

Chapter 2

Technology Issues

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Objectives

After reading this chapter one should understand

- What technologies are required to build a fusion power plant
- What are the main achievements and difficulties in these areas.

2.1 The Issues

The purpose of this chapter is to provide a summary of fusion technology issues that will be covered in the rest of the book. The main technologies of magnetic confinement fusion (MCF) are listed in Table 2.1.

2.2 Magnets

Pulsed magnets energized by capacitor banks have been used for fast pinch experiments. The capacitors can be charged at low input power for minutes, and then discharged suddenly at high power. In effect they amplify the power, but the high power output only lasts for microseconds or milliseconds. They are used for experiments like Z pinches, theta pinches, and plasma focus devices. The central solenoid (CS) coil of a tokamak is pulsed in milliseconds to induce the toroidal plasma current.

Figure 2.1 shows the three coil sets of a tokamak (Fig. 1.14 repeated).

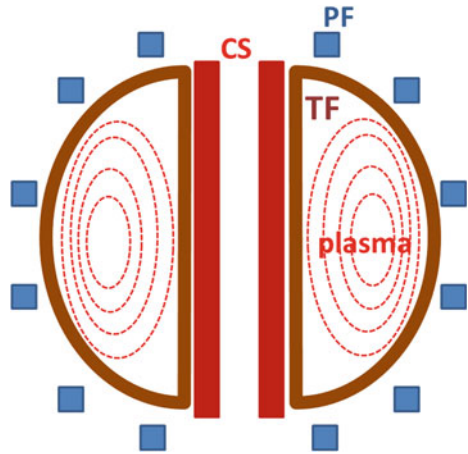
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Table 2.1 Technologies of magnetic confinement fusion

Pulsed and water-cooled magnets
Superconducting magnets
Plasma heating and current drive
First wall, blanket and shield
Power and particle control
Materials issues
Vacuum systems
Cryogenic systems

Fig. 2.1 Tokamak coil systems. The *D-shaped* toroidal field (TF) coils provide the toroidal field (as in Fig. 1.8). The central solenoid (CS) is pulsed to induce the plasma toroidal current J (Fig. 1.9). The poloidal field (PF) coils control the plasma shape and position. The *dashed ellipses* represent magnetic flux surfaces of the plasma



Faraday's Law states that a changing magnetic field induces an electric field:

$$\nabla \times \mathbf{E} = -(\partial \mathbf{B} / \partial t) \quad (2.1)$$

The integral form of this equation states that the integral of the electric field around a boundary curve equals the integral of $(\partial \mathbf{B} / \partial t)$ over the enclosed surface. (Please see Appendix D for vector relations.)

$$\oint d\ell \bullet \mathbf{E} = - \int d\mathbf{S} \bullet (\partial \mathbf{B} / \partial t) \quad (2.2)$$

Consider a CS coil with internal area S , Fig. 2.2.

The electric field induced at radius R by B changing inside S is

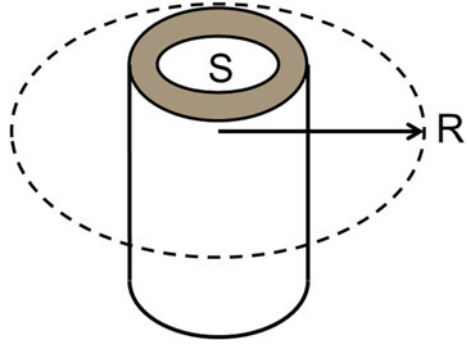
$$2\pi R E \approx S(dB/dt) \quad (2.3)$$

If $R = 2$ m, $S = 0.5$ m², and $dB/dt = 10$ T/s, then $E = 0.4$ V/m.

Ampere's law is

$$\nabla \times \mathbf{B} = \mu_0 \mathbf{J} + \mu_0 \epsilon_0 (\partial \mathbf{E} / \partial t) \quad (2.4)$$

Fig. 2.2 A CS coil with internal area S surrounded by a toroidal plasma at radius R



where $\mu_0 = 4\pi \cdot 10^{-7}$ H/m is the permeability of free space and $\epsilon_0 = 8.854 \times 10^{-12}$ F/m is the permittivity of free space. The last term may be neglected, except for very high frequency oscillations. The integral form of this equation is

$$\oint d\ell \cdot \mathbf{B} = \mu_0 \int d\mathbf{S} \cdot \mathbf{J} \quad (2.5)$$

where $\int d\mathbf{S} \cdot \mathbf{J}$ is the total current I inside the boundary. The azimuthal field B_θ at a radius r from a long, straight wire carrying a current I is

$$2\pi r B_\theta = \mu_0 I \quad (2.6)$$

This can be used to estimate the poloidal field around a plasma carrying a toroidal current I . The axial field inside a long straight solenoid with length L carrying a total current I_{tot} is

$$B_z = \mu_0 I_{\text{tot}} / L \quad (2.7)$$

In a toroidal device, $L \rightarrow 2\pi R$, and the toroidal magnetic field at major radius R generated by N toroidal field coils each carrying a current I is

$$B = \mu_0 NI / 2\pi R \quad (2.8)$$

If $N = 20$ coils, $R = 2$ m, and $B = 3$ T, then the required current per coil would be $I = 1.5$ MA. Water-cooled magnets can have current densities in the copper $J \sim 10$ MA/m², so the cross sectional area of the copper in each coil would need to be $A_c \sim 0.15$ m². If the coil radius $a_c \sim 1$ m, then the volume of copper per coil would be

$$V_c = 2\pi a_c A_c = 0.94 \text{ m}^3 \quad (2.9)$$

(The total volume would be larger to accommodate cooling water channels).

Assuming a copper resistivity of $\eta = 2 \times 10^{-8}$ Ohm-m, the power dissipated in each coil would be

$$P_c = \eta J^2 V_c \sim 1.9 \text{ MW} \quad (2.10)$$

The total power for 20 coils would be 38 MW. This simple example illustrates the fact that very high powers are required for water-cooled copper coils when

fields of several Tesla are required. New large experiments, such as tokamaks and stellarators, use superconducting coils, because the electrical power consumption of ordinary copper coils would be too high.

