ITER TOKAMAK DEVICE
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CONTRIBUTORS

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FOREWORD

Development of nuclear fusion as a practical energy source could provide great benefits. This fact has been widely recognized and fusion research has enjoyed a level of international co-operation unusual in other scientific areas. From its inception, the International Atomic Energy Agency has actively promoted the international exchange of fusion information.

In this context, the IAEA responded in 1986 to calls for expansion of international co-operation in fusion energy development expressed at summit meetings of governmental leaders. At the invitation of the Director General there was a series of meetings in Vienna during 1987, at which representatives of the world’s four major fusion programmes developed a detailed proposal for a joint venture called International Thermonuclear Experimental Reactor (ITER) Conceptual Design Activities (CDA). The Director General then invited each interested party to co-operate in the CDA in accordance with the Terms of Reference that had been worked out. All four Parties accepted this invitation.

The ITER CDA, under the auspices of the IAEA, began in April 1988 and were successfully completed in December 1990. This work included two phases, the definition phase and the design phase. In 1988 the first phase produced a concept with a consistent set of technical characteristics and preliminary plans for co-ordinated R&D in support of ITER. The design phase produced a conceptual design, a description of site requirements, and preliminary construction schedule and cost estimate, as well as an ITER R&D plan.

The information produced within the CDA has been made available for the ITER Parties to use either in their own programme or as part of an international collaboration.

As part of its support of ITER, the IAEA is pleased to publish the documents that summarize the results of the Conceptual Design Activities.
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I. INTRODUCTION

The ITER device, auxiliary systems, and facility and the associated ITER operation and research programme will make it possible to demonstrate the physics performance and technologies essential to fusion reactors. This report summarizes the considerations that lead to the ITER design and presents a general description of the tokamak reactor, supporting systems and facility. The emphasis is on the integration of the various sub-systems into a feasible engineering design that can meet the ITER experimental goals. The plan of how ITER and the ITER facility will be used to accomplish the ITER programme objectives is described in the ITER Conceptual Design Report[1].
II. DESIGN BASIS

The design parameters of ITER reflect both a distillation of the world's tokamak database and a comprehensive set of technical analyses that have been conducted by the ITER team over the past three years. The key features and parameters of the design as presented in Figs. II-1, II-2 and Table II-1 are easily traceable to the technical objectives of the ITER programme.

TABLE II-1 ITER PLASMA PARAMETERS AND PERFORMANCE

<table>
<thead>
<tr>
<th>Nominal Fusion Power, GW</th>
<th>1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operating characteristics:</td>
<td></td>
</tr>
<tr>
<td>Pulse Length, (s)</td>
<td>&gt;200 to continuous</td>
</tr>
<tr>
<td>Energy multiplication, (Q)</td>
<td>&gt;5 to infinity</td>
</tr>
<tr>
<td>Plasma major radius, (m)</td>
<td>6.0</td>
</tr>
<tr>
<td>Plasma half width, mid-plane, (m)</td>
<td>2.15</td>
</tr>
<tr>
<td>Nominal maximum plasma current, (MA)</td>
<td>22</td>
</tr>
<tr>
<td>Toroidal field, on axis, (T)</td>
<td>4.85</td>
</tr>
<tr>
<td>Toroidal coil, outer radius, (m)</td>
<td>11.5</td>
</tr>
</tbody>
</table>

The confinement requirement for plasma ignition at a density $10^{20} \text{m}^{-3}$, at an effective impurity level, $Z_{\text{eff}} = 1.7$, and at a helium concentration up to 10%, sets the needed plasma current, 22 MA, and mandates a divertor plasma configuration. Considerations of confinement optimization and plasma stability at high beta and control of axisymmetric instabilities respectively, establish design requirements for an edge safety factor, $q_{\text{C}} = 3$, and plasma elongation, $k (95) = 2.0$.

The requirement for a sustained fusion burn and the possibility of steady-state operation mandates the use of superconducting magnet technology. The plasma current, safety factor, and elongation requirements, when combined with the need for a 200 second inductively-driven burn, foreseeable performance of superconducting magnet materials, and the thickness of nuclear shielding needed to protect the magnets, establish the overall device size of 6.00 meter major radius and 2.15 meter minor radius.

The requirement for steady-state operation mandates a current drive scheme, which for ITER is provided by a combination of tangential neutral beam injection (NBI), with 1.3 MeV negative-ion beams, and lower hybrid (LH)
Fig. II-1. The ITER Basic Device
Fig. II-2. Basic Device Reference Layout (Equatorial Plan View)
current drive. Electron cyclotron (ECH) waves are also provided for current profile control and disruption avoidance/control. An alternative employing ion cyclotron waves for heating and central current drive is possible if favorable experimental results are realized.

The need for ITER to be nearly self-sufficient with regard to tritium (breeding ratio > 0.8) establishes the need for a segmented in-vessel breeding blanket. Maintenance requirements for the blanket segments, plus access requirements for the heating and current drive systems, for insertion of nuclear testing modules and for exhaust of D-T and helium from the divertor, combine to establish the device layout, major access port locations and PF coil configuration.

Requirements for alpha-particle confinement establish the allowable TF field ripple and, when combined with requirements on the location of the outboard PF coils imposed by the need to accommodate a range of operational conditions (plasma pressure and current profile), set the number of toroidal field (TF) coils and the location of the outboard leg of the TF coil.
III. TOKAMAK, AUXILIARY SYSTEM AND FACILITY OVERVIEW.

The equipment necessary to conduct the ITER operation and research programme is divided into three categories: 1) the Tokamak Reactor Device, where the D-T plasma fuel is confined and where the fusion reaction takes place; 2) the Auxiliary Systems, which support operation of the tokamak device and initiate and control the fusion reaction, and 3) the Facility, in which the tokamak and auxiliary systems are assembled, operated and, as required, maintained and repaired.

The demarcations among these three categories are somewhat arbitrary, but, for purposes of understanding the ITER concept, are best defined by two key physical boundaries, the tokamak device cryostat and the nuclear-shielded reactor and auxiliary systems cells. The major ITER systems located within the three regions so defined are shown schematically in Table III-1.

TABLE III-1. MAJOR ITER SYSTEMS

<table>
<thead>
<tr>
<th>Tokamak Device</th>
<th>Auxiliary Systems</th>
<th>Facility/Maintenance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cryostat</td>
<td>NB Injection</td>
<td>Reactor Cell</td>
</tr>
<tr>
<td>TF Magnets</td>
<td>Lower Hybrid rf</td>
<td>Aux. Sys. Sub-cells</td>
</tr>
<tr>
<td>Torus Vessel</td>
<td>Vacuum Pumping</td>
<td>Cryo. Refrigeration</td>
</tr>
<tr>
<td>First Wall</td>
<td>Tritium Recovery</td>
<td>Magnet Power</td>
</tr>
<tr>
<td>Divertor</td>
<td>Primary Cooling</td>
<td>NBI, LH, EC Power</td>
</tr>
<tr>
<td>Trit. Breeding</td>
<td>Plasma Diagnostics</td>
<td>Plant Water</td>
</tr>
<tr>
<td>Blanket</td>
<td>In-vess. Maint. Eq</td>
<td>AC Power Dist.</td>
</tr>
<tr>
<td>In-vess. Shield</td>
<td></td>
<td>Control/Data Acquis.</td>
</tr>
<tr>
<td></td>
<td>Decon./Repair/Disp.</td>
<td></td>
</tr>
</tbody>
</table>

Test Modules

A brief and somewhat non-technical description of these systems and how they contribute to ITER operation is given below. More detailed technical information about the major ITER sub-systems are given in later sections of this report. More detail about the Facility is covered in the ITER Plant Report[2].

III.1. TOKAMAK DEVICE.

The ITER tokamak device is the fusion reactor core within which the high temperature plasma used to produce thermonuclear power is confined. The key components of this reactor core are a primary, or torus vacuum vessel, and superconducting toroidal field (TF) and poloidal field (PF) magnet systems.
These systems are in turn located within an evacuated cryostat, or secondary vacuum vessel, that allows the TF and PF magnet systems and their supporting mechanical structure to be cooled to liquid-helium temperatures, 4.5 K. At these temperatures, the magnet windings are capable of producing the intense magnetic fields, up to 4.85 Tesla, needed for stable confinement of the plasma.

Four major in-vessel systems needed for plasma operation (in-vessel nuclear shielding, tritium breeding modules, the plasma-facing first wall, and the plasma divertor plates) are all enclosed within the torus vacuum vessel. The torus vessel establishes the high-vacuum conditions needed for the fusion reaction, and is also the primary containment for the gaseous D-T fuel from which the plasma is formed.

In the plasma confinement region defined by the in-vessel systems, the low-density (10⁻⁷ atmosphere) D-T fuel, initially injected in gaseous and frozen pellet form, is first ionized, and then heated to temperatures of up to 10 keV (10⁸ K). Under these conditions, a self-sustaining fusion reaction begins, in which the fusion energy produced is sufficient to maintain the plasma temperature.

The magnetic fields provided by the TF and PF magnet systems are insufficient in themselves to provide stable confinement of the plasma pressure and energy. The necessary final increment of magnetic field is provided by current flow in the plasma itself, up 22 MA under normal ITER operating conditions. This current, initially induced in the plasma by the PF magnet system, is subsequently sustained and controlled with the assistance of the plasma heating and current drive systems described below.

Under typical operating conditions, the plasma produces approximately 1000 MW of fusion power, 800 MW in high-energy neutrons, and 200 MW in alpha (α) particles (high-energy helium ions). The α-particle heat is absorbed by the plasma, sustaining its temperature, and ultimately flows to the plasma boundary. Here it flows, in part, to the divertor plates, located at the top and bottom of the torus vessel, and, in part, radiates to the first-wall, which is formed by special tiles located on the plasma-facing surfaces of the blanket/shield modules. The 200 MW of heat absorbed in the divertor and first wall are then removed from the torus and cryostat by primary cooling water.

The neutron power produced by the plasma is absorbed by the in-vessel nuclear shield and tritium-breeding blanket segments. The breeding blanket contain lithium, and the neutrons absorbed in the lithium breed tritium that is ultimately recovered and used to fuel the plasma. Under optimum conditions, ITER can breed over 80% of the tritium fuel it requires to operate.

The in-vessel shield, breeding blanket and vacuum vessel attenuate the neutron and gamma radiation flux reaching the TF and PF coils and ensure that these critical components remain at the low temperatures required for their operation.
III.2. AUXILIARY SYSTEMS.

The tokamak device requires nine auxiliary systems to support its operation. Seven of these systems provide direct support needed to initiate and sustain the fusion reaction in the plasma. The two remaining systems (plasma diagnostics and the in-vessel remote maintenance system) respectively provide diagnostic data for control and study of the reacting plasma, and a means for servicing components within the torus between plasma operation periods. The use of the in-vessel maintenance system is described in the Maintenance Section V.6. that follows.

Three of the auxiliary systems contribute to heating the plasma to thermonuclear temperatures and sustaining or controlling the toroidal plasma current that is essential for tokamak operation. The nine Neutral Beam Injection (NBI) modules provide the majority of the plasma heating and current drive power. Each negative-ion beamline module produces up to 10 MW of energetic neutral deuterium and/or tritium atoms with energies of up to 1.3 MeV. These energetic atoms, injected through tangential access ports into the torus, heat the plasma to fusion temperatures, and in addition drive part of the plasma current. This NBI-driven current is localized in the center of the plasma.

The heating and current drive power provided by the NBI modules is supplemented by two radio-frequency (rf) systems: a 50 MW, 5 GHz lower hybrid (LH) system, and a 20 MW, 120 GHz electron cyclotron (EC) system. These systems, which occupy radial ports at the torus midplane, use specialized antenna structures to couple rf waves into the plasma. The LH waves are directed to drive current in the plasma edge, and also contribute to the overall plasma heating. The EC waves are directed to penetrate to the surface q=2, and are used for fine control of the plasma current profile and suppression of magnetic instabilities (disruptions) that would otherwise abruptly terminate the plasma current and the fusion reaction.

Three other auxiliary systems provide support for plasma fueling and fuel processing. These systems inject, exhaust and recover the D-T plasma fuel and maintain the high vacuum environment needed to sustain the reacting plasma. The D-T fuel is injected in gaseous and solid (D-T ice) form by gas and ice pellet injectors. Unburned D-T fuel and the helium ash from the fusion reaction are exhausted from the plasma by the vacuum pumping system, which is connected to the pumping ducts located just outboard of the divertor plates. The vacuum pumping system also establishes the ultra-high vacuum and low-impurity (water and oxygen) conditions needed to prevent impurities from quenching the fusion reaction.

