy deposition, wall temperature</li> <li>- hardware temperatures</li> <li>- edge plasma density</li> <li>- chamber and divertor gas analysis</li> <li>- strike point detection</li> <li>- electron density and temperature at probe</li> </ul>   | <ul style="list-style-type: none"> <li>- infra-red visible cameras</li> <li>- thermocouples</li> <li>- pressure gauges</li> <li>- residual gas analysers</li> <li>- infra-red thermography</li> <li>- Langmuir probes</li> </ul>   |

One of the main problems is the integration of the diagnostics into the tokamak, which requires a high level of coordination with all the other parts of the machine design process. Diagnostics are installed both in-vessel and ex-vessel, and the transmission systems follow paths from the vacuum vessel all the way to the diagnostic building [113]. Diagnostic system components are installed in several upper, equatorial and lower ports. Access to diagnostic systems in the region of the tokamak area is generally quite limited and, in some cases, will be almost impossible after initial assembly. This is a key driver in the design of the systems and gives rise to a requirement for redundancy or very high reliability or both. Typically to access light from the plasma while minimising activation outside of the chamber, it is necessary to use mirrors and a labyrinth. This puts some critical mirror surfaces in direct line of sight of the plasma and as a result, energetic particles from the plasma can add a layer of unspecified material to the mirror surface, thus degrading the performance of the system. In some cases, a certain amount of this can be tolerated, but specific mirror cleaning techniques are being developed.

The integrity of the confinement boundaries is another critical requirement in ITER. For diagnostics, the boundaries are typically defined by port plugs, feed-throughs, and window assemblies. For port plugs, controlling neutron leakage and minimising deflections due to large electromagnetic loads is a particular challenge while for windows, a strong boundary is required.

Table 4.11 shows indicative values of neutron flux and fluence expected for different diagnostic systems in different parts of the machine.

*Tab. 4.11. Indicative values of neutron flux and fluence for different systems in different parts of the machine [112].*

| Diagnostic   | Location        | n-flux (n/m <sup>2</sup> s) | n-fluence (n/m <sup>2</sup> ) |
|--|-----------------|-----------------------------|-------------------------------|
| Neutron activation system head, microwave antennas                           | Near first wall | 0.15–3x10 <sup>18</sup>     | 0.25–5.1x10 <sup>25</sup>     |
| Inner vessel magnetics, Bolometry, wiring, waveguides, microfission chambers | VV inner skin   | 0.2–1.5 x10 <sup>17</sup>   | 0.38–2.54 x10 <sup>24</sup>   |
| Outer vessel magnetics, steady-state sensors                                 | VV outer skin   | 0.005–0.77x10 <sup>15</sup> | 0.009–1.3x10 <sup>12</sup>    |

While radiation effects in the magnetics diagnostics and first mirror degradation in optical systems are the principal technical risks that potentially challenge multiple diagnostics, and hence multiple plasma measurements, there are other risks that have to be handled in diagnostic design. For example, neutron heating and radiation damage in bolometers [21] magnetic field effects in microfission chambers, and spurious mode generation due to bends in microwave waveguides. These risks, and the avoidance and mitigating methods that are being adopted, are assessed as part of the conceptual design review process for the individual diagnostics concerned.

## 4.6.6 Tritium Systems

### *Tritium Plant*

ITER is designed for operation with equimolar deuterium–tritium mixtures [114]. The tokamak vessel will be fuelled through gas puffing and pellet injection, and the neutral beam heating system will introduce deuterium into the machine. Employing deuterium and tritium as fusion fuel in addition to generating a large amount of thermonuclear power will have important consequences. Due to the small burn-up fraction in the vacuum vessel a closed mixed deuterium–tritium loop is required, along with all the auxiliary systems necessary for the safe handling of tritium. The ITER inner fuel cycle systems are designed to process considerable and unprecedented deuterium–tritium flow rates with high flexibility and reliability. High decontamination factors for effluent and release streams and low tritium inventories in all systems are needed to minimise chronic and accidental emissions. A multiple barrier concept is needed to assure the confinement of tritium within its respective processing components; atmosphere and vent detritiation systems are essential elements in this concept.

The tritium plant is indispensable for the operation of the ITER tokamak beyond the initial hydrogen phase, as tritium will be produced from DD fusion reactions. However, the fuelling and heating systems need to be fed with gases from day one of machine plasma operation, which requires certain parts of the tritium plant to be available even before the deuterium phase [115].

Experience with the relevant technologies exists, but not at the scale and with the processing demands required by ITER, and an extensive R&D programme is underway to finalise the design of the ITER system [116].

A key technology and safety challenge for fusion reactors is the quantity of tritium fuel in the tritium plant (2–3 kg) and the rate at which this tritium can be processed (maximum 200 Pa·m<sup>3</sup>/s) while at the same time minimising tritium release to the environment during operation and under accident conditions. The majority (~80%) of this tritium resides in the fuel processing plant's cryogenic isotope separation system (ISS) (~60%) and on the reactor's vacuum vessel and cryopumps (20%).

ITER represents a tremendous leap in tritium processing compared to current technologies due to its higher inventory and processing flow rates [117], [118], [119]. The tritium supplied per pulse in TFTR and JET was small, roughly equivalent to a fraction of a gram per hour. In the case of ITER, the numbers are at least 3 orders of magnitude higher.

The main functions of the tritium systems of ITER can be summarised as follows:

- handling of incoming and outgoing tritium shipments;
- storage of tritium and deuterium, and transfers to and from the fuel cycle;

- management of airborne contaminants;
- extraction of decay  $^3\text{He}$ ;
- delivery of deuterium, tritium and mixtures for fuelling, supply of divertor impurity seeding gases and measurement and determination of tritium inventories;
- torus vacuum pumping and transfer of gases to tritium processing systems;
- processing of tokamak exhaust gas, of tritiated gas streams generated during tritium recovery from plasma-facing components, and of other tritiated off-gases for recycling of tritium and deuterium along with separation of hydrogen into specific isotopic species for refuelling at specified flow rates and isotopic compositions;
- detritiation of water and recovery of the tritium;
- extraction of tritium from test blanket modules (TBMs) and tritium recovery;
- atmosphere detritiation during normal, maintenance and accidental conditions and decontamination of gases prior to controlled release into the environment.

Considering the list of functional requirements, the complexity of the ITER fuel cycle systems, and taking the criteria for safe handling of tritium into account, it becomes obvious that design integration of the ITER tritium plant systems will be one of the major tasks during design and construction.

The tritium plant, essentially a nuclear gas processing plant, receives deuterium–tritium (DT) gases from the torus and neutral beam vacuum systems, removes impurities, separates isotopes, and delivers  $\text{D}_2$ , DT and  $\text{T}_2$  to the tokamak fuelling systems. The tritium plant also includes equipment for detritiating gases and water. Because only a small fraction of the injected fuel is burned on any given pass through the plasma, the exhaust contains a mixture of all the hydrogen isotopes (H, D, T) fed into the plasma. In general, this mixture must be isotopically separated to: 1) remove protium (H) that enters the plasma exhaust from outgassing or leaks; and 2) separate the D and T into products suitable for reinjection. In general, the tritium inventory in the isotope separation system (ISS) will increase with the amount of tritium separation required, which in turn increases with tritium throughput and with the degree of tritium enrichment.

An important size driver of the ISS is quantity of protium present in the plasma exhaust. The ITER ISS is designed for complete processing of the plasma exhaust due to the intermittent operating schedule which is likely, especially in the early phases of operation and which would tend to elevate background levels of protium. If the protium outgassing rate decreases in future power reactors (e.g. due to fewer maintenance openings), then continuous processing of the full plasma exhaust stream may not be necessary. Instead, only some side stream would need to be processed, and the ISS could be reduced in size and tritium inventory. ITER will be a good test of the amount of protium to be expected because of its reactor-relevant size and much higher operating capacity than previous fusion machines.

Experiments carried out on ITER will clarify the usefulness of isotopically enriched fuel (as distinct from 50:50 DT mixed fuel) and may result in a reduction of the ISS requirements for a reactor. Going from ITER to power reactors, another important difference is the higher-than-average tritium throughput due to the expected near-continuous plasma operation. However, ITER has been designed to handle the plasma exhaust on a steady-state basis. In principle, the separation duty for ITER could have been reduced by sizing the ISS for the time-averaged flow. This approach does not necessarily reduce overall system-wide tritium inventory, but keeps the amount in the ISS itself lower, in line with the goal of segregating tritium inventory. This is not possible with a power reactor.

For the separation process itself, in the throughput range relevant for ITER and power reactors, the most effective process for isotope separation of hydrogen isotopes is cryogenic distillation. The only real disadvantage of distillation is that a large fraction of the hydrogen (and T) is in liquid form, and the process therefore has relatively large inventories. Gas phase continuous membrane columns have been proposed for the protium stripping section of the isotope separation system [120]. This process offers potential for inventory reduction and greater stability of operation under fluctuating feed conditions, but further development is necessary to adapt the process to higher feed rates, and a large pumping system is needed. Recent advances (1980s and 1990s) in hydrogen distillation [121] have resulted in designs that are much more compact with lower inventories than in the past. In large part, this has been through careful understanding of the process and the packings, so that the systems can be optimised to run at lower inventories than were previously practical. The design tools that have supported this improvement are well-advanced, but there are still areas of experimental research where further improvements may be possible.

#### *Fuelling and shattered pellet injection for disruption mitigation*

The ITER fuelling system mainly consists of the pellet injection system (PIS), gas injection system (GIS) and disruption mitigation system (DMS).

Pellet injection from the inner wall is planned for use in ITER as the primary core fuelling system since gas fuelling is expected to be highly inefficient in fuelling the centre of burning plasmas. Tests of the inner wall guide tube have shown that 5 mm pellets with speeds up to 300 m/s can survive intact and provide the necessary core fuelling rate. Modelling and extrapolation of the inner wall pellet injection experiments from the present day's smaller tokamaks leads to the prediction that this method will provide efficient core fuelling beyond the pedestal region. Using pellets for triggering frequent small edge localised modes is an attractive additional benefit that the pellet injection system can provide. A description of the ITER pellet injection system's capabilities for fuelling and ELM triggering, and performance expectations and fusion power control aspects, are discussed in Volume 2.

Shooting pellets down a precise path is another challenge. Because the tubes that direct the pellets into the tokamak will be unusually narrow at ITER, the flying pellets can deviate no more than one degree from an arrow-straight path. The bigger ITER pellets will require more force to send them into the tokamak. Applying that force too fast, however, will break the pellet. Another engineering challenge awaits at the other end of the flight path. By design, the pellets slam into a curved section of the tube wall at 250 m/s, breaking into pieces. Those fragments will disperse across the plasma, cooling it down as the plasma expends energy stripping electrons off the cold atoms. The ORNL team is now investigating how pellet speed, pellet material and tube angle affect the all-important fragment distribution.