Researchers are developing superconductors that can carry high current densities in high magnetic fields. Most high-field superconductors operate at $T < 10$ K and $B < 14$ T. At higher magnetic fields superconductivity may be lost or the stresses may be too high. Some “high-temperature superconductors” (HTS) can be superconducting at $T \sim 80$ K, but at lower magnetic fields. However, the HTS materials can produce very high magnetic fields at lower temperatures.

High temperature superconductors are used for current leads into superconducting coils. Superconductors are also used for many applications other than fusion research, including motors, generators, transmission lines, magnetic resonance imaging, and research, so developments in one of these fields may have benefits in others.

[Chapter 3](#) discusses pulsed and water-cooled magnets, and [Chap. 4](#) describes superconducting magnet systems.

2.3 Plasma Heating and Current Drive

Plasma heating can be done by the following methods:

- Ohmic heating
- Plasma compression
- Magnetic induction
- Electromagnetic waves
- Particle beam injection
- Plasma guns.

2.3.1 Ohmic Heating

When a current is pulsed in the CS coil, a toroidal current is induced in the plasma. This is called “inductive current drive”. The current density J (A/m^2) flowing through plasma with resistivity η (Ohm-m) generates an ohmic heating power density

$$P_{oh} = \eta J^2 \text{ (W/m}^3\text{)} \quad (2.11)$$

in the plasma. (Exercise for students: verify that the units are W/m^3). The duration of the current is limited by the magnetic flux (number of Volt-seconds) that can be provided by the CS coil. If the current is to be sustained for longer times, then “non-inductive” current drive is needed.

Since η is proportional to $T^{-3/2}$, at high temperatures η becomes very small, and ohmic heating is unable to heat the plasma adequately.

Plasma compression heating was demonstrated in several tokamaks, but it reduces the plasma volume relative to the magnet coil volume, so it is not currently used in large magnetic confinement fusion devices.

2.3.2 Charged Particle Injection

It is possible to inject electrons or ions along B field lines into an “open” magnetic confinement system, shown in Fig. 2.3.

Many of the ions or electrons would simply flow through the plasma and out the other end, but sometimes a plasma instability can trap injected electrons and provide good heating.

It would be difficult to heat a *toroidal* plasma by injecting electrons or ions, because they cannot flow easily across a strong toroidal magnetic field, as can be seen from Fig. 2.4.

Fig. 2.3 The circular magnet coils on the right produce magnetic fields similar to those of the bar magnets on the left, namely a magnetic mirror field (top) and a magnetic spindle cusp field (bottom)

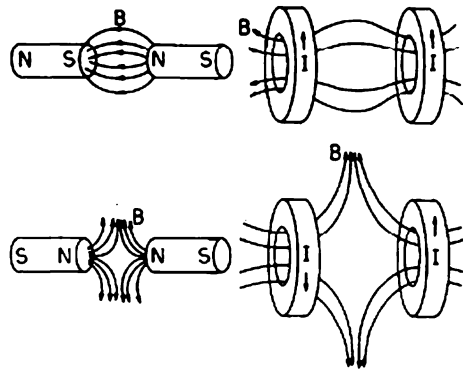
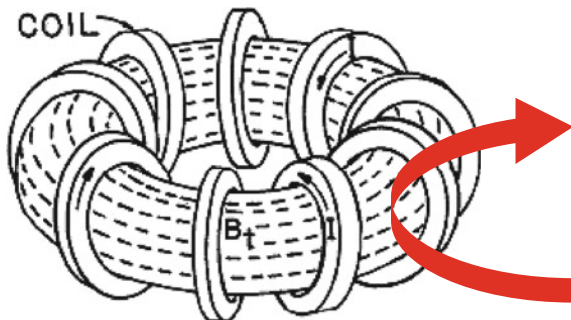


Fig. 2.4 A simple toroidal magnetic field (dashed lines) produced by many circular magnet coils. The curved arrow represents charged particles coming from the outside that are reflected by the magnetic field



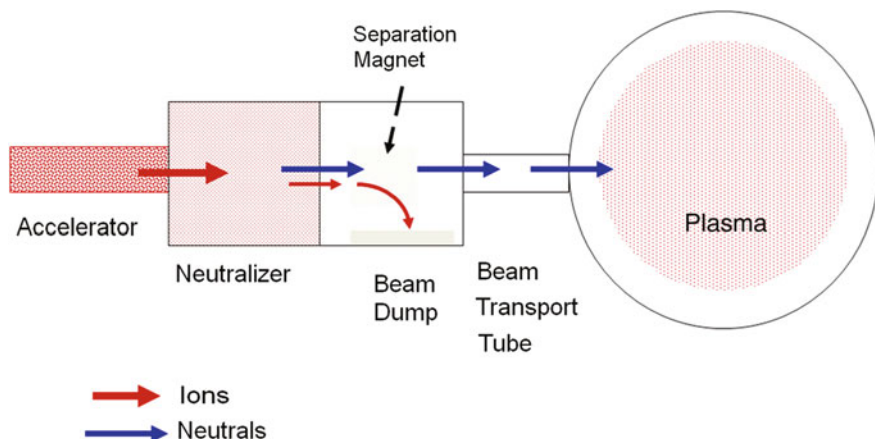


Fig. 2.5 Schematic diagram of a neutral beam injector

2.3.3 Neutral Beam Injection

However, an accelerated ion beam can be neutralized and the neutral atoms can easily cross the magnetic field into the plasma, where they are ionized and trapped.

Figure 2.5 illustrates the main components of a neutral beam injector.

The accelerator has an ion source and high voltage grids to accelerate and focus the ion beam. In a fusion power plant the ions would probably be deuterons or tritons. The beam passes through the neutralizer gas cell, where many of the accelerated ions pick up electrons to become fast neutral atoms. The un-neutralized ions are separated out by a bending magnet and directed into a beam dump. The fast neutral atoms then pass through the beam transport tube into the hot plasma, where they are ionized and trapped.

Beam energies of 20–50 keV are adequate for small plasmas, but for large high-pressure plasmas, beam energies ~ 1 MeV are required for adequate beam penetration into the plasma core. At high energies negative ions are used, because they are easier to neutralize than positive ions.

In addition to plasma heating, the neutral beams impart momentum to the plasma, which can cause plasma current drive and plasma rotation. Thus, the technology of neutral beam injection (NBI) is very important for tokamaks, such as ITER. Neutral beam injection requires large ports and straight paths for the neutral atoms. Neutrons can stream out from the fusion plasma and make external components radioactive.