A tritium-recovery system removes and recovers the tritium bred in the blanket-shield modules located within the torus, and also recovers unburned tritium from the vacuum exhaust stream and blanket/first wall/divertor primary cooling circuits.

The final auxiliary system needed for plasma operation support is the blanket / shield / divertor primary cooling system, which provides and circulates cooling water for heat removal from the in-vessel blanket, shield and divertor.
The plasma diagnostic systems provide specialized measurements of the plasma and torus operating conditions. Key measurements include plasma current, size and magnetic configuration, plasma temperature, density and composition, thermal and neutron radiation levels, and monitoring of the operational parameters of the in-vessel systems.

III.3. FACILITY AND MAINTENANCE.

The balance of the equipment needed for ITER operation and the specialized equipment and work areas needed to enclose, maintain and repair the tokamak device and auxiliary system components are contained within the ITER Facility shown in Fig. III-1 and Fig. III-2.

The cryostat, tokamak device, and the reactor-support auxiliary systems are located within thick-walled, hermetically-sealed portions (cells) of the Tokamak Building. These reactor and auxiliary-system cells serve as primary biological, or personnel protection shields to attenuate neutron and gamma radiation not absorbed by the tokamak device. They also serve as back-up containment barriers for any tritium or fusion activation products accidentally released from the reactor or auxiliary systems.

The sub-divided arrangement of the reactor and auxiliary system cells minimizes migration of contamination, and also facilitates simultaneous servicing of the respective components between ITER operation periods.

A number of more conventional plant systems also support ITER reactor operations. These non-nuclear systems include:

(1) a 100 kW cryogenic refrigeration plant to provide liquid helium for cooling the tokamak magnet systems,
(2) direct-current power supplies for the TF and PF magnets,
(3) specialized direct-current and rf power supplies for the NBI, LH and EC plasma current-drive/heating systems,
(4) a primary AC electrical power distribution system, and
(5) a water-cooling plant that provides cooling water for the secondary (non-tritium) circuits of the in-vessel cooling and heat removal systems, and for cooling of the cryo-refrigeration plant and DC and rf power supplies.

The total AC power demand and heat disposal during typical ITER operation are respectively about 700 MW and 1800 MW.

A facility-wide control system monitors and coordinates operation of the systems throughout the facility and also provides the means by which scientific and engineering data obtained during ITER operations is evaluated and archived.

The ITER facility and the arrangement of the tokamak device and auxiliary systems within the facility are configured to support maintenance and repair of these systems throughout the course of ITER operations. Once
Level R

Fig. III-1. Reactor Building Plan View
Fig. III-2. Reactor Building Elevation view
operations with D-T fuel commence, the entire tokamak device, and to a lesser extent the auxiliary systems, will become activated, and remote maintenance procedures become mandatory.

Within the torus vessel, this maintenance will be done with a pair of in-vessel manipulators and a pair of in-vessel transport vehicles. These manipulators and vehicles, equipped with specialized handling fixtures and remotely operable tools, will be used to remove and re-install in-vessel components such as first wall tiles and divertor plates. These maintenance operations are planned as a routine part of ITER operation.

Remote handling facilities are also provided for ex-vessel maintenance in the reactor and auxiliary systems cells. In the tokamak cell, the maintenance will be done using specialized manipulators for small components, and a remote heavy-lift (800 tonne) crane capacity for the more massive tokamak components. A system of airlocks and transportation devices is provided for removing activated components from the tokamak cell and transporting them to the decontamination and repair or disposal facilities. Other nuclear-shielded sub-cells in the facility (e.g., the NBI cell and tritium cell) are equipped with similar remote-handling cranes, manipulators and transport devices.

Components removed from the reactor and auxiliary cells will be transferred to specialized areas within the ITER facility for decontamination and subsequent inspection, repair or disposal.

The designs of the ITER tokamak device and auxiliary systems are specified with safety[3], maintainability and repairability as a fundamental requirement.
IV. ITER TOKAMAK DEVICE CONCEPT AND LAYOUT.

IV.1 INTEGRATION

The tokamak device is a large tritium burning machine. The main parameters and requirements of the reactor sub-systems are summarized in Table IV-1. The device configuration and major components are shown in Fig. IV-1.

The tokamak design has been made as compact as is reasonable while meeting the system requirements. The inboard legs of the TF coils have as large a radius as possible, to meet shielding requirements, to maximize the space available for the central solenoid and to maximize the PF flux-swing for burn. To minimize the PF ampere-turn and power requirements and the overall device radius, the outboard legs of the TF coils are kept at as small a radius as possible consistent with the needs for radial access, nuclear shielding requirements, and a maximum TF ripple at the plasma edge of 2.5%.

The resulting horizontal bore in the TF magnets is about 7 meters and the vertical bore is about 15 meters. The peak field in the TF windings is just over 11 Tesla and the magnet design and structural parameters are tractable.

The number and locations of coils in the PF system have been optimized. The divertor coil is positioned as close to the plasma as possible while satisfying the requirements on the distance between the divertor plate and the plasma X-point, requirements for shielding and for structure around the TF coil. The plasma in this configuration is quite close to the outer PF coils, so that the required currents for plasma control are moderate and the corresponding out-of-plane loads on the TF coils are structurally tractable.

The central solenoid and winding pack are designed to achieve maximum volt-second capability and to satisfy the constraints for structural and superconductor performance and quench protection. The result is a PF system with a reasonable margin to satisfy the requirements for volt-seconds, burn duration and operational flexibility.

The torus vacuum vessel concept provides a feasible solution for the primary-vacuum and tritium-containment boundaries and provides sufficient margins against the mechanical stresses from the reference plasma disruption conditions. The reference vacuum vessel is toroidally segmented, with electrically insulating structural connections. These connections are located in the cryostat vacuum space and sealed from the primary vacuum by resistive elements. The toroidal resistance is 20 $\mu$Ohm. The option of a light weight vessel with the same toroidal resistance is being maintained.

The vacuum vessel and magnet systems are housed entirely inside the cryostat vessel. This is a fully-metallic, welded design, with reinforcing ribs supporting a thin-shell body. It incorporate four local high resistive metallic inserts into a cylindrical wall which increase cryostat toroidal resistance. The effective thickness is about 10 mm and the stray field produced by eddy currents during start-up appear acceptably low.

The first-wall/blanket/shield is an integrated structure. Austenitic stainless steel (316) is the structural material. During the Physics Phase, the
TABLE IV-1. KEY ITER DESIGN PARAMETERS AND REQUIREMENTS (PHYSICS PHASE).

<table>
<thead>
<tr>
<th>Magnets</th>
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</thead>
<tbody>
<tr>
<td>PF system flux swing, (V-s)</td>
<td>325</td>
</tr>
<tr>
<td>Plasma Major radius, (m)</td>
<td>6.0</td>
</tr>
<tr>
<td>Plasma Minor radius, (m)</td>
<td>2.15</td>
</tr>
<tr>
<td>Elongation, k(95 %)</td>
<td>1.98</td>
</tr>
<tr>
<td>Safety factor, q (95 %)</td>
<td>3.0</td>
</tr>
<tr>
<td>Plasma current, (MA)</td>
<td>22</td>
</tr>
<tr>
<td>Field at plasma, (T)</td>
<td>4.85</td>
</tr>
<tr>
<td>TF plasma edge ripple, (%)</td>
<td>2.5</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>First Wall</th>
<th></th>
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<tbody>
<tr>
<td>Average thermal Load, (MW/m²)</td>
<td>0.15</td>
</tr>
<tr>
<td>Average neutron Load, (MW/m²)</td>
<td>1.1</td>
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<table>
<thead>
<tr>
<th>Divertor</th>
<th></th>
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</thead>
<tbody>
<tr>
<td>Peak Heat Flux, (MW/m²)</td>
<td>15</td>
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<table>
<thead>
<tr>
<th>Containment Structures</th>
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<tbody>
<tr>
<td>V.V. Toroidal Resistance, (µΩ)</td>
<td>20</td>
</tr>
<tr>
<td>Design Overpressure</td>
<td></td>
</tr>
<tr>
<td>Internal, (bar-abs)</td>
<td>2</td>
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<tr>
<td>External, (bar-abs)</td>
<td>1.5</td>
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</table>

<table>
<thead>
<tr>
<th>Heating and Current Drive</th>
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<tr>
<td>Neutral Beams D&amp;T</td>
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</tr>
<tr>
<td>Power, (MW)</td>
<td>75</td>
</tr>
<tr>
<td>Energy, (MeV)</td>
<td>1.3</td>
</tr>
<tr>
<td>Lower Hybrid RF</td>
<td></td>
</tr>
<tr>
<td>Power, (MW)</td>
<td>50</td>
</tr>
<tr>
<td>Frequency, (GHz)</td>
<td>5</td>
</tr>
<tr>
<td>Electron Cyclotron RF</td>
<td></td>
</tr>
<tr>
<td>Power, (MW)</td>
<td>20</td>
</tr>
<tr>
<td>Frequency, (GHz)</td>
<td>120</td>
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<table>
<thead>
<tr>
<th>Blanket/Shield</th>
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<tbody>
<tr>
<td>Blanket Tritium Breeding Ratio</td>
<td>.8</td>
</tr>
<tr>
<td>Shield</td>
<td></td>
</tr>
<tr>
<td>Dose in Reactor Room 24 hrs after shutdown, (mrem/h)</td>
<td>0.5</td>
</tr>
<tr>
<td>Insulator Irradiation, (rad)</td>
<td>&lt;3 x 10^9</td>
</tr>
<tr>
<td>Max. magnet nuc. htg.,(kW)</td>
<td>60</td>
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<table>
<thead>
<tr>
<th>Fuelling/Vacuum</th>
<th></th>
</tr>
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<tbody>
<tr>
<td>Pellet Fuelling Rate, (moles/hr)</td>
<td>75</td>
</tr>
<tr>
<td>Torus Operating Pressure, (mbar)</td>
<td>4 x 10^-7</td>
</tr>
<tr>
<td>Primary Vacuum Pump Speed, (m³/s)</td>
<td>1500</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Maintenance</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>All reactor systems remotely maintainable</td>
<td></td>
</tr>
<tr>
<td>Maximum radiation dose for re-weld of Stainless Steel, (He - appm)</td>
<td>0.1</td>
</tr>
</tbody>
</table>
Fig. IV-1. Basic Device Reference Layout (Elevation View)
The entire first wall will be protected by carbon-fiber-composite tiles, attached mechanically to the first wall, and cooled by radiation and/or conduction to the structure. Most of tiles will be deleted in the Technology Phase substituted by the bare or tungsten coated stainless steel wall. The blanket/shield design provides cooling to the first wall using poloidal flow paths in the inboard sections and toroidal flow paths in the outboard sections. The coolant is water at a temperature less than 100°C.

The same blanket and shield components may be used for both the Physics and Technology Phases. Three breeding material options have been investigated, and a lithium ceramic (solid breeder) concept was selected as the "first option" with the lithium-lead breeder blanket concept as an alternate.

The blanket design is highly segmented. The inboard blanket is electrically divided into 32 segments; each segment is further divided into three subsegments to minimize disruption-induced electromagnetic effects. The outboard blanket is divided into 64 segments of two types: 32 poloidal segments placed on either side of an equatorial port, and 32 shorter segments placed above and below the port. The outboard blanket/shield segments also incorporate passive current loops for stabilization of the plasma vertical position.

The inboard shield design features a steel/water body backed by a thin layer of Pb/B₄C. The layer is attached to the vacuum vessel, which provides additional shielding. Thinner shielding is provided behind the divertor and near the NBI duct on the back side of the outboard TF leg. The thickness of the outboard blanket/shield/vacuum vessel is enough to permit personnel access outside the tokamak 24 hours after shutdown, provided extra shielding is placed around penetrations.

The shape of the divertor plate is optimized to minimize infringement on the shielding space behind it. The inclination of the divertor surface to the separatrix is 15 degrees at the outboard strike point and 45 degrees at the inboard strike point. The poloidal distances from the X-point to the strike points are respectively 1.4 meters and 0.6 meters.

The divertor plate design features armor brazed to poloidally-oriented cooling tubes. In the Physics Phase, the armor will be carbon based material. The divertor will be cooled by high-velocity (10 m/s) water. To accommodate maintenance and assembly procedures, the divertor will be segmented toroidally into two modules per sector, an alternative with poloidal segmentation is also considered.

The tokamak device design provides 16 large equatorial ports. These ports are occupied by the plasma heating and current drive systems, plasma diagnostics, plasma fueling, maintenance equipment and, nuclear test modules.

Maintenance of all components in the tokamak device will be possible using fully remote methods. Remote maintenance capability will also extend to the reactor components where replacement during the machine lifetime is not expected. The maintenance approach and time required for completion will, however, vary with the type of component.