First-of-a-kind achievements have already emerged from the US ITER and ORNL team, including the largest cryogenic frozen hydrogen pellet ever formed and fired by a pellet injector "gun" (25 mm in diameter) and a novel three-barrel pellet injector for delivering the pellets to the plasma. The pellet guns use a gas propellant to fire pellets at 300 m/s or about 1080 km/h.

### *Vacuum pumping*

The ITER vacuum pumping system is a first-of-a-kind in size and complexity, comprising (in addition to the major cryopumps) at least 300 mechanical pumps, as well as 10km of vacuum lines. Orders-of-magnitude improvements in vacuum reliability are required compared to existing and past fusion devices [122].

The vacuum systems for ITER are characterised by the requirements for tritium compatibility, tolerance of high magnetic and radiation fields and remote maintainability. In addition, although the vacuum levels are relatively modest, high pumping speeds are needed to achieve the high gas throughputs required. While the ITER torus base pressures are like JET, the ITER pumping duties are significantly more demanding because the torus volume is approximately 10 times larger and ITER plasma shots will be up to 300–500 s (short pulse operation) and up to 3000 s in long pulse mode, compared with a few seconds for JET. The tritium throughput during the main DT campaign (DTE) in 1997 in JET was roughly equivalent to one day of planned ITER DT operation.

The main pumping systems are the six torus exhaust pumps, the four cryopumps for the neutral beam injection systems used in plasma heating, and the two cryopumps for the ITER cryostat to maintain the low pressure required for the operation of the superconducting magnets. In addition, there is a mixture of standard and custom pumps in the roughing pump system (RPS), diagnostic, transmission line, and service vacuum system (SVS). The complex pumps have been tailored for the very specific applications and requirements at ITER. All are based on cryopanel, cooled with supercritical helium and coated with activated charcoal as sorbent material. Research and development have shown that charcoal from finely ground coconut shells has the right density and porosity for imprisoning the helium particles in ITER. The outgassing rates from plasma-facing components and the gas species evolved are strongly dependent on the selection of plasma-facing materials.

It is uncertain if cryopumps will prove to be an effective VV pumping option for a steady-state fusion reactor like DEMO. This uncertainty relates to possible reliability concerns for cryopumps given their transient mode of operation, i.e. cycled fuel loading and unloading modes. A solution called the “direct internal recycling” (DIR) approach has been proposed that has the potential for reducing the DEMO tritium processing plant size to that of ITER’s, or 75% smaller [123]. A key technology proposed for the DIR approach is called a “superpermeable” metal foil pump (MFP) [124], [125], [126]. The MFP is a steady state, high temperature vacuum pump that works by directly extracting the unburnt hydrogen fuels from the plasma exhaust, instead of condensing them. Because this extracted fuel is free from plasma exhaust impurities, it can be sent directly to the reactor’s fuelling system for reinjection into the plasma instead of to the fuel processing plant.

#### 4.6.7 Remote Handling

The main experience of remote handling in fusion facilities comes from the JET device. JET has routinely used teleoperated robots, controlled by human operators from the safety of a control room. The remote maintenance system in JET [127], [128] has been the most developed and comprehensive of its type, having accumulated over 30,000 hours in operation [129]. It used two snake-like booms to enter the vacuum vessel which moved around the vessel controlled remotely by operators, carrying other subsystems and payloads such as the MASCOT tele-manipulator. The booms were reconfigurable to adapt to a wide range of missions. A similar deployment principle is used for the Articulated Inspection Arm, used in the EAST and WEST devices [130]. Although limited to 10kg payloads, it can operate under vacuum conditions.

The ITER remote handling system involves a significant extrapolation from JET experience [131]. For instance, the dose rate inside of JET has typically been a few mGy/h, compared with that anticipated in ITER of several hundred Gy/h, a factor of about two orders of magnitude higher. A dedicated hot cell with specialised remote equipment is required to process components leaving the reactor [132]. In future systems it is expected that an entire active maintenance facility will be required to process the highly activated components, with many large areas being out of bounds for human access [133]. This expanding boundary implies that the number of maintenance activities that must be performed remotely will increase dramatically. Similarly, with respect to the weight of components requiring remote maintenance, the JET system manipulated weights of about 50 kg, compared to several tens of tonnes expected in ITER, two orders of magnitude higher. These differences have required the identification and development of new RH approaches and RH tooling solutions.

In-vessel maintenance of ITER also involves the regular occasional of plasma-facing components with cooling interfaces, and therefore requires the cutting and joining, of cooling pipes [134], [135]. Most of these pipe joints are at locations under high radioactivity but low accessibility and visibility. No direct human intervention is possible inside the vacuum vessel and other peripheral areas of the reactor [136]. In addition, joints are subjected to elevated temperature and cyclic

thermal loads, and leak tightness is key to protecting the tokamak vacuum during operation [137]. Therefore, specifically developed tools with safe and reliable remote maintainability are required to be deployed through entry ports and carry out maintenance tasks in space-constrained environments.

Pipe maintenance related topics is a critical area of R&D and there are many technical challenges and unresolved issues. Relevant tooling and systems for such processes have also been under development and a review of pipe cutting, welding, and NDE technologies for use in fusion device is provided in Ref. [138].

Numerous cutting and re-welding jobs on the primary circuits of a nuclear power plant (NPP) have been undertaken over the years. However, they are primarily one-off repairs which were not necessarily foreseen when the plant was designed. Examples of this are nozzle replacements or repair to weld-roots which are found to be corroded during routine inspections. In the case of ITER however, for unavoidable reasons, several tens of in-vessel primary cooling circuit cutting and re-welding operations might have to be carried out to maintain its basic operation in case of corrective maintenance.

For the divertor alone there are 108 cooling pipes to cut, re-weld and inspect for every divertor exchange- so this activity could be carried many more times compared to a few times in the life of an NPP. For the blanket there are 400+ connections to make under fully remote conditions even during first installation.

The RH System of ITER is described in detail in Vol. 2 and it involves several systems:

- blanket RH system used to replace individual first wall panels and blanket shield blocks.
- divertor RH system used for the replacement of divertor cassettes. Cassettes are retrieved by an initial toroidal translation, sliding on toroidal rails, and are then withdrawn from the vacuum vessel in the radial direction via three lower ports;
- cask and plug RH system; large transfer casks carry the strongly activated and contaminated in-vessel components from the tokamak complex to the hot cell complex (HCC). These casks provide confinement (for dust and tritium), but not radiation shielding. Consequently, they are fully robotic, typically transferring equipment through the tokamak complex and HCC at night, which necessitates the evacuation of personnel from these facilities during the transfer.
- NBI RH system used to maintain neutral beam components in the neutral beam cell.

In ITER, remote operations, maintenance equipment, and reactor components have been designed all together, as all associated constraints must be considered in an integrated way to achieve a successful design satisfying the lifetime operational requirements [139]. As a result, the verification and validation of RH solutions, which requires physical mock-up testing, have been done since the early phase of the design (see Section 7.3) to show that they work as intended in

a representative setup. The ITER Divertor Test Platform 2 facility provides an example in this regard. This has allowed extensive testing to ensure that the design concept for the divertor RH system is feasible [140], given that it must be able to operate in a highly constrained space and in the presence of gamma radiation.

The RH approach implemented in ITER together with the detailed description of the RH systems and tools is detailed in Vol. 2. As an example, the reference design of the divertor RH system [141] (see Fig. 4.8) consists of two movers. The Cassette Multifunctional Mover transports the 10 t divertor cassettes radially to and from the vessel through the maintenance port. Several interfacing end-effectors also allow it to transport a range of other hardware and RH equipment. The Cassette Toroidal Mover locates the cassettes inside the vessel, moving them along toroidal rails. A variety of tools are used to cut cooling pipes, lock the cassettes in place, and collect contaminated dust from the work area. These tools are deployed with a dexterous, teleoperated manipulator arm, mounted on the movers.

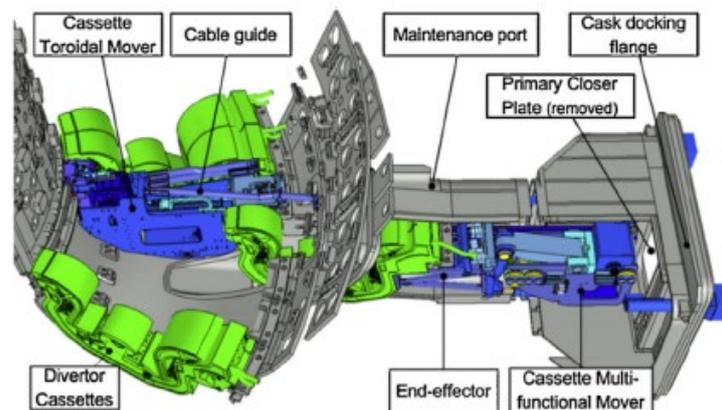


Fig. 4.8. Overview of ITER's divertor RH system [142].

For the maintenance of the blanket first wall and shield blocks (up to about 1m x 1.4m x 0.5m, 4.5t in the latter case), ITER is considering the use of an In-Vessel Transporter (IVT), which is composed of a manipulator mounted on a monorail-type vehicle and an articulated rail [142], [143]. During the EDA, the JAEA fabricated a prototype of the IVT system [144] (see Chapter 7) and confirmed its feasibility including reduction method of vibration induced by blanket load [145], automatic positioning of the blanket [146] and rail deployment procedure of the articulated rail [147]. However, some risks remain, and R&D is still in progress to confirm the feasibility.

In the 2016 baseline, the principal device for installing blanket shield blocks and first wall panels was the in-vessel tower crane (IVTC). During the development of the 2024 baseline, several studies were conducted to identify ways to speed up the in-vessel installation processes and have more overlap or parallelisation of the installation of the different in-vessel components. One of the possibilities identified was to parallelise the installation of the blanket and divertor systems.

However, since the IVTC operates on rails installed in the lower part of the vacuum vessel (normally occupied by the divertor), it cannot be used when the divertor cassettes are installed, so it was clear that an additional tool would be needed to allow the installation of the blanket and divertor to proceed in parallel. Hence, it was decided to proceed with the design and procurement of two blanket assembly transporters (BATs) which can be deployed and operated fully independently from the IVTC. BATs are long-reach, serial robotic devices deployed in a cantilevered fashion from equatorial ports (with no need to rely on the rails in lower part of the vacuum vessel). During the pre-SRO assembly phase, they will be used to install a certain number of lighter shield blocks (<3 t) and the temporary first wall panels. BATs are configured to deploy specific end-effectors (e.g., for bolting and torquing), and for the welding of hydraulic connectors (should the opportunity to proceed with this scope during the first assembly phase be taken). These end-effectors will be provided by Japan.

