2.4 Divertor

2.4.1 Divertor Function and Main Components

The main function of the divertor system is to exhaust the major part of the alpha particle power as well as He and impurities from the plasma. As the main interface component under normal operation between the plasma and material surfaces, it must tolerate high heat loads while at the same time providing neutron shielding for the vacuum vessel and magnet coils in the vicinity of the divertor. Although good progress has been made in the understanding of divertor plasma physics, there continues to be some uncertainties, and hence the divertor remains an experimental device, which it is anticipated will need to be replaced and upgraded several times during the life of ITER. To facilitate rapid replacement, remote maintainability of the divertor has been given a high priority.

The main components of the divertor system are (see Figure 2.4-1):

- a divertor cassette body, that is reusable to minimise activated waste and provides neutron shielding and a mechanical support for different possible arrangements of plasma interfaces;
- inner and outer vertical targets, which are the plasma-facing components (PFCs) which in their lower part interact directly with the scrape-off layer (SOL) plasma and in their upper part act as baffles for the neutrals;
- the private flux region (i.e. the space below the separatrix which has no flux line connections to the main plasma) PFC which in turn consists of:
    - i) a dome, located below the separatrix X-point, seeing mainly radiation and charge exchange (CX) neutrals - the dome additionally baffles neutral particles and protects the liner and the neutral particle reflector plates from the SOL plasma;
    - ii) inner and outer neutral particle reflector plates that together with the lower ends of the vertical targets form a “V” shape that confines particles in the divertor channels to aid in reduction of peak heat flux by encouraging partial plasma detachment from the plate;
    - iii) a semi-transparent liner that protects the cassette body from direct line-of-sight of the plasma, while allowing He and other impurities to be pumped away;
- support pads integrated into the cassette to provide locking and alignment of the divertor cassettes on the rails;

- divertor to VV gas seals, to prevent back-streaming of gas from the divertor into the main plasma chamber;
- cooling pipe interfaces connecting the divertor cassettes to the radial cooling pipes at each divertor port;
- special diagnostic cassettes providing access for diagnostics;
- rails supporting the cassettes, part of the VV.

Three PFCs (inner and outer vertical targets and a private flux region assembly) are mounted on each cassette body in the hot cell by special semi-automatic tools.

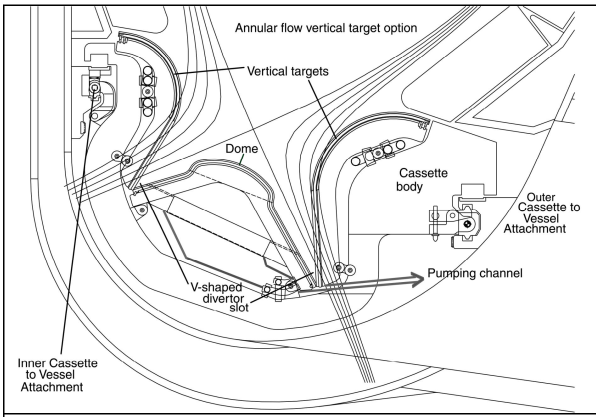

Figure 2.4-1a
Cross-section Through the Divertor
This shows the V-shaped strike zones and the pumping path as well as the attachment to the toroidal rails.

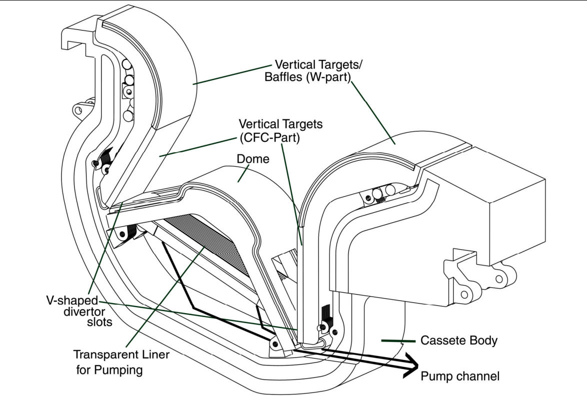

Figure 2.4-1b
Divertor Cassette Showing the Large Openings in the Dome Support
Also shown is the transparent liner allowing pumping access from the private flux region.

2.4.2 General Description

The geometry of the divertor is based on simulations obtained using the B2-EIRENE Monte Carlo code and by extrapolation from results of tokamak experiments[1]. The reference configuration for the ITER divertor is a vertical target/baffle with an open private flux region and a dome below the X-point (Figure. 2.4-1). The vertical target is inclined so as to intercept the magnetic field lines of the separatrix at an acute angle, giving deep inboard and outboard channels in which to establish a partially detached plasma regime. In this regime, while the plasma remains attached in the outer region of the SOL, the plasma is detached from the PFCs in the region near the separatrix, causing the power profile to broaden and power to be radiated to other surfaces. Together with the lower end of each vertical target, a neutral particle reflector plate forms a “V” shape that confines neutral hydrogenic particles in the divertor channels and aids partial plasma detachment.

Compared to a fully attached operating mode, the peak heat flux onto the water-cooled target PFCs is reduced by a factor of two, bringing the target incident heat flux into the range of engineering feasibility (see 2.4.4). For inductive operation, up to 100 MW of thermal power is delivered via the SOL to the inner and outer channels of the divertor, and hence to the PFCs that delimit the divertor channels. According to code simulations, partial plasma detachment will occur and a maximum incident heat flux on the vertical targets of

1ITER Physics Basis, Nuclear Fusion 39 (1999) 2137

< 10 MWm^{-2} is to be expected. For steady-state scenarios up to 136 MW of thermal power is delivered via the SOL to the targets, and preliminary simulations indicate a peak target heat flux in the order of 15 MW m^{-2}. Although the targets can certainly sustain this heat load since they are rated to handle transient (10 s) heat flux perturbations of up to 20 MWm^{-2}, it will be with a reduced lifetime.