Regular maintenance and/or replacement of the in-vessel plasma-facing components is planned. This type of maintenance operation will be
accomplished without warm-up of the magnet systems by using in-vessel transporters and manipulators inserted through four equatorial ports disposed at 90 degree azimuths around the torus. These in-vessel operations will be performed in an inert-gas atmosphere to preclude contamination of the first wall.

It may also be necessary to replace entire blanket segments. Removal and replacement of blanket segments will be accomplished through the vertical access ports at the top of the tokamak using dedicated handling devices from above the reactor. Minor in-situ repairs to the blankets can also be accomplished, without removal, using the in-vessel service equipment.

Provisions for hands-on maintenance will be made wherever possible. Such maintenance will be feasible for the tokamak device up until tritium is introduced, and may be feasible (depending on the system in question) for auxiliary systems in the D-T phase. The tokamak design and service layout places many components (e.g., electrical, water, and cryogenic service connections) outside the reactor-cell shield, where limited hands-on maintenance will be possible.

The reactor hall layout must consider the allocation of equatorial entry ports, the need to locate auxiliary equipment (including in-vessel maintenance and inspection devices) near the machine, and the required access for the main crane during the installation and maintenance of heavy components. Other considerations include routes for transportation of activated components to the hot cells, the location of auxiliary services, and contamination control.

IV.2. ITER EQUATORIAL PORT ALLOCATIONS

The 16 equatorial ports in the ITER device provide the most useful access to the machine for auxiliary systems, testing, diagnostics and maintenance. These ports are relatively large and provide direct access to the reacting plasma. The allocation of these access ports is important to making the machine technically feasible, satisfactorily reliable and suitable for the ITER physics and technology experimental testing programs.

The operating program for ITER calls for a physics phase and a technology phase. As the name implies, during the physics phase the major emphasis will be on operating ITER as a plasma physics experiment starting with a hydrogen plasma and carrying on to a D-T Plasma. During the technology phase the emphasis will be on nuclear testing of components such as developmental power reactor blanket modules.

There will be a difference in the allocation of ports in each phase with the emphasis on plasma diagnostics in the physics phase changing to nuclear testing in the technology phase. There will be nuclear testing ports during the physics phase and diagnostic ports during the technology phase. The conceptual design takes both of these operational phases into account by planning for an efficient transition involving the replacement of activated components.

The systems that require access at the equatorial plane are:

- Nuclear Testing
- Diagnostics
Fig. IV-2. Basic Device Composition Designation and Port Allocations (Technology Phase)
Fuelling
Heating and Current Drive
Maintenance

For nuclear testing it is desirable to maximize the access to the first wall of the reactor in order to provide test module spaces large enough to minimize edge effects the testing program. Large ports are also desirable for the introduction of maintenance equipment. These characteristics are in opposition those that favor high tritium breeding ratios and simple nuclear shielding design. The size of the equatorial port also affects the design of the PF and TF coil systems. The width of the port affects the TF coil ripple and the space available for vertical support of the PF coils. The height of the port affects the PF coil design and limits the space available for the TF coil intercoil support for the outboard leg. The dimensions of the equatorial ports in ITER are 2.0 m wide and 3.4 m high. This port size is a compromise that meets the minimum requirements for all of the sub-systems.

Physics diagnostics requirements in the equatorial plane are generally less demanding on the size of ports and therefore should be adequately served by ports sized for nuclear testing.

The 16 ports are numbered in the clockwise direction (Fig. IV-2) and are specifically allocated as shown in Table IV-2 for the physics and technology phase.

This allocation scheme concentrates the auxiliary systems on one side of the reactor making a relatively clear area on the opposite side for testing facilities.

The allocation listing of the ports only addresses the major systems. In several of the ports there will be space for additional usage such as small test units, diagnostics and inspection equipment.

IV.3. REACTOR BUILDING LAYOUT

This section describes the arrangement of systems in the ITER Reactor Building. This space is part of the overall ITER building facilities but is described here because it is an integral part of the ITER device and its auxiliary systems. More detailed documentation of the Reactor Building can be found in the ITER Plant Report [2].

The Reactor Building is the partially inert gas filled containment volume that surrounds the reactor device and its directly connected auxiliary systems. These systems are:

Maintenance
Neutral Beam
Electron Cyclotron RF
Lower Hybrid RF
Fuelling
Nuclear Testing
Vacuum Pumping

A brief description of each of these systems and its functions is given later in this report.
### Reactor Building Requirements

The Reactor Building houses all the system components that are directly connected to the torus vacuum. The Reactor Building not only provides the normal support services but provides a final confinement barrier for the reactor systems. Because a failure in one of these systems can lead to the release of contaminants outside the primary confinement walls of the reactor system, a secondary confinement enclosing all these systems is required.

An inert gas atmosphere is maintained in the Reactor Building in order to reduce the fire hazard, minimize the production of unwanted radioactive nuclides, prevent the ingress of contaminants to the plasma vacuum vessel and to simplify the building's atmospheric clean-up system.

The Reactor Building must have adequate space to house the reactor systems and space to allow necessary maintenance and repair activities. In addition, the reactor building provides much of the local shielding that is required in order to allow manned access to the building spaces 24 hours after reactor shutdown.

### Reactor Building Arrangement

The arrangement of systems in the Reactor Building are shown in Fig. III-1, Plan View of the Reactor Room at the Equatorial Port Level, and Fig. III-2, Vertical Cross Section of the Reactor Building.

There two principle levels in the Reactor Building, from the top down they are the Crane Hall and the Reactor Hall.

The Crane Hall houses the main building cranes and provides space for the assembly and maintenance of major components.

The Reactor Hall is made up of several levels: at the equatorial plane of the reactor, it provides the space for the major reactor auxiliary systems and for the nuclear testing facilities; at the next lower level are located the primary vacuum pumps and below that is a maintenance space.

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### TABLE IV-2 ITER EQUATORIAL PORT ALLOCATIONS

<table>
<thead>
<tr>
<th>Physics Phase</th>
<th>Technology Phase</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Neutral Beam</td>
<td>Neutral Beam</td>
</tr>
<tr>
<td>2 Diagnostics/Nuclear Testing</td>
<td>Diagnostics/Nuclear Testing</td>
</tr>
<tr>
<td>3 Maintenance</td>
<td>Maintenance</td>
</tr>
<tr>
<td>4 Lower Hybrid RF</td>
<td>Lower Hybrid RF</td>
</tr>
<tr>
<td>5 Lower Hybrid RF</td>
<td>Lower Hybrid RF</td>
</tr>
<tr>
<td>6 Electron Cyclotron RF</td>
<td>Electron Cyclotron RF</td>
</tr>
<tr>
<td>7 Maintenance</td>
<td>Maintenance</td>
</tr>
<tr>
<td>8 Diagnostics/Nuclear Testing</td>
<td>Nuclear Testing</td>
</tr>
<tr>
<td>9 Diagnostics</td>
<td>Nuclear Testing</td>
</tr>
<tr>
<td>10 Diagnostics</td>
<td>Nuclear Testing/Diagnostics</td>
</tr>
<tr>
<td>11 Maintenance</td>
<td>Maintenance</td>
</tr>
<tr>
<td>12 Nuclear Testing</td>
<td>Nuclear Testing</td>
</tr>
<tr>
<td>13 Nuclear Testing</td>
<td>Nuclear Testing</td>
</tr>
<tr>
<td>14 Fuelling/Diagnostics</td>
<td>Fuelling/Diagnostics</td>
</tr>
<tr>
<td>15 Neutral Beam/Maintenance</td>
<td>Neutral Beam/Maintenance</td>
</tr>
<tr>
<td>16 Neutral Beam</td>
<td>Neutral Beam</td>
</tr>
</tbody>
</table>

---

30
Crane Hall

The Crane Hall provides access to the top of the tokamak for maintenance and assembly. The Crane Hall provides the primary route for transporting large components such as blanket/shield modules and neutral beam modules to the hot cells for repair or disposal. In order to handle the major components in ITER a crane capacity of 800 tonne is required. Other systems, such as operational and physics diagnostics, will be located here where vertical access to the reactor is available.

The height of the Crane Hall is determined by the maximum mass and vertical height of the reactor components and handling tools.

Reactor Hall

At the mid-plane, the circular shape of a Tokamak leads naturally to arranging the associated systems in radial positions corresponding to the 16 equatorial ports. As seen in the plan view, all of the auxiliary systems except for one of the articulated boom enclosures are located together on one side of the reactor. The remaining space on the opposite side is dedicated to the nuclear testing program.

The vacuum pumping is located in eight rooms around the perimeter of the tokamak. These rooms are designed in such a way that the pumps can be isolated from the reactor and transported to the hot cells for repair or disposal.

On the maintenance level of the Reactor Hall there are spaces for remote tooling to service the bottom of the reactor and a lateral tunnel that is large enough to allow replacement of the lower divertor coil (PF5).
V. ITER SYSTEM DESCRIPTIONS

This chapter summarizes the key characteristics of the major ITER sub-systems. More detailed descriptions may be found in the individual ITER Design Unit Technical Reports. (Ref. [4 through 11])

V.1. MAGNET SYSTEMS [4].

The ITER magnet systems are 1) the Toroidal Field (TF) magnet system, 2) the Poloidal Field (PF) magnet system, and 3) the in-vessel Plasma Control Coils. These magnet systems respectively 1) provide the toroidal field needed for stable plasma operation, 2) define and control the shape of the plasma configuration and inductively establish and maintain the plasma current, and 3) stabilize the vertical and horizontal position of the plasma. The TF and PF systems are superconducting, and are cooled to liquid helium temperatures by a cryogenic refrigerator. The plasma control coils are water-cooled normal copper coils, and are installed in modular form within the torus vacuum vessel. Fig. V-1 is the reference layout of the magnet system.

Achieving the magnet performance required to meet ITER mission goals requires state-of-the-art superconducting technology for the TF and PF magnets. The concepts selected for these systems have been chosen after an iterative process of detailed analyses and design optimization.

Both the TF and PF systems will use force-cooled cabled conductors with a steel jacket around the cable space to provide additional distributed structure. In addition to ensuring positive heat removal, this design approach offers good rigidity in the winding pack and is compatible with the high-voltage design requirements imposed by system operation and quench protection. Nb$_3$Sn is selected as the superconductor because it provides a higher temperature margin. NbTi is an option for the superconductor for the outer PF coils.

The winding packs of the TF and PF magnets are designed on a consistent basis that assumes state-of-the-art performance (current density) from the wire, but provides comfortable margins in performance with regard to heat removal and stability. Ample margin is provided in each coil design for protection in the event of a quench.

The magnet structures (TF cases, the distributed structure in the TF and PF windings, the TF intercoil connections, and the gravity supports) are designed to accommodate the full range of expected operational and credible fault loads. The loads accommodated in normal operation include gravity, thermal expansion/contraction, preloading and magnetic force. The loads accommodated in off-normal, but expected conditions, include additional magnetic loading from plasma disruptions and from magnet quench from a worst-case plasma operation condition.
Fig. V-1. Magnets TF Coil and PF Coils
Fault loading conditions are presently being studied. As a minimum, the goal is to accommodate credible failure modes, such as imbalance among the TF coils and electrical faults at the coil terminals. The goal of the fault analysis is to demonstrate no fault propagation to the systems (e.g. vacuum vessel).

V.1.1. Toroidal field coils.

The primary requirement of the TF system is to produce 4.85 tesla at a radius of 6.0 meters. When other tokamak design requirements (i.e. space at the center of the machine for the central solenoid and space in the bore of the TF coils for plasma, blanket and shield, divertor, and vacuum vessel) are taken into account, the result is a field of just over 11 Tesla at the windings of the TF coils. Design considerations for the amount and efficient distribution of structure in the inboard legs of the TF coils lead to a winding-pack current density of 30.5 A/mm². The average current density in the central vault region formed by the wedged TF magnet cases is 14 A/mm². Table V1. summarizes the important characteristics of the TF magnets. Fig. V-2 is a perspective view of the TF coil arrangement.

The number of TF coils, 16, and the radius of the outer leg, 11.1 meters, are chosen to keep field ripple at the plasma edge below 2.5%. This outer leg location also minimizes the ampere-turn and power requirements for the outer PF coils, and provides adequate clearance for access ports and shielding inside and around the TF outboard legs.

The 15.6 meters height of the TF coil is set by the space required for the plasma, the divertor plasma scrape-off region, and the divertor plate and associated in-vessel nuclear shielding. These vertical clearance requirements, when combined with the outer leg position, lead to an elongated, racetrack-like TF coil shape. Although this shape departs substantially from the classic constant-tension shape, the overall structural loading penalty is not significant. The reason is that while the in-plane and bending loads are higher, the out-of-plane loads from the PF coils are lower because of the closer proximity of the PF coils to the plasma.