Fusion reactors following ITER will face further challenges, related to the additional requirements as designs approach commercial deployment. Higher fusion power will lead to increased activation and to higher radiation levels during maintenance, which will be sufficient to damage electronic systems that are not properly shielded. Deployment of robotic systems inside the vessel, while still possible for ITER, will be unfeasible for a DEMOnstration Fusion Power Plant (DEMO) and future power plants unless adequate radiation-resistant technology is developed [148]. Another key challenge for future fusion reactors is their larger size, which means very large components need to be handled with high precision. For instance, the largest components in EU DEMO could be 10 m tall, have masses in the order of several tens of tonnes, and require a positioning accuracy in the order of 20 mm [149].

Inside fusion power plants, a number of maintenance tasks will have to be performed remotely. This, in combination with the increasing size and mass of reactor components and the need to achieve commercially relevant plant availability, would mean that future remote maintenance systems will need to be faster, more capable and more automated. Future fusion plants will then need to evolve highly integrated designs where the maintenance strategy is considered from the outset and built into the very architecture of the plant. The handling strategy for such large components is a fundamental driver for the architecture of the entire plant.

To reach commercially relevant levels of plant availability, the speed of maintenance systems and the ability to perform tasks in parallel must improve significantly. It will be necessary to automate many maintenance activities, leveraging the productivity improvements already demonstrated in manufacturing industries worldwide. However, deployment of automated and autonomous systems with reduced human supervision in such harsh environments is a significant technology challenge, particularly considering the need for acceptance by regulators.

#### 4.6.8 Test Blanket Modules

Since the very beginning, one of the key ITER objectives was to test reactor-relevant blanket modules and associated subsystems up to a given neutron fluence to demonstrate the capability to reach tritium self-sufficiency and also to produce and extract high grade heat for electricity generation in a reactor.

Prior to the start of the ITER CDA, there was a debate as to whether to consider reactor-relevant technology developments for DEMO separately from physics research. This was advocated in 1985 and later [150], [151]. Specifically, plasma-based Volumetric Neutron Source (VNS) designs based on mirrors, reversed field pinches and tokamaks were identified and proposed to address the fusion nuclear technology R&D aspects [152]. These device concepts were small with major radius of about 2–3 meters, low power (<100 MW), low tritium consumption, and with designs emphasising maximum access to the fusion core.

The decision was taken not to have a major dedicated technology testing facility in Europe and other regions, but to use ITER for both physics and technology developments, including extensive nuclear component testing relevant to a future fusion reactor. A problem with this approach is that physics testing requires relatively large fusion power (e.g., 300–600 MW in ITER) to have performing plasmas, whereas nuclear technology testing only requires low fusion power (~ 20–50 MW) but needs high n-fluence (>10 dpa) [153]. The combination of high power for physics testing and high fluence for nuclear technology testing in a single device leads to high tritium consumption (for example >110 kg per full power year in DEMO). Such large amounts of tritium can only be provided if the device has its own self-sufficient tritium breeding blanket that operates reliably from the very beginning of operation.

ITER operation relies on utilising available external supplies of tritium that are very limited [154]. Consequently, achieving tritium self-sufficiency in the so-called breeding blanket will be an unescapable requirement for any next-step fusion nuclear facility beyond ITER. Developing a breeding blanket that meets the required performance and reliability goals is essential for the deployment of future fusion reactors.

Breeding blanket systems have been under development since the start of civil fusion investigation in the early 1950s [155]. Many concepts were investigated, ranging from more conservative concepts to advanced and riskier concepts for future reactors. The major candidate breeding materials consist of lithium-based liquid breeders, mainly liquid metals, and lithium ceramic breeders. The degree of conservatism in the concept is often linked with the choice of structural material since more advanced concepts generally require operation at high temperatures to provide for high cycle efficiency and power production performance and, thus, a greater degree of extrapolation in structural material properties and technology. The choice of structural material, in turn, influences the choice of breeding material based on the accommodation of key issues such as material compatibility and temperature limits.

Self-sufficiency in tritium production was never an objective of ITER nor its precursors (INTOR/NET). In this respect, mock-ups of breeding blanket elements, either in the form of module or poloidal segment tests were considered adequate to allow reliable extrapolations. However, initially a tritium breeding blanket was introduced in the INTOR and NET basic machine designs in order to reduce the demands on tritium supply and thus the operating cost. For this breeding blanket, the goal of full reactor relevance was replaced by the condition not to interfere with the reliable operation of the basic machine. To achieve this objective and to maximise the shielding capability on the inboard of the torus, the breeding blanket was restricted to the outboard and upper sections of the plasma chamber, covering about 50–60%, of the available space.

In the case of NET, the adopted blanket maintenance approach (i.e. removal of the banana-like segments through the upper vessel ports) required a strong full steel structure in the inboard segments to increase the stiffness of the segments during removal thus avoiding a dangerous flexibility and oscillations. In the early design phase, in 1995 ITER followed the same strategy by creating the TBM Program where several concepts of breeding blanket were proposed for installation in a second phase of operation aiming at extensive nuclear testing to achieve the production of a significant amount of tritium and significant levels of accumulated n-fluence.

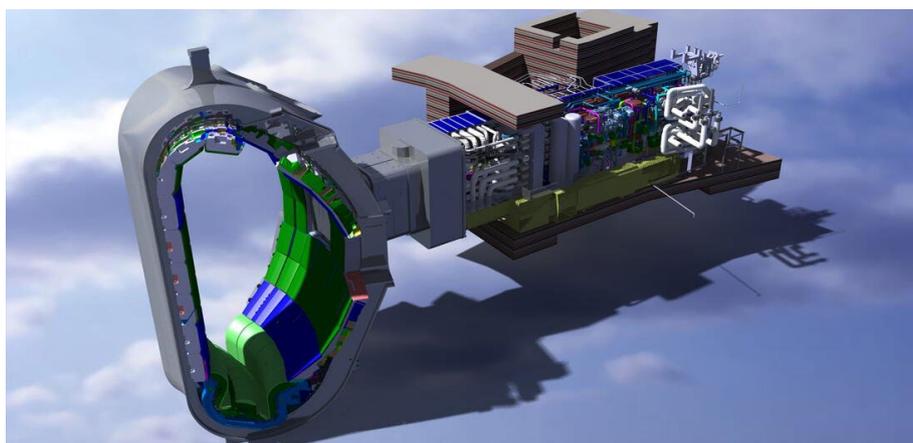
For the initial phase of operation (Basic Performance Phase (BPP)), focused primarily on achieving satisfactory plasma physics performance, a shielding blanket was considered for the inner coverage of the machine. During this phase, partial testing and qualification of breeding blanket modules and associated subsystems (e.g., high-temperature, high-pressure coolant system, tritium extraction system) to be deployed in equatorial ports was considered. The blanket shield system was designed to offer the possibility of replacing the shield by a breeding blanket within the same dimensional, maintenance, and coolant constraints, to provide the tritium to meet the technical objectives of the more technology oriented nuclear technology phase (Enhanced Performance Phase (EPP)). Provisions were also made to install and test more reactor-relevant tritium breeding blanket in specially designated equatorial ports.

However, after thorough technical assessments the strategy to replace the shielding blanket with a breeding blanket was found to be unfeasible or at least very risky. For example, a specific problem was related to the choice of the coolant conditions, that in the case of ITER, was water at relatively low temperature and low pressure. Tritium release and recovery from the breeding blankets is a critical issue that is very much temperature dependent. This is particularly the case for ceramic breeders in which tritium transport is strongly dependent on the temperature of operation of the breeder and for which, below a certain temperature threshold, tritium release would be inhibited resulting in very high tritium inventory levels.

Installing and operating a breeding blanket cooled with low-temperature water would make the process of tritium release very slow and the goal of maintaining acceptable tritium inventory levels very challenging.

Special design provisions, relying on thermal barriers to maintain the breeder region at sufficiently high temperature levels were not considered to be sufficiently reliable. Calculations for ITER show that the blanket, its primary/intermediate cooling circuit and tritium recovery system requirements added significantly to plant unavailability. Replacing these items with a breeding blanket would reduce machine availability. Also, the choice of a low temperature low pressure driver blanket for ITER with the aim to provide reliable and safe operation while maximising tritium breeding would not be reactor relevant. Performance results on such a blanket could not be extrapolated to DEMO which would utilise a high temperature blanket. However, the use of a high temperature blanket on ITER was also not a viable alternative, since it would introduce more complexity and risk, as it would be more likely to fail and be potentially more damaging when it fails, thereby significantly impacting the availability of the whole machine. Also, the level of maturity of such blanket concepts was very low and their development and validation would be riskier and require substantial time and resources.

These considerations led to a substantial curtailing of the nuclear mission of ITER and the implementation of a strategy that relies on utilising a shielding blanket in ITER and testing breeding blanket modules and associated subsystems (so-called Test Blanket System or TBS) in equatorial ports (i.e., the TBM Program mentioned above). Even more so, when the wall loading and neutron fluence objectives of ITER were reduced for the final design. Extrapolation to DEMO component conditions for materials irradiation data now relies on dedicated materials irradiation facilities, such as IFMIF/DONES156 in Europe. Chapter 10 in this volume, whilst some of the options being considered for minimising the risks remaining with the performance and the reliability of the breeding blanket that represents undoubtedly one of the largest gaps to be filled beyond ITER can be found in volume 2. The design work for ITER's test blanket modules and associated subsystems is being conducted in the Parties, according to boundary conditions established by the ITER project.



*Fig. 4.9. Test blanket module and its auxiliary systems in one of the two port cells dedicated to testing tritium breeding [161].*

Tritium breeding blanket testing still represents a critical element of the ITER mission, as much can be learnt despite the reduced performance. The ITER TBM Program represents the principal strategy by which ITER will provide the first experimental data on the potential of fusion as an energy source. TBS tests in ITER will provide essential information toward resolving this challenge. The major testing objectives are:

1. validation of theoretical predictions of structural integrity and response under combined relevant thermal, mechanical and electromagnetic loads;
2. validation of tritium breeding predictions;
3. validation of tritium recovery process efficiency and tritium inventories in blanket materials;
4. validation of thermal predictions for heterogeneous breeding blanket concepts with spatially dependent volumetric heat sources;
5. demonstration and understanding of the integral performance of the blanket systems.