2.3.4 Electromagnetic Waves

The “cyclotron frequency” at which electrons and ions spiral around magnetic field lines is

$$\begin{aligned}\omega_c &= qB/m\gamma \quad (\text{radians/s}) \\ f_c &= \omega_c/2\pi \quad (\text{Hz})\end{aligned}\tag{2.12}$$

where m = particle mass (kg), B = magnetic field (T), q is the particle's charge (C) and $\gamma = (1 - v^2/c^2)^{1/2}$. Here we assume, $\gamma = 1$ unless otherwise specified. For example, the electron cyclotron frequency in a field of 1 T is $f_c = 28$ GHz. (Exercise: Verify this value and the units.)

Several types of electromagnetic waves can be used for plasma heating and current drive, including:

- Electron cyclotron resonance (tens to hundreds of GHz)
- Ion cyclotron resonance (tens of MHz)
- Lower hybrid resonance (a few GHz).

These can be tuned to the particle mass and magnetic field region where heating or current drive is needed.

Radio waves require antennas, which can introduce impurities into the plasma. Microwaves can be transmitted through waveguides, with windows separating the plasma chamber from the klystron or magnetron tubes that generate the waves. The windows must be able to transmit high power fluxes with little energy deposition in the windows, which could cause windows to crack. Diamond and sapphire make excellent windows, because of their very high thermal conductivity, but they are expensive.

Engineers have done much work to develop efficient, reliable, high-power generators, transmission lines, windows, and coupling antennas or grills. For example, the provision of 1 MW sources of 170 GHz microwaves that can operate for many seconds is a major technology development program for ITER.

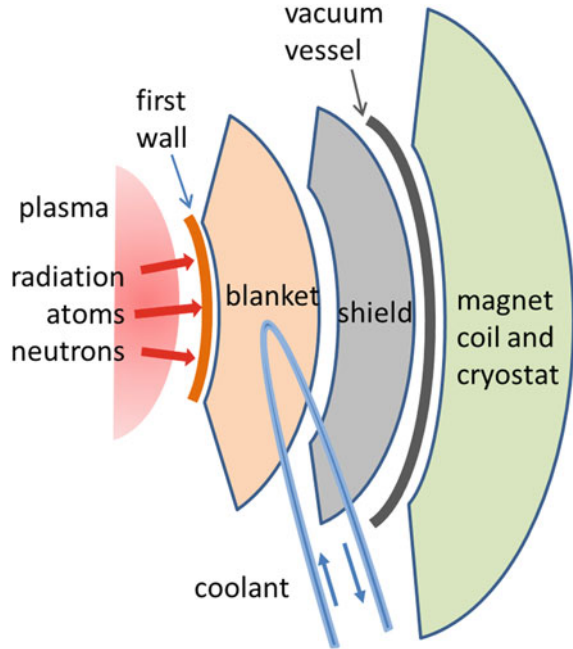
2.3.5 Plasma Guns

Coaxial plasma guns can accelerate plasma blobs to high velocities, pushing through the magnetic field and adding energy and electric current to a plasma. Current drive may be described mathematically as addition of “helicity” to the plasma, where “helicity” is defined in terms of an integral of $\mathbf{A} \cdot \mathbf{B}$ over the plasma volume, \mathbf{B} is the magnetic field, and \mathbf{A} is the magnetic vector potential. Plasma heating and current drive are discussed in [Chap. 5](#).

2.4 First Wall, Blanket, and Shield

Figure 2.6 shows the main elements of a fusion reactor blanket and shield.

Fig. 2.6 A segment of the first wall, blanket, and shield. The blanket contains lithium to breed tritium. The shield must attenuate both neutrons and gamma rays



Much of the fusion energy would be absorbed in the first wall- blanket-shield subsystem, and this energy is carried by coolants to a steam generator (to drive a steam turbine and electrical power generator, Rankine cycle) or by gaseous coolant (He, CO₂, ...) to a gas turbine/generator (Brayton cycle). There will probably be an intermediate coolant loop between the reactor and the steam generator (or gas turbine) to minimize tritium permeation. The efficiency of converting thermal energy into electricity is always less than the Carnot efficiency

$$\eta_c = (T_h - T_c)/T_h, \quad (2.13)$$

where T_h and T_c are the hot and cold temperatures of the coolant. For example, if $T_h = 800$ K and $T_c = 300$ K, then $\eta_c = 62.5$ %. A typical steam cycle can achieve about 64 % of the Carnot efficiency, which would be about 40 % for this case. Thus, it is important to use structural materials in the first wall, blanket and shield that can operate reliably at high temperatures for years.

The first wall-blanket-shield design must cope with many issues simultaneously

- First wall design issues
 - Access ports
 - High heat flux
 - High neutron flux
 - High temperature operation
 - Degradation due to sputtering, heat, stresses, creep and radiation damage.

- Breeding materials
- Tritium breeding, control, inventory
- Coolants
- Structural materials
- Cooling of first wall, blanket, and shield
- Stresses and loss of properties
- Flow rate and pumping power
- Neutronics calculations, including gamma transport
- Energy conversion methods
- Maintenance scheme, including remote handling of radioactive components
- Hundreds of connections for coolant, plasma heating, tritium control, diagnostics, etc.

The first wall must withstand high fluxes of neutrons and heat, radiation damage, creep, swelling, embrittlement, and stresses, yet operate reliably for several years (Merola 2008).

The blanket region behind the first wall of a fusion power plant will probably contain lithium to multiply neutrons via the endothermic ${}^7\text{Li}(n, 2n)$ reaction and to breed tritium by neutron capture in ${}^7\text{Li}$ and ${}^6\text{Li}$ (Table 1.3). Neutron multiplication can also be done by Be, Pb, and PbLi, which is a potential reactor coolant. On the average each neutron from a DT fusion reaction must breed more than one tritium atom, to sustain the fuel cycle. Since some neutrons are absorbed in other materials (such as structure) without breeding tritium and some are lost, a local tritium breeding ratio (TBR) > 1.1 is needed in the lithium-containing regions of the blanket to produce a net TBR > 1 .