The structure of the TF coils encloses the windings in thick steel cases that resist the self-generated bursting forces and the out-of-plane distortions that arise when the PF fields are applied. The noses of the inboard legs of these cases are wedged together in a vault at the center of the magnet set to react the centering loads on the coils. The overturning loads are reacted by friction in the vault, by shear keys at the top and bottom of the vault, and by the outer shear frame, which also partially restraints the outward bursting forces.

Most of the TF design analyses to date have focused on the electromagnetic in-plane and out-of-plane loads. The loads imposed by fault, quench, and disruption conditions will be somewhat more severe, but they also should occur much less frequently. This will mitigate their impact on the design, since fatigue is otherwise the primary concern.
TABLE V-1. KEY PARAMETERS AND FEATURES OF THE TF COILS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of TF coils</td>
<td>16</td>
</tr>
<tr>
<td>Total current per coil, (MA)</td>
<td>9.1</td>
</tr>
<tr>
<td>Coil dimensions:</td>
<td></td>
</tr>
<tr>
<td>Mean radius, inboard leg, (m)</td>
<td>2.55</td>
</tr>
<tr>
<td>Mean radius, outboard leg, (m)</td>
<td>10.50</td>
</tr>
<tr>
<td>Mean height above/below midplane, (m)</td>
<td>7.81</td>
</tr>
<tr>
<td>Horizontal clear bore, (m)</td>
<td>7.10</td>
</tr>
<tr>
<td>Vertical clear bore, (m)</td>
<td>14.76</td>
</tr>
<tr>
<td>Radius of clear bore for central solenoid, (m)</td>
<td>2.13</td>
</tr>
<tr>
<td>Field on axis (T) (at 6.0 m)</td>
<td>4.85</td>
</tr>
<tr>
<td>Maximum field at the windings, (T)</td>
<td>11.1</td>
</tr>
<tr>
<td>Winding pack description</td>
<td></td>
</tr>
<tr>
<td>Conductor,</td>
<td></td>
</tr>
<tr>
<td>Pack dimensions (m x m)</td>
<td></td>
</tr>
<tr>
<td>Mean pack current density, (A/mm^2)</td>
<td>0.306 x 0.846</td>
</tr>
<tr>
<td>Number of turns</td>
<td>30.5</td>
</tr>
<tr>
<td>Conductor current, (kA)</td>
<td>240</td>
</tr>
<tr>
<td>(10 radial x 24 toroidal)</td>
<td></td>
</tr>
<tr>
<td>Mean composition (%)</td>
<td>38</td>
</tr>
<tr>
<td>Steel</td>
<td>43</td>
</tr>
<tr>
<td>Conductor (70 % copper)</td>
<td>29</td>
</tr>
<tr>
<td>Helium space</td>
<td>20</td>
</tr>
<tr>
<td>Insulation</td>
<td>8</td>
</tr>
<tr>
<td>Cable space, current density, (A/mm^2)</td>
<td>63</td>
</tr>
<tr>
<td>Mass of the windings, (t)</td>
<td>70</td>
</tr>
<tr>
<td>Mass of the case, (t)</td>
<td>400</td>
</tr>
</tbody>
</table>

V.1.2. Poloidal field coils.

The number and locations of the PF coils have been optimized to provide a satisfactory compromise between issues of plasma operational flexibility and control, access, port location and size, and credible coil fabrication and structural load paths. In the main (ex-TF) system, there are seven up/down symmetric coil pairs; four pairs are in the central solenoid, one pair is above/below the divertor, and two pairs are outboard of the TF. Fig. V-2 illustrates the location of the PF coils.
Fig. V-2. Sectionnel View of the Combined ITER Superconducting Magnet System
TABLE V-2. KEY PARAMETERS AND FEATURES OF THE PF COILS

<table>
<thead>
<tr>
<th>Coil No.</th>
<th>R (m)</th>
<th>Z (m)</th>
<th>DR (m)</th>
<th>DZ (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PF1 U/L</td>
<td>1.725*</td>
<td>0.950</td>
<td>0.650</td>
<td>1.900</td>
</tr>
<tr>
<td>PF2 U/L</td>
<td>1.725*</td>
<td>2.850</td>
<td>0.650</td>
<td>1.900</td>
</tr>
<tr>
<td>PF3 U/L</td>
<td>1.725*</td>
<td>4.750</td>
<td>0.650</td>
<td>1.900</td>
</tr>
<tr>
<td>PF4 U/L</td>
<td>1.725*</td>
<td>6.650</td>
<td>0.650</td>
<td>1.900</td>
</tr>
<tr>
<td>PF5 U/L</td>
<td>3.900</td>
<td>9.000</td>
<td>0.900</td>
<td>0.900</td>
</tr>
<tr>
<td>PF6 U/L</td>
<td>11.500</td>
<td>6.000</td>
<td>0.500</td>
<td>1.500</td>
</tr>
<tr>
<td>PF7 U/L</td>
<td>11.500</td>
<td>3.000</td>
<td>0.500</td>
<td>0.900</td>
</tr>
</tbody>
</table>

*For the central solenoid, this is not the current center, only the geometric center, since the winding pack is structurally graded.

Conductor: \( \text{Nb}_3\text{Sn/Cu composite, force-cooled by LHe, encased in steel jacket (NbTi a candidate for outer PF coils)} \)

Central solenoid module description (PF1 - PF4):
- Pack dimensions, (m x m) \(0.640 \times 1.840\) (excluding joints, cross-overs, and spacers)
- Number of turns 480 (12 radial x 40 vertical)
- Mean Composition in the winding pack (%):
  - Steel 56
  - Conductor (60 % copper) 19
  - Helium space 12
  - Insulation 13
- Total current, (MA) 20.6 at IM, 22.4 at end-of-burn
- Maximum field, (T) 13.5 at IM, 12.5 at end-of-burn
- Mean pack current density, (A/mm\(^2\)) 17.5 at IM, 19.0 at end-of-burn
- Cable-space current density, (A/mm\(^2\)) 55.4 at IM, 60.2 at end-of-burn
These coils provide the principal plasma shaping and induction fields. In addition, there is a pair of water-cooled copper coils imbedded in the vacuum vessel/blanket structures for rapid control of the vertical and horizontal position of the plasma. The main features of the PF coils are listed in Table V-2.

The radial build and winding pack design of the central solenoid are optimized to provide maximum flux-swing (volt-second) for inductive plasma current drive. The key considerations in the optimization are: 1) allowable stresses (from a fatigue perspective) in the central solenoid cable conduit and distributed structure, 2) temperature allowance above the coolant inlet temperature to accommodate increases from heat absorption and coolant Joule-Thompson expansion effects, 3) sufficient temperature margin for stable and reliable performance of the superconductor, and 4) quench protection provisions.

The optimization process used to address these considerations was iterative, with initial attention given to maximizing the flux at initial magnetization. Subsequently, the winding pack design was adjusted slightly to account for the fact that the protection constraint is more severe at end-of-burn, where the currents in the central solenoid are higher and the fields are lower. This optimization resulted in maximum fields in the windings of about 13.5 Tesla at initial magnetization and 12.5 Tesla at end-of-burn.

With these fields, a total central solenoid flux swing capability of 250 volt-seconds is achievable with a radially-uniform current density winding. When structural grading is applied to the central solenoid winding pack design (with the same constraints as the ungraded design) the central solenoid flux swing increases to approximately 265 volt-seconds. The plasma shaping and radial equilibrium fields provide an additional 60 volt-seconds. The total PF system volt-second capability is 325 volt-seconds. This volt-second capability allows full inductive plasma current ramp-up to 22 MA and a minimum fusion burn of 200 seconds for the full range of beta poloidal, $b_p$, and internal plasma inductance, $l_{\psi}(3)$ (respectively 0.4-0.8 and 0.55-0.75) anticipated for ignition operation.

The PF system volt-second capability and central solenoid design also satisfy the auxiliary design requirement that the flux swing available for burn be at least 10% (33 volt-seconds) of the total PF volt-second capability.

Because the overall size and performance of ITER is so closely tied to the performance of the central solenoid, it is essential to adopt a highly optimized design that takes full advantage of the foreseeable performance of Nb$_3$Sn superconductor technology. This highly optimized design approach is justifiable, since the central solenoid can be completely tested to full operating levels before installation in the tokamak, and can be (relatively) easily removed for repair or replacement if necessary.

The central solenoid conduit hoop stress is large: about 450 MPa hoop tension when peaking factors are taken into account. At this stress level, the operational lifetime of the distributed structure and conduit jacket is finite. Here the character of the operating scenarios planned over the expected device lifetime is taken into account. A preliminary analysis of crack growth in the
critically stressed portions of the central solenoid shows that the desired lifetime for both the Physics and Technology Phases can be achieved.

The space constraints and peak fields of the outer PF coils are less demanding than those for the central solenoid, and here a more conservative design is possible. This is prudent, because these coils are much harder to replace or test than the central solenoid. The maximum fields at the windings of the outer PF coils lie in the 5 to 9 Tesla range, making NbTi a possibility, but Nb₃Sn is still the superconductor of choice because of the larger temperature margins it provides.

V.1.3. Cryogenic systems.

The heat loads to the cryogenic systems in ITER are large by comparison with traditional standards, but are well within the capabilities of existing cryo-refrigeration technology. The standby heat load in the TF and PF magnets from conduction, thermal radiation and lead heat influx will be about 17 kilowatts. During normal operation for the Physics Phase, AC losses in both systems, nuclear heating in the TF system, and joint losses and ohmic loss in the current leads will raise the total pulse-average heat load to almost 70 kilowatts. With contingencies and other non-magnet loads, a refrigeration plant having a total capacity of about 100 kilowatts is needed.

The most effective way to achieve this capacity will be with modules with individual capacities in the 20 to 30 kilowatt range. Modularization at this level will result in cold-boxes of transportable size, and will have the added benefit of permitting greater flexibility of operation under the variable heat loads.

V.2 CONTAINMENT STRUCTURES [5].

The ITER Containment Structures provide the controlled environments needed to conduct ITER operations. The torus vacuum vessel forms the primary vacuum boundary for the plasma and provides containment of tritium. The vacuum vessel also provides mechanical support for the in-vessel nuclear components, which are attached to the vacuum vessel by an attaching locks system. The cryostat vacuum vessel provides the secondary vacuum used as thermal insulation for the TF and PF superconducting coils. The machine gravity supports support the TF and PF magnet systems and vacuum vessel within the cryostat vessel. The "blanket-to-blanket" attachment system ("attaching locks") employed in the design allow compensation of the principal electromagnetic loads inside the cylindrical blanket structures rather than transmitting them to the vacuum vessel. The passive loops and active coils for plasma stabilization are discussed here because they are an integral part of the above systems.

V.2.1. Vacuum vessel

A reference vacuum vessel (VV) design was chosen and developed in detail during the CDA phase. The torus vacuum vessel is assembled from 16 sectors, with one sector per TF coil (see Fig. V-3). This design can withstand the reference electromagnetic loads, accidental overpressures, and other impacts.
The assembly joints are protected by a sufficient thickness of nuclear shielding to meet requirements for stainless steel rewelding after irradiation.

A few critical points in the VV design still exist, including high local stress concentrations in the resistive element and electroinsulating connections, which must be solved with future design optimization.

The most critical future needs are the feasibility studies of the VV manufacturing, VV assembly and VV repair/disassembly/reassembly by remote handling tools.

Alternative, thin-wall VV options were also analysed during the CDA phase as attractive, simplified design solutions. If the feasibility questions relating to the total structural integration, such as compatibility with passive/active elements and blanket attaching locks, and shielding performance can be successfully answered, then the thin-wall VV concept has a good chance to be adopted on ITER. The final VV design solution may be a combination of the reference and thin-wall options, using the best features of each concept.

V.2.2. Cryostat vessel

A few feasible cryostat design options were compared during CDA phase on the basis of established requirements. A full metallic welded design, was chosen.

The cryostat design is feasible and meets all existing requirements. The necessity of limited technological R&D for new, non-standard components (resistive inserts, large flanges and flexible bellows) was identified.

As a possible alternative approach, a thick prestressed concrete cryostat with thin inner metallic liner has also been proposed, which combines the cryostat, biological shielding, and building wall functions. Preliminary analysis have shown, however, that biological shielding 24 hours after shutdown can be provided by the reactor itself (blanket and VV) with extra shielding around all penetrations.

V.2.3. Machine gravity supports

The flexible multiplate gravity supports concept (without any kinematic or rolling mechanisms) was chosen both for the magnet system and vacuum vessel supports. Two possible sub-options of compressed verses tensile flexible plates are still under consideration for design optimization, but the compressed flexible plates option was used for ITER design integration based on engineering experience.