Until recently, many ITER Parties viewed the TBS testing in ITER as their only component-level testing step before a DEMO reactor, deferring the materials testing to DEMO. New ideas are now emerging and are described in Chapter 10. Originally three equatorial ports were allocated to test six different types of TBMs. In November 2018, the ITER Council endorsed the proposal of an optimisation of the TBSs configuration based on the use of only two equatorial ports. Such an optimisation required the reduction of the number of simultaneously operating TBSs to a maximum of four, of which the in-vessel part, the TBMs, is fitted into the two ports, two TBMs per port.

All ITER Members contribute to the TBM Program, and a testing strategy has been developed for approximately the first 10 years of ITER operation. Tests will be carried out on mock-ups not only of the tritium breeding blankets themselves, but of complete systems (the TBSs), each of which consists of a TBM together with several subsystems providing coolant to the TBM, a dedicated tritium extraction system, a coolant purification system, a tritium accountancy system and associated instrumentation and control (I&C) systems. All the four TBSs are independently controlled and operated from the centralised ITER Control Room. A schematic view of a TBS and its locations in the ITER Tokamak Complex can be seen in Fig. 4.10 that shows the example of the helium-cooled ceramic breeder TBS.

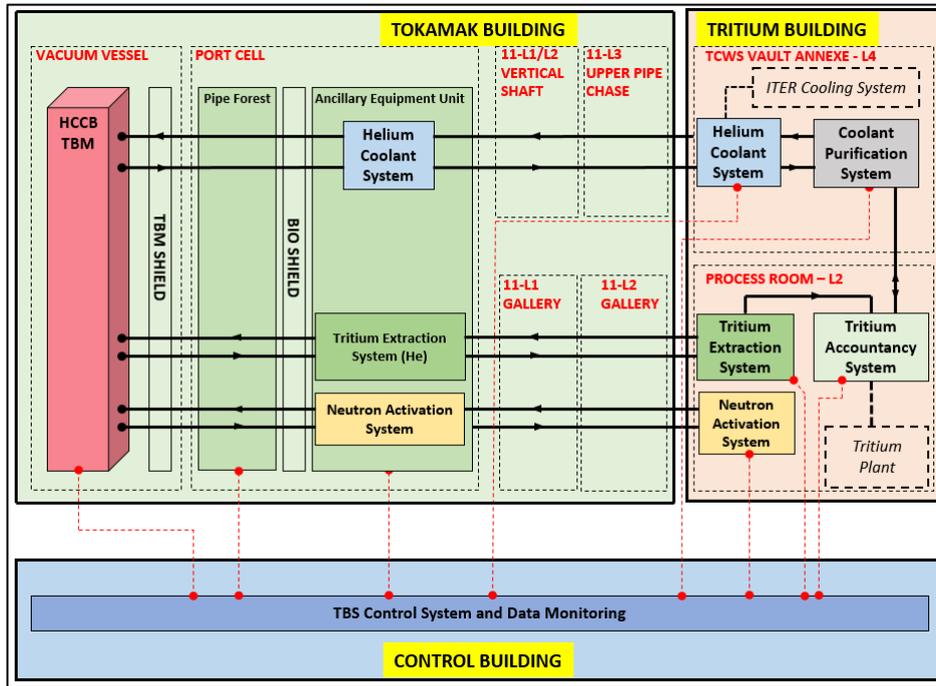


Fig. 4.10. Schematic view of a TBS and its locations in the ITER Tokamak Complex (example of the helium-cooled ceramic breeder TBS) [166].

The TBSs' functional characteristics are dictated by the operational conditions and requirements expected in DEMO and, in this sense, they differ from other major ITER components that are designed specifically in compliance with requirements derived from ITER's target performance. However, the TBSs must be fully integrated into the ITER facility and therefore they must be compatible with the systems and operational procedures of ITER, as well as with the ITER operating plan. Moreover, it is required that operation of the TBSs should not impact ITER's operational performance, safety, and availability. More details on the ITER TBM Program can be found elsewhere (see for example [157], [158] and in volume 2).

The main characteristic of the four TBMs to be tested in ITER based on information available today are shown in Table 4.11. The four TBMs are planned to be installed before the beginning of the DT-1 operation phase and it is assumed that they will be operated throughout. TBSs operations are expected to be continued during the DT-2 operation phase either for the same TBMs or for more advanced TBSs.

Tab. 4.11. Main characteristics of the four TBMs.

| Water cooled TBMs   | Helium cooled TBMs   |
|---|--|
| <p>Water-cooled lithium lead (WCLL) developed and procured by Europe</p> <p>TBM structure made by EUROFER97 steel, Pb<sub>16</sub>Li (multiplier/ breeder), 90% <sup>6</sup>Li enrichment</p> <p>Coolant: H<sub>2</sub>O at 15.5 MPa, 295/328 °C inlet/outlet temperatures</p> <p>T-purge gas: Helium +0.1 H<sub>2</sub> at 0.4 MPa</p>                                 | <p>Helium-cooled ceramic pebble (HCCP), jointly developed and procured by Korea and Europe</p> <p>TBM structure made by EUROFER97 steel, Be pebble (multiplier), Li<sub>4</sub>SiO<sub>4</sub> or Li<sub>2</sub>TiO<sub>3</sub> pebbles (breeder) 30-60% <sup>6</sup>Li enrichment</p> <p>Coolant: He at 8 MPa, 300/500 °C inlet/outlet temperatures</p> <p>T- purge gas: Helium +0.1 H<sub>2</sub> at 0.3 MPa</p> |
| <p>Water-cooled ceramic breeder (WCCB) developed and procured by Japan</p> <p>TBM structure made by F82H steel, Be pebbles (multiplier), Li<sub>2</sub>TiO<sub>3</sub> pebbles (breeder) 60% <sup>6</sup>Li enrichment</p> <p>Coolant: H<sub>2</sub>O at 15.5 Pa, 280/325 °C inlet/outlet temperatures</p> <p>T- purge gas: Helium &lt;0.1 H<sub>2</sub> at 0.4 MPa</p> | <p>Helium-cooled ceramic breeder (HCCB), developed and procured by China</p> <p>TBM structure made by CLF-1 or CLAM steel, Be pebbles (multiplier), Li<sub>4</sub>SiO<sub>4</sub> pebbles (breeder) 90% <sup>6</sup>Li enrichment</p> <p>Coolant: He at 8 MPa, 300/500 °C inlet/ outlet temperatures</p> <p>T- purge gas: Helium +0.1 H<sub>2</sub> at 0.3 MPa</p>   |

The entirely novel aspect of the TBM Program at ITER is underlined by that fact that a complete TBS with all the ancillary sub-systems has never been operated, not even at laboratory scale. A fortiori, the ITER testing program for the TBS will be the first time that a TBS will have been operated in the tokamak environment. Therefore, most of the TBS components must be considered as technological innovation and, as such, they require demonstration and validation. The testing of TBSs in ITER is viewed as an essential step to reduce the technical risks and uncertainties associated with the demonstration of power extraction, tritium breeding technologies and tritium management essential for a DEMO fusion power plant. Large modelling development activities are also part of the TBM Program for achieving reliable data extrapolation to DEMO breeding blankets. The licensing, operation, and maintenance of the TBSs are deemed to be valuable in providing information and support for licensing a blanket that breeds tritium. However, the TBMs will operate only up to low n-fluence (< 1dpa) and mainly at low duty factors

(with the exception of some dedicated days). Thus, large gaps will remain even with a successfully completed TBM Program. Risk mitigation strategies to accelerate the deployment of fusion power, relying on the availability of qualified breeding blanket systems, are being discussed between the ITER members and are described in Chapter 10.

#### 4.6.9 Integrated System Plant Design and Safety

A power-producing tokamak reactor, like ITER, is a highly complex device embodying the results of innumerable assumptions and decisions. The ITER design process has clearly shown that one cannot choose parameters in isolation. The plasma design conditions affect the engineering design, and vice versa. Even among the engineering design conditions themselves, a constraint arising from a system can affect another system. For example, even if high field superconductors could be applied to increase the fusion power density and reduce the plasma size, the dimensions of the core structure might not become smaller because of the increase of the coil structure size to overcome the higher forces and stresses that roughly scales like  $B^2$ . In addition, even if a high field and high current density superconductor could be applied, the plasma-facing components could not handle the resultant high wall load conditions, unless new high heat flux materials are developed. Therefore, there exists some optimum magnetic field satisfying these combined conditions for known reliable technologies.

The design of ITER (and in general of any other fusion reactor) was affected by a high degree of complexity/system interdependencies and multiple design drivers across various systems that impacted the design and performance. There were several design choices that had to be made and pervasively affected the overall design layout and the performance of the nuclear fusion plant and its maintainability and safety, because of the interfaces with all key nuclear systems. These challenges had to be addressed by the scientists and engineers responsible for designing and developing robust and reliable solutions, enabling safe machine operation. Such an integrated approach is essential to the successful development of the knowledge base for attractive fusion energy sources because of the complex nature of fusion systems and the multi-disciplinary aspects of the underlying science and engineering. This approach includes the identification and the subsequent evaluation and impartial assessment of multiple design options through parallel investigations for systems and/or technologies with high technical risk or novelty.

For example, robust structural concepts were needed to enable the set of TF coils (18 in ITER) to withstand the centring and vertical forces. Several concepts have been investigated in the past including full wedging, or partial wedging, and bucking against a CS or bucking post. However, all concepts relying on bucking are very complicated in practice, as they require the TF coil to slide against the bucking cylinder or the CS coil to accommodate motion during operation. Very complex structural designs are needed to support out-of-plane loads on the TF coils without transferring this twist to the CS. For the bucked and wedged configuration, detailed analyses showed that the stress on the TF leg is extremely sensitive to the assembly tolerance between TF

coils and CS (or whatever bearing surface is placed in between). Sub-millimetre tolerances are to be achieved over the large contact surface for the stress distribution to be predictable, which is impractical. After in-depth assessments conducted during the design phase, ITER opted for full wedging of the TF coils.

The integrated systems must operate reliably and safely in a nuclear environment, implying that the facility had to be designed to withstand the harsh conditions arising from a powerful thermonuclear plasma and to meet the stringent nuclear safety requirements characteristic of modern regulatory frameworks. Radiation shielding and nuclear safety have played a major role in the design of the ITER device from the early days, not only to protect the most sensitive equipment from the effects of neutrons, but also to protect workers and public from radiation exposure, and to minimise activation and radiation damage to peripheral equipment. This required, for example, the adoption of thick shielding in areas where the space is limited, i.e. the inner side of the torus, to protect sensitive equipment essential to the functioning of the reactors and special shielding for major penetrations to minimise radiation streaming.

ITER has definitely raised awareness of the importance of the integration aspects in the design process and that formidable integration challenges exist, made even more challenging by the nuclear environment, which is subject to specific regulations.