In the reference design, inner and outer targets are inclined at 35.5° and 24.5° respectively with respect to the separatrix. Inclining the targets further, to 17.75° and 12.25°, to reduce the target heat flux during Type I ELMs can be achieved by changing the vertical target PFCs, while leaving the rest of the divertor unchanged. In fact, with these "steep" targets the ELM heat flux on the targets is expected to be reduced by 40 to 50% with respect to the reference design (this takes into account toroidal peaking factors). Based on present estimates of Type I ELM energy of 12 MJ, the peak energy deposited on the targets is estimated at 1.7 and 0.8 MJm^{-2} for the reference and steep targets respectively. Some broadening of the SOL is expected to reduce this peak, say a factor 2, to < 0.4MJm^{-2} for a steep target. This is comparable to the melt limit for tungsten[1]. This value might mean Type I ELMs will have little effect on a steep tungsten target lifetime. However, there are large uncertainties in the estimates of the ELM energy to each target and the extent to which broadening will mitigate the peak flux,. For partially attached plasma operation (normal operation) the heat flux is reduced by ~30% with respect to the reference design. Overall, B2-EIRENE code simulations indicate that operation with steep targets is tenable[2]. However, further study is needed to optimise the geometry and to gain confidence that the minor changes in the re-cycling pattern will not impair good H-mode confinement. The downside with this geometry is that narrower divertor channels are less accommodating to variations in the magnetic configuration than the reference design. Hence the steep target design is at present only considered for installation after initial operational experience has been gained with the more open divertor channel geometry of the reference design.

The baffle region is designed to provide a neutral particle pressure reduction of 10^4 between the divertor and the main chamber and of 10 between the private flux region below the dome and that above the dome. The baffle, positioned immediately below the X-point, is formed by the upper part of the divertor vertical targets and by the divertor dome. The baffle constitutes a toroidally continuous high heat flux surface following the 6 cm flux surface (measured at the outer plasma equator). Using the vertical targets extended upwards to form the baffle means that all the components, with the exception of the limiters, that are required to sustain surface heat loads > 1 MWm^{-2} are integrated into the divertor. This division of PFCs allows a uniform design of blanket modules and is expected to minimise overall in-vessel costs. Furthermore, incorporating the baffles in this way should localise future configuration changes to the divertor cassette alone. In addition, gas seals are employed between the divertor cassettes and the divertor cassette toroidal support rails in order to prevent neutrals from streaming back into the main plasma chamber.

Operating in the partially attached regime also eases the pumping, as the neutral pressure and the helium partial pressure in the private flux region will be increased considerably (0.3 to 10 Pa). This allows sufficiently rapid exhaust of the He ash to keep the He concentration in

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[1] G. Federici, J.N. Brooks, D.P. Coster, G. Janeschitz, A. Kukuskhin, A. Loarte, H.D. Pacher, J. Stober, C.H. Wu, "Assessment of erosion and tritium codeposition in ITER FEAT", J. Nucl. Mater. **290-293** (2001) 260-265.

[2] A. Kukushkin et al.,"Operational space of a shaped divertor in ITER", 28th EPS, Madiera, 2001

the main plasma below 6%. This exhaust gas passes through the semi-transparent liners located beneath the dome (see Figure 2.4-1b), from where it flows radially outwards beneath the outboard PFCs, then through a pumping slot (500 mm high by 100 mm wide) formed by matching cut-outs in adjacent cassette bodies, and from there to the cryo-pumps. The slot in the cassette body is behind the outer vertical target where there is low neutron streaming. To maintain a high enough gas pressure to achieve detachment in the outer divertor channel, without recourse to excessive gas puffing, a large unrestricted opening in the private flux region PFC is provided, beneath the dome and above the semi-transparent liner, to connect the inner and outer channels. This allows free re-circulation of neutrals from the inboard to the outboard private flux region and their re-ionisation in the outer divertor plasma (higher pressures in the inner channel are observed in experiments and predicted by modelling).

To maximise the divertor channel lengths, and hence maintain the peak heat flux on the vertical targets to ~ 20 MWm^{-2}, it is necessary to have the lower face of the cassette body closely follow the internal profile of the vessel, especially in the regions immediately beneath the targets. A gap of 70 mm between the bottom of the cassette body and the vessel is maintained to accommodate build tolerances, differential movement of the vessel and divertor under thermal or electro-magnetically induced loads, and to leave space for diagnostic cable runs as well as for pellet injector guide tubes.

The cassette concept employed for the RH class 1 divertor is fundamental for the maintenance strategy as it allows installation to be limited to a few integrated components inside the vessel, thus minimising the maintenance operations in-vessel and allowing short RH intervention times and high reliability. A few complete replacements of the divertor PFCs are anticipated throughout the lifetime of ITER, because of armour erosion and possibly because of changes in the divertor configuration. In these cases, only the PFCs will be substituted, while the cassette body can be re-used, thus minimising the amount of activated waste produced. The divertor is segmented into 54 cassettes, 3 per port. This is based on the maximum size of cassette that can be handled via the maintenance ports and still have an integer number of cassettes per port. Of the 18 lower ports, 3 equally spaced ports are allocated to divertor remote maintenance. These ports are 2 m high, which is large enough to allow the cassettes to pass through them during installation. The ports are inclined so as not to interfere with the inter-coil structure or with the building slab between the equatorial and lower ports. Each cassette is 3.5 m long, ~ 2 m high and 0.4 – 0.9 m wide, and weighs ~ 10.6 t.

The replacement of the PFCs is performed ex-vessel in a hot cell, where the refurbishment of 54 cassettes will take about half a year.

Of the 54 cassettes, 5 are designated diagnostic cassettes and accommodate waveguides and optical diagnostics for studying the plasma. These are located immediately in front of the maintenance ports 3, 9 and 15, and in front of ports 12 and 18. In addition, there are a further 10, “instrumented cassettes”, that carry diagnostics such as thermocouples, which require only minor adaptation of the cassette body and PFCs.

2.4.3 Armour Selection & Armour Issues

Carbon-fibre composite (CFC) is the reference armour for the strike point regions of the targets. Tungsten has been selected for all other plasma-facing surfaces of the divertor, for the baffle regions of the target and the surface of the dome where there are charge-exchange (CX) neutrals, because of its low sputter yield, and for the private region PFCs because of its low T retention and high melting temperature. Using this armour combination, the erosion lifetime of the PFCs is expected to meet the goal of sustaining 3,000 full-power discharges of 400 s duration, with one in ten discharges ending in a disruption where the SOL becomes fully attached to the target. Although CFC has been selected, an all-tungsten divertor is possible. The merits and demerits of both materials are discussed below.

Carbon is forgiving as an armour material since it will ablate as a result of disruption events or if target misalignments cause leading edges to intercept the SOL (~ 100 MWm^{-2}). There are no concerns over what happens to a melt layer. R&D has developed the technology to manufacture carbon-armoured PFCs with a demonstrated heat flux capability > 20 MWm^{-2}.

