# Divertor Design and its Integration into the ITER Machine

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**Abstract:** The physics of the edge and divertor plasma is strongly coupled with the divertor and the fuel cycle design. Due to the limited space available the design as well as the remote maintenance approach for the ITER divertor are highly optimized to allow maximum space for the divertor plasma. Several auxiliary systems (e.g. in vessel viewing, glow discharge electrodes…) as well as a part of the pumping and fuelling system have to be integrated together with the divertor into the lower level of the ITER machine. Two main options exist for the choice of the plasma-facing material in the divertor, i.e. W and CFC. Based on already existing R&D results one can be optimistic that the material choice will be mainly based on physics considerations and material issues (e.g. C-T co-deposition). The requirements for the ITER fuel cycle arise from plasma physics as well as from the envisaged operation scenarios. Due to the complex dynamic relationship of the fuel cycle subsystems among themselves and with the plasma, codes are employed for their optimization. This paper elaborates these interacting issues and gives the latest design status.

## 1. Introduction

The divertor design and its integration into the ITER machine is on the one hand driven by the physics of the edge and divertor plasma and on the other hand by boundary conditions arising from the space available in the vacuum vessel below the X-point as well as in the divertor level ports. The latter is in particular important for integration of the high vacuum pumping system (6/10 cryopumps), for the divertor cooling pipes (6 per port), for diagnostic access to the divertor plasma (6 ports) and to provide sufficient access for the remote handling (RH) tools (3 RH ports) which are needed to remove or install the divertor from/into the vacuum vessel. In addition, access has to be provided at the divertor level (through the divertor ports) for 3 gas injection systems, for 3 pellet injector guide tubes, for 6 evenly distributed in-vessel viewing (IVV) systems, 6 glow discharge probes (GDC), and possibly for several light manipulator arms (up to 6) suitable for deployment into the vessel while it is under vacuum and with the toroidal magnetic field switched on.

The divertor itself consists of cassettes (54) onto which the high heat flux (HHF) components are mounted. The divertor geometry for the ITER machine was initially defined by simple rules [1] based on the experience from modelling the 1998 ITER divertor design, and was then refined using the B2-EIRENE edge plasma code. Due to the reduced space available between the X-point and the vacuum vessel compared to the 1998 ITER design, a modified RH concept for the transport of cassettes through the divertor ports and a relocation of the toroidal rails was necessary [1]. In addition the pumping channel was moved from below the cassettes into the space between cassette body and HHF components.

Two plasma facing materials are used in the divertor, namely CFC for the highly loaded strike zones and W for the upper vertical targets (VT) and for the private region liner and dome HHF components. However, R&D performed on C-chemistry in a plasma environment similar to that expected for ITER suggests that a significant amount of hydrocarbon molecules will be produced, some with very low sticking coefficients which are thus able to travel large distances before being deposited [2,3,4]. Therefore C-T co-deposition can arise far away from the divertor cassettes, aggravating the T retention problem. Two possible solutions for this problem exist and will be briefly discussed below, namely to find a method for trapping these hydrocarbon molecules close to the divertor cassette or to avoid the use of CFC-clad components in ITER. More R&D and analysis will be needed before a final decision can be made.

The design of the fuel cycle, which includes plasma fuelling by gas puffing and pellet injection, plasma exhaust by in vessel cryo-pumps together with a set of remotely located roughing pumps, as well as exhaust processing and isotope separation in a tritium plant, is strongly coupled to the edge plasma physics, the divertor design, and in particular to the requirement for plasma He exhaust. The fuelling cycle sub-systems have a complex dynamic relationship among themselves as well as with the plasma. Thus plasma edge modelling codes (e.g. B2-EIRENE) as well as a code describing the dynamic behaviour of the complete fuel cycle, are employed to optimize the different subsystems of the fuel cycle [5].