Therefore, a thorough examination of system design and integration aspects was essential from the early phase of the design to ensure that the integrated view of the plant is maintained from the very beginning and all factors affected by the numerous design choices to be made are identified, evaluated, and properly weighted. Implementation of this approach provided an opportunity for overall design convergence and reduction of integration risk. Not everything went perfectly, but the approach was correct. One of the lessons of ITER emerging from its long design and R&D phases is that integration of technologies into a coherent design is a central challenge for fusion and cannot be postponed in the hope of unleashing innovation.

At the time of writing, there are many discussions in various ITER Member states to develop a specific regulatory framework for fusion energy that maintains appropriate protections for people and the environment, proportionate to the hazards of fusion energy. Due to their fundamental differences, fusion energy must not be regulated like nuclear fission. It is unlikely that these future improvements will have an impact on ITER, but they would be beneficial for the development of fusion energy and eventual commercialisation.

ITER was designed, and is being constructed, with a high level of safety as an essential requirement. It is a main goal of ITER to demonstrate the safety and environmental potential of fusion and thereby provide a good precedent for the safety of future fusion power plants. However, it is necessary to account for the experimental nature of the ITER facility, the related design and material choices, and operational requirements.

In ITER, the principal source of radiation is the fusing D and T plasma which will produce neutrons of 14 MeV during plasma operations. The interaction of these particles with matter will give rise to secondary sources of gamma radiation. During the maintenance of the ITER machine, when the plasma is off, the gamma radiation will be the result of the decay of activated materials. Dust will also be generated inside the ITER VV due to interaction with the PFCs. This is an area where there are still large gaps, and research is conducted to narrow down existing uncertainties. The unique feature of tritium contamination is the very high mobility of tritium. It will readily be absorbed into many materials and then subsequently outgassed giving rise to chronic airborne contamination. Tritium contamination can readily spread from one area or object to another. It can diffuse through many materials resulting in contamination of the bulk as well as on the surface.

The confinement of radioactive and hazardous material and the limitation of exposure to ionising radiation are the two most important of the fundamental safety functions. This is achieved by implementing a succession of physical barriers and proper shielding provisions in the design. Multiple provisions are made so that, in accordance with a defence-in-depth approach, the failure of one barrier does not result in a release to the environment or to rooms in which personnel could be exposed. Ventilation systems maintain a pressure cascade between rooms so that air flow is always towards the more contaminated zone.

Provision of the confinement function by passive barriers is preferred over active systems, and the number of barriers, their leak tightness, and reliability is specified to achieve the required performance for the safety function. Where active components such as isolation valves are required, multiple components are sometimes required to achieve the required overall reliability. Effluents are filtered and detritiated in such a way that their release to the environment is as low as reasonably achievable.

A strong safety design is required to minimise the frequency of plant failures that could initiate an accident sequence, and to eliminate or reduce the potential consequences of all off-normal situations. Extensive safety studies of ITER have been performed over a long period as the project has progressed through its design stages. The safety analyses culminated in the preparation of the preliminary safety report (RPrS) submitted to the French nuclear safety authorities in March 2010 and the following licensing process has emphasised the importance of certain topics.

Codes and Standards- ITER is the first fusion facility which is under construction in accordance with safety and licensing requirements for a Basic Nuclear Installation in France. To fulfil these requirements a coherent set of Codes and Standards (C&S) 159 for various components is identified to provide rules for the design and manufacture of components within nuclear facilities such as pressure vessels, piping, pressure retaining portions of structures as well as pumps, valves, heat transfer systems, and support structures and confinement structures. These rules also contain requirements for quality assurance, materials specifications, design, fabrication, testing, examination, inspection, certification and stamping. Further, the rules provide guidance on how

to accommodate the degradation of materials, such as fracture, high temperature and cyclic operations. The principal safety requirement applicable to equipment fulfilling a confinement function is structural integrity and leak tightness. The design of the equipment must take into account the loading specified for each item. In accordance with regulation for nuclear equipment, the loads (pressure, electromagnetic forces, seismic, etc.) at foreseeable conditions shall be taken into account for classification and design. To ensure coherency between design, manufacturing, inspection and testing for each specific component one single code shall be used.

When designing ITER, the structural integrity of its component systems, has been, from the very beginning, of paramount importance for safety and investment protection. The need for specific rules for designing and manufacturing components inside the tokamak vacuum chamber was recognised at the early phase of the ITER design. The special environmental conditions (including high energy flux neutron radiation, fast and slow variation of electromagnetic fields and, consequently, induced currents and electromagnetic loads, high heat fluxes, and vacuum) were not at all covered by a single existing code.

The ITER team decided to concentrate on the use of the existing fission codes rather than pursuing the gradual development of a standard which would take into account the specificity of structural containment structures such as the VV, the high heat flux components, and the magnet systems. In the absence of well-defined C&Ss for fusion a multi-code approach was adopted in the selection of the codes for the ITER components [160]. Industrially available codes like ASME, RCC-MR, and EU harmonised standards were selected for various equipment.

For equipment which were not covered by existing codes, ITER specific design criteria were developed, and this includes for example Magnet Structural Design Criteria, Structural Design Criteria for In-vessel Components (ISDC-IC), specification for non-metallic components and design by experiment rules. For example, the ISDC-IC developed by ITER is not a code but rather a set of design criteria for the in-vessel components, which specifically indicates how to use ASME and RCC appropriately for design and construction [161]. ISDC-IC sets out how to achieve the Essential Safety Requirements (ESR) set by the nuclear authority in France with the use of RCC-MRx in rules for the design, materials, fabrication, inspection, testing and marking amongst other aspects.

The classification process of structures, systems and components (SSCs) in ITER and any fusion power station (independent of the licencing process adopted) is the first step on the path forward to understand the needs for codes and standards. It underpins the entire understanding of the fusion reactor and how each SSC operates. Consequently, the failure pathways can be drawn up to determine which components need the greatest level of quality control and assurance in order to achieve the As Low As Reasonably Achievable/Practicable (ALARA/ALARP) approach to safety around the world. With this, a designer and a licensor of fusion power stations will understand the criteria for materials, developing appropriate and proportionate design rules, and quantifying the radiological inventory and risk to the public.

The approach to classification is a top-down process starting with an understanding of the fusion reactor design, its safety analysis (such as normal, accident and unmitigated scenarios) and how the main functions will be substantiated. Once the classification of each SSC in the system was conducted, a complete set of engineering rules for design should be specified. A method similar to what has been used in the nuclear industry [162] is being applied in ITER to scrutinise a component's safety function and to identify safety class components.

This safety classification is important as it determines the quality of design and manufacturing requirements for each component within the system 163. Safety Importance Class (SIC) describes a classification scheme for structures, systems and components (SSC) of ITER that perform a safety function and contribute towards meeting the General Safety Objectives at ITER during incident/accident situations. Those SSCs assigned a Safety Importance Class will receive adequate attention during the design, fabrication, installation, commissioning and operational stages. The objective is to ensure and demonstrate that they will meet the minimum performance and reliability requirements throughout their intended lifecycle so that the safety function is provided when required. Two classes of SIC (SIC-1 and SIC-2) are adopted in ITER to graduate the SIC components. This is directly related to the function provided in preventing/mitigating the impact. SIC-1 are SSCs required to bring to and to maintain ITER in a safe state; SIC-2 are SSCs used to prevent, detect or mitigate incidents or accidents, but not required to maintain ITER in or to bring it to a safe state.

The experience of ITER will be extremely important for new reactors. However, future devices beyond ITER will use a number of different materials, such as reduced-activation ferritic-martensitic steels and oxide dispersion strengthened steels, and the operating conditions, notably the nuclear flux and fluence will be very different than those in ITER. The purpose of materials qualification is to ensure the material properties are well understood in both a virgin and degraded state (due to environmental conditions such as radiation, temperature, chemical, and/or cyclic effects) in order to design for mitigation of failure. The foreseen fluences to be achieved are orders of magnitude higher than experimental fusion devices, thus placing greater emphasis on radiation resistant materials and high temperatures.

As the existing nuclear codes and standards for construction do not adequately cover the design, manufacture or construction of fusion energy devices that are currently being considered for future constructions, there is an urgent need to develop such C&S that will have a critical role in enabling the commercialisation of fusion energy [164].

## Summary

The extrapolation to ITER is large and has required careful study, development and analysis of different design choices, considerable R&D in the corresponding technologies, as well as much supporting plasma physics investigation to enhance understanding of the conditions to be faced. Although this necessarily but somewhat unexpectedly has taken decades, this time has allowed the emergence of workable solutions that appear to have a good chance of success in achieving ITER's objectives. The ITER Organization is now addressing a number of risks of technical and licensing nature with the 2024 baseline that have arisen during the construction of this complex first-of-a-kind project and in particular the fabrication of some of the critical components like the vacuum vessel. The remaining open design issues need to be resolved, and construction completed for the subsequent ITER operation to maximise the lessons learned.

Fusion power offers the prospects of an almost inexhaustible source of energy for future generations. However, deploying reliable magnetic confinement fusion power plants relies both on the future success of ITER operation to achieve the full project specifications, and the ability to overcome the remaining design, physics and engineering gaps and development needs for key fusion technologies that are essential for reliable and efficient operation of a fusion reactor.

There is a general agreement within all the countries involved in the ITER Collaboration of the pivotal role of the ITER device to validate the physics and part of the technology to be used in a next-fusion reactor aimed at harnessing fusion power. However, there are still differences of opinions around the world on how to bridge the gap between ITER and a fusion power plant, and there are outstanding issues common to any next major facility after ITER, whether a component test facility, a Pilot Plant, DEMO, or other. These include the need to develop foreseeable sound technical solutions for the problems of plasma confinement, power exhaust, tritium breeding, cooling and extraction of high-grade heat from the breeding blanket, remote maintenance for the in-vessel components, robust magnet designs, qualified structural and PFC materials, and nuclear safety. The development of a fusion power reactor beyond ITER is the subject of Chapter 10.

The contribution of ITER to fusion is immense. Many of the lessons learned will not be forgotten and the next generation of fusion devices will be better because of what ITER has pioneered. Details of the design, manufacture, assembly and operation could never have been learnt in any other way.