Carbon has one major drawback, the potential to chemically trap tritium in co-deposited layers at a rate estimated at 5 g ± 50% T/pulse of 400 s. Besides the presence of He and DT gas in the divertor, there are likely to be significant quantities of carbon and hydrocarbons that have been chemically and physically eroded from the vertical target. It has been shown in laboratory and tokamak experiments[1] that the nature of these hydrocarbons, i.e. the volatile and active species, is such that they can form thick, hydrogenated coatings on cool surfaces. (If they are deposited at high temperature, they form diamond-like hard layers and contain relatively little hydrogen.) Of real concern is the T that will be trapped in soft hydrogenated layers, which have a high proportion of hydrogen, that will build up on cold surfaces of the private flux region and, because of low sticking coefficients of the species that produce these soft layers (e.g with CH_3 radicals), can migrate downstream of the divertor (and elsewhere in the tokamak).

In order to overcome this, it was suggested that the liner should operate at a high temperature (800 to 1,000°C) and provide sufficient area for surface collisions to recombine the active components, such as H^0 and C_xH_y radicals, into volatile compounds that can be pumped safely away. Although this promises to reduce the amount of co-deposited T, maintaining the divertor exhaust pumping capability is not compatible with providing the required number of wall collisions (~ 1,000) to transform the whole amount to stable compounds that can be pumped.

A second approach considered a hot duct (> 350°C) leading to a cold trap for the hydrocarbons (~ 70K) located in a region sheltered in the cryo-pump ports where the lower neutron flux makes the cryogenic cooling system feasible. The intention is to capture the hydrocarbons that would otherwise contaminate the in-vessel surfaces and in particular, the cryo-pump. The cold traps would need regeneration to reclaim the T, and valves to close off the cold trap from the rest of the in-vessel system during regeneration. Detailed evaluation of this design is underway, but the complexity of including such a system is to be avoided if at all possible.

For both the above options R&D is underway to establish the range and proportions of the C_xH_y species that will be present, and the sticking probabilities versus temperature of these species in the presence of a large flux of atomic hydrogen. It is still possible that results from

[1] G. Federici, et al., Issues arising from plasma wall interactions in reactor-class tokamaks, to appear in Nuclear. Fusion.

this R&D could make one or other of the above options workable, or at least reduce the frequency at which clean-up operations need to be performed. Furthermore, it will be prudent to have available a method than can thoroughly clean C deposits from the in-vessel surfaces of the machine.

Possible methods of using oxygen to release the tritium are under investigation and consider the use of atomic O (molecular oxygen being not efficient enough). The possibilities are baking in ozone, and glow discharge cleaning in O. The results of these studies are as yet inconclusive. Whatever the chosen method, the whole clean-up operation, including re-conditioning, should fit with a tenable machine operation scenario. Introducing O into a tokamak raises concerns over the need to re-condition the first wall of the vessel and also over the damage that might be done to the beryllium surfaces.

On the other hand there are no concerns over T inventory if an all-tungsten armoured divertor is adopted. Encouraged by this possibility, tungsten armour technology has been developed to a level where the feasibility of building reliable targets capable of handling incident heat flux > 20 MWm^{-2} has been demonstrated[1]. Concerns over plasma contamination when operating with tungsten as a plasma-facing material have been partially allayed by the good plasma performance obtained while operating with high-Z, armoured PFCs in Alcator C-MOD[2] and more recently in ASDEX Upgrade[3]. However, there remain concerns over what will happen to the melt layer of the tungsten target (up to 80 μm deep during a disruption) and what effect an uneven re-solidified surface might have on subsequent operations and target life-time. Assuming that 50% of the melt layer is lost in each disruption, then a target lifetime similar to that of carbon is calculated (in fact disruption experiments show a much lower loss fraction). Finally, there are concerns over surface cracking of tungsten due to repeated disruptions and blistering of the tungsten surface due to hydrogen implantation. However, in experiments with tungsten at divertor relevant surface temperatures these effects appear be unimportant. R&D is continuing to study the performance of tungsten armour and the ASDEX Upgrade programme aims to operate with increasing amounts of the first wall covered with tungsten.

During the hydrogen operation of the machine, co-deposition is not an issue, but its importance and possible level can be fully ascertained. Hence, the most prudent scenario is to be able, if appropriate, to remove all carbon deposits at the end of the H phase, and to install an all-tungsten divertor before DT operation begins.

Finally on the issue of plasma-facing materials, the plasma wall interactions in ITER with materials such as Be, W, and CFCs is expected to generate substantial amounts of "dust", most of which will end up in the divertor region. This dust may be tritiated, radioactive, chemically reactive and/or toxic. Dust on the hot tungsten surfaces of the dome and upper part of the vertical target may promote reactions with steam during a water leak (Be-dust) producing sizeable quantities of hydrogen, or give rise to the possibility of explosion during

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[1] M. Merola, et al., Manufacturing and Testing of a Prototypical Divertor Vertical Target for ITER, 9th Int. Conf. on Fusion Reactor Materials, October 10-15, 1999, Colorado Springs, to appear in J. Nucl. Materials. A. Makhankov et.al. Development and Optimization of Tungsten Armour Geometry for ITER Divertor. Proceed. of 20 Symposium on Fusion Technology, Marseille, September 1998, p.267-270

[2] M. Greenwald, H Mode confinement in Alcator C-MOD, Nuclear Fusion, 37 (1997) 793

[3] K. Krieger, H. Maier, R. Neu, and the ASDEX Upgrade Team, J. Nucl. Mater. **266-269** (1999) 207

sudden air leaks (carbon dust). The dust on these hot surfaces will be held in the gaps between tiles. The general limit for in-vessel dust, based on allowable release to the atmosphere, is 100 kg for W, Cu, steel and Be, and 200 kg for C because of explosions. The actual rate of dust generation and its distribution in the machine will be studied during the hydrogen operation phase of ITER. In the meantime, R&D has been instigated aimed at finding methods to both measure the quantity of dust inside the ITER machine and to remove it during maintenance periods.

In summary, carbon remains the choice for the lower part of the vertical target where its ablative property makes it a very forgiving material against disruptions and target misalignments that can result in very high heat flux on leading edges. At present all the parties within ITER are investigating the physics of hydrocarbon radicals to decide whether designs incorporating carbon can result in acceptably low T deposition rates or if carbon has to be avoided altogether in a long pulse machine operating with T. In this case, an all-tungsten target is foreseen to be used for the strike point regions of the target.