## 2. Plasma Edge and Divertor Modelling Results influencing the Divertor Design

By modelling the behaviour of the divertor plasma with B2-EIRENE for different divertor target geometries it was found that the peak power load on the vertical targets depends strongly on the existence of a V-shaped geometry near the strike zone. Thus, if the bottom part of the divertor targets form a distinct corner (V-shape), and if the separatrix strike-point is located near this corner, then the hydrogenic neutrals become locked in the vicinity of the strike-point, favouring a partial detachment of the separatrix [6,7]. The reduction of the peak power load arises from an increased neutral density near the strike zone, yielding higher charge exchange (CX) and radiation losses and possibly from changes in the plasma flow pattern inside the divertor. For a given set of plasma parameters the peak power load on the outer VT can be reduced by as much as ~ 30% in the presence of a V-shaped target [7] when compared to a straight VT.

Another way of lowering the peak power load is to reduce the pumping speed and thus again to increase the neutral density inside the divertor. However, in contrast to the geometry variation where the He exhaust remains largely unchanged, a reduction of the pumping speed causes an increase of the He concentration in the main plasma.

A second important result found from the modelling studies is the understanding of the need to provide a large conductance for the passage of neutrals from the inner to the outer divertor channel. This is supported by evidence from JET when comparison is made of experiments with a septum between inner and outer strike zone and those with no septum [8]. The larger neutral pressure in the inner divertor channel compared to the outer, caused by the lower plasma temperature in the inner strike zone and pressure constancy along field-lines as well as possibly by poloidal drifts, can provide a significant supply of neutrals to the outer strike zone. This again helps to reduce the peak power load there on the outer vertical target [7]. To minimise the power to the outer vertical target it has been found that, as a minimum, a 50% probability of entering the private flux region is required for each neutral streaming from the inner divertor channel towards the private flux region as well as a similar probability for neutrals streaming from the private flux region through the outer liner to the outer strike zone. A more detailed discussion of all the findings from the modelling study is given in [7].

### 3. Divertor Geometry and HHF Component Design

Based on the above modelling results, the geometry of the private flux region plasma-facing components (PFCs) has been modified to provide a narrow V-shaped channel in the area of the strike zones as well as a large opening in the inner and outer support of the dome (FIG. 1) in order to facilitate the required large conduction for hydrogenic neutrals. The walls of this penetration and the surfaces facing the divertor plasma are lined with radiatively cooled ~ 5 mm thick W tiles. The divertor cassette body is shielded from the plasma radiation, which would shine through these large openings by a louvered (and thus transparent) structure connecting the bottom of the inner and outer opening in the dome support (FIG. 1b). These louvers provide sufficient conductance for pumping [5] while protecting the cassette body from the divertor plasma radiation. The large opening between inner and outer strike zones yields the required conductance for neutrals and thus contributes to a reduction in the outer VT peak power load.

Beyond the louvers the pumped gas is conducted between the bottom of the outer divertor channel HHF components and their attachments to the cassette body, and then through a hole in the cassettes (reduced cassette width below the outer attachment provides large holes) as shown in FIG. 1b. In the case of a CFC cladding on the lower part of the VT, a cold trap (~ 70K) for hydrocarbon molecules will have to be introduced as close as possible to the source (strike zone), while the walls of the pumping channel from the strike zones to the cold trap should be at elevated temperature (~ 300˚C) in order to reduce T co-deposition [4]. This cold trap is at present under study and will most likely be located outside the cassettes in the cryopump ports. To keep the walls of the pumping channel at elevated temperature the louver will be removed in this case and the pumping channel will be clad with radiatively cooled W tiles heated by the divertor plasma radiation.

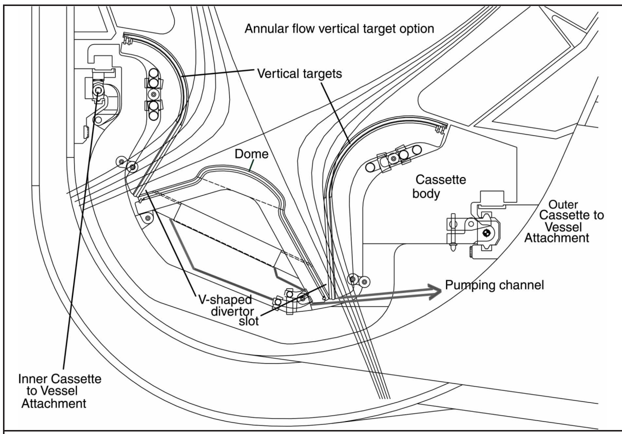



*FIG. 1a: Cross-section through the divertor showing the V-shaped strike zones and the pumping path as well as the attachment to the toroidal rails.*