V.2.4 Attaching locks

No final reference design solution was chosen for the blanket attaching locks. The choice of the welded connectors option was made for the purpose of design integration and it is based on the prediction that a successful remote welding technique will be developed. There are four other possible design options that are under investigation, but none of these options fully satisfy the established requirements.

Significant efforts were made to resolve this problem and a few original, attractive engineering ideas and guidelines for solutions to conflicts between design and requirements were proposed. Five principal design approaches will be advanced for future design improvements and verification by
Fig. V-3. Vacuum Vessel Details
the technology R&D during the earliest part of the EDA phase. In addition, each proposed solution has some chance to be used, because different designs may be more suitable for attachments of different elements (i.e. blanket segments, shielding plugs or divertor plates, inboard versus outboard blanket, lateral-to-lateral versus central-to-lateral segment connections, etc.).

V.2.5 Passive loops and active coils

**Passive loops**

It was shown that the conventional saddle-passive loop concept inadequate to the ITER requirements. A new twin-loops concept satisfying both design integration and plasma stabilizing requirements was adapted as a reference.

One potentially serious technological problem for both previous and new passive loop designs is the development of a reliable process for bonding the high conductivity plates to the stainless steel structures.

**Active coils**

The active coil position is near the equatorial port and does not interfere with blanket removal trajectories. Current joints are located in the secondary vacuum or in the atmosphere of the reactor hall and are repairable by remote handling tools. Coil "single-turn self-inductance" is reduced by factors of 1.5 to 2 compared with the initial options.

Three or four workable active coil design options have been identified as candidates for a final reference choice. This choice will be made between more conventional (but less electromagnetically optimum designs) and some potentially more attractive proposals, which are not verified yet by detailed R&D. In all cases, careful development and testing of the AC conductor (and especially conductor current joints) is needed to provide a reliable basis for a final design choice.

V.3. FIRST WALL [6].

The ITER First-Wall (FW) system is located on the plasma-facing surface of the in-vessel blanket/shield modules and covers most of the region within which the plasma is produced. The FW is covered by an array of armor tiles mounted on the front face of the blanket/shield modules. The FW structure and coolant are integrated with the tritium-breeding and nuclear-shielding portions of these modules. The principal functions of the FW are to 1) define the ultra-clean (low-impurity) region within which the plasma is produced, 2) absorb the electromagnetic radiation and charged-particle flux emanating from the plasma, and 3) protect the underlying blanket/shield components from direct contact with the plasma and high-energy electrons, particularly during periods of plasma start-up and shutdown and during plasma disruptions.

V.3.1. FW design requirements

The FW system must satisfy a complex and sometimes contradictory set of design and operational requirements. The plasma-facing armor must be able
to 1) accommodate the high thermal heat and neutron fluxes (up to 0.6 MW/m² and 1.1 MW/m², respectively) produced during normal operation, 2) withstand the impulse surface energy depositions (2 MJ/m² in 1 ms) produced during plasma disruptions, and 3) survive impingement by high-energy (runaway) electrons, with energies up to 300 MeV and localized energy depositions of up to 100 MJ/m² that may be produced in the aftermath of plasma disruptions.

In addition to these thermal and nuclear requirements, the material of the first wall must be compatible with the surface-impurity conditions (low absorbed water and oxygen) needed for plasma operation, should not in itself introduce impurities that significantly degrade plasma performance, and should minimize the in-vessel inventory of tritium retained in the first-wall armour and bulk materials.

The basic structural material is 316 stainless-steel. The required segmentation must be sufficient to allow the blanket modules on which the FW is mounted to be extracted through the upper access ports. Additional electrical segmentation within the inboard modules and further subdivision of the armor tiles are provided to minimize the electromagnetic disruption forces. The segmentation of the outboard modules is determined by the configuration of the passive stabilization coils.

Low-pressure water (60°C inlet at 1.5 MPa) is chosen as the basic coolant. Redundant coolant circuits and natural convection shutdown cooling are provided for improved passive safety.

V.3.2. FW design and integration with blanket/shield modules

The FW panel on which the FW armor tiles are mounted is integrated with the underlying blanket/shield modules. Two design concepts have been developed.

The inboard blanket modules (Fig.V-4 top) will employ poloidal (vertical) FW cooling, since this flow geometry is well matched to the elongated configuration of the module and requires far fewer cooling channels than if toroidal cooling were used. The continuous FW and back-wall structures are manufactured as flat modules by brazing or electron beam welding of two plates. Cooling tubes (10 mm inside diameter) are brazed into the outer steel structure, and thus provide double-walled containment of the coolant in order to achieve a virtual "zero" leakage.

The outboard blanket modules (Fig.V-4 bottom) will employ toroidal FW cooling. Here, owing to interruption of the modules by the equatorial access ports, a toroidal (horizontal) coolant flow results in minimal complexity. The FW panels will be assembled from U-shaped continuous FW and side wall segments, manufactured by hot isostatic pressing, of rectangular coolant tubes between a thin front and a thicker back plate.

Structural analysis of both designs has been performed. Thermal stresses in the structure at the peak heat flux (0.4 MW/m²) are less than 400 MPa. This estimate includes the effects of tile attachment, but does not include the effects of integration of the FW-panel into the blanket box. For these
INBOARD 1/32 FIRST WALL BLANKET SEGMENT
WITH POLOIDAL COOLING (SEC. 1-1)

INTEGRATION OF PLASMA FACING COMPONENTS

ITER WITH A 22 MA PLASMA

RADIATIONS
COOLED TILE

ELECTRICAL
INSULATION

CONDUCTIVELY
COOLED TILE

T-BREEDING
Fig. V-4. First Wall Design Concepts Integrated with Blanket Segments
thermal stresses, the present nuclear design codes allow $2 \times 10^4$ to $8 \times 10^4$ cycles. Electro-magnetic forces due to disruptions are estimated to create dynamic stresses exceeding 100 MPa, and may thus create a potential for progressive deformations. Further study of this effect is needed.

V.3.3. FW armor

Based on experience in present large tokamaks, for the Physics Phase a low-Z, plasma-facing armor on the FW steel structure is required, such as a carbon-fibre-composite or beryllium. For the Physics Phase, a carbon-fibre-composite material has been adopted as a reference choice. Carbon-fibre-composite materials offer superior thermal shock resistance. The enhanced reliability of carbon-fibre-composite materials relative to ordinary (non-composite) carbons is already demonstrated in present tokamak experiments.

A design concept based on mechanically-attached carbon-fibre-composite tiles has been selected for detailed study. This design, shown in Fig. V-5, features tiles with fully-mechanical attachments using carbon-fibre-composite nuts and studs in grooves of the FW steel structure. The attachment scheme allows the tiles to be replaced by remote handling.

The tile cooling method depends on the expected heat flux. Radiation-cooled tiles are used in hot-spot areas with high ($0.3-0.6$ MW/m$^2$) heat flux, and conduction-cooled tiles, with the same basic design, are used for the areas of the FW with lower heat fluxes. The limited use of radiation cooling (10% of the FW area) is to minimize plasma impurities and reduce tritium inventory. Conduction-cooled tiles operate at low temperatures (<1000 C) relative to radiation-cooled tiles (1600-1800 C), and thus result in a reduced plasma impurity source and a lower tritium inventory in the FW tiles. The total tritium inventory in the FW is estimated to be about 200 grams.

Several open issues about the FW design remain. Like all carbon materials used in tokamaks, the carbon-fibre-composite tiles must be baked to temperatures greater than 350 C following exposure to air or water, and also require glow discharge cleaning between periods of tokamak operation to maintain the clean surface conditions needed for successful plasma operation. Scenarios for baking and discharge cleaning are being developed. Options include circulating hot gas in the FW cooling circuits, and the use of rf power to heat the tiles while the underlying panel structure remains at lower temperature.

The operational life of the FW tiles is being evaluated. The key issues are irradiation effects and tile surface erosion from plasma disruptions. Data on the irradiation lifetime due to swelling of carbon-fibre-composite materials is needed to evaluate the ultimate lifetime of the tiles.

The response of the tiles to disruption conditions also requires further study. Here the ability of the carbon-fibre-composite material to withstand the erosion and thermal shock associated with the disruption energy flux appears acceptable; for the number of disruptions anticipated in the Physics Phase, all but the most highly-load portions (10%) of the FW should survive for the entire
Fig. V-5. Mechanically attached carbon tiles
Physics Phase. Limited portions of the FW may have to be replaced several times, depending on the magnitude of the localized disruption energy.

The maximum allowable tile temperature is the subject of continuing study. The FW tile design is based on a maximum allowable surface temperature of 1800°C. This limit is set by radiation-enhanced sputtering, which results in unacceptable erosion at higher temperatures. However, reactions between the carbon-fibre-composite material and air or water (as could occur in a loss of vacuum or in-vessel coolant leak) become significant for temperatures > 1000°C, and further analysis is required to study the possible safety risks and to determine if a corresponding limit on the maximum operating temperature is required.

For the Technology Phase the aim is to reduce the coverage with CC-tiles due to considerations of lifetime, conditioning, and tritium inventory. If tungsten is selected as a divertor armor, it is envisaged that a tungsten coating (about 1mm thick) would be applied by plasma spray on the steel FW. The tungsten armor will protect the steel structure from melting during a disruption and very little if any vaporization or melting of the tungsten armor will occur during the disruptions. If any damage to the coating occurs, it can be readily repaired by in-situ recoating by plasma spray. Other advantages for the tungsten armor in addition to the protection and simple repair include lower bakeout and conditioning temperatures (<150°C), low tritium inventories, and radiation damage resistance. With this proposal several issues should be investigated in more detail during the EDA such as high Z plasma contamination, damage of the W-coating and steel substrate by disruption and run-away electrons.

V.4. DIVERTOR PLATE [6].

The divertor plates, located above and below the plasma region within the torus vacuum vessel, establish the interface between the plasma and the material surface of the tokamak device. During normal operation, these plates must withstand and remove approximately 100 MW of heat that is conducted to them from the plasma boundary. These plates are also the location where the plasma (including the helium produced by the fusion reaction) is neutralized for subsequent exhaust by the torus vacuum pumping system. The operational requirements associated with these functions are among the most challenging of the ITER design.

Table V-3. presents a summary of important specifications.

V.4.1. Divertor design and materials

Details of the divertor plate design and the support structure are shown in Fig. V-6.

As DP armor for the Physics Phase, carbon fibre composites (CC) are selected primarily because of the extensive operating experience of carbon based plasma facing materials in current tokamaks, the expected wide range of allowed
TABLE V-3. DIVERTOR DESIGN AND OPERATION SPECIFICATIONS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Design Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design heat flux, (MW/m²)</td>
<td>15</td>
</tr>
<tr>
<td>Pulse duration</td>
<td>continuous</td>
</tr>
<tr>
<td>Allowable surface temperature, (°C)</td>
<td>1000</td>
</tr>
<tr>
<td>Coolant:</td>
<td></td>
</tr>
<tr>
<td>Inlet temperature, (°C)</td>
<td>60</td>
</tr>
<tr>
<td>Velocity, (m/s)</td>
<td>10</td>
</tr>
<tr>
<td>Plate toroidal segmentation</td>
<td>32 modules</td>
</tr>
<tr>
<td>Maximum dimension, (m x m)</td>
<td>1.2 x 3.5</td>
</tr>
</tbody>
</table>

plasma edge conditions for carbon, its unique thermal shock resistance, the potential for tailoring CC to specific requirements such as achieving high thermal conductivity.

The allowed maximum CC-surface temperature of 1000°C should avoid run-away erosion by self sputtering at plasma edge temperatures up to 100 eV.

Beryllium is considered as back-up DP-armor in view of promising results at JET. Of particular interest is the potential for in-situ repair by plasma spray, lower tritium inventory and reduction of baking requirements.

The DP-armor has to be bonded (brazed) to the heat sink in view of the high heat loads. For the heat sink, alloys based on copper, molybdenum or niobium are proposed.

As DP-armor for the Technology Phase, tungsten is considered as alternative to carbon mainly due to prospects for significantly lower sputtering erosion rates, better irradiation resistance, potential for in-situ repair, lower bake-out and conditioning temperatures, and better protection of the heat sink. A 2 mm thick W coating is diffusion bonded onto a niobium (alternatively Mo or Cu) alloy heat sink with rectangular coolant channels. This W coating would permit in-situ repair by plasma spray.

A flexible attachment scheme is provided to accommodate thermal expansion between the divertor plate and the underlying support structure. The coolant flows first through the divertor plate and is then returned through channels which provide cooling for the support structure.