2.4.4 Vertical Target

Each vertical target is based on a number of thin poloidal elements ~ 23 mm width; 27 elements for the outboard vertical target and 21 for the inboard. The upper part of the element is clad with W tiles and the lower part with CFC monoblocks. All the PFCs are constructed using a similar range of manufacturing techniques, demonstrated to be viable by R&D. This approach is intended to simplify manufacture and minimise costs, as it allows the critical fabrication steps, particularly those involving the armour-to-heat-sink joints, to be performed and qualified on small units. Each poloidal element employs a 10 mm bore, 12 mm outer diameter, copper tube with swirl tape (twist ratio = 2) inserted in the lower part of the vertical target. The swirl tape is required to enhance the critical heat flux (CHF) limit and provides an ~ 1.5 margin on 20 MWm^{-2} slow transients. Uncertainties in the precise location of the SOL strike point require the insertion of the swirl tape over an ~ 500 mm length. High velocity swirl flow (~ 10 ms^{-1}) is employed over the minimum length required in order to minimise the overall pressure drop through the vertical target and maximise the CHF capability (see 2.4.8). For the upper vertical target (heat flux < 5 MWm^{-2}) the swirl tape is not employed and instead the coolant channel is a smooth tube. Figure 2.4-2 shows the geometry of the inner vertical target. The poloidal plasma-facing elements are mounted onto a single steel support structure that incorporates toroidal cooling, resulting in a limited temperature difference (~ 50°C) within the structure, thus providing a stable base for the plasma-facing elements. In fact, finite element calculations show that the maximum deformation caused by the nuclear heating and worst case asymmetric heating of the plasma-facing surface is only 0.25 mm.

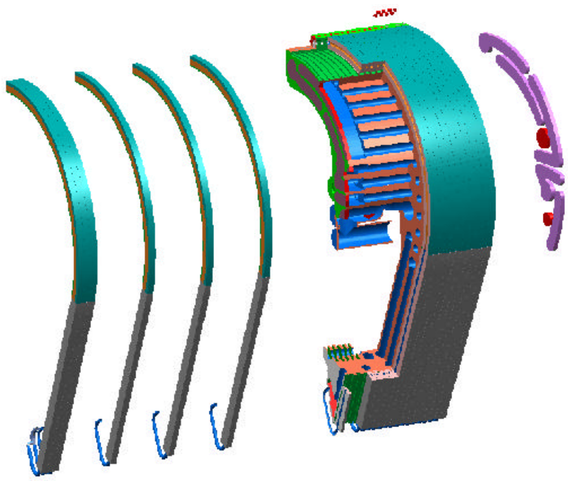


Figure 2.4-2
Exploded View of the Inner Vertical Target

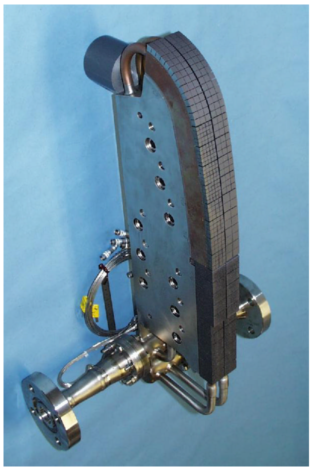


Fig 2.4-3
EU Prototype with CfC & W Armour

The most critical aspect of a PFC is the armour-to-heat-sink joint and the EDA R&D has seen impressive progress made in the development of both CFC to Cu and tungsten to Cu joints.

For the carbon armour joints, the bore of the CFC monoblocks are lined with a pure Cu layer cast onto a laser-textured and Ti-metallised surface, so-called active metal casting (AMC). The Cu in the bore of the monoblocks is machined to size prior to them being low temperature (~ 500°C) hot-isostatically pressed (HIP) to a CuCrZr tube. The precipitation hardened CuCrZr alloy has been selected over other Cu alloys because of its good post-irradiation fracture toughness. In fact, several techniques might be used for making the Cu-CuCrZr joint (furnace braze with fast quench, rapid brazing using ohmic or inductive heating, or HIP-ing), but the HIP process gives optimised mechanical and thermal properties, and minimises the residual strains in the critical Cu-CFC joint. The CFC monoblock has been shown to be a robust design for the CFC armour, and in tests[1] a prototype (Fig. 2.4-3) has survived 2,000 cycles at 20 MWm^{-2}. It is preferred over the less expensive flat tile design, because of concerns over the observed tendency for flat tiles to suddenly and totally detach.

For the upper part of the target, 10 x 10 x 10 mm tungsten tiles, with a pure Cu layer cast onto the tungsten to accommodate the differential thermal expansion of tungsten and carbon, are brazed to a CuCrZr alloy rectangular hollow section, a design that has sustained > 20 MWm^{-2} for 2,000 cycles in HHF testing[2]. An added benefit of this design is that, by faceting the surface of the Cu heat sink, the armour can be applied to the curved upper part of the target.

The vertical targets have been analysed for the various operational structural loads. Figures 2.4-4 & 2.4-5 show the temperature distribution due to the transient nuclear heating for the ITER duty cycle and the corresponding von Mises equivalent stress. The main loads

[1] M. Merola, et al, Manufacture & Testing of a Prototypical Divertor Vertical Target for ITER, 9th Int. Conf. On Fus. Reactor Materials, October 1999, Colorado Springs.

[2] S. Chiocchio et al; "The Divertor for the Reduced Technical Objective/Reduced Cost ITER" SOFE, Albuquerque, Oct. 1999.
R. Tivey et al, "ITER Divertor, Design Issues and R&D", Fus. Eng. & Des. 46(1999) 207-220

on the targets are the eddy current loads due to a fast VDE. Finite element analysis shows that the combined loads, secondary plus primary membrane and bending, are within the ITER structural design criteria allowable for category III event.

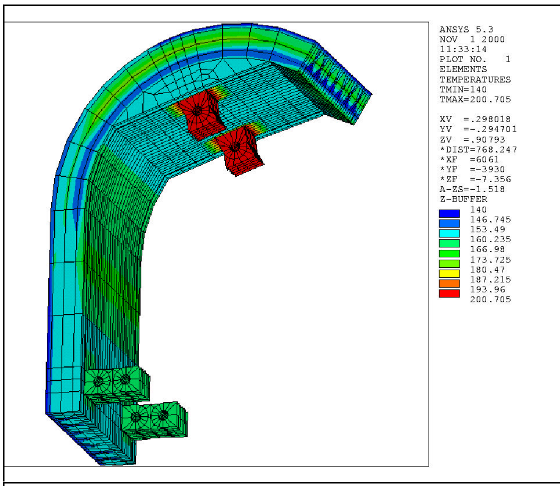


Figure 2.4-4
Temperature Distribution in the Outer Vertical Target for Transient Nuclear Heating