## Glossary

|         |  |
|---------|--|
| AGHS    | Active Gas Handling System   |
| ALARA   | As Low As Reasonably Achievable  |
| ASN     | Autorité de Sûreté Nucléaire et de Radioprotection (French Nuclear Safety Authority) |
| BPX     | Burning Plasma Experiment  |
| BPP     | Basic Performance Phase  |
| CDA     | Conceptual Design Activities   |
| CICC    | Cable-In-Conduit Conductor   |
| CIT     | Compact Ignition Tokamak   |
| CS      | Central Solenoid   |
| DEMO    | Demonstration fusion power plant   |
| DMS     | Disruption Mitigation System   |
| DN      | Double Null (divertor configuration)   |
| DT      | Deuterium-Tritium  |
| DTE1    | Deuterium-Tritium Experiment 1   |
| DTE2    | Deuterium-Tritium Experiment 2   |
| DTE3    | Deuterium-Tritium Experiment 3   |
| EAST    | Experimental Advanced Superconducting Tokamak  |
| EC      | Electron Cyclotron (heating)   |
| ECCD    | Electron Cyclotron Current Drive   |
| EDA     | Engineering Design Activities  |
| ECH     | Electron Cyclotron Heating   |
| ELM     | Edge Localised Mode  |
| EPP     | Enhanced Performance Phase   |
| FEAT    | Fusion Energy Advanced Tokamak (reduced-cost ITER design)                            |
| FIRE    | Fusion Ignition Research Experiment  |
| FOAK    | First Of A Kind  |
| H&CD    | Heating & Current Drive  |
| HTS     | High-Temperature Superconductor  |
| IAEA    | International Atomic Energy Agency   |
| IC      | Ion Cyclotron (heating)  |
| IO      | ITER Organization  |
| INB     | Installation Nucléaire de Base   |
| INTOR   | International Tokamak Reactor  |
| ISS     | Isotope Separation System  |
| IVT     | In-Vessel Transporter  |
| JET     | Joint European Torus   |
| JT-60SA | Japan Torus-60 Super Advanced  |



|        |  |
|--------|--|
| JT-60U | Japan Torus-60 Upgrade   |
| KSTAR  | Korea Superconducting Tokamak Advanced Research                    |
| LH     | Lower Hybrid (waves)   |
| LOCA   | Loss Of Coolant Accident   |
| MCF    | Magnetic Confinement Fusion  |
| MHD    | Magnetohydrodynamic  |
| MGI    | Massive Gas Injection  |
| NET    | Next European Torus  |
| NBI    | Neutral Beam Injection   |
| NTM    | Neoclassical Tearing Mode  |
| PCS    | Plasma Control System  |
| PF     | Poloidal Field   |
| PFCs   | Plasma-Facing Components   |
| QA     | Quality Assurance  |
| QC     | Quality Control  |
| Q      | Fusion energy gain factor (ratio of fusion power to heating power) |
| RH     | Remote Handling  |
| RTO/RC | Reduced Technical Objectives / Reduced Cost                        |
| RWM    | Resistive Wall Mode  |
| SN     | Single Null (divertor configuration)                               |
| SPI    | Shattered Pellet Injection   |
| SRO    | Start of Research Operation  |
| TBM    | Test Blanket Module  |
| TBR    | Tritium Breeding Ratio   |
| TF     | Toroidal Field   |
| TFTR   | Tokamak Fusion Test Reactor  |
| VS     | Vertical Stability   |

## References

- [1] J. W. Van Dam, “Progress towards burning plasmas,” *Plasma and Fusion Research: Review Articles*, vol. 4, 2009, p. 035.
- [2] ITER Physics Basis, *Nuclear Fusion*, vol. 39, no. 12, p. 2137, 1999. ITER Physics Basis, *Nucl. Fusion* 39 (12), 2137 (1999).
- [3] Progress in the ITER Physics Basis, *Nuclear Fusion*, vol. 47, no. 6, p. S1, 2007.
- [4] Special Issue: On the Path to Tokamak Burning Plasma Operation: A collection of papers prepared by the ITPA Topical Physics Groups reviewing progress in the development of the physics basis for burning plasma operation. D.J. Campbell et al 2025 *Nucl. Fusion* 65 093002.
- [5] D. J. Campbell, “The physics of the International Thermonuclear Experimental Reactor FEAT,” *Physics of Plasmas*, vol. 8, pp. 2041–2049, 2001, doi:10.1063/1.1348334. D. J. Campbell, The physics of the International Thermonuclear Experimental Reactor FEAT, *Phys. Plasmas* 8, 2041–2049 (2001), <https://doi.org/10.1063/1.1348334>.
- [6] R. J. Hawryluk, “An overview of fusion research,” *Reviews of Modern Physics*, vol. 70, p. 537, 1998.
- [7] M. Keilhacker *et al.*, *Nuclear Fusion*, vol. 41, p. 1925, 2001.
- [8] T. Fujita and the JT-60 Team, *Nuclear Fusion*, vol. 43, p. 1527, 2003.
- [9] C. F. Maggi *et al.*, *Nuclear Fusion*, vol. 64, p. 112012, 2024.
- [10] X. Litaudon *et al.*, *Nuclear Fusion*, vol. 64, p. 112006, 2024.
- [11] “Prediction of the thermal energy confinement time for ITER size experiments,” based on ELMy H-mode statistical scaling analyses (ITER design documentation). The prediction of the thermal energy confinement time,  $\tau_{th}$ , for ITER size experiments is based on power law scalings obtained using data sets of present tokamak results in specific regimes, the most relevant being the ELMy H mode regime. A thorough statistical approach has provided a best fit to these data with an estimation of the error bars which forms the basis for the ITER design parameters.
- [12] P. Barabaschi, *Nuclear Fusion*, vol. 41, p. 155, 2001.

- [13] G. Federici et al 2019 Nucl. Fusion 59 066013.
- [14] M. Siccinio et al 2019 Nucl. Fusion 59 106026
- [15] G. Federici *et al.*, “Relationship between magnetic field and tokamak size,” *Nuclear Fusion*, vol. 64, p. 036025, 2024.
- [16] G. Federici, L. Boccaccini, F. Cismondi *et al.*, “EU breeding blanket design strategy,” *Fusion Engineering and Design*, vol. 141, pp. 30–42, 2019.
- [17] M. A. Abdou, “Radiation considerations for superconducting fusion magnets,” *Journal of Nuclear Materials*, vol. 72, pp. 147–167, 1978.
- [18] M. A. Abdou, “Radiation considerations for superconducting fusion magnets,” *Journal of Nuclear Materials*, vol. 72, pp. 147–167, 1978.
- [19] J. Wesson, D. J. Campbell, Tokamaks, 4<sup>th</sup> Edition, Clarendon Press, 2011, ISBN 978-0-19-959223-4.
- [20] Description of H-mode and L-mode confinement regimes (ITER Physics explanatory text). In plasma physics and magnetic confinement fusion, the high-confinement mode (H-mode) is a phenomenon and operating regime of enhanced confinement in toroidal plasma such as tokamaks. When the applied heating power is raised above some threshold, the plasma transitions from the low-confinement mode (L-mode) to the H-mode where the energy confinement time approximately doubles in magnitude. Prior to the discovery of H-mode, all tokamaks operated in what is now called the L-mode. The L-mode is characterized by relatively large amounts of turbulence, which allows energy to escape the confined plasma.
- [21] Reduced Technical Objectives/Reduce Cost ITER (RTO-RC ITER referred to as ITER-FEAT).
- [22] R. Andreani *et al.*, “The Frascati Tokamak Upgrade,” *Fusion Technology*, vol. 218, 1991.
- [23] I.H. Hutchinson *et al.*, Overview of the ALCATOR-C MOD results, 2001 *Nucl. Fusion* 41 1391.
- [24] S. Itoh, et al., “Steady state current drive by lower hybrid wave in TRIAM-1M Tokamak”, IAEA-CN-50/E-III-2, pp. 629-635.

- [25] W. Rauch, W. A. Stark, "BPX System Design Description", BPX CDR Document O-910311-PPL-14, 1991.
- [26] D. M. Meade, "Fusion Ignition Research Experiment (FIRE)," *Fusion Technology*, vol. 39, pp. 336–342, 2001.
- [27] B. Coppi and the Ignitor Project Group, "Highlights of the Ignitor experiment," *Journal of Fusion Energy*, vol. 13, pp. 111–119, 1994.
- [28] B. Coppi and the Ignitor Project Group, "Highlights of the Ignitor experiment," *Journal of Fusion Energy*, vol. 13, pp. 111–119, 1994.
- [29] P. Barabaschi *et al.*, "ITER progresses into new baseline," *Fusion Engineering and Design*, vol. 215, 2025, Art. no. 114990.
- [30] S. Imagawa *et al.*, "Concept of magnet systems for an LHD-type reactor," *Nuclear Fusion*, vol. 49, p. 075017, 2009.
- [31] T. Rummel *et al.*, "The superconducting magnet system of Wendelstein 7-X," *IEEE Transactions on Plasma Science*, vol. 40, pp. 769–776, 2012.
- [32] Equipe TORE SUPRA (1989) TORE SUPRA: A tokamak with superconducting toroidal field coils-status report after the first plasma. *IEEE Transactions on Magnetics* 25: 1473–1480.
- [33] K. Kim, et al. (2005) Status of the KSTAR superconducting magnet system development. *Nuclear Fusion* 45: 783–789.
- [34] M. Kwon, Y.K. Oh, et al. (2011) Overview of KSTAR initial operation. *Nuclear Fusion* 51: 094006 (12pp).
- [35] S. Wu and EAST team (2007) An overview of the EAST project. *Fus. Eng. Des.* 82: 463–471.
- [36] J. Wei, W.G. Chen, W.Y. Wu, et al. (2010) The superconducting magnets for EAST Tokamak. *IEEE Transactions on Applied Superconductivity* 20: 556–559.
- [37] K. Yoshida, K. Tsuchiya, K. Kizu, et al. (2010) Development of JT-60SA superconducting magnet system. *Physica C* 470: 1727–1733.
- [38] D.G. Whyte *et al* 1999 *Nucl. Fusion* 39 1025.