A shield outboard of the blanket attenuates neutrons and gamma rays to protect the magnet coils and other systems. The required thickness of a power reactor blanket-shield would probably be in the range 1–1.5 m. Some designs vary this thickness around the torus, using a thinner blanket where the plasma must be close to the coils for good confinement. The shield may be designed to operate at high temperature, to provide structural support for the first wall and blanket, and to last for the lifetime of the plant (~ 60 years).

The majority of the ITER modules will not contain tritium breeding blankets. They will simply contain stainless steel and water to shield the magnet coils from neutrons and gamma rays. It is planned to have some test blanket modules to evaluate their tritium breeding capability. The ITER non-breeding blanket-shield modules are shown in Fig. 2.7.

Each of the 440 ITER blanket-shield modules is comprised of a first wall and a blanket-shield region. Most of the first wall of the main chamber will have

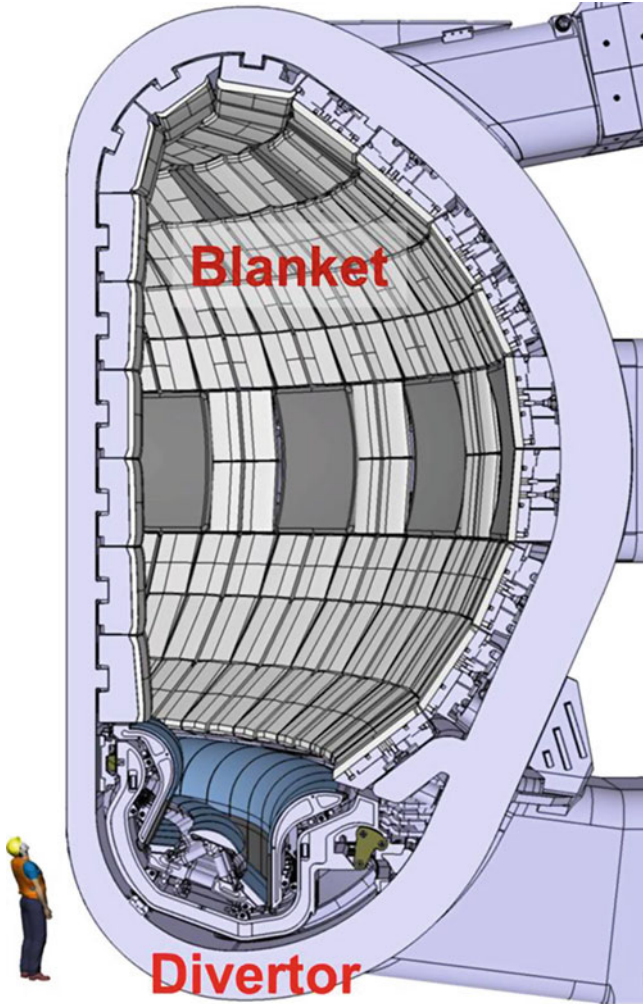


Fig. 2.7 Some of the 440 ITER blanket-shield modules (*rectangular boxes*) and the single-null divertor (the *W-shaped* region at the *bottom*). Courtesy of ITER Organization

beryllium tiles bonded to a copper substrate with stainless steel tubes containing pressurized, low-temperature water coolant, similar to the tiles of Fig. 2.8.

The first wall, blanket, and shield issues are discussed in [Chap. 6](#).

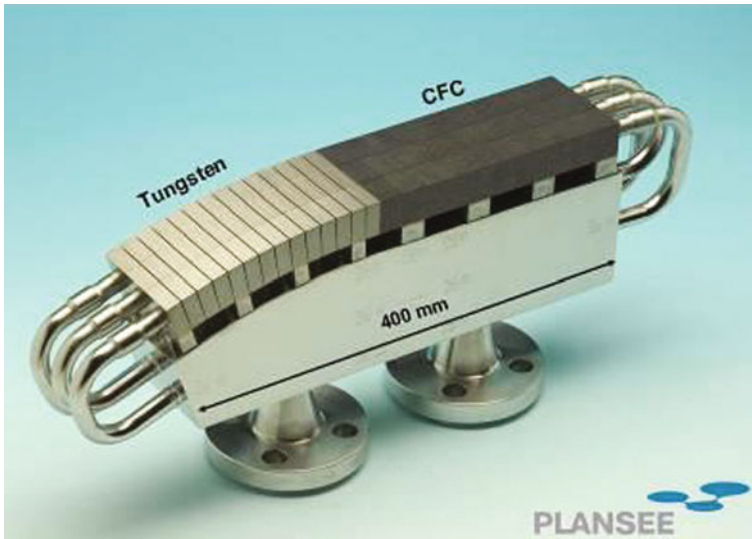


Fig. 2.8 Armor tiles of W and C bonded to copper substrate containing stainless steel coolant tube (Merola 2008)

2.5 Control Systems

The fusion power plant control systems must regulate many processes, including:

- Magnet coils and possible quenches (Chaps. 3, 4)
- Vacuum systems (Chap. 9)
- Cryogenic systems (Chap. 10)
- Plasma density, temperature, fusion power, position, stability, purity (Chap. 7)
 - Plasma diagnostics (Chap. 11)
 - Plasma heating and current drive (Chap. 5)
 - Plasma fueling and gas recycling (Chap. 7)
 - Divertor operation (Chap. 7)
- First wall-blanket-shield heat removal (Chap. 6)
- Tritium flow, recovery from coolant, inventory (Chap. 12)
- Remote handling maintenance systems (Chap. 13)
- Radioactive material inventories (Chap. 13)
- Routine emissions (Chap. 13)
- Accidents (Chap. 13)
- Heat exchangers and steam generators
- Steam turbines and electric generators

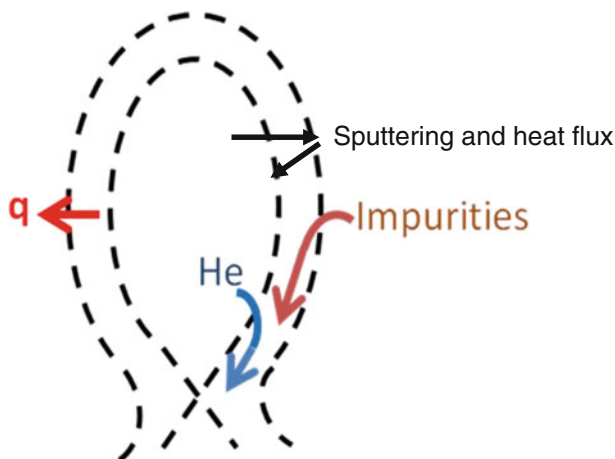


Fig. 2.9 Divertor functions: reduction of heat flux, reduction of sputtering, channeling of helium and sputtered impurities to divertor chamber at the bottom (not shown)

- Electricity switching and distribution.