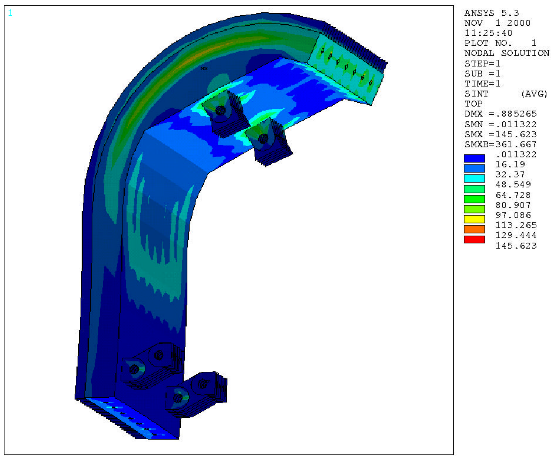


Figure 2.4-5
Stress Intensity (MPa) in the Outer Vertical Target under Thermal Load and Dead Weight

With regard to the low cycle fatigue (LCF) life-time of the heat sink for the reference monoblock design and the coolant parameters described in 2.4.8, analysis indicates a life-time of 2.8×10^5 cycles at 10 MWm^{-2}, and 806 cycles at 20 MWm^{-2}.

An alternative to this reference design that has been investigated is an annular flow heat sink. The advantages of this concept are the removal of expansion pipes at the base of the monoblocks bringing improved reliability and relieving space in an over-congested region and the potential for cost savings due to the lower number of plasma-facing elements (17 instead of 27 elements for the outer vertical target). Unlike the reference design (12mm OD swirl tube), which needs optimum CuCrZr mechanical properties to satisfy LCF requirements obtained using a low temperature HIP cycle during manufacture, the annular flow monoblock (tube outer diameter 20-22 mm) meets these requirements with the non-optimum CuCrZr properties obtained in a furnace brazed construction. During testing a CFC armoured annular flow mockup, with furnace brazed CFC/CuCrZr joints, survived 3000 cycles at 20 MWm^{-2}, thus demonstrating the viability of the concept.

2.4.5 Private Flux Region PFCs

As mentioned in 2.4.3, there are several options still under consideration for the private flux region PFCs, and the final choice will depend on the armour selected and the success of on-going R&D. However the divertor cassette approach has the flexibility to accommodate any of the likely design solutions. The PFC design described here is the simplest version without cold trap, etc (see Fig 2.4-6).

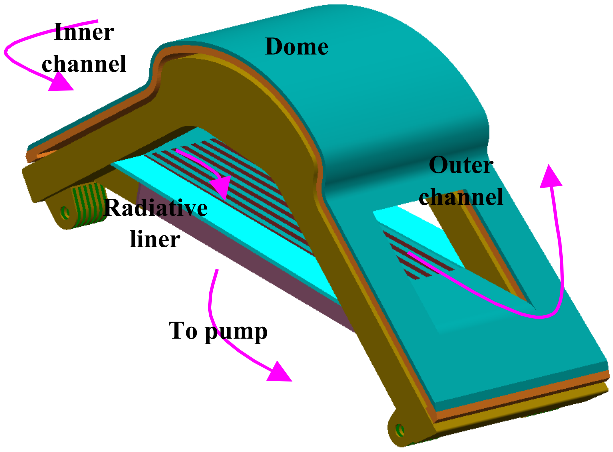


Figure 2.4-6 Open Dome Design with Radiative Liner

The dome and inner and outer short targets of the private flux region PFCs are clad with tungsten tile armour that is attached using a similar technique as used for the upper parts of the vertical targets. The dome is supported on four posts which are protected with a combination of tungsten tile armour (surfaces facing the divertor channels) and radiatively cooled tungsten plates. As part of the PFCs, a semi-transparent liner clad also with radiatively cooled tiles is suspended above the cassette body. In this way, an open duct is formed beneath the dome, connecting the inner and outer channels of the divertor, that is clad completely in radiatively cooled tungsten plates. This allows the surfaces of the duct to be maintained at temperatures > 350°C for the majority of the 400 s discharge.

2.4.6 Divertor Cassette Body

The stainless steel cassette body that supports the PFCs is designed to withstand the electromagnetic forces, provide shielding for the vacuum vessel and coils, and incorporates internal coolant channels, which cool the cassette body and act as manifolds for the PFC coolant. Construction of the cassette body both from steel castings and from forged steel plates have been considered. R&D has shown that either approach has the capability to produce a robust vacuum-compatible component. However, cast 316LN material has ~ 30% lower yield strength than forged material, and hence fabrication from forged steel is preferred, as it allows the cassette thickness capable of sustaining the electromagnetic loads to be ~ 50 mm thinner in the critical region beneath the inner divertor channel, a thickness that can be allocated to the inner channel depth. Two cassette-to-vacuum-vessel support options have been considered, one with the cassette attached to the vessel by a pinned support both inboard and outboard of the cassette, and the second with a pinned connection at the outboard but at the inboard a support that allows radial translation but no vertical displacement (sliding support). The worst case loads are caused by halo currents flowing through the cassette as a result of VDE II and VDE III in combination with internal pressure and thermal loads.

The stress levels are such that only the pinned inboard and outboard option is acceptable, even if preliminary analysis indicates that only ~ 50% of the halo current is carried by the cassette. The resulting maximum primary stress is 64 MPa, assuming all the halo current is carried by the cassette (I_{cass}= $0.29 I_p Pf / N$, where Pf is a peaking factor taken as 2 and N is the number of cassettes).

Figure 2.4-7 shows the von Mises equivalent stress distribution for the combined load case of 100% halo current, internal pressure, and nuclear heating. Table 2.4-1 summarises the different stresses in the cassette body with the corresponding margins on the allowables. For this pessimistic load condition the pinned inner and outer supports provide a margin of > 10%.

Table 2.4-1 Stress Estimations from ANSYS Analysis For the Case of Inboard and Outboard Supports Pinned.