V.4.2. Divertor performance and lifetime issues

The thermal performance (peak power handling) and operational lifetime of the divertor plates are important operations issues. The design goals for the Physics Phase are to be able to accommodate a peak surface heat flux of 15 MW/m² and to achieve a period of 1-calendar year or greater between a replacement of divertor plates. In both cases, preliminary analysis shows that there are reasonable prospects to achieve these goals. However, significant physics and engineering uncertainties exist, and present estimates of divertor
**Fig. V-6. Main Divertor Plate Concepts for Physics and Technology Phases**
plate thermal performance and lifetime are provisional. The lifetime is mainly limited by the number and magnitude of disruptions.

The peak heat load expected on the divertor plate is 10 MW/m$^2$. When allowance is made for local peaking of the heat fluxes due to toroidal field ripple and plate misalignment, the resulting design heat load is 15 MW/m$^2$, near the limit of engineering feasibility. Limiting the peaking factor to 1.5 will require precise divertor plate alignment with respect to the toroidal field lines.

The maximum allowable peak heat flux to the divertor plate is limited by tile surface temperature and by burnout of the coolant channel caused by film boiling. The burnout limitation is more restrictive. For water coolant velocities of 10 m/s, burnout occurs at a critical heat flux level of 40 to 60 MW/m$^2$. For the proposed design, which uses a round channel with a twisted tape insert to enhance heat transfer, the estimated burnout safety margin is 2 to 4 for a surface heat flux of 15 MW/m$^2$.

The 6-mm thickness of the divertor plate CFC armor tile is chosen to be consistent with a 1000°C maximum surface temperature at a static heat load of 15 MW/m$^2$. By varying the magnetic configuration of the plasma, it is possible to sweep the energy deposition across the strike-point location, at a frequency of 0.1 Hz, and thereby permit either a doubling of the tile thickness or a doubling of the allowable heat flux. In the latter case, however, the coolant flow velocity would have to be increased to maintain adequate burnout margin.

Neutron irradiation is expected to reduce the thermal conductivity in carbon, but the performance loss so produced may be offset by thinning of the tile due to erosion.

The lifetime of the divertor plate will be determined by surface erosion. The principal erosion mechanisms include plasma sputtering (physical and chemical) and sublimation due to thermal loading and runaway electron impingement produced by plasma disruptions. When the effects of re-deposition are taken into account, sputtering erosion is relatively modest, and sublimation from disruption thermal loading is the lifetime limiting factor. There are, however, considerable uncertainties in the factors that lead to this conclusion.

Although carbon exhibits high gross erosion rates due to sputtering, modeling of the conditions at the divertor plate surface indicates that most of the sputtered material will be re-deposited locally. When the effects of chemical sputtering and periodic shifting of the strike point (where the erosion peaks) are taken into account, the net sputtering erosion appears to be much lower. The major concern is a huge tritium inventory in the redeposited material resulting up to 1 kg at the end of the Physics Phase.

Erosion due to plasma disruptions is significantly more severe. For worst-case assumptions of 20 MJ/m$^2$ energy deposition, the peak erosion is predicted to be up to 0.7 mm per disruption.

High-energy ("runaway") electrons generated during disruptions may strike the divertor plate with energies potentially as high as 300 MeV, and penetrate through carbon armor to deposit their energy in the metal cooling structure. The threshold level sufficient to structure damage is estimated to be 50 to 100 MJ/m$^2$, assuming an electron energy of 300 MeV.
For the tungsten armour (Technology phase) it is estimated a 2 to 4 mm allowable thickness according to the heat sink material and 1400 °C maximum temperature. During the technology phase the erosion during normal operation is the dominant factor due to the long integral burn time and reduced number of disruptions. Carbon would require a far too high number of replacements and beryllium frequent in situ plasma spray recoatings. Only tungsten is then considered at present and it is estimated that 3 to 4 replacements would be needed (with sweeping) or 3-4 full in-situ repair.

V.4.3. Other design considerations

Thermal stresses in the divertor plate are a potential lifetime limiting issue and are presently being analyzed. Preliminary estimates show that primary stresses due to coolant pressure and stresses due to electromagnetic loading, are low. Thermal stresses and residual stresses at the braze interface are high and further study and testing are required to verify that the fatigue lifetime of the divertor plate will be adequate.

Baking of the divertor armor at 350 °C, which will be necessary after air or water exposure, will be done by temporarily changing the coolant from water to hot helium gas.

The tritium inventory in the divertor appears manageable. However, the tritium trapping in co-deposition layers of eroded carbon is estimated to be about one kilogram of tritium per year. Keeping the tritium inventory to acceptable levels (< 100 g) will require removing tritium from the redeposited carbon surface layer. Techniques such as in-situ baking with oxygen present or glow discharge cleaning are promising, but need to be studied.

The consequences of a divertor loss-of-cooling accident (LOCA) have been evaluated. The ability of the proposed design to withstand a LOCA is acceptable. During a complete LOCA, the delay to melting of the cooling tube is estimated to be at least five seconds if burn is not stopped. The high tile surface temperature and resulting carbon sublimation, according to preliminary calculations, should provide passive plasma shutdown before tube melt-through would occur.

V.5. BLANKET/SHIELD SYSTEM [7].

The ITER in-vessel nuclear systems are 1) the tritium-production (or "driver") blanket modules, and (2) the non-breeding shield components. The primary function of the driver blanket is to supply tritium for plasma operation, particularly during the Technology Phase of the device. The blanket also recovers most of the thermal heat resulting from the capture of 14-MeV neutrons. The blanket and shield, along with the torus vacuum vessel, also provide neutron shielding for the TF magnets. In most regions, the blanket and shield are integrated with the vacuum vessel into a single component. Specialized
non-breeding shielding is provided behind the divertor plates and around the equatorial port penetrations.

Austenitic steel (type 316 annealed) is selected as the structural material for the blanket/shield because of its extensive database and ease of fabrication. Low-temperature water (Tinlet = 60 C, Toutlet = 100 C) is selected as the coolant.

V.5.1. Blanket segments

Figures V-7 and V-8 show general configuration and cross-section views of the options for the inboard and outboard blanket/shield components. Key considerations in the design include 1) incorporation of lithium containing material (for tritium breeding), 2) incorporation of beryllium for neutron multiplication (to achieve a tritium breeding ratio as close to unity as possible), 3) heat removal and tritium recovery, and 4) blanket maintenance and replacement. The last consideration is addressed in the discussions of the tokamak device layout and maintenance approach. A discussion of the first three considerations follows.

Three options have been investigated to incorporate lithium containing material into the blanket. These include the use of lithium ceramics (solid-
Fig. V-8. In-vessel component cross sections
Fig. V-9. Ceramic Blanket Pebble Bed Concept Design
Fig. V-10. Ceramic Blanket Layered Block Concept Design
Fig. V-11. Ceramic Blanket Poloidal BIT Design
breeder), lithium-bearing salts in the water coolant (aqueous salt), and lithium-
lead eutectics. The solid-breeder concept is selected as the "first option".

Details of the solid-breeder design are being developed. The lithium
ceramic and beryllium may be incorporated into the blanket as small spheres
(1 mm diameter) or as layers of sintered material. Several lithium ceramics are
possible (Li_2O, LiAlO_2, Li_2SiO_4, Li_2ZrO_3). Tritium is recovered in-situ by
flowing low pressure (0.1 MPa) helium through the ceramic material.

A key design issue for the solid breeder blanket is how to keep the
ceramic material at a temperature (400-1000 C) high enough to enhance tritium
release, while still being compatible with a low-temperature, water-cooled steel
structure. The thermal isolation needed is provided by a small, helium-filled gap
between the coolant pipes and the ceramic or heat insulation is provided by
layers of beryllium placed between the coolant channels and the lithium ceramic.

Figures V-9 V-10 and V-11 show three options for the outboard blanket:
(1) poloidal cooling with a sphere-pac layers of the lithium ceramic and
beryllium, (2) radial/toroidal cooling with a layered block configuration of
beryllium and ceramic and (3) poloidal cooling with a breeder inside tubes (BIT)
configuration and beryllium outside the coolant tube.

Figure V-10 shows a plan view of the outboard blanket segments for a
layered block design. In general, all outboard segments have equal toroidal
widths at the midplane. This is compatible with all but two ports, which are 1300
mm wide at the midplane for maintenance purposes. The outboard, side blanket
segments at these locations are specially designed to accommodate these larger
ports.

Estimates of the achievable tritium breeding ratio with the proposed
solid-breeder and module designs are in the range of 0.8 to 0.9. The variation
depends on 1) the FW coverage of the blanket, 2) the amount of FW carbon-
fibre-composite tile coverage and 3) the structural requirements of the
FW/blanket. External supplies of tritium should be adequate to make up the
difference.

Tritium produced in the driver blanket will be distributed within the
structural material, the breeding material, and the multiplier, with the latter two
being the most important from the tritium inventory perspective. The tritium
inventory in the ceramic breeder is strongly design dependent, with estimates of
the inventory in the breeding materials ranging from less than 10 grams to
100 grams. The most important factor affecting the tritium inventory is the
temperature distribution, since most of the inventory of tritium will be located in
the cooler regions of the blanket.

The amount of recoverable tritium depends on the concept selected for
thermal isolation. If beryllium blocks are used to provide isolation between the
coolant tubes and lithium ceramic, the beryllium temperature will be relatively
low (100 to 400 C) and much of the tritium will be retained. If (in a worst-case
situation) all of the tritium were retained, the end-of-life inventory in beryllium
at the conclusion of the Technology Phase (1-3 MWa/m^2) would be about 0.4-
1.2 kg. The inventory at end of the Physics Phase would, however, be about 20
grams.
If beryllium temperature could be kept over 600 C (difficult because of radiation swelling), more than 90% of the tritium can be recovered, with a corresponding reduction of in-blanket inventory.

An alternate blanket concept with 83 Pb-17 Li eutectic alloy as the breeding material has also been developed. The lithium-lead blanket has poloidal breeding channels which follow the first wall geometry. Each channel consists of coaxial pipes where the breeder is separated in individual chambers. The mass of Pb-Li in the entire blanket is about 1000 tonne. During operation, the eutectic alloy is in the solid form. The alloy is melted and transported out of the reactor for tritium recovery during the reactor down time. The lead serves as a neutron multiplier; therefore, beryllium is not used in this concept. A tritium breeding ratio of 0.7-0.8 is attainable. The main issues with this concept relate to problems associated with melting the breeder and safety problems associated with tritium containment and generation of polonium. The elevation view and mid-plane cross section of the lithium-lead blanket concept are shown in Figure V-12.

V.5.2. Shielding

The design of the inboard blanket and vacuum vessel that lie between the inboard breeding region and the TF coils is optimized to minimize nuclear heat loading and radiation damage to the TF coil. Studies of the composition of the inboard region show that a steel/water shield backed with about 5 cm of Pb/B4C and a 30 to 40 cm thick vacuum vessel is an optimum choice. With this shielding configuration, the total radial build needed is 84 cm. The total nuclear heating in the TF coils in the inboard region will be 13 to 20 kW. The maximum insulator dose in the Technology Phase is estimated to be about 3 x 10^9 rads (these numbers include appropriate design safety factors).

The shielding in the region behind the divertor has also been optimized. Here, minimization of the total vertical build is the most important consideration. With about 60 cm of shield/vacuum vessel, the estimated total nuclear heating in the localized region of the TF coils behind the divertor is on the level of 30 kW with similar shield materials as in the inboard region (insulator doses are less than in the inboard region because of thicker TF coil cases in the divertor region). The total nuclear heating in the TF coils in both the inboard and divertor region is 60-70 kW or 50-60 kW in the Physics and Technology Phases correspondingly.

Shielding analysis of the effects of blanket assembly gaps and port penetrations has been carried out. For one cm-wide assembly gaps, safety factors of two on integrated responses and three on local responses are adequate to account for the effects of neutron streaming. The minimum shield for TF coils protection in the area of the NBI duct (at the place of closest approach to the backside of the coil) is 35 cm. The TF coil winding pack, in the region of the divertor/vacuum pumping duct, is adequately shielded by 35 cm of shield and vacuum vessel.
Fig. V-12. Lead Lithium Blanket Concept Design
The radial build of the outboard blanket/shield/vacuum vessel (150-170 cm) is adequate to limit the dose rate outside of the tokamak to 0.5 mrem/hour, 24 hours after shutdown, provided adequate shielding (typically one meter thickness for large ducts) is provided around all penetrations.

V.6. MAINTENANCE EQUIPMENT [8].

All components of the ITER tokamak device and auxiliary systems are designed with the need for maintenance and repair in mind. These fundamental requirements are reflected in the basic layout and configuration of the tokamak device. The facility is equipped with specialized equipment to perform the maintenance needed during operations.