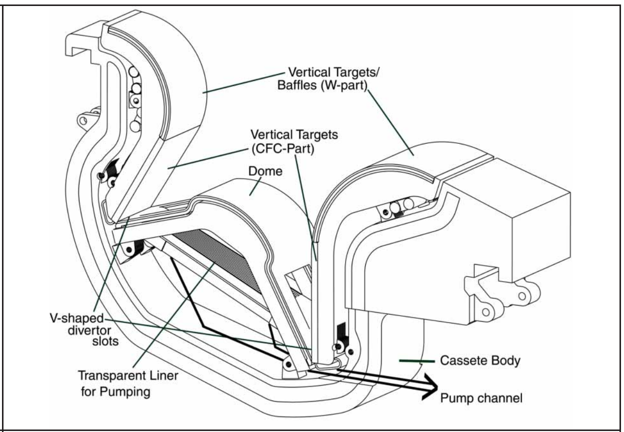



*FIG. 1b: Divertor cassette showing the large openings in the dome support . Also shown is the transparent liner allowing pumping access from the private flux region.*

Regardless of the pumping channel design, the new divertor private flux region geometry will allow the peak power load at the outer strike zone to be kept below 10 $MWm^{-2}$ for most envisaged operation scenarios as long as the power conducted into the scrape-off layer (SOL) does not exceed 100 MW significantly [6,7]. However, as outlined in [2] there are a few reference operation scenarios (hybrid and steady-state) where the heat flux into the SOL is larger (up to 130 MW). In these cases the limit of 10 $MWm^{-2}$ peak power load will be somewhat exceeded even when applying all the above means of peak power load reduction.

Recent R&D results suggest that under certain circumstances, and for a limited number of pulses, peak power loads of ~ 15 $MWm^{-2}$ can probably be accepted in particular if the divertor is clad with W [9-13].

The design of the VT HHF Components for the ITER divertor has progressed significantly during the past years and continues to develop driven by the need to reduce costs and improve reliability, as well as provide a high heat flux W clad target as a back-up to the CFC reference. A W back-up is sought in case the efforts to mitigate the impact of C-T co-deposition by employing the cold trap prove ineffective. FIG. 2a shows the lower end of the VT for the reference CFC monoblock option mounted on 10 mm ID swirl tubes with cooling tubes catering for thermal expansion routed into the SS support structure. As an alternative, an annular flow CFC monoblock is under development, using coaxial tubes with the inner tube held in place in the 25mm diameter outer tube by a swirl tape turbulence enhancer. The hairpin return of the annular flow eliminates the vulnerable return cooling tubes of the reference design and as a result reduces the space required beneath the VT. The monoblock is ~50% wider than the reference version reducing the parts count and hence the component cost. FIG. 2b shows a flat tile target based on hypervapotron heat sinks which is suitable for a HHF tungsten target, and as in the case of the annular flow, eliminates the need for expansion pipes.

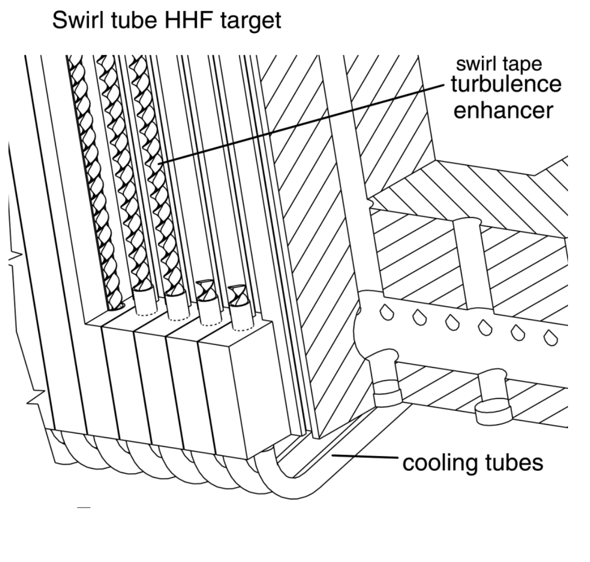



*FIG. 2a: Detail of the reference CFC monoblock swirl tube design. The expansion pipes on the bottom and the swirl tape turbulence enhancer inside the tube can be seen.*