The last three systems are common to other types of power plants and will not be discussed in this book.

A “divertor” is a region at the bottom or top of the torus where poloidal magnetic field lines lead plasma to be neutralized and pumped away by vacuum pumps (to be discussed in [Chap. 7](#)). The purposes of the divertor are to:

- Reduce the **heat flux** on the first wall of the main chamber by moving much of the heat and particle load to the divertor.
- Reduce **sputtering** by having cooler temperatures near the wall.
- Remove **helium ash** from the outer layers of the plasma, so that it does not build up to high levels and dilute the fuel ion density.
- Prevent **impurity** atoms sputtered from the wall from entering the plasma core.

These are illustrated in [Fig. 2.9](#).

An ITER divertor cassette is shown in [Fig. 2.10](#).

The ITER divertor cassettes are designed for replacement, using a robotic transporter on a removable rail. Other types of magnetic confinement, such as stellarators, will probably also use divertors. Divertor issues are discussed in [Chap. 7](#).

Particles are removed by the divertor and vacuum system and injected by the fueling system and by sputtering.

Plasma fueling may be done by hydrogen gas “puffing”, neutral beam injection, solid fuel pellet injection, or other methods.

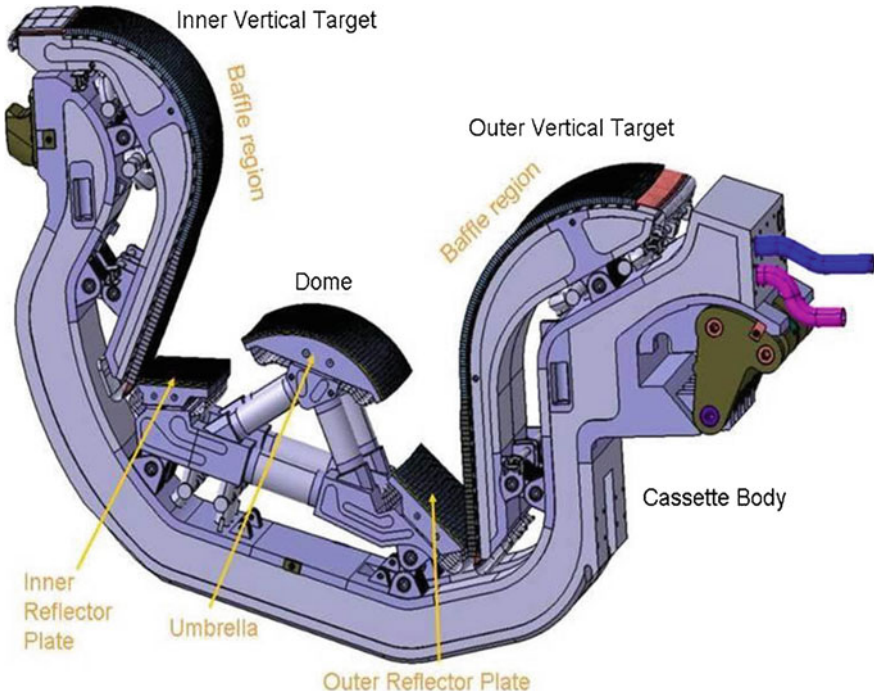


Fig. 2.10 An ITER divertor cassette. The plasma-facing materials in the dome and targets will be water-cooled tiles with tungsten surfaces. Courtesy of ITER Organization

Gas puffing is easiest, but the gas is ionized at the edge of the plasma, the fuel is needed in the plasma core, and the edge density spike may destabilize the plasma. Gas jets injected through nozzles can penetrate slightly better than gas puffs.

Hydrogen atoms adsorbed on the chamber wall or trapped inside the wall may be desorbed during a plasma pulse, causing an additional source of hydrogen ions that may increase the plasma density too much and cause a plasma density limit “disruption” (rapid termination of the tokamak discharge). To avoid excessive hydrogen recycling the walls must be carefully cleaned, heated to about 300 °C to remove most water molecules, and conditioned by preliminary low-density plasma pulses or by coating the walls with special materials, such as titanium or boron. Elastomer (rubber-like) O-rings are not good at high temperatures, so metal gas-gaskets of Cu or Al are widely used.

Plasma blob injection by coaxial plasma guns has been demonstrated in small tokamaks, but for complete fueling of large tokamaks this method would require many rapidly pulsing guns.

NBI can also be used for plasma fueling, but much energy may be expended in accelerating the beam. Therefore, this method limits the attainable Q .

Solid pellets of frozen DT fuel may be injected by a light gas gun or by a rotating arm (like pitching a baseball or cricket ball). Production of the pellets requires a cryogenic system to freeze the DT fuel ($T < 20$ K).

Control systems are discussed in [Chap. 7](#).

2.6 Materials Issues

Fusion power plants need advanced materials for the first wall, divertor surfaces, coolant tubes and insulators, blanket structure, superconducting magnets, and so on. The materials issues include:

- High temperature operation inside the shield
- Surface erosion by particles and photons
- Plasma chamber dust
- Tritium trapping
- Compatibility of coolant with walls and structure
- Stresses
 - Thermal stress
 - Pressure stress
 - Gravity
 - Cyclic stress and fatigue
- Radiation damage
 - Creep
 - Swelling
 - Embrittlement
- Induced radioactivity
- Hydrogen and helium effects on materials
- Fabrication and durability of superconducting materials
- Lifetime of insulators.

It is desirable for all the reactor components to last the lifetime of the power plant (~ 60 years), but the first wall will probably need periodic replacement. It is also desirable to develop advanced materials whose radioactivity decays away to tolerable values in less than 100 years. The leading candidate structural material is reduced activation ferritic or martensitic (RAFM) steel. Silicon carbide composite might be a good candidate for the blanket structure, but it has not been fabricated in large sizes and tested at high temperature under intense neutron irradiation.