V.6.1. Design requirements for maintenance

All components of the tokamak device are classified according to their requirement for maintenance. The classification is according to the need for scheduled or unscheduled maintenance, by the likelihood of maintenance, and by the impact of the maintenance procedure on operations and availability.

The first class of components is those that are known to require frequent scheduled maintenance (e.g., the divertor plates, first wall tiles). These components, and the associated remote handling equipment and service procedures, are designed together to minimize the replacement time. This will maximize availability.

The second class of components is those, which, while designed to last the life of ITER, may require unscheduled maintenance, or replacement (e.g. the in-vessel breeding blanket modules). These semi-permanent components are designed for full remote repair or replacement, but minimization of repair and replacement time is subordinate to considerations for component design, such as nuclear performance and operational reliability.

The third class of components is the basic machine systems, such as the torus vacuum vessel and TF and PF magnets. These components are expected to last the life of ITER, and major maintenance or upgrading are not anticipated. Here, if major maintenance operations should be needed they will require substantial disassembly of at least part of the tokamak device, and the projected maintenance time will be long. Although these components are designed to make disassembly and repair/replacement feasible by remote handling means, their design emphasize reliability and performance optimization.

Reliability and fail-safe design of the maintenance equipment itself are also required. All maintenance equipment will be designed so that no single-point failure can, (a) preclude the withdrawal of the equipment to its maintenance area, (b) result in the inadvertent release of the payload, or (c) cause any loss of confinement or accidental release of radioactivity.
V.6.2. Equipment.

The main item of equipment for handling heavy payloads within the reactor hall will be the main crane. Different lifting rigs will be used with the crane depending on the component handled. The main crane will be used for the initial assembly of the tokamak and for all subsequent replacement operations. The major components handled by the crane are: TF and PF coils, the TF intercoil structure, the vacuum vessel sectors, and the biological shield.

The remaining maintenance equipment consists of combinations of subsystems. Each subsystem typically includes a transporter, an end-effector, and a tool. Using a combination of such subsystems allows the number of large handling devices (e.g. transporters) to be kept to a minimum. The end-effectors and tools, of more limited dimensions, will be available to handle a large variety of components.

Transporters for in-vessel operations (Fig. V-13) include two 90-degree articulated booms, two 180-degree vehicle systems, and two transfer units for rapid transfer of plasma-facing components into and out of the vacuum vessel. Servo-manipulator and divertor plate end-effectors are provided for in-vessel operations. Special purpose tools for in-vessel operations include tools for tile replacement, leak detection, and cutting and welding.

Remote replacement of blanket modules will be carried out by dedicated devices operating from above the reactor. An illustration of this is given in Fig. V-14.

The details of the ex-vessel maintenance equipment and procedures are being developed in parallel with the final definition of the total device configuration and auxiliaries. However, here the primary items of equipment will include bridge- and floor-mounted transporters that will allow access from both the top or bottom of the machine, as well as from radial positions through the cryostat penetrations.

Preliminary engineering design of the in-vessel remote handling equipment is in progress. The design concentrates on feasibility, minimization of intervention time, recovery in case of failure, and interface compatibility with machine component design. Preliminary design of ex-vessel remote handling equipment is being carried out in parallel with the definition of the basic machine configuration.

V.7. CURRENT DRIVE AND HEATING [9].

The current drive and heating systems are intended to fulfill several functions: plasma initiation, current ramp-up assist, current drive, plasma profile control, heating, and burn control.

V.7.1. Concept description

The reference system is comprised of a 20 MW, 120 GHz Electron Cyclotron (EC) system for plasma initiation and profile control; a 50 MW, 5
Fig. V-13. In-Vessel Manipulator
GHz Lower Hybrid (LH) system for ramp-up assist and current drive in the outer region of the plasma; and a 75 MW, 1.3 MeV Neutral Beam (NB) system for current drive in the plasma core and profile control. During steady-state conditions, the combined power injected into the plasma will be about 110 MW and for an average efficiency of 0.4, the electrical power required from the grid is about 300 MW.

The EC system consists of 28 rf channels which transmit the waves to the tokamak where they are focussed by mirrors into the plasma; one of the mirrors is moveable. Four channels are used as active spares to ensure reliable operation. Each channel consists of an 80 to 100 kV, 3 MW, DC power supply; a 120 GHz, 1 MW gyrotron with integrated mode convertor; initializing optics; a corrugated waveguide approximately 45 meters in length; fast valves; and an rf window assembly. The EC waves will be injected into the plasma at the equatorial plane, at an angle of from 15 to 25 degrees to a machine radius and in the direction opposite to the plasma current. To ensure that the desired q = 2 magnetic surface will be tracked, a turning mirror is used capable of sweeping the EC waves in the horizontal direction at a rate of 10 degrees per 1 second (see Fig. V-15).

The LH system is similar to systems constructed for JT-60, Tore Supra and JET. The LH system will use two launchers made of 16 rows of 100 waveguides, and will occupy the lower half of the two adjucted ports. A grill launchers, are fed by a multi-junction waveguide arrays which in turn are fed by about 100 rf transmitters, each rated at 0.7 to 0.8 MW. The required frequency is 5 GHz. A good power coupling is expected when the grill apertures are flush to the first wall. However, due to the uncertainties in the prediction of the edge density as well as to accommodate various operating conditions, provisions are made in the design for a radial motion of the launchers and for high power load guard limiters. A representative layout of the LH wave system is given in Fig. V-16.

The NB system consists of nine modules, arranged in three vertical arrays of three modules. Each vertical array is aimed through a port aligned with a tangent to the magnetic axis of the tokamak. Each module nominally generates 8.3 MW of deuterium or hydrogen neutrals with an energy of 1.3 MeV. In the event of a source module failure, each module can be operated up to 10 MW which assures the ability of the system to reliably inject the required 75 MW. A module consists of one or more neutral beam sources; each source includes a negative-ion plasma generator with pre-accelerator, an accelerator, a beam profile controller, a gas neutralizer, and an ion dump system. A source module is about 4 meters in diameter and 15 meters long and will be connected to the torus through a 25 meters long duct (see Fig. V-17).

V.7.2. Alternate concept descriptions

An Ion Cyclotron system is made up of linear array of 42 antennas distributed around the torus. Two antennas are located side by side in the top half of a standard port and an additional antenna set is integrated into the blanket
segments of either side of a port (Fig.V-18). A port services six antennas, and seven ports will be required to service all 42 antennas. The antennas are 0.5 m wide by 1.25 m height and form a continuous array around the torus which is 24 m in extent. The nominal output for the antenna array is 130 MW. Each antenna, rated at 3.0 MW, is driven by a dedicated high power rf power source with a nominal rating of 3.25 MW. A low loss coaxial transmission line, which contains tuning and matching elements, will connect the antennas to the rf sources. The rf sources will be tunable over a frequency range of 15-80 MHz.

There are two supplementary systems. The first is a 20 MW, EC system very similar to the reference EC system but with a higher frequency, 140 to 160 GHz. The system would share the same port with the reference EC system. The second system is a 20 MW, 70 to 110 MHz IC system that features a single input launcher.
Fig. V-15. Electron Cyclotron RF System
Fig. V-16. Lower Hybrid RF System

Fig. V-17. Neutral Beam Injector System
Fig. V-18. Ion Cyclotron RF System
V.8 FUEL CYCLE SYSTEM [10].

V.8.1 Scope and primary requirements

The main elements of the fuel cycle system are shown in Fig. V-19. This system supports fuelling of the torus, torus vacuum pumping, processing of exhaust fuel, and recovery of tritium from the breeder blanket and test sectors. The system also supports several common plant operation processes such as hydrogen isotope separation, storage and management of fuel gases, and treatment of solid, liquid and gaseous tritiated wastes. Key requirements are listed in Table V-4.

V.8.2 System design

Plasma fuelling will be provided primarily by gas puffing in the divertor region. For fuelling during plasma current ramp-up, pellet injectors capable of producing a limited number (< 100/pulse) of pellets at velocities in excess of 2 km/s will be provided to produce peaked profiles. Additionally, to enhance flexibility during operation, pellet injectors will be provided which are able to provide continuous fuelling beyond the scrape-off layer (1.2 to 2 km/s).

Plasma chamber vacuum pumping during plasma operation and during dwell periods between pulses will be achieved using argon-spray compound cryopumps. The required 24 cryopumps are arranged in eight stations. For initial pump-down and torus conditioning, a system of turbomolecular pumps (TMP's) will be used (eight pumps, each with a capacity of 15 to 25 m³/s, integrated into the cryopumping stations).

The possibility of replacing the compound pumps with turbopumps is also being evaluated. The pump-layout will provide sufficient space for an all-turbopump option.

Roughing and backing will be provided by Normetex-type 600 plus 1200 m³/s scroll pumps. Here minor design modifications will be needed to reduce tritium inventory.

The design of the fuel processing system for torus exhaust will incorporate two process methods. In the first method, impurities will be cryosorbed on molecular sieves and subsequently oxidized to tritiated water. The water is then reduced to tritium plus oxygen using an electrolysis cell. NBI and pellet injector gas impurities will also be processed using this method. The second method, which avoids inventories of tritiated water, uses a permeation membrane followed by catalytic steps to "crack" hydrocarbons and reduce water vapour using the water-gas-shift reaction.

The design of the processing system for blanket tritium recovery supports the lithium-ceramic blanket option. The tritium in the blanket purge gas stream is recovered by cryosorption on molecular sieves, along with the swamping hydrogen and any impurities present. The impurities are subsequently separated and the tritium recovered using processes similar to those found in the plasma exhaust fuel processing system. Hydrogen and tritium are recovered on
Fig. V-19. Fuel Cycle Block Diagram
<table>
<thead>
<tr>
<th><strong>TABLE V-4. KEY DESIGN PARAMETERS OF FUEL CYCLE</strong></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>FUELLING AND EXHAUST</strong></td>
</tr>
<tr>
<td>Fuelling and exhaust rate, (moles/h)</td>
</tr>
<tr>
<td>Pump speed, (m³/s)</td>
</tr>
<tr>
<td>Effective pumping speed at torus during burn, (m³/s)</td>
</tr>
<tr>
<td>Ultimate pressure in torus (mbar)</td>
</tr>
<tr>
<td>Installed TMP capacity, (m³/s)</td>
</tr>
<tr>
<td><strong>BLANKET TRITIUM PROCESSING</strong></td>
</tr>
<tr>
<td>Tritium production rate, (g/full-power-day)</td>
</tr>
<tr>
<td>Ceramic breeder hydrogen swamping ratio, (H/T)</td>
</tr>
<tr>
<td>Maximum H + T recovered as oxide, (%-water)</td>
</tr>
<tr>
<td>Nominal production of tritiated water, (moles/d)</td>
</tr>
<tr>
<td><strong>COMMON PROCESSES</strong></td>
</tr>
<tr>
<td>Max. flow to isotopic separation, (mol/h)</td>
</tr>
<tr>
<td>protium</td>
</tr>
<tr>
<td>deuterium</td>
</tr>
<tr>
<td>tritium</td>
</tr>
<tr>
<td>Effluent water de-tritiation rate, (kg/h)</td>
</tr>
<tr>
<td>Discharge concentration, (Ci/kg)</td>
</tr>
<tr>
<td>Plant volumes requiring air-detritiation,</td>
</tr>
<tr>
<td>Air</td>
</tr>
<tr>
<td>helium</td>
</tr>
<tr>
<td>Max. local tritium inventory, design target, (g)</td>
</tr>
</tbody>
</table>

getter beds. The process uses generally proven concepts, although not yet demonstrated on a large-scale.

Alternate processing approaches are being examined. If a large fraction of the hydrogen and tritium can be recovered in oxide form, it may be efficient to oxidize the entire hydrogen flow and process it using liquid phase and vapour phase catalytic exchange. This process capability will likely be required in any event to process tritiated water wastes.

Finally, if the blanket purge-gas flow requirements are modest, a permeator concept can be used. This process requires a scale up of more than a factor of ten from today's parameters, and a consequent major re-design in the permeator mechanical structure. However, since it is continuous, this approach would permit a reduction of two to three in the tritium inventory in the blanket.

Tritiated water processing required to detritiate water collected in the facility (primarily from building driers) to concentrations suitable for environmental discharge will be accomplished with either distillation of water or
combined-electrolysis-catalytic-exchange. For water to be re-used in the plant, detritiation to a much less stringent level is required, and distillation of water is the process of choice.

Tritiated atmosphere processing will use proven catalytic oxidation and drier technology. Solid waste tritium recovery will involve heating (melting) and vacuum degassing of components such as metal components exposed directly to the plasma. Graphite waste containing substantial quantities of tritium will be either burned under controlled conditions with subsequent collection of the combustion products, or subjected to high temperature vacuum-degassing.