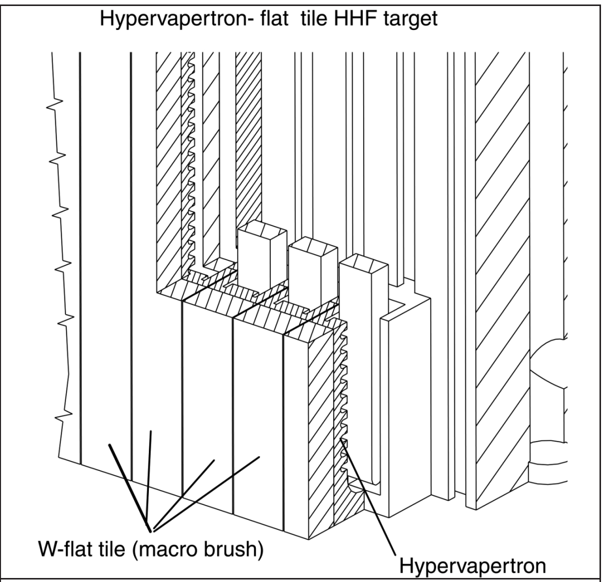



*FIG. 2b: Detail of the W flat tile on hypervapotron heat sink. The rectangular cooling channels and the teeth to provide turbulence are shown.*

The aim of the design and R&D effort presently ongoing is to have a sufficiently advanced design for any of these options to allow a choice based only on cost, reliability, physics and materials issues. For all three options a variable number of positive R&D results exist already [9-13] allowing some optimism that the above goal of bringing all options to a similar level of engineering performance will be achieved before the end of the ITER EDA. Many mockups with CFC monoblocks and with flat tile W macro-brush cladding as well as with W monoblocks have reached or exceeded the ITER goals (i.e. 20 $MWm^{-2}$ for > 1000 pulses) [12,13]. Several new mockups aimed at testing VT designs based on annular flow and hypervapotron heat sinks are in the pipeline and results are expected within the EDA timeframe.

However, even if engineering design and R&D allows the switch from partial CFC-clad VTs to full W-clad VTs, several physics issues need to be clarified before such a decision can be made. Even a moderate release of tungsten from the divertor into the main plasma can be a problem (W concentration < $10^{-5}$), and the loss of natural radiation from carbon in the divertor plasma has to be compensated by impurity seeding. This requires an assessment of different radiating impurities (Ne, Ar) by modelling and experiment, together with elaboration of these new operation modes to ensure an adequate lifetime of the divertor HHF components. An additional concern, when using W, is the melting of the surface during disruptions (~ 80μm) and the possible loss of this melt layer which would significantly reduce the target disruption (and thus the overall) lifetime.

## 4. Integration of the Divertor, the Pumps, the IVV- System and Remote Maintenance

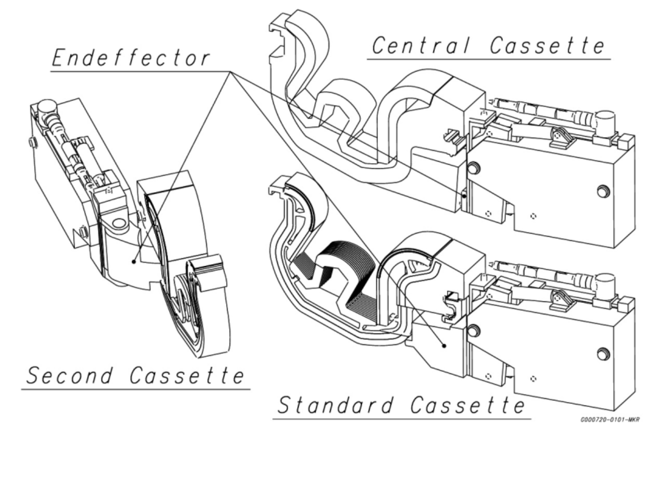
A large number of systems have to be integrated together with the divertor into the lower part of the vacuum vessel and into the divertor level ports. An allocation of divertor level ports to major systems can be seen in Table I.