Development of materials that can survive 14-MeV neutron bombardment for years will be a challenging problem, especially since no adequate neutron source is yet available. Japan and Europe are collaborating to build the International Fusion

Materials Irradiation Facility (IFMIF), which could test materials under intense neutron irradiation. [Chapter 8](#) deals with Materials Issues.

2.7 Vacuum Systems

Vacuum technology began in the 17th century with experiments on barometers and vacuum pumps by Galileo, Torricelli, Pascal, Von Guericke, and Boyle. In the 20th century industrial development of vacuum tubes for radios, x-ray tubes, oscilloscopes, televisions, radars, and accelerators brought great advances.

Fusion experiments require an ultrahigh vacuum to get rid of impurity atoms that could spoil plasma confinement. ITER will use mainly turbomolecular pumps, roughing pumps, and cryogenic pumps. The technology for vacuum gages, chambers, flanges, valves, windows, and flexible bellows is well developed. Vacuum system designers can calculate conductances of each element, effective pumping speeds, gas flow rates, and the time required to reduce the pressure. The control of tritium is important because of its radioactivity hazard.

Large fusion experiments require complex vacuum systems including all these elements. For example, the ITER vacuum vessel (Fig. 2.11) has many ports for heating, current drive, vacuum pumping, and diagnostics and maintenance.

Elaborate procedures are established for fabricating these large and complex vacuum vessels

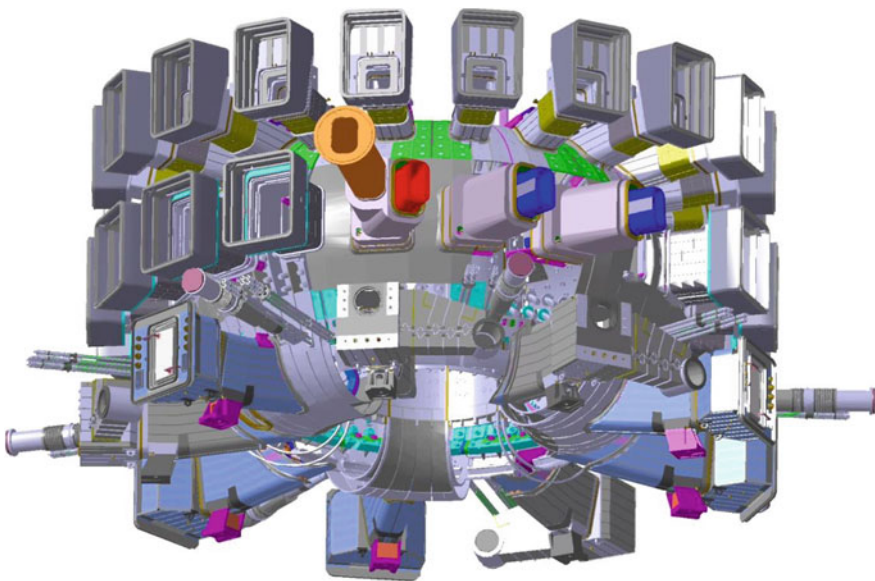


Fig. 2.11 The ITER vacuum vessel ports. Courtesy of ITER Organization

- High quality welding techniques
- Manufacturing components and polishing surfaces to minimize gas trapping
- Cleaning surfaces with degreasing agents, acids or alkalis, deionized water, and alcohol
- Baking the chamber under a vacuum to remove adsorbed water vapor
- Leak testing using a helium gas jet and a mass spectrometer
- Coating the surfaces with special materials, such as lithium, beryllium or boron
- Running low-pressure plasma discharges to further clean the surfaces.

The vacuum chamber for ITER is quite large, as can be seen in Fig. 2.12.



Fig. 2.12 One segment of the ITER vacuum chamber model. Note the person at the *bottom*. Courtesy of ITER Organization

This chamber must be fabricated with great precision, including allowance for dimensional changes due to welds. Vacuum technology is described in [Chap. 9](#).

2.8 Cryogenic Systems

The word “cryo” means “cold”, and “genes” means “that which generates”, so “cryogenics” deals with systems that produce low temperatures. Cryogenic systems are required for many applications, including:

- Industrial gas production
- Food preservation
- Biomedical applications, such as magnetic resonance imaging
- Bearings
- Electronics
- Motors and generators
- Physics research
- Space technology
- High quality vacuum systems
- Magnets
- Electrical power transmission lines.

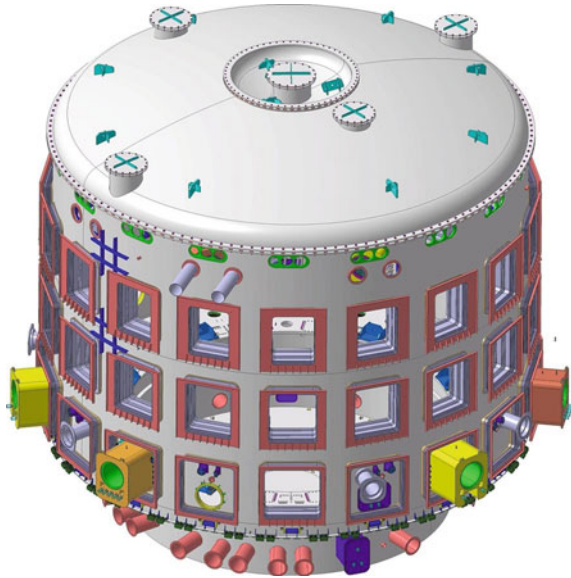
Most superconducting applications operate at $T \sim 4$ K (the boiling point of liquid helium), so cryogenic systems are required to maintain that temperature during operation, typically with liquid helium coolant. Liquid nitrogen ($T \sim 77$ K) may also be needed for high temperature superconductors and staged cooling of helium systems.

Materials properties change at low temperatures. Some materials become brittle, such as a banana peel or a flower cooled by liquid nitrogen.

Cryogenic refrigerators were developed by Kapitza in 1934 and Collins in 1947. About 300 W of input power are required to remove 1 W of heat inflow at $T \sim 4$ K. Modern cryogenic refrigerators using multiple stages of cooling and energy dissipation can operate reliably for many months.

Cryogenic engineers can use established practices and databases of materials properties to design reliable systems, taking into account dimensional changes, heat capacities, thermal conductivities, insulation, conduction, convection,

Fig. 2.13 ITER cryostat.
Courtesy of ITER
Organization



radiation, vapor shielding, flow rates, etc. A large magnet system may take many days to cool down from room temperature to operating temperature.

The massive cryostats surrounding the ITER magnets are shown in Fig. 2.13. Cryogenic systems are discussed in [Chap. 10](#).

2.9 Plasma Diagnostics Systems

Plasma diagnostic systems are needed to measure the following basic plasma parameters (Hutchinson 2002; Hacquin 2008):

A complete four-dimensional space–time mapping of plasma parameters, with spatial resolution of millimeters and accuracies of a few percent, would be desirable, but is not practical to attain at this time. Two or three different techniques may be used for important parameters like electron density and temperature. The agreement of redundant methods provides assurance of their accuracy.

The diagnostic methods may be classified as

- Electric Probes
- Magnetic Probes
- Passive Particle Methods

Table 2.2 Some techniques for measuring electron density

<i>Electron density</i>
Langmuir probe
radiofrequency (RF) conductivity probes
Microwave, far infrared (FIR), and optical interferometers
Microwave cavity resonance
Heavy-ion beam probe
Neutral atom beam probe
Spectroscopy, such as stark broadening
Holographic interferometry
Thomson scattering
Alfven wave and sound wave propagation
Charged particle collectors
Photography

- Active Particle Methods
- Passive Wave Methods
- Active Wave Methods.

Passive methods measure particles or waves emitted by the plasma, and active methods inject particles or waves into the plasma. Table 2.2 illustrates the large variety of methods used for measuring one plasma parameter.

The locations of some ITER diagnostics systems are shown in Fig. 2.14. Improved plasma diagnostics are continually being developed. The quantity and quality of diagnostics for an experiment are limited by the ports in the chamber, by space around the torus for the instruments, and by the available funds. Plasma diagnostic systems are discussed in Chap. 11.

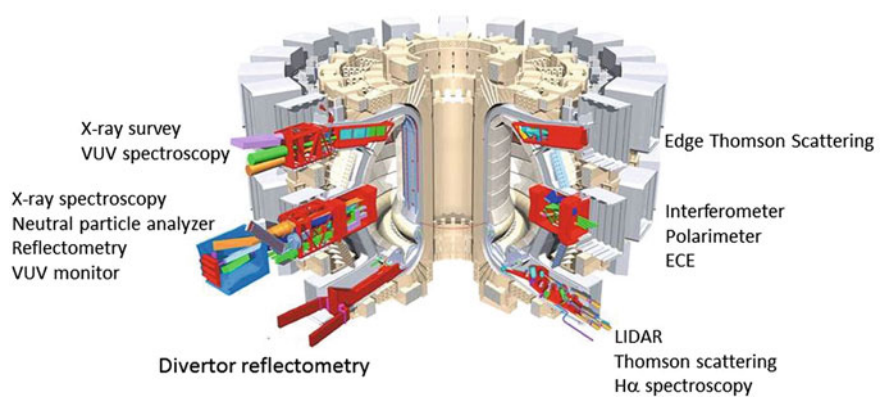


Fig. 2.14 Cutaway view of ITER showing where some diagnostics instruments will be mounted (Hacquain 2008). For simplicity, only a few are labelled. The names will be explained in Chap. 11. Courtesy of ITER Organization

2.10 Safety and Environment

The safety and environmental issues of fusion power include

- Tritium properties, inventory and transport
- Other radioisotope generation and transport
- Routine emissions
- Safety hazards
- Accident scenarios and analyses
- Radiation management approaches
- Materials resources.

Tritium is a radioactive beta emitter, with a half-life of 12.3 years and mean beta energy of about 6 keV. Tritiated water in the human body has a residency half-life of about 10 days, as it is eliminated via sweat, urine, and exhalation. HTO (tritiated water) is more dangerous to humans than gaseous tritium, which disperses more easily and is not so easy to ingest as HTO. The tritium inventory is very difficult to measure and control in a large facility.

Under bombardment by 14 MeV neutrons most structural materials and some coolants will become highly radioactive. The reactor management should calculate the inventories of each radioisotope, manage their safe containment and disposal, ensure that staff are trained in radiation safety, and promote a safety culture in the organization. Special “hot cells” with remote handling equipment will be used to manage the radioactive components.

There will be constant low-level emissions of gases and particulates from the reactor building, usually through a filtered stack. The plant operators will monitor the stack emissions to ensure that they stay within regulatory safe limits. For example, tritium emissions are carefully monitored at CANDU heavy water fission power plants.

Fusion power plant potential safety hazards include

- Fire, such as possible lithium/water reactions
- Explosion (hydrogen)
- Earthquake
- Flood
- People falling from elevated areas
- Electrical shock
- Eye hazards like breaking glass
- Toxic gases
- High magnetic fields
- Magnet quench arcing and pressure

- Severe plasma disruptions
- Structural fatigue and failures
- Exposure to intense electromagnetic waves
- Exposure to ionizing radiation
- Accidental release of tritium
- Accumulation of frozen oxygen in cryogenic systems.

Fusion power plant design studies consider potential accident scenarios, such as rupture of a high-pressure coolant tube, and estimate the possible consequences to the plant, to plant workers, and to the public offsite. It is highly desirable to ensure that no offsite evacuation would be required, even in the case of a severe accident, to gain public acceptance.

The power plant design would include plans for decommissioning, decontamination, and disposal of radioactive materials. Possible shortages of key materials, such as helium and niobium, should be taken into account when planning for deployment of hundreds of fusion power plants.

Safety and environmental issues will be discussed in [Chap. 12](#).

2.11 Power Plant Designs

Power plant design issues include

- Criteria for attractive power plants
- Reliability, availability, and maintenance issues
- Economics estimates
- Fusion-fission hybrids
- Design studies in various countries.