- [39] R. J. Hawryluk, Phil. Trans. R. Soc. Lond. A 357 (1999) 443.
- [40] M. Keilhacker, M.L., Watkins, and the JET Team, J. Nucl. Mater. 266{269 (1999) 1.
- [41] G. Federici et al., 20001, Nucl. Fusion 41, 1967.
- [42] The inner surface of the tokamak: plasma facing materials (incl. impact of conditioning requirements as function of the material).
- [43] S. Brezinsek et al., Nucl. Fusion 53 (2013) 083023.
- [44] H. Zohm et al., Nucl. Fusion 49 (2009) 104009
- [45] H. Zohm *et al* 2001, The physics of neoclassical tearing modes and their stabilization by ECCD in ASDEX Upgrade, Nucl. Fusion 41 197DOI 10.1088/0029-5515/41/2/306.
- [46] R.J. La Haye, A. Isayama, M. Marascheck, Prospects for stabilization of neoclassical tearing modes by electron cyclotron current drive in ITER. Nucl. Fusion 49, 045005 (2009).
- [47] M. Prokopas et al., ITER Control System Model: a full-scale simulation platform for the CODAC infrastructure. Fusion Eng. Des. 128, 86 (2018).
- [48] D.A. Humphreys et al., Experimental vertical stability studies for ITER performance and design guidance, paper IT/2-4b, in Proceedings of 22nd IAEA Fusion Energy Conference, Geneva (2008).
- [49] A. Portone et al., ITER plasma vertical stabilization, paper IT/2- 4a, in Proceedings of 22nd IAEA Fusion Energy Conference, Geneva (2008).
- [50] A. Loarte et al., Progress on the application of ELM control schemes to ITER scenarios from the non-active phase to DT operation. Nucl. Fusion 54, 033007 (2014).
- [51] L.R. Baylor et al., Reduction of ELM intensity using high repetition rate pellet injection in Tokamak H-mode plasmas. Phys. Rev. Lett. 110, 245001 (2013).
- [52] P.T. Lang et al., ELM frequency control by continuous small pellet injection in ASDEX Upgrade. Nucl. Fusion 43, 1110 (2003).
- [53] S. Zoletnik, E. Walcz, S. Jachmich, U. Kruezi, M. Lehnen et al. Disruption Shattered pellet technology development in the ITER DMS test laboratory, Fus. Eng. Des. 190 (2023) 113701.

- [54] B. Stein-Lubrano *et al* 2024 *Nucl. Fusion* 64 036020.
- [55] M. Dibon *et al.*, Design of the shattered pellet injection system for ASDEX Upgrade *Rev. Sci. Instrum.* 94, 043504 (2023).
- [56] T.E. Gebhart, *et al.*, *Nucl. Fusion* 61 (2021), 106007.
- [57] S. Jachmich, *et al.*, Advances in Shattered Pellet Injection Technology for the ITER Disruption Mitigation System in: 32nd Symposium On Fusion Technology, 18–23 September, Dubrovnik, Croatia, 2022.”, *Proc. 32nd Symp. On Fusion.*
- [58] R.J. Hawryluk *et al.*, Principal physics developments evaluated in the ITER design review. *Nucl. Fusion* 49, 065012 (2009).
- [59] Chapter 9.2: Physics basis and design implications.
- [60] G. Federici, C. Bachmann, V. Corato, S. Jimenez and J.H. You, Magnetic Confinement Fusion—Technology—Fusion Core Encyclopedia of Nuclear Energy (2021), Pages 554-575.
- [61] P. Bruzzone (2015) Superconductivity and fusion energy - The inseparable companions. *Superconductor Science and Technology* 28: 024001 (6 pp).
- [62] M. Huguet (1997) The ITER magnet system. *Fus. Eng. Des.* 36: 23–32.
- [63] L. Lao (2020) Magnetic Confinement Fusion - Plasma Theory: Magnetohydrodynamic Stability. This encyclopedia, ch. 01206.
- [64] N. Mitchell *et al* 2008 The ITER magnet system *IEEE Trans. Appl. Supercond.* 18 435–40.
- [65] A. Devred *et al* 2014 Challenges and status of ITER conductor production *Supercond. Sci. Technol.* 27 044001.
- [66] N. Mitchell *et al.*, The ITER Magnets: Design and Construction Status, *IEEE Trans. Appl. Supercond.* 22(3) 4200809 (2012).
- [67] A. Ballarino, P. Bauer, and Y.E. Bi (2012) Design of the HTS current leads for ITER. *IEEE Transactions on Applied Superconductivity* 4800304(4 pp): 22.

- [68] D.J. Campbell et al., Innovations in Technology and Science R&D for ITER, *Journal of Fusion Energy* (2019) 38:11–71.
- [69] Y. Gribov et al 2015 *Nucl. Fusion* 55 073021
- [70] Y. Shimomura *et al* 1999 *Nucl. Fusion* **39** 1295
- [71] B. Steibl, P. Lang, F. Leuterer, J-M. Noterdaeme, and A. Stabler, *Fus. Sci. Tech.* 44(3) (2003) 578–592.
- [72] G. Duesing, The vacuum systems of the nuclear fusion facility JET. *Vacuum* 37(3,4) (1987) 309–315.
- [73] C. Bachmann, F. Arbeiter, L. Boccaccini et al. Issues and strategies for DEMO in-vessel component integration, *Fus. Eng. Des.* 112 (2016) 527–534.
- [74] M. Loughlin, E. Polunovskiy, K. Ioki, M. Merola et al., Nuclear shielding for the toroidal field coils of ITER, *Fus. Sci. Tech.* 60 (2011) 81–86.).
- [75] F. Farfaletti-Casali, D. Booker, U. Buzzi, et al., The interaction of systems integration, assembly, disassembly and maintenance in developing the INTOR-NET mechanical configuration. *Nuclear Engineering and Design. Fusion* 1 (1984) 115–125.
- [77] F. Maviglia, R. Albanese, R. Ambrosino, W. Arter, et al., Wall protection strategies for DEMO plasma transients *Fus. Eng. Des.* 124 (2017) 385–390.
- [77] D. Wilson, N. Bernard, and A. Mariani (2015) Alignment of in-vessel components by metrology defined adaptive machining. *Fus. Eng. and Des.* 98–99 (2015) 1688–1691.
- [78] F. Escourbiac, T. Hirai, S. Carpentier-Chouchana, A. Fedosov, et al. (2012) Effort on Design of a Full Tungsten Divertor for ITER, IAEA-CN—197 ITR/P5-08.
- [79] R. Albanese, M. Mattei and F. Villone Prediction of the growth rates of VDEs in JET. *Nuclear Fusion* 44: F (2004) 999.
- [80] (Buzio M (1998) Structural effects of plasma instabilities on the JET Tokamak. PhD thesis London: Imperial College.
- [81] Choi CH, Sborchia C, Ioki K, et al. (2014) Status of the ITER vacuum vessel construction. *Fus. Eng. Des.* 89: 1859–1863.

[82] Lloyd B, Carolan PG, and Warrick CD (1996) ECRH-assisted start-up in ITER. *Plasma Physics and Controlled Fusion* 38: 1627.

[83] de Vries PD and Gribov Y (2019) ITER breakdown and plasma initiation revisited. *Nuclear Fusion* 59: 096043.

[84] M. Merola et al., Engineering challenges and development of the ITER Blanket System and Divertor. *Fusion Eng. Des.* 96 -97, 34 (2015).

[85] R. Tivey et al., ITER divertor, design issues and research and development, *Fus. Eng. Des.* 46 (1999) 207–220.

[86] R. Toschi, M. Chazalo, V. Engelmann, J. Nihoul, J. Raeder, and E. Salpietro (1988) Next European torus: Objectives, general requirements and parameter choices. *Fusion Technology* 14: 19.

[87] G. Janeschitz, K. Borrass, G. Federici, Y. Igitkhanov, A. Kukushkin, H.D. Pacher, G.W. Pacher, and M. Sugihara (1995) The ITER divertor concept. *Journal of Nuclear Materials* 220-222: 73–88.

[88] A. Cardella, M. Chazalon, G. Vieider G, et al. (1991) The Divertor System for NET. In: *Fusion Technology 1990*, pp. 258–262. Amsterdam: Elsevier.

[89] A. Cardella et al. (1993) Design Manufacturing and Testing of the Monoblock Divertor. In: *Fusion Technology 1992*, pp. 211–215. Amsterdam: Elsevier.

[90] E. Visca et al., Hot radial pressing: An alternative technique for the manufacturing of plasma-facing components, *Fus. Eng. Des.* 75–79 (2005) 485–489.

[91] V. Kuznetsov et al., Status of the IDTF high-heat-flux test facility, *Fusion Eng. Des.* 89 (2014) 955–959.

[92] M. Akiba, M. Araki, S. Suzuki, S. Tanaka, M. Dairaku, M. Seki, K. Yokoyama, *Plasma devices oper.* 1 (1991) 205–212.

[93] Bobin-Vastra et al., *Fus. Eng. Des.* 75–79 (2005) 357–363.

[94] A.R. Raffray et al., The ITER blanket system design challenge, *Nucl. Fusion* 54 (2014) 000000 (18pp).

- [95] M. Merola et al., Engineering challenges and development of the ITER Blanket System and Divertor. *Fusion Eng. Des.* 96–97 (2015) 34.
- [96] R.A. Pitts et al., A full tungsten divertor for ITER: physics issues and design status. *J. Nucl. Mater.* 438, (2013) S48.
- [97] F. Wagner, On the heating mix of ITER, *Plasma Phys. Control. Fusion* 52 (2010) 124044 (14pp).
- [98] R. Wilhelm, Plasma heating — A comparative overview for future applications, *Fus. Eng. Des.* 11 (1989) 167-180.
- [99] Yevgen Kazakov et al. plasma heating in present-day and future fusion machines, 2015. Forschungszentrum Jülich GmbH.
- [100] K. Sakamoto, A. Kasugai, K. Takahashi, et al., Achievement of robust high efficiency 1 MW oscillation in the hard-self-excitation region by a 170 GHz continuous-wave gyrotron, *Nat. Phys.* 3 (6) (2007) 411–414.
- [101] S. Alberti et al., "Status of development of the 2MW, 170GHz coaxial-cavity gyrotron for ITER," 2008 IEEE 35th International Conference on Plasma Science, Karlsruhe, Germany, 2008, pp. 1-1, doi: 10.1109/PLASMA.2008.4590755.
- [102] B. Piosczyk, et al., A 2-MW, 170-GHz, coaxial cavity gyrotron, *IEEE Trans. Plasma Sci.* 32 (2004) 413–417.
- [103] R. Heidinger et al., , Conceptual design of the ECH upper launcher for ITER, *Fus. Eng. Des.* 84 (2009) 284-289.
- [104] K. Takahashi, K. Kajiwara, N. Kobayashi, A. Kasugai, K. Sakamoto, Improved design of an ITER equatorial EC launcher, *Nucl. Fus.* (2008) 054014.
- [105] A. Argouarch et al., Validation of an ICRF ITER-Like Antenna on Tore Supra, *AIP Conf. Proc.* 1187, 209–212 (2009) <https://doi.org/10.1063/1.3273730>.
- [106] Y. Takeiri, et al., *Nucl. Fus.* 46 (S199) (2006)., K. Tsumori, et al., *Rev. Sci. Instrum.* 79 (2008) 02C107.