			Dynamic loading factor	Allow. Stress MPa	Halo[1] MPa	Margins	Halo + Nuclear Heating MPa	Margins
CASSETTE BODY	Primary + Secondary Stress	Category I VDE I	1.2	340[3]	32	8.8	291	1.2
		Category II VDE II	1.2	340[3]	48	5.9	297	1.1
		Category III VDE III	1.2	340[3]	64	4.4	304	1.1
	Secondary Stress	Category I VDE I	NA	340[3]	-	-	279	1.2
		Category II VDE II	NA	340[3]	-	-	-	-
		Category III VDE III	NA	-	-	-	-	-
	Primary Stress P_m+P_b	Category I VDE I	1.2	200[4]	32	5.2	12	14
		Category II VDE II	1.2	200[4]	48	3.5	18	9.2
		Category III VDE III	1.2	240[4]	64	3.1	24	8.3

Maximum Displacements

	Axial Directions	Positive direction mm	Negative direction mm
U_{MAX}	Radial	1 (H+NH)[5]	4.5 (H+NH)
	Toroidal	1.03 (H+NH)	-
	Vertical	1.8 (H+NH)	5.5 (H+NH)

1 *Internal pressure + Halo (slow VDE).*
2 *These stresses come from a transient analysis.*
3 *Allowable @Temperature 350°C*
4 *Allowable @Temperature 200°C*
5 *Halo + Nuclear Heating (the analysis with the maximum displacement)*

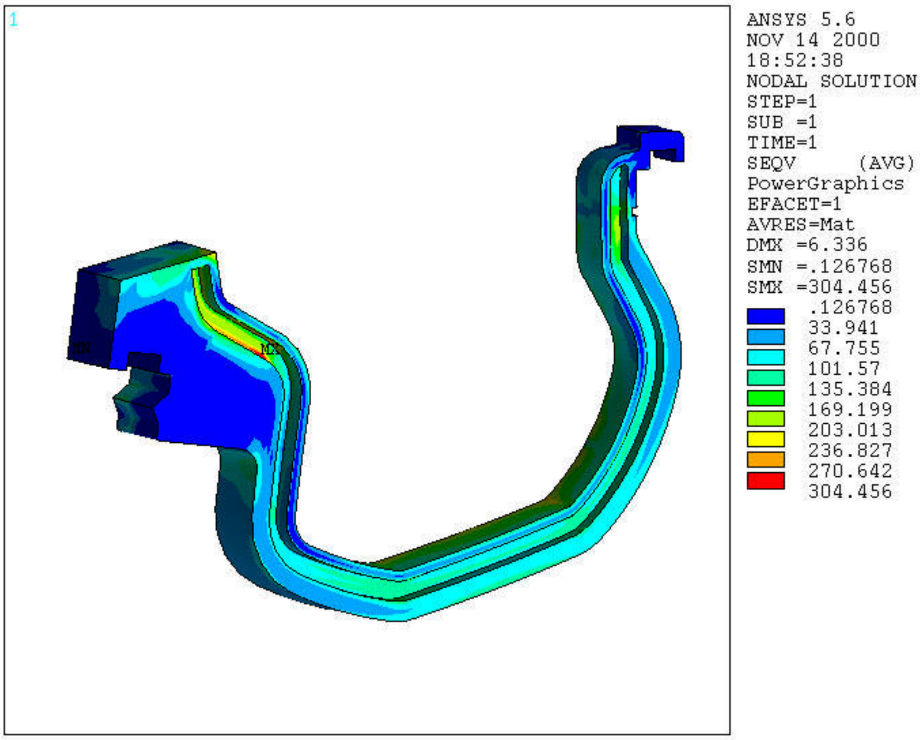


Figure 2.4-7 Von Mises Equivalent Stress in the Central Cassette Body for Primary Plus Secondary Stress with the Inner and Outer Supports Pinned.

The loads caused by eddy currents generated in the PFCs, as a result of disruption types I and II, are less severe for the cassette body than those caused by halo currents during VDEs. However, they determine the cassette body design local to the PFC-to-cassette attachments. A cassette minimum thickness of 240 mm is maintained in order to provide neutron shielding of the VV and coils. The coolant channels within the cassette body that service the PFCs combine to represent an average of 21% of the volume of the cassette body (the acceptable level for both coils and VV being in the range 10-30% water).

2.4.7 Alignment

The scrape-off layer strikes the vertical targets at a glancing angle (< 2°). Targets on adjacent cassettes must be aligned accurately with respect to one another and angled slightly so as to shield the leading edges of the adjacent components from direct incidence of the field lines in the scrape-off layer. A maximum radial step in the toroidal direction between adjacent targets of 4 mm is taken as a requirement. The overall alignment of the divertor is less critical, provided the deviations from the nominal are of a long wavelength (i.e. occurring in an arc spanning several cassettes), and an overall tolerance of ± 10 mm is taken as a requirement. These demanding tolerances are met by a combination of accurate manufacture of the cassettes and in-vessel toroidal rail support points, and if necessary, adjustment of the attachments prior to installation in order to suit the as-built dimensions of the toroidal rails.

2.4.8 Divertor Cooling

The main driver for the cooling layout is the need to maintain an adequate margin (> 1.4 based on experimental results) to the critical heat flux (CHF) limit, and not, as may at first be thought, the total input power to the divertor.

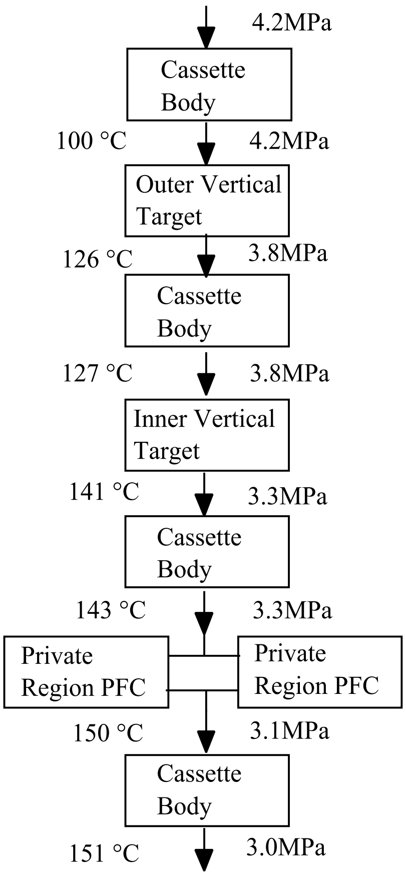


Figure 2.4-8 Schematic of Coolant Circuit in a Divertor Cassette

Table 2.4-2 Divertor Coolant Parameters, and Margins on CHF for each PFC

Inlet pressure	4.2	MPa
Inlet temperature	100	°C
Flow rate	734	kgs^{-1}
Total pressure drop	1.2	MPa
Total temperature increase	~50	°C
Axial velocity in the swirl section of the outer target	9.0	ms^{-1}
Axial velocity in the swirl section of the inner target	11.9	ms^{-1}
Minimum margin to CHF in the outer vertical target (based on 20 MWm^{-2})	1.65	
Minimum margin to CHF in the inner vertical target (based on 20 MWm^{-2})	1.53	
Minimum margin to CHF in the dome (based on 3 MWm^{-2})	2.84	

Data on the mechanical properties of irradiated CuCrZr indicate that the sub-cooling that a 100°C inlet temperature provides allows the coolant to be fed in series through the PFCs of the divertor (Fig. 2.4-8 & 2.4-9). A higher coolant velocity is employed in the downstream inner vertical target in order to compensate for its lower sub-cooling and to achieve a similar margin on CHF as the outer vertical target. The coolant parameters and margins on CHF for each PFC are given in Table 2.4-3 for the reference Inductive Operation Scenario I plasma.

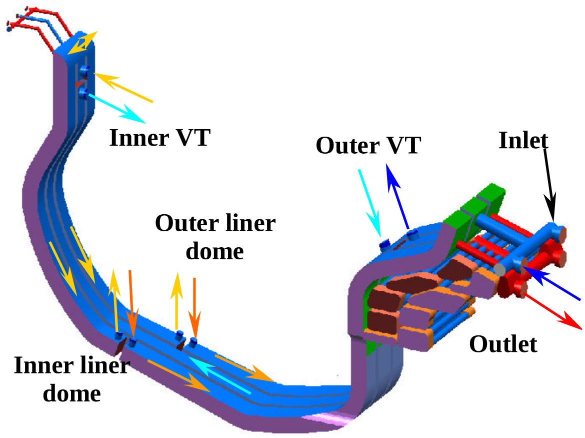
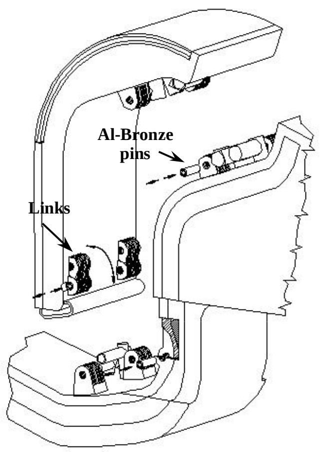


Figure 2.4-9 Coolant Routing within the Cassette Body.

2.4.9 Scheme of PFC Attachment to Divertor Cassette Body

Each of the 3 PFCs are supported from the cassette body using two pairs of remotely maintainable attachments (Fig. 2.4-10). One pair of attachments, the lower ones in the case of the vertical targets, accommodates both rotation and translation. A series of links engage at one end in slots in the PFC and at the other in slots in the cassette body. The holes in the links are aligned at one end with holes in the cassette body and at the other with holes in the stainless steel support structure of the PFCs. Thick aluminium-bronze tubes (~ 30 mm outer diameter and ~ 20 mm bore) are then inserted into each of the holes of the inter-linked components (clearance of 0.5 mm) and one at a time the pins are expanded by drawing a mandrel through them. Aluminium-bronze is selected because of good wear characteristic when operating in vacuum against stainless steel. The second attachment uses only one expanded pin and is capable of rotation only. In this case no links are needed, and instead a series of tongues (part of the cassette body) are inserted into slots in the PFC. These links react the loads caused by eddy currents in the PFCs induced during disruptions. For PFC replacement in the hot cell, the attachments are removed by simply drilling out the pins using the central hole as a guide. Coolant pipe connections (60 mm dia) between the PFCs and the cassette body are located adjacent to the pinned supports, where the minimum relative displacement occurs. Helium generation in these tubes is estimated to be ~ 5 appm over the life of ITER, which is beyond the assumed re-weldability limit. However, by ensuring that, at each PFC replacement, part of the tube belonging to the most recently replaced PFC remains with the cassette, the He content in welded material will remain acceptable. Analytical assessments indicate that the attachments can carry the currents flowing between the PFCs and the cassette body during disruptions or VDEs. Preliminary R&D performed in air simulating the electrical and mechanical loads and the applied joint rotation shows no joint degradation and indicates that separate earth straps are unnecessary.



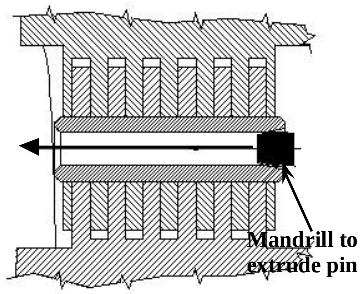


Figure 2.4-10
Attachment Scheme of the PFCs Using Copper Alloy Pin Extruded into Cassette Body and PFC Attachment Lugs.

2.4.10 Cassette to VV Attachments

During operations, the cassettes are accurately held in position on two toroidal rails (Fig. 2.4-1), attached to the VV. During shutdowns, these rails are used for remote maintenance of the divertor. The cassette is positively and remotely locked to both the inboard and outboard rails (see above), except in the case of the 3 cassettes positioned immediately in front of the maintenance ports. These are positively locked at the outboard and held at the inboard by applying a radial pre-load (~ 30 t), which, taking advantage of the radial spring stiffness of the cassette body ($k = 2.5$ t mm^{-1}), radially compresses the cassette by 13.7 mm and results in a peak stress in the body structure of 230 MPa (VDE III + pre-load). The pre-load ensures that the cassette remains in contact with the inner support attached to the VV under all loading conditions, including that caused by the radial differential thermal expansion between cassette and vacuum vessel. Each support allows rotation around a toroidal axis and can sustain all normal and off-normal vertical, horizontal and radial loads. As in the case of the PFC to divertor attachments, R&D is expected to show that separate earth straps are an unnecessary complication.

2.4.11 Divertor Gas Seal

The divertor-to-main-chamber gas "seal" (actually a baffle) is achieved by the cassette-to-toroidal-rail attachments. It is designed to meet a leak tightness of < 5 Pam^3s^{-1} for a ratio of 10^4 in pressure between the divertor and main chamber at a pressure of 1 Pa in the divertor area.

2.4.12 Diagnostic Cassettes

The 3 diagnostic cassettes located in front of the maintenance ports have an appendage that protrudes into the port. This supports wave-guides and optical transmission lines. As a result these cassettes weigh an additional 1.7 t making them 12.3 t.

All the diagnostic cassettes carry diagnostics on the side wall of the cassette body (Fig 2.4.11). These diagnostics include Langmuir probes, thermocouples, pressure gauges etc. The diagnostics are supported on localised stainless steel or copper alloy support structures bolted (or possibly riveted) to the cassette sides and are cooled without water, radiatively and by conduction through the attachments to the cassette body. The side plates, shown in the figure schematically, are in fact carried on strong-backs/jigs that support and correctly position the diagnostics during handling prior to their attachment to the cassette. The diagnostic support plates then remain with the cassette and use a minimum of material to keep down the nuclear heating. They are locally split so as to allow thermal expansion without distortion and take their strength during operation from the cassette body and PFCs to which they are rigidly attached. As far as possible, the diagnostics, associated cabling, and support plates, are inserted in recesses or grooves in the side plates of the cassette body and the PFCs.. In this way the recessed diagnostics are protected during divertor maintenance, and the neutron shielding performance of the divertor as a whole is unaffected.