Electric power utility companies have many criteria for selection of power plants to build, including:

- Output of power plant, MWe
- Capital cost per kW
- Cost of electricity, mills/KWh
- Length and rating of required transmission lines to cities and industries
- Size and flexibility of connected grid system
- Location away from areas of high seismic activity and flood danger
- Availability of fuel and its transportation systems (tritium is bred on-site)
- Fuel, operations and maintenance costs
- Feasibility of shipping large components, such as pressure vessels
- Acceptance by local population
- Security requirements and costs

- Availability and reliability of power plant
- Ease and speed of maintenance
- Waste disposal
- Staff numbers, skills, and costs
- Government laws and regulations
- Possible delays due to licensing and opposition.

The availability of fusion power plants is a difficult issue. Current experiments have many equipment failures because they are pushing their performance boundaries, and some components were not designed to be highly reliable or maintainable. This low overall availability of experiments would be unacceptable to utilities. The availability of power plants can be improved by designing and testing highly reliable components and by providing redundant components. The components must be designed for quick efficient repair or replacement. Repair of large power core subsystems, such as the superconducting magnets or vacuum vessel, would be very difficult and time consuming to accomplish, so they should last for the lifetime of the plant. Remote handling will be required for highly radioactive components, such as the first wall and blanket. All power core and hot cell subsystems must be capable of being maintained and replaceable with remote handling equipment.

We calculate the cost of each plant component or subsystem to estimate the total capital cost of the plant. This estimate is escalated according to the assumed interest rate and inflation rate to get an effective annualized capital cost. This is added to the annual cost of the fuel cycle and the operating and maintenance cost, and the sum is divided by the estimated annual net energy output (MWe, including down time for maintenance and equipment failures) to estimate the cost of electricity (COE) in units of cost per kWh for the country of operation (cents/KWh, Yen/kWh, etc.) Design studies for 1,000 MWe fusion power plants, assuming 70 % availability, typically estimate that the COE from fusion power would be higher than the COE from fission or coal plants. This is due mostly to the high capital cost and uncertain availability of the fusion reactors. The fusion fuel costs would be very low, while the operating and maintenance costs could be comparable to those of other power plants. The capital costs might be reduced by more compact designs, or by developing alternative confinement concepts.

Fusion reactors have a strong economy of scale, which means that a 3,000 MWe power plant would have a much lower COE than a 300 MWe plant, but utilities may not wish to build very large power plants (Dolan 1993).

Studies of hypothetical fusion power plants have been done in several countries (Europe, USA, Japan, China, ...), and their results are discussed in [Chap. 13](#) (ARIES Team; Dolan et al. 2005).

In addition to electrical power generation, fusion reactors could also be used to

- Produce hydrogen and other fuels for transportation and industry
- Desalinate seawater, which could alleviate water shortages that cause international strife

- Provide heat for industrial processes, such as distillation of alcohol and mining oil shale and tar sands
- “Incinerate” radioactive wastes, which are a major barrier to public acceptance of nuclear fission power plants
- Breed fuel for fission power plants, which could prolong the capability of fission power plants to meet world demand.

2.12 Fusion-Fission Hybrids

Fusion-fission hybrids would use uranium or thorium in the fusion reactor blankets, in addition to lithium. The blanket could be optimized either to produce more heat and electricity from the hybrid plant, to “incinerate” radioactive wastes, or to breed fissile fuel (^{239}Pu or ^{233}U) for use in satellite fission reactors. Such hybrids could improve fusion power economics (Chap. 14).

2.13 Problems

- 2.1. A central solenoid with area $= 0.9 \text{ m}^2$ is pulsed at 8 T/s. What electric field is induced in the plasma at $R = 3 \text{ m}$? If the plasma resistivity is twice that of copper, what average current density J is induced in the plasma? (Use $E \approx \eta J$) If the plasma minor radius is 0.3 m, estimate the total plasma current.
- 2.2. A tokamak with $R = 1.5 \text{ m}$, coil radius $a_c = 0.7 \text{ m}$, is to have $B = 2.2 \text{ T}$, provided by 16 copper coils with $J = 9 \text{ MA/m}^2$. What are the required current per coil, cross sectional area of each coil, and total power dissipated?
- 2.3. If the central magnetic field of ITER is 5 T, what wave frequency (MHz) would be needed to heat deuterons there using the ion cyclotron resonance?
- 2.4. If a steam system can achieve 60 % of the Carnot efficiency and the cold temperature is 30 °C, what hot temperature (K) would be required to achieve a thermal-to-electrical conversion efficiency of 39 %?

2.14 Review Questions

1. What do the parameters in the following equation represent? How is it related to the toroidal field of a tokamak? $B = \mu_0 NI / 2\pi R$.
2. What do the parameters in the following equation represent? $P_c = \eta J^2 V_c$.
3. About what temperatures are required for the superconductors used widely now?

4. Name four methods of plasma heating.
5. Sketch a tokamak, showing the toroidal field coils, central solenoid, toroidal field, plasma current, and poloidal field.
6. Why is ohmic heating ineffective at high temperatures?
7. For what type of confinement system could charged particle injection be used? For what type would it not succeed?
8. Sketch a neutral beam injector and explain how it works.
9. What NBI energies would be needed for large plasmas, such as ITER?
10. What frequencies would be good for plasma heating by electromagnetic waves?
11. What do the following equation and its parameters represent? $\eta_c = (T_h - T_c)/T_h$.
12. What materials make good neutron multipliers?
13. What thickness is required for the blanket-shield region?
14. What structural material and coolant are used for most places in ITER?
15. What are the functions of a divertor?
16. What materials are used for plasma-facing components in the ITER divertor?
17. What fueling methods can give good penetration into the plasma?
18. What is the leading candidate structural material for a fusion reactor?
19. What special materials may be used to coat the walls of a tokamak?
20. What does “cryogenics” mean, and why are such systems required for fusion research?
21. About how much power is required to remove 1 W of heat at 4 K?
22. What are the six categories of plasma diagnostic methods?
23. What factors limit the number of diagnostic systems on a large experiment?
24. What are the half-life and mean beta energy of tritium?
25. What materials limitations should be considered when planning for hundreds of fusion reactors?
26. Why would the cost of electricity from fusion power plants probably be higher than from fission or coal power plants?
27. What is “economy of scale”?
28. What are three possible applications of fusion reactors, in addition to generation of electricity?
29. What are possible uses of “fusion-fission hybrids”?

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