- [107] L. R. Grisham, P. Agostinetti, G. Barrera, P. Blatchford, D. Boilson, J. Chareyre, et al., Recent improvements to the ITER neutral beam system design, *Fus. Eng. Des.* 87 (11), 1805–1815.
- [108] A. Masiello, The European contribution to the development of the ITER NB injector, *Fus. Eng. Des.* 86 (2011) 860–863.
- [109] P. Sonato, The ITER full size plasma source device design, *Fus. Eng. Des.* 84 (2009) 269–274.
- [110] U. Fantz, Achievement of the ITER NBI ion source parameters for hydrogen at the test facility ELISE and present Status for deuterium, *Fus. Eng. Des.* 156 (2020) 111609.
- [111] A.E. Costley, T. Sugie, G. Vayakis, C.I. Walker, Technological challenges of ITER diagnostics, *Fus. Eng. Des.* 74 (2005) 109–119.
- [112] M. Walsh et al., ITER diagnostic challenges IEEEINPSS 24th Symposium on Fusion Engineering (2011), DOI:10.1109/SOFE.2011.6052210.
- [113] A. Encheva, L. Bertalot, B. Macklin, G. Vayakis, C. Walker, Integration of ITER in-vessel diagnostic components in the vacuum vessel, *Fus. Eng. Des.* 84 (2009) 736–742.
- [114] M. Glugla, A. Antipenkov, S. Beloglazov, C. Caldwell-Nichols, I.R. Cristescu, et al, The ITER tritium systems, *Fus. Eng. Des.* 82 (2007) 472–487.
- [115] D.K. Murdoch, Ch. Day, P. Gierszewski, R.-D. Penzhorn, C.H. Wu, Tritium inventory issues for future reactors: choices, parameters, limits, *Fus. Eng. Des.* 46 (1999) 255–271.
- [116] M. Glugla, A. Antipenkov, S. Beloglazov, The ITER tritium systems, *Fus. Eng. Des.* 82 (2007) 472–487.
- [117] I. R. Cristescu et al., 2007 *Nucl. Fusion* 47 5458, M. Glugla et al, *Fusion Eng Des*, 82, (2007), 472.
- [118] C. Day, 2004 "Validated design of the ITER main vacuum pumping systems", 20th IAEA Fusion Energy Conference, paper IT/P3-1.
- [119] B.J. Peters, C. Day, Analysis of low pressure hydrogen separation from fusion exhaust gases by the means of superpermeability, *Fus. Eng. Des.* 124 (2017) 696–699.

- [120] P.J. Dinner, M. Iseli, An advanced fuel cycle design for fusion power reactors, Proceedings of 17<sup>th</sup> SOFT, Rome, Italy, 1992, Fusion Technol. (1992) 1152–1156.
- [121] D.K. Murdoch, J. Blevins, P. Gierszewski, R. Matsugu, Canadian contribution to the European Home Team Programme for ITER, in: The 11th Pacific Basin Nuclear Conference, Banff, Canada, May 1998, Canadian Nuclear Society, Toronto, Canada, 1998, pp. 813–820.
- [122] D Murdoch et al 2008, Vacuum technology for ITER, J. Phys.: Conf. Ser. 100 062002.
- [123] C. Day et al., The pre-concept design of the DEMO tritium, matter injection and vacuum systems Fus. Eng. Des. 179 (2022) 113139.
- [124] A. Güntherschulze, H. Betz, and H. Kleinwächter, 1939 Z. Physik 111 657.
- [125] A. L. Livshits, 1976 Sov. Phys. Tech. Phys. 21 187.
- [126] A. L. Livshits, 1997 J. Nucl. Mater. 241 1203.
- [127] T. Raimondi (1989) The JET Remote Maintenance System, Vienna: IAEA Technical Document IAEA-TECDOC-495.
- [128] O. David, A. Loving, J. Palmer, S. Ciattaglia, J.P. Friconneau, Operational experience feedback in JET RH, Fus. Eng. Des. 75–79 (2005) 519–523.
- [129] R. Buckingham and A. Loving, Remote-handling challenges in fusion research and beyond. Nature Physics 12(5): (2016) 391.
- [130] E. Villedieu, V. Bruno, P. Pastor, et al. (2016) An articulated inspection arm for fusion purposes. Fus. Eng. Des. 109–111(Part B): 1480–1486.
- [131] A. Tesini and J. Palmer (2008) The ITER remote maintenance system. Fus. Eng. Des. 83(7–9): 810–816.
- [132] J-P. Friconneau, V. Beaudoin, A. Dammann, et al. (2017) ITER hot cell—RH system maintenance overview. Fus. Eng. Des. 24: 673–676.
- [133] J. Thomas, A. Loving, O. Crofts, R. Morgan, and J. Harman, DEMO active maintenance facility concept progress, Fus. Eng. Des. 89 (2014) 2393–2397.

- [134] K. Keogh, S. Kirk, W. Suder, I. Farquhar, T. Tremethick, A. Loving, Laser cutting and welding tools for use in-bore on EU-DEMO service pipes, *Fus. Eng. Des.* 136 (2018) 461–466, <https://doi.org/10.1016/j.fusengdes.2018.02.098>.
- [135] B. Macklin, L. Bao, P. Chappuis, F. Escourbiac, S. Gicquel, J.D. Palmer, et al., Assembly and RH of ITER plasma facing components, in: 2015 IEEE 26th Symposium on Fusion Engineering (SOFE), Austin, TX, USA, IEEE, 2015, pp. 1–8, <https://doi.org/10.1109/SOFE.2015.7482266>.
- [136] C. Damiani, J. Palmer, N. Takeda, C. Annino, S. Balagué, P. Bates, et al., Overview of the ITER remote maintenance design and of the development activities in Europe, *Fusion Eng. Design* 136 (2018) 1117–1124, <https://doi.org/10.1016/j.fusengdes.2018.04.085>.
- [137] T. Tremethick, S. Kirk, K. Keogh, A. O’Hare, E. Harford, B Quirk, Service joining strategy for the EU DEMO, *Fusion Eng. Design* 158 (2020), <https://doi.org/10.1016/j.fusengdes.2020.111724>.
- [138] Y. Ren, R. Skilton A review of pipe cutting, welding, and NDE technologies for use in fusion devices, *Fus. Eng. Des.* 202 (2024) 114396.
- [139] A. Tesini and J. Palmer (2008) The ITER remote maintenance system. *Fus. Eng. Des.* 83(7–9): 810–816.
- [140] H. Saarinen, T. Kivelä, L. Zhai, V. Hämäläinen, J. Karjalainen, et al. (2011) Results of CMM standalone tests at DTP2. *Fus. Eng. Des.* 86: 1907–1910.
- [141] S. Esqué, C. van Hille, R. Ranz, et al. Progress in the design, R&D and procurement preparation of the ITER Divertor RH system. *Fus. Eng. Des.* 89 (2014) 2373–2377.
- [142] N. Takeda et al., Mock-up test on key components of ITER blanket remote handling system *Fus. Eng. Des.* 84 (2009) 1813–1817.
- [143] K. Shibnuma, T. Honda, M. Kondoh, T. Munakata, S. Murakami, N. Sasaki, et al., *Fusion Eng. Des.* 18 (1991) 487.
- [144] K. Shibnuma, T. Honda, *Fusion Eng. Des.* 55 (2001) 249.
- [145] S. Kakudate, K. Oka, T. Yoshimi, K. Taguchi, M. Nakahira, N. Takeda, et al., *Fusion Eng. Des.* 51–52 (2000) 993.

- [146] S. Kakudate, K. Shibamura, *Fusion Eng. Des.* 65 (2003) 33.
- [147] S. Kakudate, K. Shibamura, *Fusion Eng. Des.* 65 (2003) 133.
- [148] C. Bachmann, S. Ciattaglia, F. Cismondi, T. Eade, et al., Overview over DEMO design integration challenges and their impact on component design concepts. *Fus. Eng. Des.* 136 (2018) 87–95.
- [149] J. Keep, S. Wood, N. Gupta, M. Coleman, and A. Loving, Remote handling of DEMO breeder blanket segments: Blanket transporter conceptual studies. *Fus. Eng. Des.* 124 (2017) 420–425.
- [150] M. Abdou and et al, “FINESSE: A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research and Development (Interim Report),” UCLA-ENG-84-30, Rep. PPG-821.
- [151] P. Komarek and G. Kulcinski, “Meeting the Technology Needs for a Tokamak DEMO: A Strategy Including Mirror Based Nuclear Test Facilities,” *Fusion Technology*, 8:1P2B, no. DOI: 10.13182/FST85-A39915, pp. 1075-1080, 1985.
- [152] These concepts were all based on a beam-to-target fusion approach where a near Maxwellian background plasma is sustained against energy and particle losses by neutral beam injection, and fusion reactions principally occur between the fast and target ions as the beam thermalises via Coulomb collisions.
- [153] M. Abdou, “Overview of Finesse Effort on Fusion Nuclear Technology,” *Fusion Technology*, 8:1P2B, no. DOI: 10.13182/FST85-A39916, pp. 1081-1090, 1985.
- [154] R.J. Pearson, A. B. Antoniazzi, W.J. Nuttalla Tritium supply and use: a key issue for the development of nuclear fusion energy, *Fus. Eng. Des.* 136 (2018) 1140–1148.
- [155] A.R. Raffray, M. Akiba, V. Chuyanov, L. Giancarli and S. Malang. 2002 Breeding blanket concepts for fusion and materials requirements *J. Nucl. Mater.* 307–311 (2002) 21–30.
- [156] A. Ibarra et al., The European approach to the fusion-like neutron source: the IFMIF-DONES project, 2019 *Nucl. Fusion* 59 065002.
- [157] L.M. Giancarli et al., Status of the ITER TBM Program and overview of its technical objectives, *Fus. Eng. Des.* 203 (2024) 114424.

[158] L.M. Giancarli et al., Test blanket modules in ITER: An overview on proposed designs and required DEMO-relevant materials, *J. Nucl. Mat.* 367–370 (2007) 1271–1280.

[159] The main purpose of codes and standards (C&S) is to establish national or international criteria based on state-of-the-art knowledge, experience, and experimental feedback from nuclear facilities to ensure structural integrity is maintained

[160] V. Barabash et al. , Codes and standards and regulation issues for design and construction of the ITER mechanical components, *Fus. Eng. Des.* 85 (2010) 1290–1295.

[161] G. Sannazzaro, Development of design Criteria for ITER In-vessel Components *Fus. Eng. Des.* 88 (2013) 2138– 2141.

[162] IAEA, “SSG-30 Safety Classification of Structures, Systems and Components in Nuclear Power Plants,” IAEA, 2014.

[163] IAEA, “Integrated Approach to Safety Classification of Mechanical Components for Fusion Applications” (IAEA, TECDOC- 1851, 2018)

[164] T.P. Davis, The Need for Codes and Standards in Nuclear Fusion Energy, *Journal of Fusion Energy* (2023) 42:13 <https://doi.org/10.1007/s10894-023-00350>.