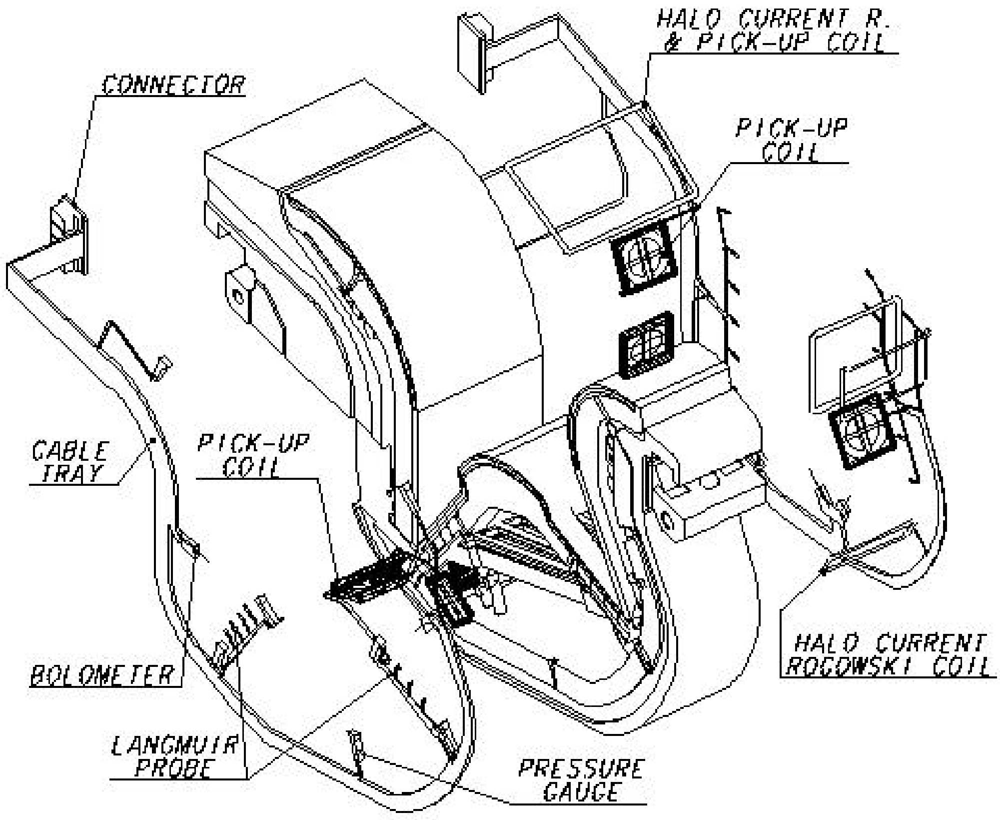


Fig. 2.4.11 Exploded View of Instrumented Cassette

Similar side-plates accommodate the wave-guides. However, there are two types of wave-guide (10 mm and 20 mm wide) and the 20 mm wide version will require a deeper cut into the cassette body side wall.

The cassettes that incorporate optical diagnostics have a mirror assembly in the private region beneath a semi-transparent liner with slots are cut in it to allow viewing of the divertor channels. A hole running radially through the outboard region cassette body provides the route for the optical transmission lines.

2.4.13 Further Divertor Integration Issues

Of the 18 lower ports, 10 contain cryo-pumps (see 2.7), 3 are allocated to remote maintenance (see 2.9) and contain diagnostic blocks supporting wave-guides and optical transmission lines (see 2.6), 2 are allocated primarily to diagnostics, and 3 accommodate part of the pellet and gas injection systems. In addition, the cryo-pump ports host the glow discharge cleaning heads (see 2.7) and the laser-based, in-vessel viewing and metrology system (see 2.9). Detailed integration studies have generated designs that accommodate all the requirements for the maintenance and cryo-pump ports, including remote handling needs and locations for divertor coolant feed pipes (6 per port).

Fig 2.4-12 shows the maintenance ports with diagnostic racks carrying waveguides or optical transmission lines installed. The rack is held on the radial rails that carry and guide the RH transporter during maintenance interventions. Six coolant pipes that feed the divertor cassettes are located either side of the port.

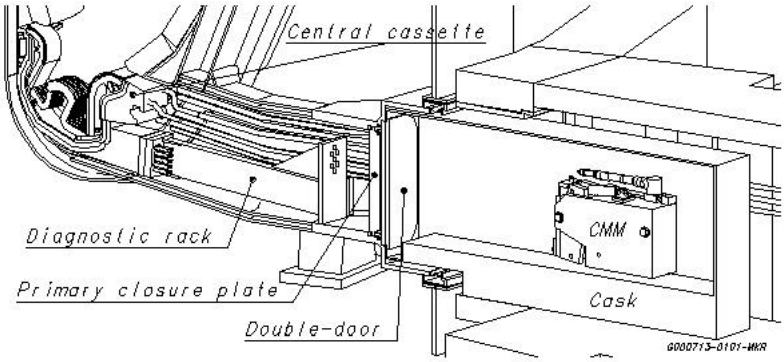


Figure 2.4-12 Maintenance Port with Diagnostic Equipment Installed

For fuelling, the divertor incorporates 3 gas valve lines in ports 1, 7 & 13. The gas is routed via the rail into the cassette, the seal between rail and cassette being achieved by contact pressure alone, which require no in-vessel RH operations. Gas lines are routed, through channels in the toroidal rails and the cassette bodies, to positions beneath the separatrix in the divertor dome. Pellet fuelling gas lines are included in these ports. These closely follow the vessel wall beneath the divertor and pass through the inner toroidal rail to the blanket.

2.4.14 Divertor Overall Assessment

The above divertor design provides an engineering solution compatible with today's plasma physics expectations. Its detailed specification has been thoroughly investigated through R&D, some of which is still underway. Given the uncertainties in the plasma physics extrapolation, and thus the component durability, the design provides a system for rapid replacement and refurbishment. The present design provides an accurate mechanical support, and flexibility to change the plasma-facing geometry as well as the plasma-facing materials on the PFCs. It is expected that R&D already active in present tokamaks (JET, JT60, DIII-D, ASDEX-U) will provide results on which an optimized material choice can be based before manufacturing starts, to be confirmed by initial experimentation in ITER (during the H phase) If needed, the divertor PFCs in the cassettes can readily be replaced to provide a different geometry and/or different plasma-facing material.