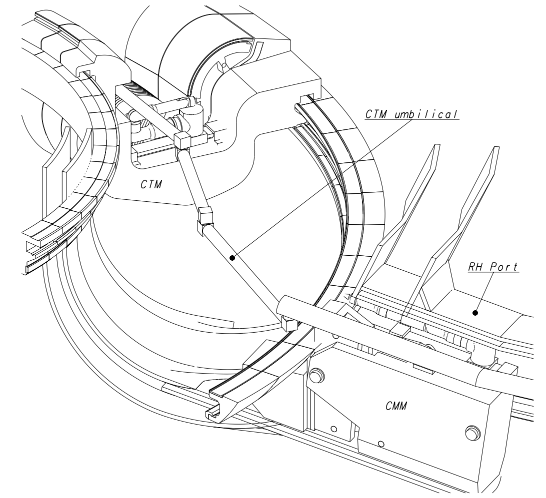
TABLE I: DIVERTOR PORT ALLOCATION TO MAJOR SYSTEMS

| Port | Main functions | Port | Main functions |
|---|---|---|---|
| **1** | Pellet & Gas Injection, Detritiation | **10** | Cryopump |
| **2** | Cryopump & GDC & IVV | **11** | Cryopump & GDC & IVV |
| **3** | **RH & Diagnostics** | **12** | Diagnostics |
| **4** | Cryopump & Diagnostics | **13** | Pellet & Gas Injection, Detritiation |
| **5** | Cryopump & GDC & IVV | **14** | Cryopump & GDC & IVV |
| **6** | Cryopump | **15** | **RH & Diagnostics** |
| **7** | Pellet & Gas Injection, Detritiation | **16** | Cryopump & Diagnostics |
| **8** | Cryopump & GDC & IVV | **17** | Cryopump & GDC & IVV |
| **9** | **RH & Diagnostics** | **18** | Diagnostics |

Each of these systems has its specific requirements and boundary conditions as well as interfaces with the RH tools. In addition, access has to be provided for the RH tools to the divertor, to the diagnostic components located in some ports, and to the cryopumps. To find a consistent solution is a design undertaking in its own right and is described in more detail than possible here in [14].

Due to the limited space available between vacuum vessel and X-point in the new ITER machine, the design of the divertor structures and of the remote maintenance scheme for the divertor has to be highly optimized (e.g. to minimize the thickness of the cassette body) in order to maximize the space for the divertor plasma [1,12,14]. Therefore the toroidal rails in the vacuum vessel were located near the inner and outer top of the divertor cassettes (FIG. 1a) such that the stresses in the cassette body are minimized during off-normal events with large electromagnetic forces. Due to the large triangularity of the plasma the divertor cassettes have to be angled from the horizontal and thus have to follow a non-linear trajectory during removal through the remote maintenance ports.




| | |
|---|---|
| *FIG. 3a: Cassette multifunctional mover (CMM) moves a divertor cassette radially through one of the 3 RH ports in a cantilevered way and is also able to place it sideways by one cassette width.* | *FIG. 3b: Cassette toroidal mover (CTM) takes the cassette from the CMM and moves it to its final location* |

This radial movement is performed by a CMM (Fig. 3a) that holds a cassette with a special end-effector in a cantilevered manner [14]. The cantilevering minimises the space between the cassettes and the vacuum vessel again giving more space to the divertor plasma. The toroidal movement of the cassettes during maintenance which is performed by a CTM (FIG. 3b), as well as the cutting, welding and inspection of the water pipes which connect the cassettes to the cooling circuit, are performed in a similar manner as in the 1998 ITER design [15-17]. Therefore the majority of the R&D performed for divertor remote maintenance and refurbishment [16] remains fully valid.

The in-vessel viewing and metrology systems (IVV) as well as the glow discharge probes (GDC) are inserted from the divertor level through 6 of the cryopump ports (FIG. 4). The IVV and GDC probes are of particular importance for machine operation. While GDC will be deployed after shutdowns, when no magnetic field is present, in order to prepare plasma start up, the IVV will be deployed shortly (hours) after an abnormal event during operation where damage to in-vessel components is suspected. This requires the probes to work with magnetic field on and in a high radiation environment.

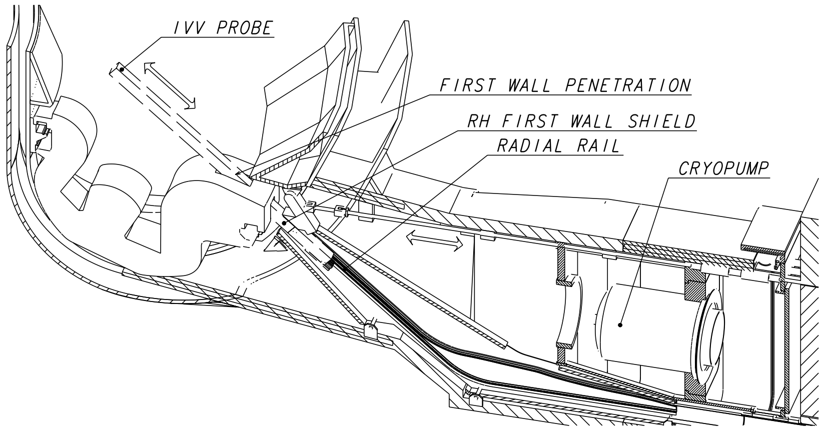



*FIG. 4: IVV system allowing the deployment of up to 3 different ~ 150 mm diameter probes through the upper part of a divertor cassette into the plasma chamber. Presently an IVV probe, a GDC probe and possibly a light manipulator (snake arm) are foreseen.*

A proposal is being studied whereby the deployment of IVV and glow discharge probes is performed from a specific IVV cask [14], which is also used to handle the plugs ensuring the continuity of the shielding at the IVV vacuum vessel and bio-shield penetrations (FIG. 4). In this design the probes are moved by a pushing chain and guided by rails, which rotate the probes to pass through the divertor baffle region.

## 5. Fuel Cycle Design

The ITER fuel cycle includes plasma fuelling by gas puffing on top of the machine (4 locations) and in the divertor (3 locations), pellet injection from the high field side (HFS), exhaust by (ultimately) 10 batch-regenerated cryopumps (6 operating, 4 in regeneration) together with remotely located roughing pumps, as well as exhaust processing and fuel storage in an also remotely located T- plant. While the requirements for the fuel cycle are governed by plasma edge and divertor physics, and are therefore partially a result of the modelling studies the dynamic interplay of all these complex systems is itself a modelling problem aimed at optimizing the integrated fuel cycle system. Initial results of the latter modelling activity are presently used to optimize fuel cycle component design solutions. A more detailed description of the fuel cycle and its dynamic behaviour as well as of the related R&D is given in [5,18]

An issue related to the fuel cycle is also the cryogenic cooling power required for rapid cool down (from 90K to 4.5K in < 75sec) of the cryopumps at the end of their regeneration cycle in order to provide a quasi-stationary operation without accumulating excessive H inventory inside the pumps (deflagration limit). As a result of an optimization process the required cooling power at 4.5K has been significantly reduced and is at present in the order of 13 kW. To operate these pumps, He supplies at several temperatures (4.5K, 90K, 300K) are needed. The system operates with a closed loop that requires the use of complex valve boxes to control the supply and return of cryogens via ring manifolds to the cryo-plant [5]. These subsystems also have to be integrated into the divertor level of the building just outside the bio-shield and thus can potentially interfere with IVV and RH access as well as with the water-cooling supply lines for the divertor.

## 6. Summary and Conclusions

The design and integration of the divertor and its auxiliary systems into the lower part of the ITER machine is a complex problem involving physics issues as well as various engineering, material, and RH problems. Therefore, besides the resolution of detailed engineering issues in each subsystem, a significant system engineering effort is required, where the integral problems of combining several subsystems with their own requirements and boundary conditions into an optimized plant system have to be tackled. This paper has revealed the complexity of the problem as well as showing that satisfactory solutions have been found not only on the subsystem level but also for the integrated divertor plant system.

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