Isotopic separation will be based on cryogenic distillation of hydrogen isotopes, which has already been demonstrated at a scale sufficient for ITER. All hydrogen feeds may be processed in one column train, or alternatively, separate column trains, for substantially different feed compositions, may be used.

V.8.3. Tritium inventories in ITER

The principal inventories of tritium in ITER are shown in Table V-5. The total process inventory during the Technology Phase is approximately 800 grams. This inventory must be relatively mobile to facilitate processing. The maximum local vulnerable inventory concentration does not exceed 200 grams. It should be possible to significantly reduce these maximum local inventories during the Physics Phase by sizing certain components for lower throughput. Quantities of relatively immobile tritium in working storage (stored on metal beds) and tritium in solid waste awaiting processing (mainly in graphite powder) depend strongly on operating plans and conditions which have not yet been determined. For each of these categories, one kilogram may be regarded as a reasonable working assumption.

The Tritium vulnerable inventory is much lower than the total, therefore the maximum tritium release due to a hypothetical accident will not be a hazard to the general public living outside the site exclusion areas. See ref. [3].

Tritium in the driver blanket will be distributed among structural material, breeding material, and multiplier. The factors affecting blanket tritium inventory are discussed in the ITER blanket report [7]. Tritium inventories in the ceramic breeder blanket are relatively immobile under upset conditions.

TABLE: V-5. EXPECTED TRITIUM INVENTORIES IN ITER

<table>
<thead>
<tr>
<th>System</th>
<th>Physics Phase (g)</th>
<th>Technology Phase (g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Cycle</td>
<td>1000-2000</td>
<td>1200-2200</td>
</tr>
<tr>
<td>Solid Waste Processing</td>
<td>1000</td>
<td>TBD</td>
</tr>
<tr>
<td>Fuel Storage</td>
<td>300-600</td>
<td>1000</td>
</tr>
<tr>
<td>Breeder Blanket</td>
<td>20-200</td>
<td>100-1000</td>
</tr>
<tr>
<td>First Wall</td>
<td>~ 150</td>
<td>~ 150</td>
</tr>
<tr>
<td>Divertor</td>
<td>~ 1000</td>
<td>TBD</td>
</tr>
</tbody>
</table>
Tritium inventories in the divertor plate and FW armor are dependent on the design and operational parameters as given in the ITER plasma facing components report [6]. For the values given in the table, it is assumed that for the FW in the Physics Phase, about 50 m$^2$ of radiative tiles are used on the outboard wall, with the rest of the outboard and inboard tiles being conductively-cooled. For the divertor, five complete plate replacements are assumed during the Physics Phase, and the tritium inventory is due essentially to co-deposition.

The maximum tritium inventory and other activated materials that can be released during a hypothetical accident scenario are such that it does not require the population evacuation. See ref. [3].


V.9.1. Diagnostics Requirements

ITER has to provide the data base in physics and technology necessary for the design and construction of a demonstration fusion power plant. To achieve these goals, ITER must be sufficiently flexible to operate with a variety of plasma parameters, demonstrate the capability to control a burning tokamak plasma efficiently, and also must possess adequate diagnostic capability to characterize the physics of the ignited plasma thoroughly.

V.9.2. Plasma Diagnostics in ITER

Requirements for diagnostics in ITER will be demanding due to the plasma size and parameters, the limited access, the high level of background radiation, the need for remote handling, and the use of long pulse lengths. Diagnostic for ITER will be very extensive during the physics phase to provide the full profile and temporal information necessary to optimize the tokamak operation. A reduced set will remain for the technology phase to permit control of plasma performance.

Several types of abnormal behaviour of the discharge that may be dangerous for machine integrity can be expected in ITER. Uncontrolled rise of the fusion power and plasma disruptions can produce unacceptably high thermal fluxes on the first wall and divertor plates. High mechanical stresses can arise in the first wall elements due to high poloidal currents flowing through these elements during a vertical displacement event. Development of safety diagnostics capable of detecting the onset of these events and controlling them are of primary importance for ITER.

The diagnostics proposed for safety, control and plasma performance are listed in Table V-6.. This table, which also shows (in italics) additional
<table>
<thead>
<tr>
<th>Plasma parameter</th>
<th>Candidate diagnostics</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) Plasma current</td>
<td>Magnetics</td>
<td>Need to develop methods of measuring steady-state fields</td>
</tr>
<tr>
<td>2) Plasma position and shape</td>
<td>Magnetics</td>
<td>See above</td>
</tr>
<tr>
<td>3) $q(r)$ (current density)</td>
<td>Magnetics</td>
<td>Accuracy needs confirmation</td>
</tr>
<tr>
<td></td>
<td>Motional Stark effect</td>
<td>$l_t(r)$; see above</td>
</tr>
<tr>
<td></td>
<td>Faraday rotation</td>
<td>Needs diagnostic neutral beam</td>
</tr>
<tr>
<td></td>
<td>Reflectometry</td>
<td>Severe access limitations</td>
</tr>
<tr>
<td>4) Electron density Thomson</td>
<td>Interferometry</td>
<td>Needs diagnostic neutral beam</td>
</tr>
<tr>
<td>scattering</td>
<td>Reflectometry</td>
<td>Line-averaged; limited chord number</td>
</tr>
<tr>
<td>5) Ion/electron temperature</td>
<td>Neutron spectrometry</td>
<td>$n_e(r)$; density fluctuations are issues</td>
</tr>
<tr>
<td></td>
<td>Neutral particle analysis</td>
<td>Core plasma ($r &lt; 2a/3$) at DT phase</td>
</tr>
<tr>
<td></td>
<td>CHERS</td>
<td>Plasma periphery; needs diagnostic neutral beam</td>
</tr>
<tr>
<td></td>
<td>Thomson scattering</td>
<td>Needs diagnostic neutral beam</td>
</tr>
<tr>
<td></td>
<td>ECE</td>
<td>and radiation resistant optics</td>
</tr>
<tr>
<td>6) D/T density</td>
<td>Neutron particle analysis</td>
<td>$n_e(r)$; $l_t(r)$; density fluctuations are issues</td>
</tr>
<tr>
<td></td>
<td>Neutron spectrometry</td>
<td>Core plasma; issue: S/N for DD neutrons</td>
</tr>
<tr>
<td></td>
<td>Visible spectroscopy</td>
<td>Edge plasma; needs radiation resistant optics</td>
</tr>
<tr>
<td>7) Fusion power</td>
<td>Neutron yield monitor</td>
<td>Calibration methods need further development</td>
</tr>
<tr>
<td>8) Confined -particles</td>
<td>Collective Thomson</td>
<td>Needs validation on existing tokamaks and development of 1.5 THz radiation source</td>
</tr>
<tr>
<td></td>
<td>scattering</td>
<td>Needs diagnostic neutral beam</td>
</tr>
<tr>
<td></td>
<td>CHERS</td>
<td>Needs diagnostic neutral beam</td>
</tr>
<tr>
<td>9) Escaping a-particles</td>
<td>$g$-spectroscopy</td>
<td>Needs R&amp;D to demonstrate feasibility</td>
</tr>
<tr>
<td></td>
<td>Thermocouples</td>
<td>Slow response time</td>
</tr>
<tr>
<td></td>
<td>Faraday cups</td>
<td>Needs R&amp;D to demonstrate feasibility</td>
</tr>
<tr>
<td></td>
<td>Bolometers</td>
<td>Needs R&amp;D to demonstrate feasibility</td>
</tr>
<tr>
<td>10) Divertor plasma</td>
<td>Visible spectroscopy</td>
<td>Needs radiation resistant optics</td>
</tr>
<tr>
<td></td>
<td>Laser induced</td>
<td>Needs R&amp;D to demonstrate feasibility</td>
</tr>
<tr>
<td></td>
<td>fluorescence (LIF)</td>
<td>Severe erosion problems</td>
</tr>
<tr>
<td></td>
<td>Langmuir probes</td>
<td>Complicated plasma geometry</td>
</tr>
<tr>
<td></td>
<td>Reflectometry</td>
<td></td>
</tr>
<tr>
<td>Plasma parameter</td>
<td>Candidate diagnostics</td>
<td>Comments</td>
</tr>
<tr>
<td>------------------------</td>
<td>-----------------------------------------------</td>
<td>-----------------------------------------------</td>
</tr>
<tr>
<td>11) Erosion rate</td>
<td><em>Visible spectroscopy</em></td>
<td>See above</td>
</tr>
<tr>
<td></td>
<td><em>Tile markers</em></td>
<td></td>
</tr>
<tr>
<td>12) Heat loads</td>
<td>Thermocouples</td>
<td>Needs R&amp;D to demonstrate feasibility</td>
</tr>
<tr>
<td></td>
<td>IR/visible imaging Camera (for first wall)</td>
<td>Slow response time</td>
</tr>
<tr>
<td></td>
<td>IR thermometer (for divertor view)</td>
<td>Needs radiation resistant optics; access limitations</td>
</tr>
<tr>
<td>13) Helium concentration</td>
<td><em>CHERS/NPA</em></td>
<td>Needs dedicated neutral beam</td>
</tr>
<tr>
<td></td>
<td>Residual gas analyzers</td>
<td>Relation to He density in the plasma is uncertain</td>
</tr>
<tr>
<td>14) Radiative loss</td>
<td>Bolometers</td>
<td>Limited access; low S/N ratio</td>
</tr>
<tr>
<td>15) Impurity content</td>
<td><em>VUV spectroscopy</em></td>
<td>For divertor view; access problematic</td>
</tr>
<tr>
<td></td>
<td><em>X-ray spectroscopy</em></td>
<td>Issue: radiation hardening of crystals</td>
</tr>
<tr>
<td></td>
<td><em>Visible spectroscopy</em></td>
<td>Needs radiation resistant optics</td>
</tr>
<tr>
<td>16) Runaway electrons</td>
<td>ECE</td>
<td>Suitable in principle; needs validation</td>
</tr>
<tr>
<td></td>
<td><em>X-ray monitor</em></td>
<td>Analysis of likely capability is required</td>
</tr>
<tr>
<td>17) Disruption precursors</td>
<td>Magnetics</td>
<td>Possible but not universal</td>
</tr>
<tr>
<td></td>
<td>ECE</td>
<td>( T_e ) fluctuations; uncertain</td>
</tr>
<tr>
<td></td>
<td>Neutron camera</td>
<td>( n_e T_e ) fluctuations; uncertain</td>
</tr>
<tr>
<td></td>
<td>Bolometers</td>
<td>Radiation increase; not universal</td>
</tr>
<tr>
<td>18) Edge localized modes (ELMs)</td>
<td>D(T)-light monitor</td>
<td>Needs radiation resistant optics</td>
</tr>
<tr>
<td></td>
<td>Reflectometry</td>
<td>Suitable in principle; needs validation</td>
</tr>
<tr>
<td></td>
<td><em>Langmuir probes</em></td>
<td>Severe erosion problems</td>
</tr>
<tr>
<td></td>
<td>Magnetic probes</td>
<td>Need probes with good time response</td>
</tr>
</tbody>
</table>

*Diagnostics shown in italics are additional ones for the physics phase.*

diagnostics needed in the physics phase, lists the plasma parameters to be measured, the technique and some comments on the applicability of this technique to ITER.

Diagnostics that are suggested for the technology phase have to be accommodated on three large horizontal ports (i.e., ports #2, #10 and #14, shared with testing modules and pellet injectors) and some top ports and pumping ducts at the bottom. The more extensive set of diagnostics in the physics phase will be allocated on five large horizontal ports with additional access at the top and the bottom (Fig.V-20). Because of severe limitations for vertical diagnostic access, tangential sightlines are proposed (e.g., for interferometer,  \( n_e n_d \) neutron spectrometer and Thomson scattering). Many diagnostics (e.g., bolometers and IR cameras for divertor view,
Fig. V-20. ITER diagnostic allocations and sight lines
interferometer/polarimeter, vertical neutron camera, micro-fission chambers, ECE and microwave reflectometry) require minor or significant changes to be made in the design of the blanket modules. A candidate solution is the replacement of some outboard breeding blanket modules adjoining the diagnostic ports by shielding modules of a special design dictated by the diagnostic requirements.

For many of these diagnostics, conceptual designs have been developed. They show that an extensive Research and Development programme is needed to prepare the diagnostics for ITER. Experience in component behavior must be expanded greatly by extensive active in-situ radiation testing. New diagnostic techniques, to be fully tested as prototypes on predecessor tokamaks, are required because of the ITER plasma parameters and size and, possibly, because of geometrical constraints. A particular concern is the ability to calibrate instrumentation reliably after the tokamak has become activated and so a special study is required in this area.

REFERENCES:
