Interface of Blanket Testing and ITER design.

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Outline of Presentation

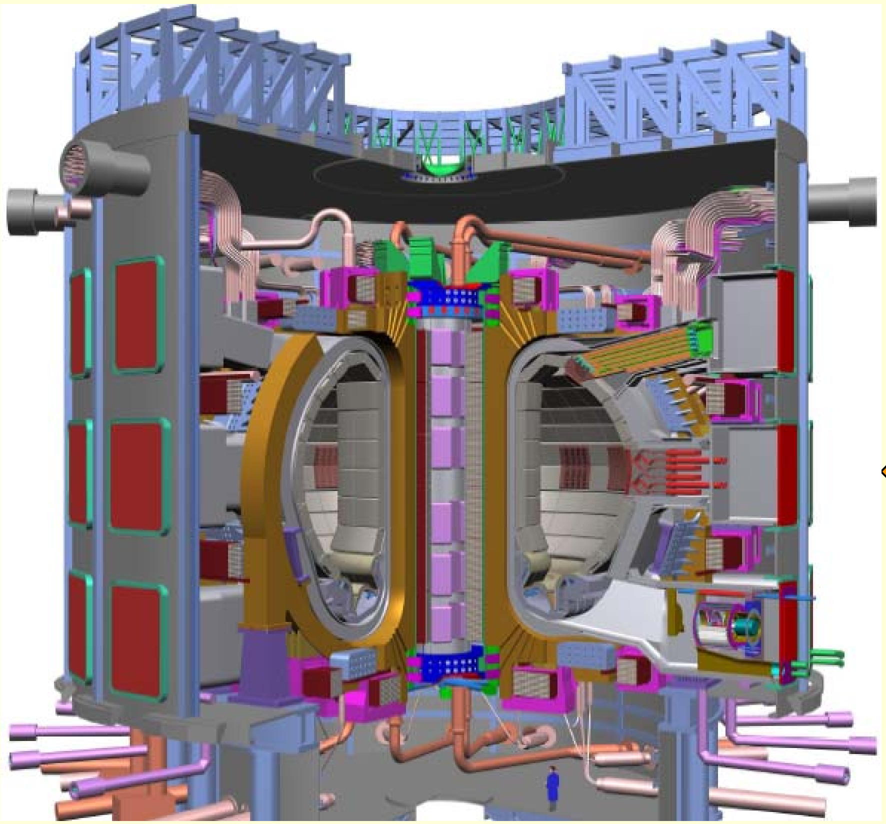
- **The ITER parameters have been significantly revised during the period of extension of the ITER EDA.**
- However , the objectives of ITER to demonstrate fusion technology in an integrated system by performing testing of nuclear components required to utilize fusion energy, in particular- **to test design concepts of tritium breeding blanket relevant to a DEMO reactor-** stays unchanged .
- In this paper I am not going to discuss to what extent ITER will be able to confirm feasibility of different blanket concepts for DEMO applications. This subject, I hope, will be covered in following presentations.
- My task here is to give you a clear portrayal what will be the conditions of testing in ITER.

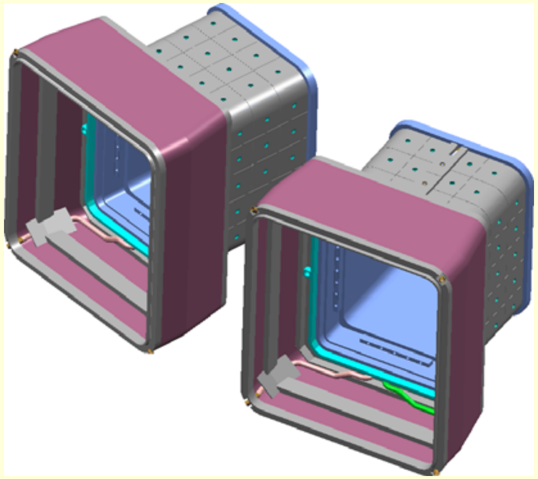
Main Parameters of ITER

Nominal Parameters in inductive operation	ITER FDR	ITER
Q	Ignition	10 (5 for CD)
Fusion Power	1500 MW	~500 MW (200-600 range)
Total rejected power	2200 MW	970MW (1080 MW max)
Plasma volume	2000 m3	837 m3 (nom)
Plasma surface	1257m2	678 m2
Plasma current	21 MA	15.0 MA
W thermal	1.1 GJ	325 MJ
W magn	1.1 GJ	485MJ
Heat flux on FW	0.25-0.5 MW/m2	0.2-0.5 MW /m2
Neutron Wall loading	1 MW/m2	0.57 MW/m2 average Max 0.8 MW/m2 (at 500MW)
Total fluence	1 MWa/m2 ,	0.3 MWa/m2 ;
Pulse length	1000s	Inductive flat top 300-500s
Total number of pulses	50000	Several 10s thousands
Breeding blanket	Must be installed	Option of later installation must not be precluded

Three of the big ITER equatorial ports are dedicated for TBM testing.

In the following presentation I describe testing conditions in these ports.





Equatorial ports

As an experimental machine ITER is designed to investigate a broad range of regimes

Three main operational scenarios are envisaged :

- An inductive operation , where the plasma current is driven by the ITER central solenoid (CS) and other poloidal coils.

- A non-inductive operation , where the plasma current is driven by injection in the plasma of particle and high frequency (Ion Cyclotron and Electron Cyclotron) energy beams.

- A hybrid operation where the plasma current is driven by a combination of inductive (CS) and non-inductive means.

Inductive operation.

In this case:,

- **-durations of plasma current are limited by the total available magnetic flux and,**
- **-for a typical plasma current ~15 MA, one can expect burn times ~400 s and**
- **-minimum repetition times >1800 s.**

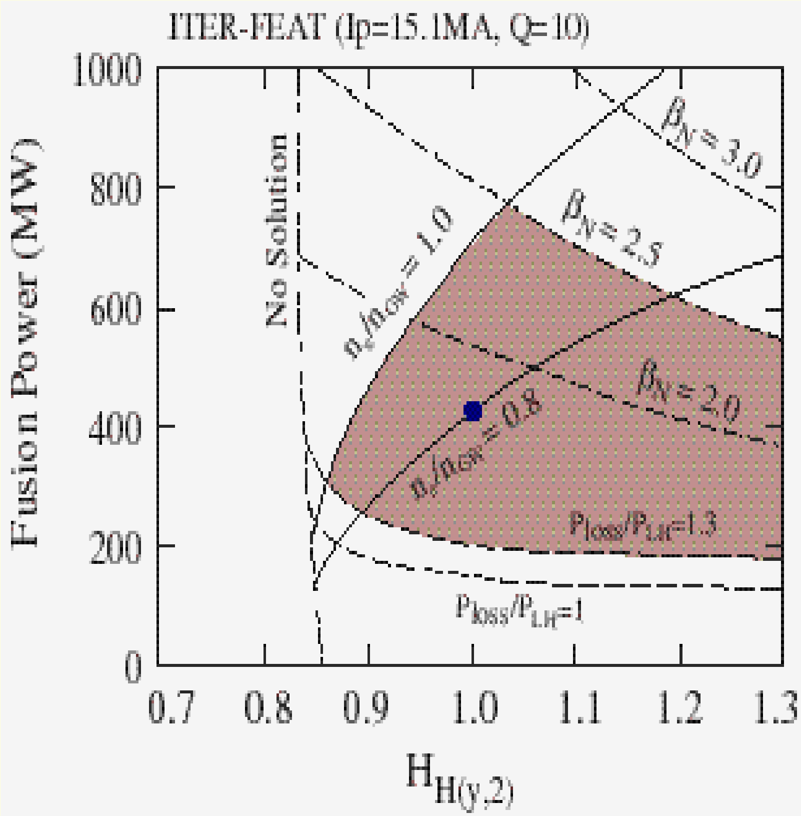
ITER is optimized for this kind of operation.

We can expect to reach Q>~10

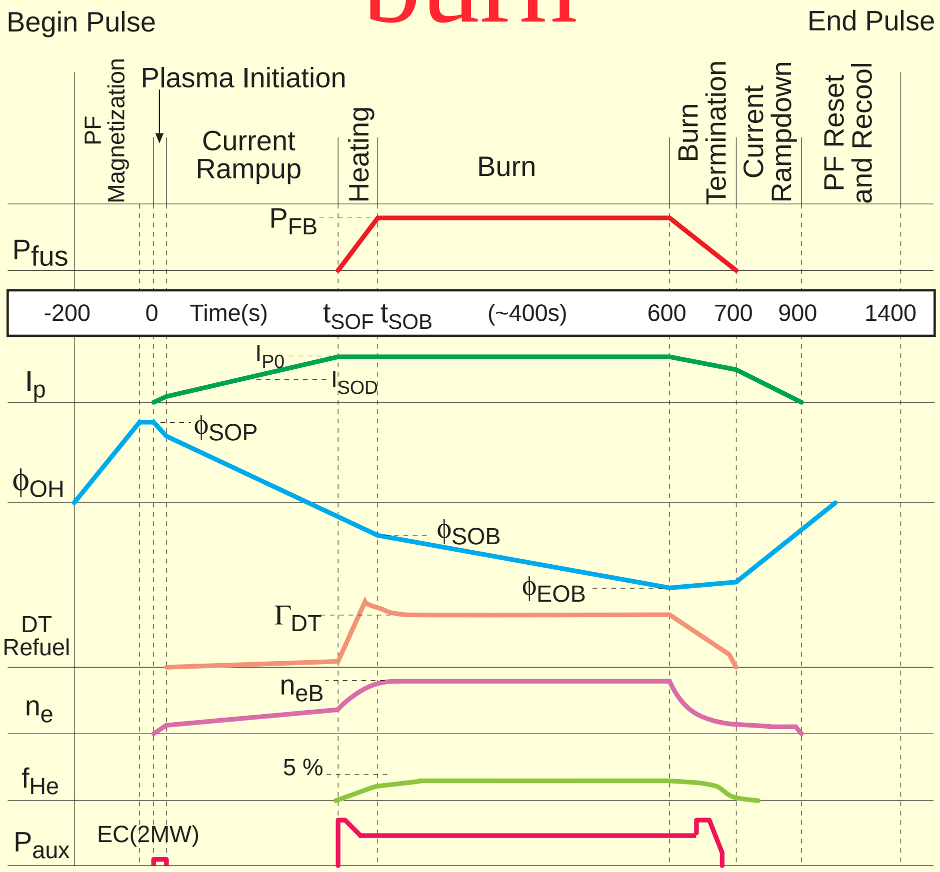
with a fusion power in the range 300-600 MW

and a neutron wall loading on test modules ~ 0.76 MW/m^2

at 500MW of fusion power



Standard Inductivelyy Driven Pulse of ITER is ~400 sec of burn



Non-inductive operation

The duration of plasma current is limited by the technical capabilities of external systems, and for the current ITER design

one can expect to obtain pulses up to 3000 sec

with the minimum repetition time > 12000 sec .

The physics of these regimes is not as well known as for the inductive operation and significant research and optimization will be needed during ITER operation

It is expected to obtain Q ~5 ,

fusion power ~ 360 MW and

neutral wall loading ~0.55 MW/m^2 .

Hybrid operation

This scenario combines advantages and limitations of two previous regimes.

In comparison with a non-inductive scenario the physics is better known .

Higher fusion powers (~400 MW) and

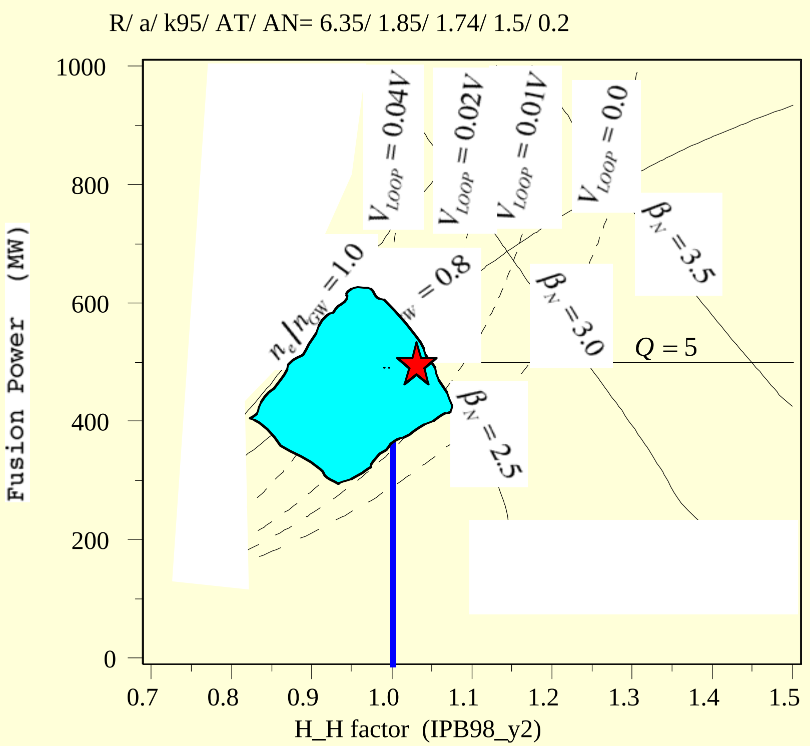
higher wall loadings (0.62 MW/ m2) may be achieved,

but

the burning duration is ~1000 s ,

the minimum repetition time is 4000 s.

This regime may be the first to be used for blanket testing., if the development of high fusion power non-inductive regimes is not successful



12 MA Pcd=100MW Hybrid Operation Space
Vloop=0.02V corresponds to a flat-top 2500 sec

ITER parameters significant for testing in different regimes of operation.

	Inductive operation	Hybrid Operation	Non-inductive operation	Comments
Burn duration sec	400	1000	3000	
Min. repetition time sec	1600	4000	12000	
Maximum burn duty cycle	25%	25%	25%	It is limited by design .
Fusion power MW	500	400	356	Highest power achievable in a short pulse ~ 700 MW
Neutron wall loading at the TBMs MW/m^2	0.78	0.62	0.55	Highest wall loading achievable in a short pulse ~1.09 Mw/m^2
Total average neutron fluence at the first wall $MW*y/m^2$	0.3			After first 10 years: 0.12 After 20 years 0.3 Maximum fluence 0.5
Surface heat flux at TBM MW/m2	0.1			For all regimes . Peak (design) value =0.27
Disruption heat load at the recessed TBM surface MJ/m^2	0.55 during 1-10 ms			For all regimes
Be First wall sputtering mm/(Mwy/m2)	~ 2 mm			For all regimes
Number of burn pulses	30000			Total for all regimes

ITER Operation Plan

Table 1.29-1 Neutron Fluence during the First Ten Year Operation (MWa/m^2)

	1~3	4	5	6	7	8	9	10	total
Equivalent number of nominal pulses	0	1	750	1000	1500	2500	3000	3000	11751
Average neutron fluence at FW	0.0	0.0	0.006	0.008	0.012	0.020	0.024	0.024	0.09

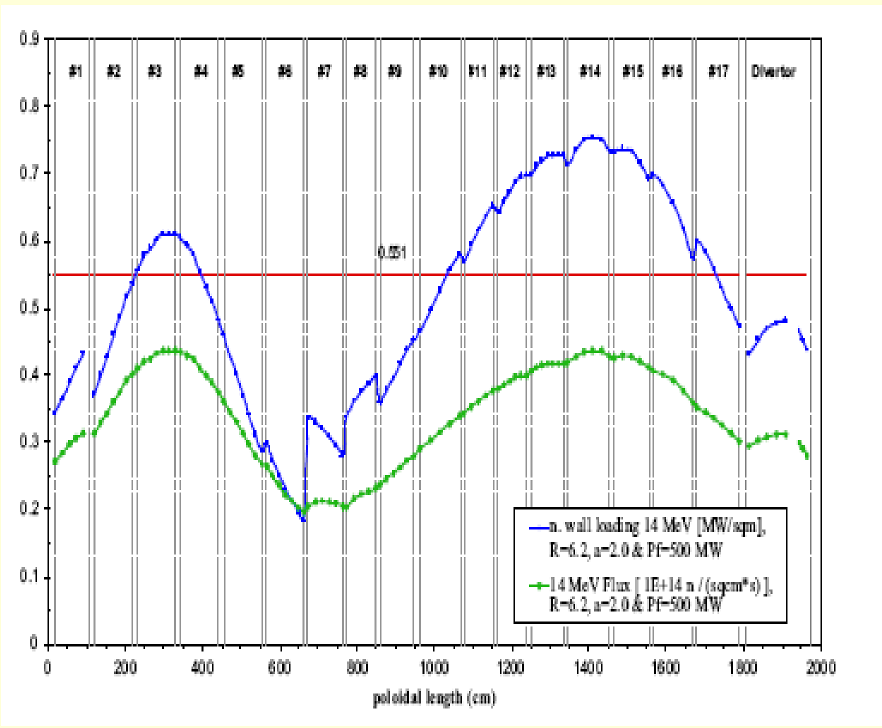
- **1 year of integration on sub system level**
- **2.5 years of initial operation in hydrogen**
- **Brief DD phase.**
- **Tritium operation**
  - **Initial operation with 400s 500 MW inductive pulses**
  - **"Hybrid" operation with longer (~1-2000 sec) pulses**
  - **Long non-inductive steady state pulses**

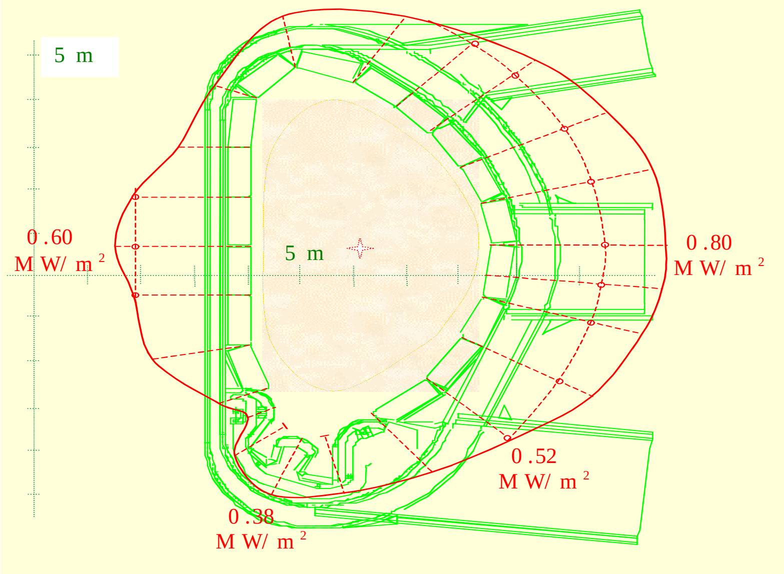
Blanket module testing will be done whenever significant neutron fluxes become available.

- **Test modules must be installed early before the DT operation**

Poloidal Distribution of Neutron Wall Loading.

Wall loading by 14 MeV neutrons up to 0.8 MW/m^2 and
14 MeV neutron flux up to 0.75E14 n/cm^{2}sec at 500 MW of fusion power.
(F_{max} / F_{mean} =1.43)





Poloidal Neutron Wall Loading Distriubtuion
(Fusion Power 500 MW)

Possible improvements

It is difficult to increase fusion power and fluence.

However, the duty factor may be improved with moderate investments:

An addition of 2 (to the current 4) cooling towers and

an addition of one (to the current 4) 18 kW liquid He process module would permit cooling and cryogenic supply for indefinitely long pulses.

These additions are possible and foreseen by the design .

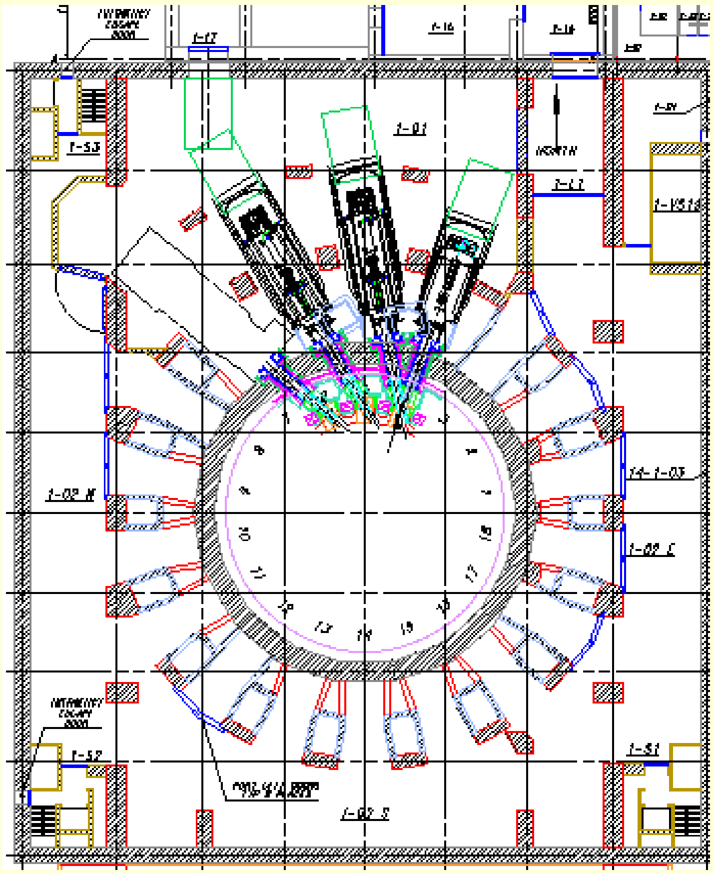
Certainly, a full design review and a new safety assessment must be done before such additional constructions may be authorized, but

One cannot rule out the possibility of very long pulses , say 12000s with a dwell time of 1200s and a duty factor 0.9 .

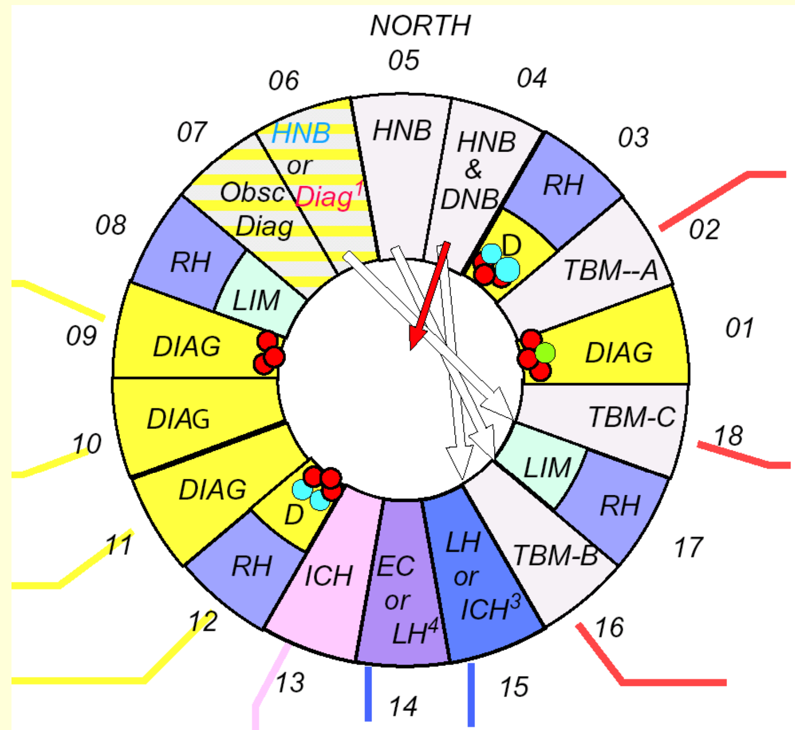
ITER has 17 ports in equatorial cross section. Thirteen of these ports are standard radial ports and 3 ports are tangential for NB injection.

Three of the standard ports (P16, P18 and P02) are dedicated for blanket test modules.

Additional space is available behind ports , in the tokamak cooling Vault and in the Tritium building



Tokamak Building Equatorial Level



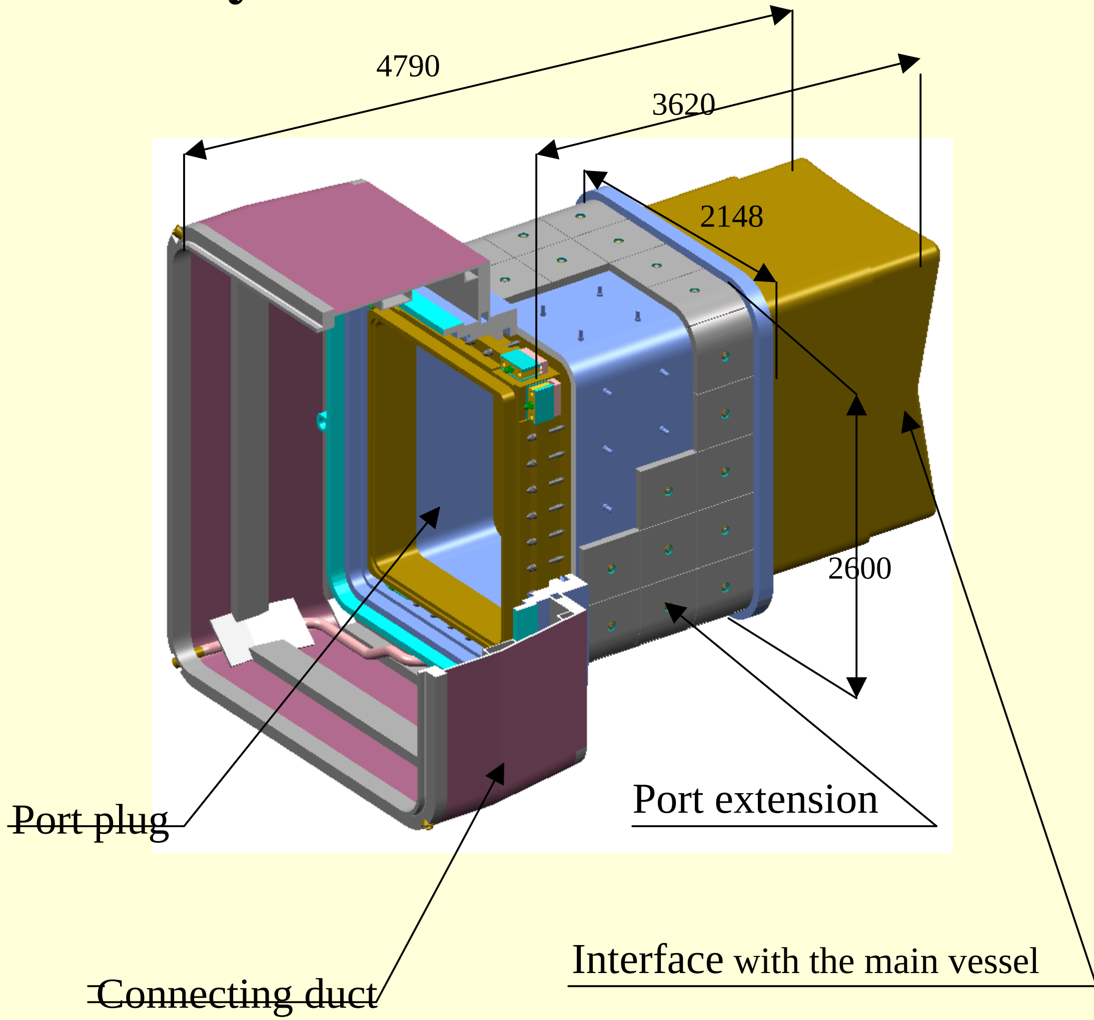
Equatorial port allocation

The volume of a port may accommodate a blanket module , a support structure , an additional shield and some equipment if necessary.

- **Rectangular cross-section :**
- **outer**
  2.6 m high x 2.148 m wide
- **inner**
  2.2 m high x 1.748 m wide
- **for TBM frame**
  2.16 m high x 1.708 m wide
- **TBM face cross section up to 1760mm high x 1310 mm wide**
- The length :

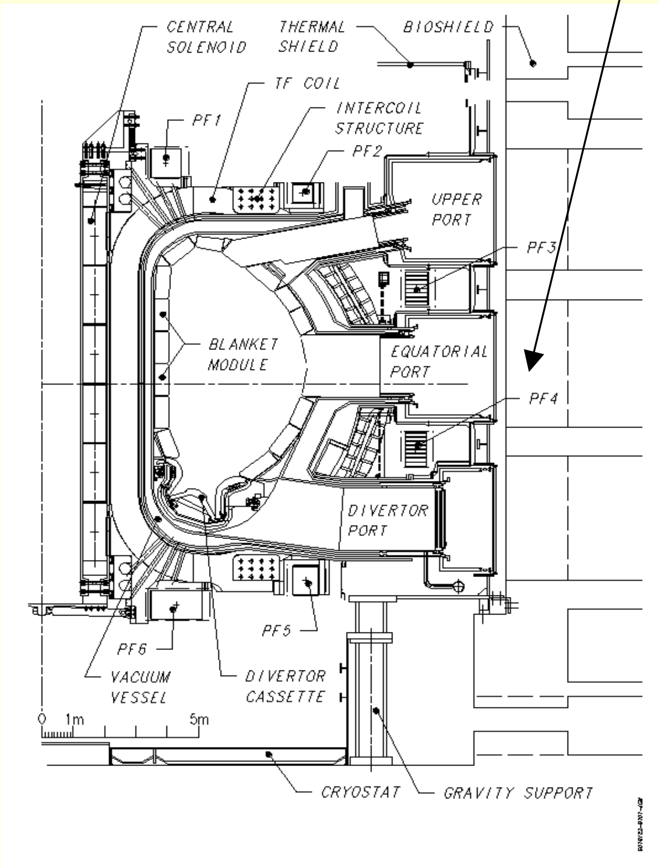
FW to the connecting duct: 4.79 m

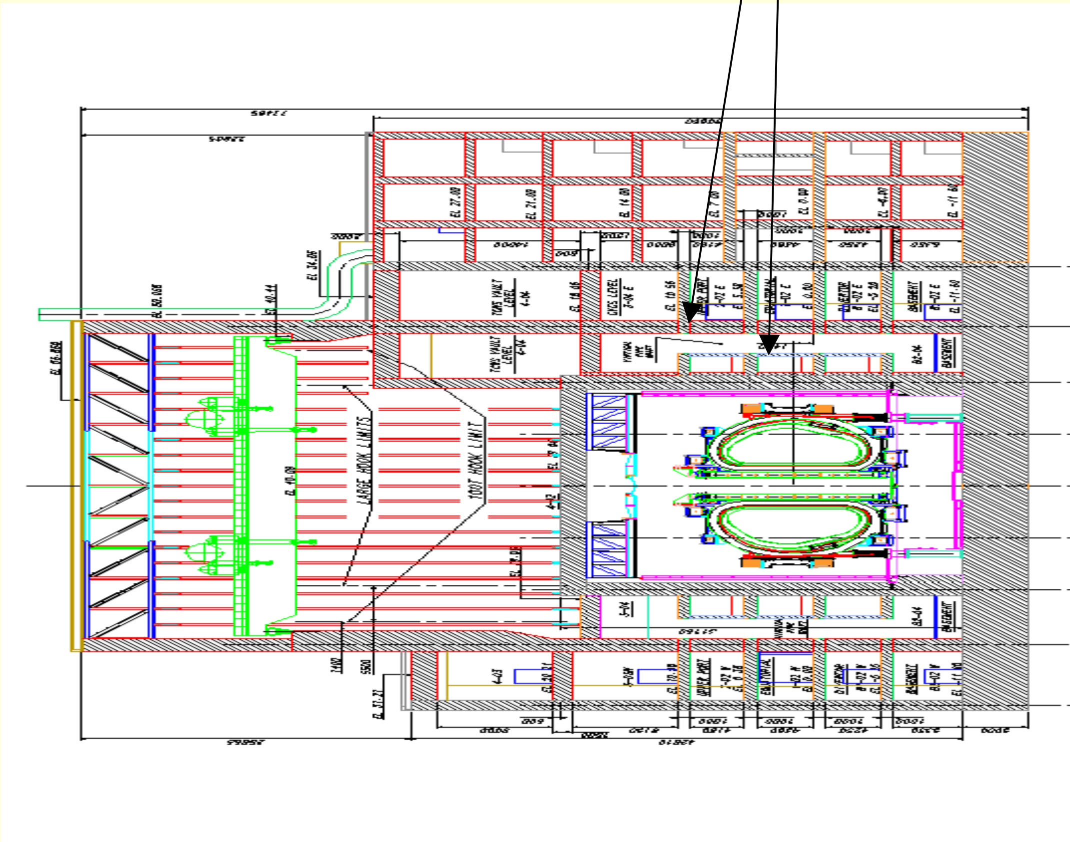
FW to end of the port extension : 3.62 m



Additional space for TBM equipment is available behind of the equatorial ports (3*(7x2.65x2.6 m^3) and in TCS Vault (120 m2).

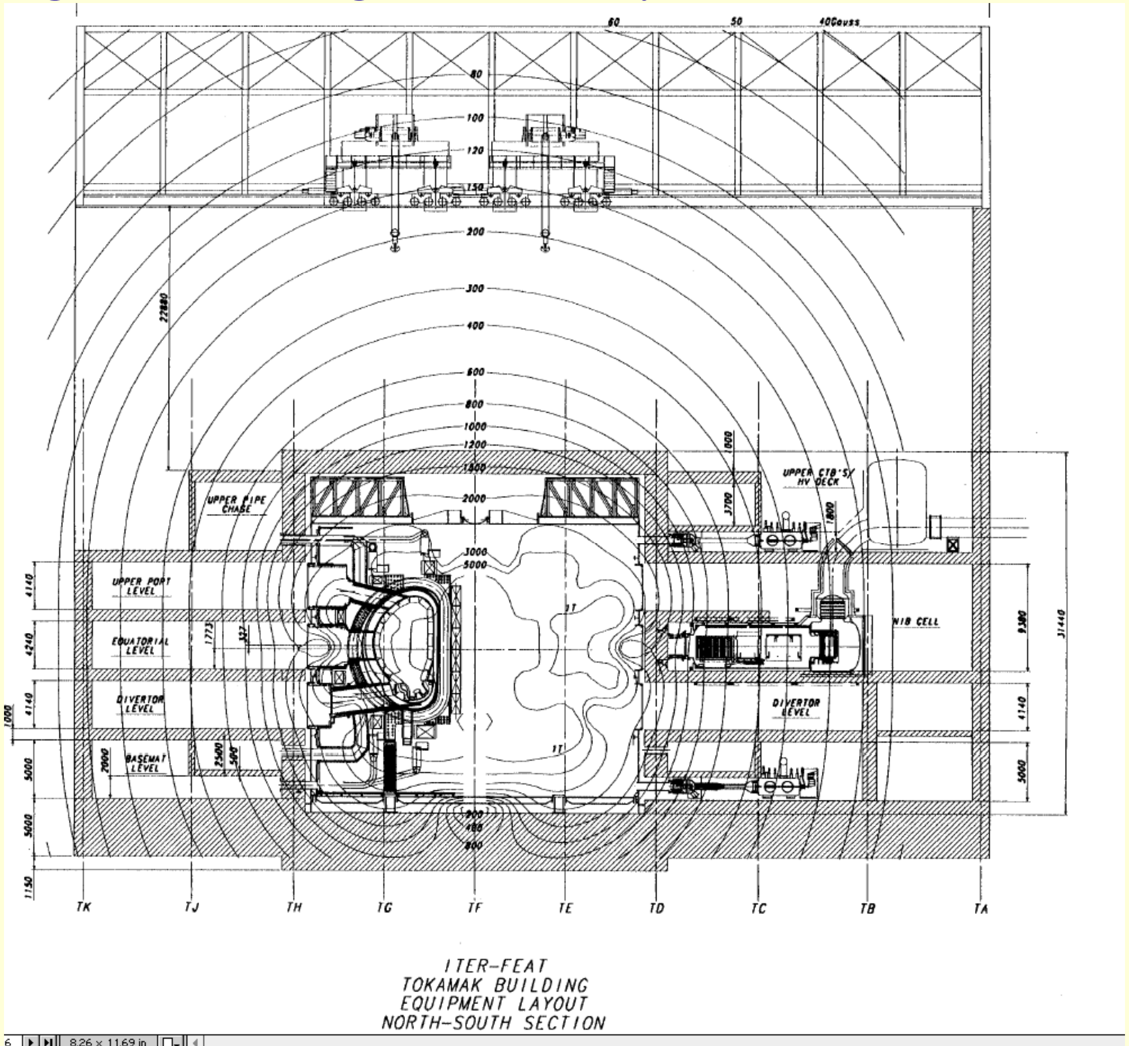
Additional space 180 m^2 is in Tritium Building





During plasma operation strong and changing magnetic fields exist inside the port cells with amplitude up to 2000 Gauss near the bioshield. This imposes severe limitations on the type of equipment that can be used inside the port cells without special precautions.

In the corner of the TCWS vault, where the interface with the heat rejection system is located, the magnetic field strength is sufficiently low to use conventional equipment.



Cooling/ baking

TBMs may use cooling water , helium or to be Li or molted sat self-cooled.

- **The test modules may use for cooling the ITER pressurized water system of the shielding blanket with the nominal pressure of 3.0 MPa at normal operating conditions and the inlet temperature 100°C. The water flow and temperature control shall limit the coolant inlet-outlet temperature difference to ≤ 50°C for normal loads.**
- **The dedicated water or helium cooling systems may be installed.**
- **The <u>common</u> heat rejection system provided by ITER for the TBMs is designed to supply cold water at 35C and accept hot water at 75°C.**
- **The first wall shall be baked, as other blanket modules, at the maximum temperature of 240°C**

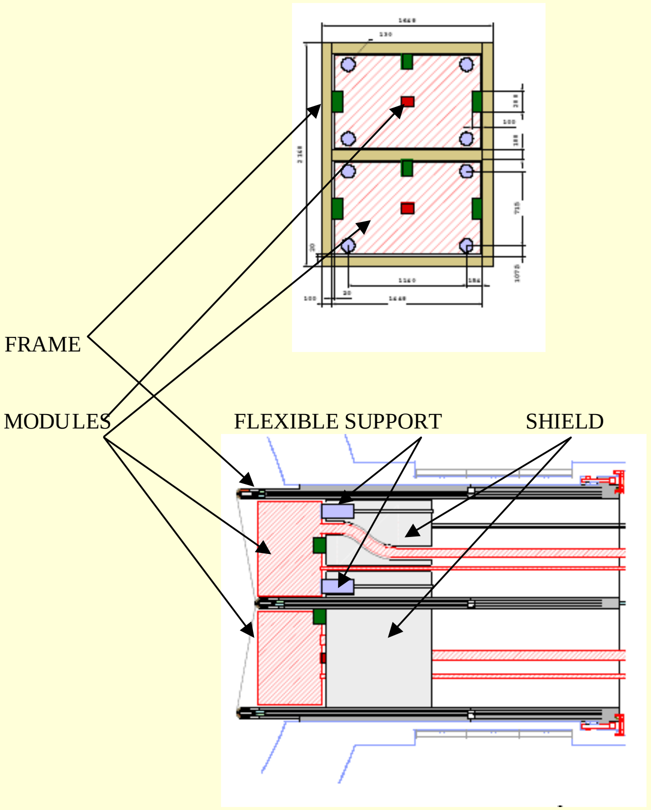
Installation of TBMs

The test blanket modules inside a port must be contained in a "frame", which provides a standardized interface with the ITER basic structure including thermal isolation from the basic machine.

The frame is built from 100 mm thick SS plates with cooling canals .

Modules (or sub modules) are mechanically connected with the frame through flexible supports , similar to the supports used for shielding blanket modules.

The dose rate behind a shield plug shall be less than 100 mSv/hr during maintenance (12 days after plasma shutdown). If the shielding capability of the test module is not enough an additional shield must be installed behind a TBM to allow hands-on access to port flanges.



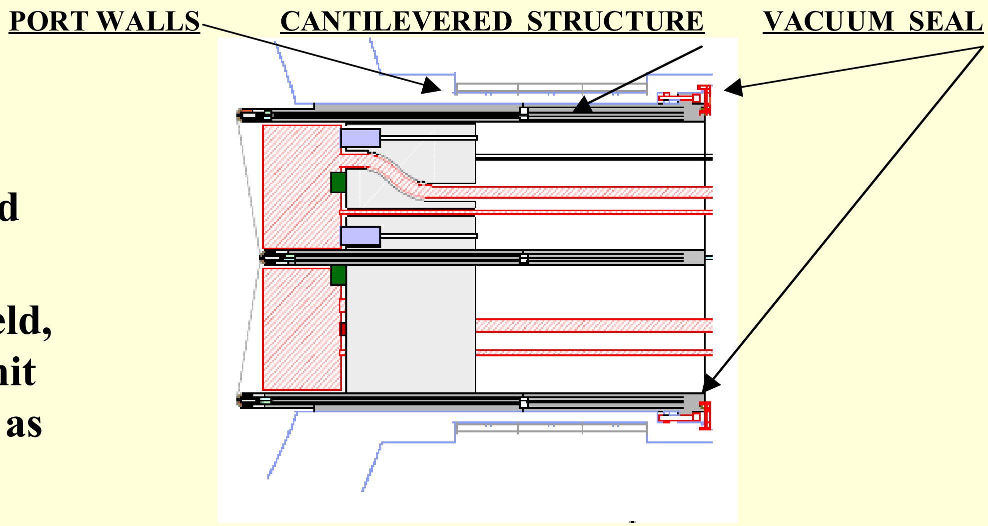
Installation of TBMs

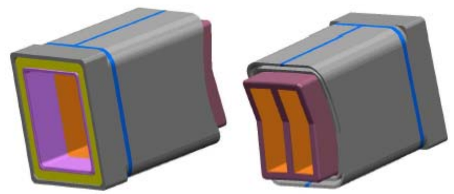
All port plug structure is cantilevered and completely supported by a vacuum vessel port flange.

A primary vacuum seal is situated between the vacuum vessel port and the flange.

The whole port plug (TBM, frame, shield, flange) must be a self-contained unit that may be installed and removed as a single piece without remote handling operations inside the vacuum vessel port extension.

To avoid any requirements on the alignment of TBMs to magnetic surfaces , the first wall of a TBM must be recessed by 50 mm below adjacent first walls of shielding blanket modules..





Equatrorial Port Structure (as an example for the blanket test module)

Remote handling and hot cell.

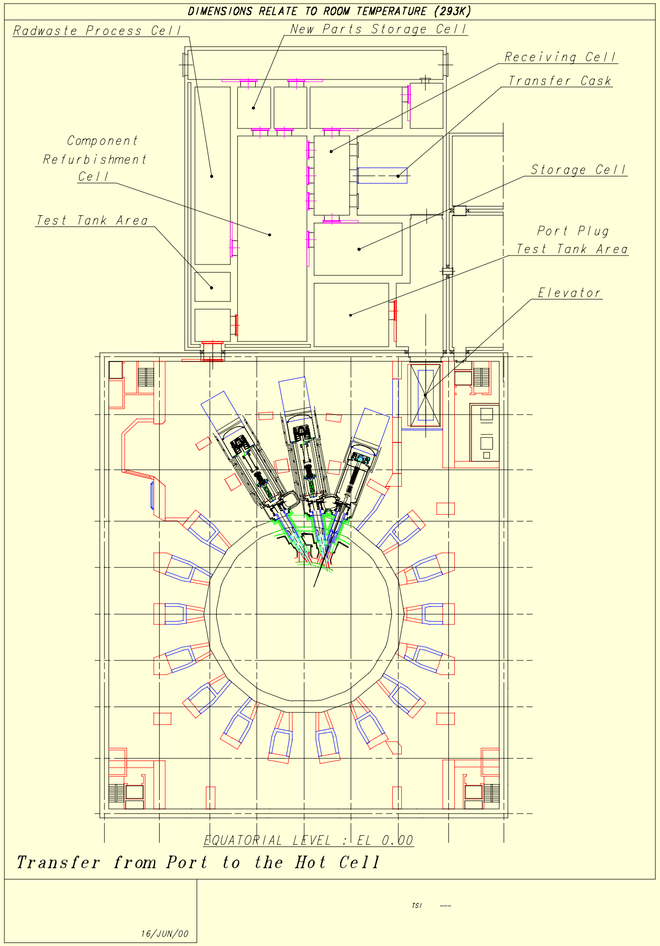
The test blanket module will be remotely installed and replaced with equipment supplied by ITER.

For this purpose the weight of a TBM (without the weight of the coolant) must be limited to 2 t.

Test blanket modules will not be repaired but refurbished or replaced in the ITER hot cell.

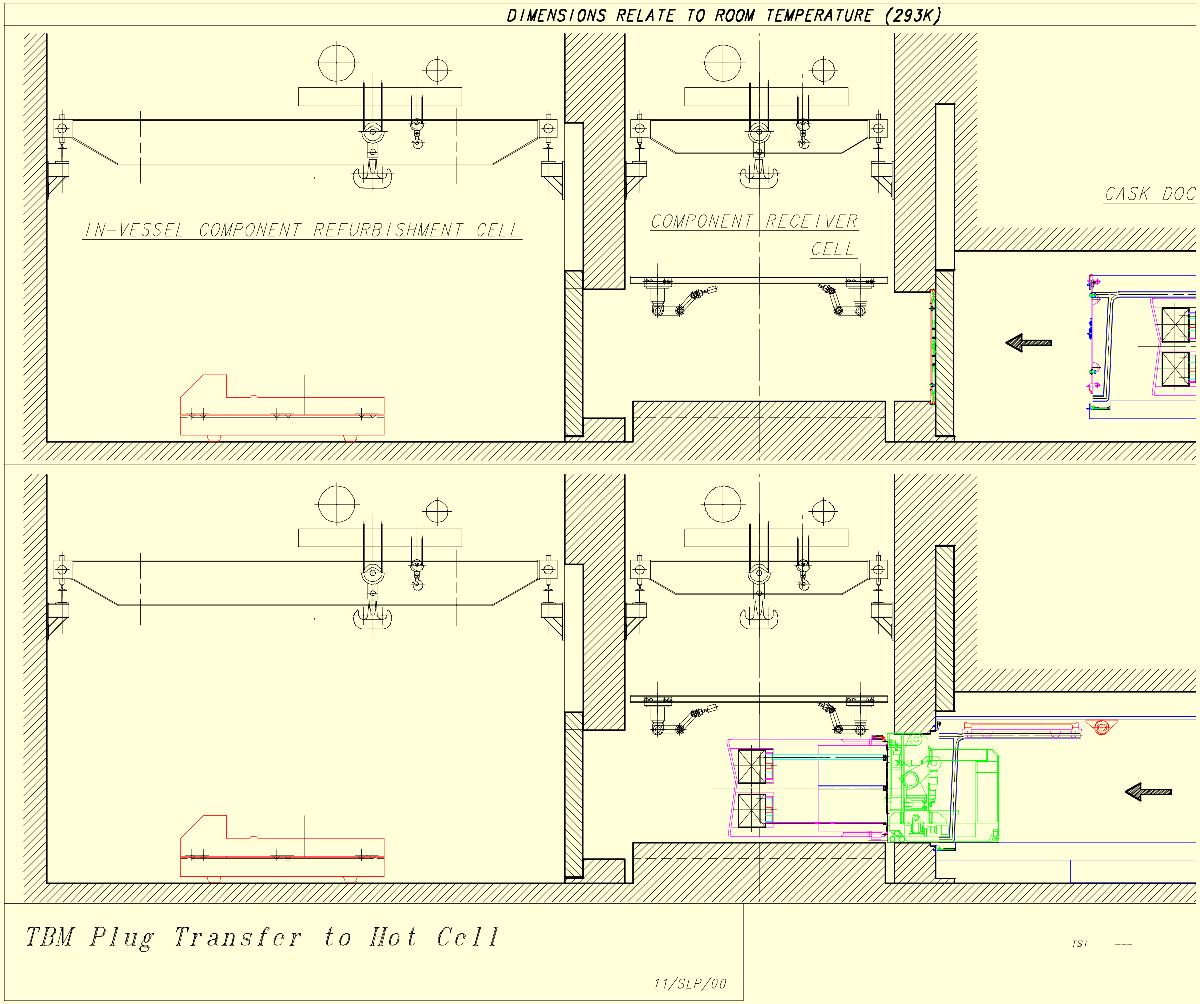
The hot cell may be used to replace irradiated test modules but at present it is not designed for post-irradiation studies of the test modules. Organization of post-irradiation studies depends on the selected site.

- **The weight of the integrated structure, consisting of TBM, frame and shield plug to be carried in a transfer cask and installed on the machine, must be limited to 40 t.**

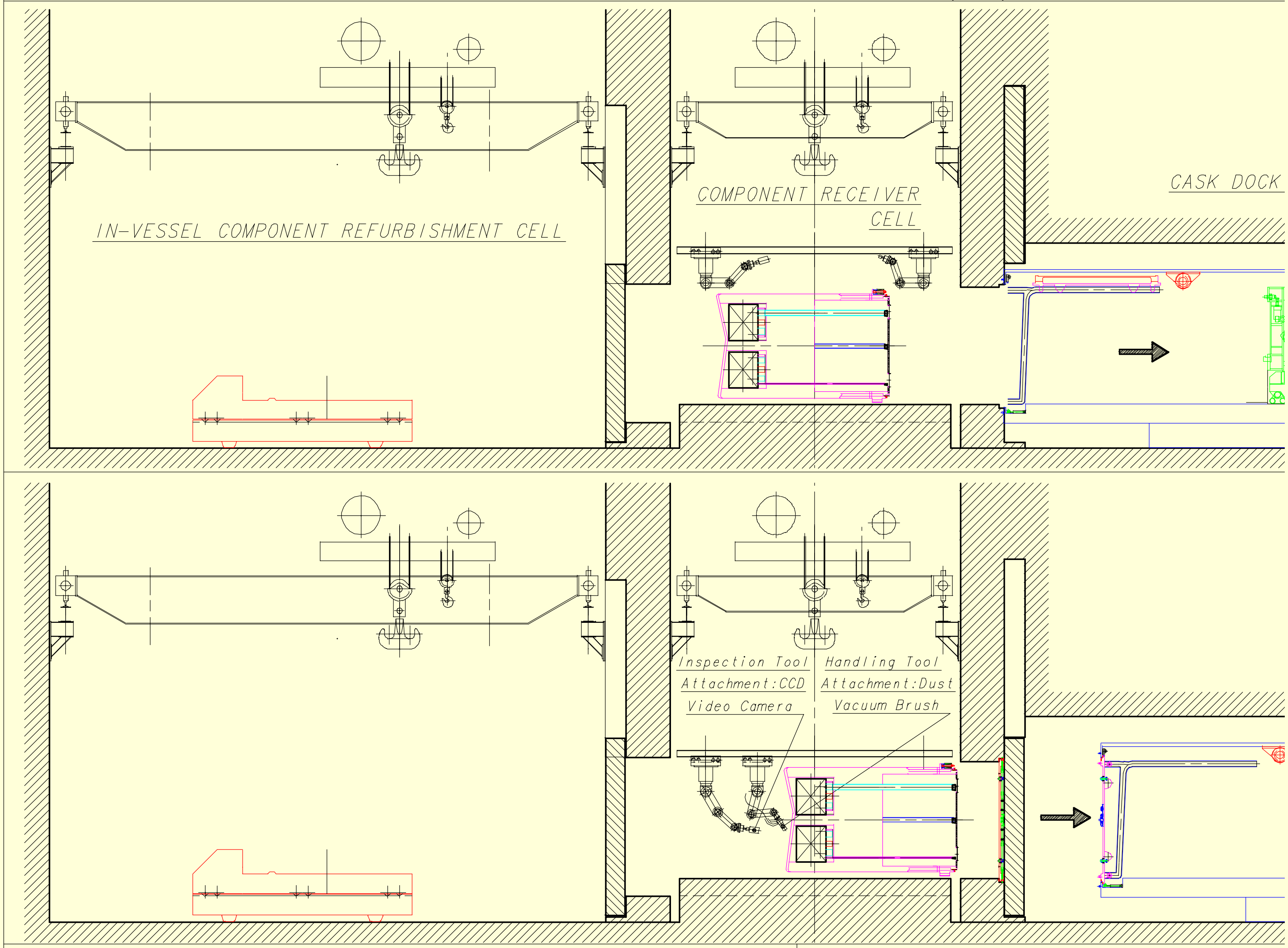


Procedures for Test Blanket Module handling in the Hot Cell

1. Dock RH transfer cask to component receiving cell and open double seal door.
2. Dust cleaning of TBM and frame/shield plug Assembly.
3. Disconnect the port plug and remove cask



DIMENSIONS RELATE TO ROOM TEMPERATURE (293K)



TBM Plug Transfer to Hot Cell

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ITER REQUIREMENTS APPLIED TO TBMs

The testing program may be successful only if ITER itself will operate successfully.

As a result, the testing of breeding blanket modules must neither interfere with ITER operation, decrease ITER reliability and compromise safety of operation, nor contradict ITER operational plans. .

However, , the presence of the TBMs may by itself interfee with ITER operation.

Let us consider just 2 examples :

Interaction of th TBM surface with particle fluxes.

Interaction of the ferro- magnetic TBMs with magnetic field.

Bare Steel First Wall of the TBM is a concern and may require Be cover of ~1 mm thickness

The analysis shows that the number of Fe atoms eroded for a bare stainless steel wall is about 3.6E19 1/s (1.34 g-Fe/400 s.) The total amount of eroded iron is a factor of 20 lower compared to the amount of Be eroded from other parts of the wall.

However the tolerable concentration of Fe in the plasma is a factor 50-100 lower than for Be and the tolerable concentration of W in the plasma is nearly three orders of magnitude lower than for Be.

The wall erosion is of concern. But this is not the only factor. The probability of the eroded atoms entering into the plasma core, their lifetime in the plasma core may vary as a function of the material and must be also taken in account considering the build-up of the impurity concentration in the plasma.

Erosion rates • $[s^{-1}]$, • [gr/400 s]	**Be** (~10 m^2)	**W** (~10 m^2)	**Fe** (~10 m^2))	Total Be wall (690 m^2)
- DT neutrals	a) 2.0 $x10^{19}$ b) 0.1	a) $9.4x10^{17}$ b) 0.12	a) 1.5 $x10^{19}$ b) 0.56	a) 1.4 $x10^{21}$ b) 8.4
- DT ions	a) $4.4x10^{19}$ b) 0.27	a) $<10^{16}$ b) <0.01	a) $1.3x10^{19}$ b) 0.48	a) $2.7x10^{21}$ b) 16.1
- Impurity ions	a) $0.9x10^{19}$ b) 0.06	a) $1.1x10^{18}$ b) 0.13	a) 0.8 $x10^{19}$ b) 0.29	a) $2.6x10^{20}$ b) 2.1
- Total	a) $5.4x10^{19}$ b) 0.43	a) $2.1x10^{18}$ b) 0.25	a) 3.6 $x10^{19}$ b) 1.34	a) $3.1x10^{21}$ b) 26.7

Interaction of the ferromagnetic TBMs with magnetic field.

Small perturbations of the axial symmetry of the poloidal magnetic field, called "error fields", having a component normal to the magnetic surface of order 10^{-4} of the toroidal magnetic field, can induce in the plasma locked (i.e. non-rotating) tearing modes.

Locked modes are not stabilized by plasma rotation.

Islands grow , degrade fusion performance and lead to disruptions.

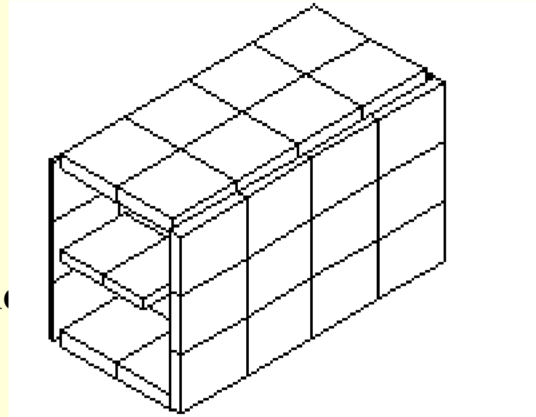
Test blanket modules are made of a ferromagnetic structural material, which can create error fields.

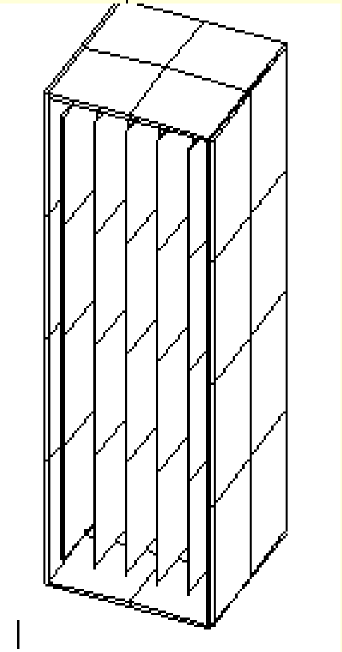
Error Fields from Test Blanket Modules

- **Six test blanket modules (2 per port)were envisaged in ITER.**
- **Five of them will be composed of a ferromagnetic structural material, which can create error fields.**
- **The ferromagnetic material is in saturation in the toroidal (B=5.3T at R=6.2m) and poloidal magnetic fields.**
- **The magnetisation of the ferromagnetic material in the saturation is 1.7 T.**
- **To simplify the study, all modules are represented as a set of ferromagnetic plates.**

Total volume and mass of TBM ferromagnetic material assumed for the error field study

Type of TBM	Mass, kg	Volume, cm^3
JA WC	1318	166796
EU WC	1095	138633
JA HC	1552	196441
EU HC	1403	177577
RF HC	521	65904


EU HC TBM


RF HC TBM

Error fields created by ferromagnetic TBMs

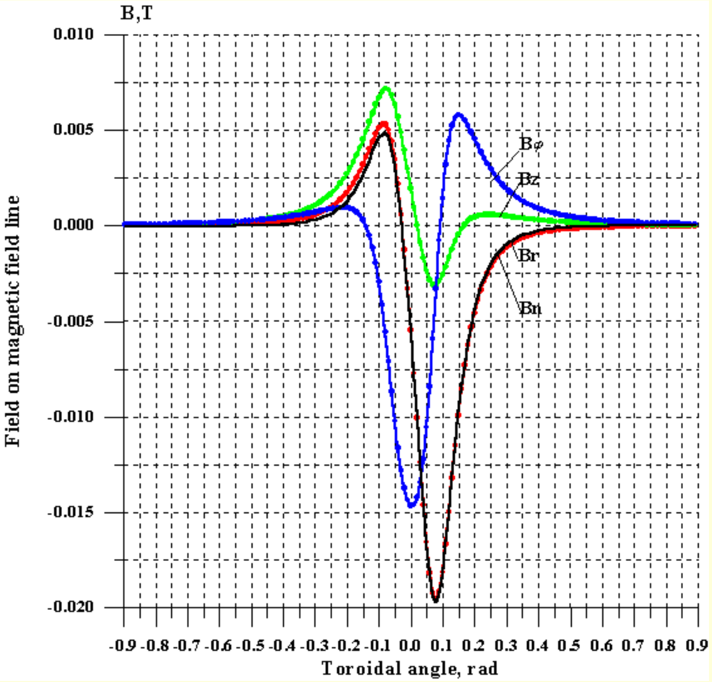
- **For a given plasma equilibrium and for a given mode (m, n), the component of the error field normal to the surface $B_\perp$ is averaged over the field line, passing the equatorial plane at a toroidal angle φ_0, and decomposed into a Fourier series.**

- **The following "3-mode" error field criterion was adopted for the ITER design:**

$$B_{3\text{-mod}\,e}/B_0 = \sqrt{(B_{2,1}/B_0)^2 + 0.8(B_{3,1}/B_0)^2 + 0.2(B_{1,1}/B_0)^2} \le 5\times 10^{-5}$$

where B_0 is the value of toroidal magnetic field and all the modes are decomposed on the q = 2 magnetic surface

- **The TBMs provide rather localized disturbance of the axisymmetrical magnetic field**



Results of error field calculations for TBMs

- **All TBMs produce about the same values of error field.**
- **The "3-mode"error field due to the five test blanket modules, having ferromagnetic structural material, is 3.1×10^{-5}.**
- **This value is comparable to the criteria and needs to be reduced.**
- **Moreover, the deviation of the field lines of toroidal magnetic field from their ideal shape in the presence of TBMs is too high (of the order of 10 mm, when the field line passes by a test blanket module).**
- **Therefore the amount of ferromagnetic material in TBMs should be reduced (at least in the places close to the plasma).**

Amplitude of the error field mode (2,1) from the TBMs (in 10^{-5})

TBM	$B^{(0)}_{2,1}/B_0$
EU HC TBM	0.52
JA HC TBM	0.61
EU WC TBM	0.49
JA WC TBM	0.56
RF HC TBM	0.26

The deviation of field lines is about:

15 mm near the outermost part of the first wall,

0.05Źmm near the strike point of the separatrix outer leg,

0.05 mm near the strike point of the separatrix inner leg.

The testing of breeding blanket modules must not interfere with ITER operation, decrease ITER reliability and compromise safety of operation, or contradict ITER operational plans.

- **Ferromagnetic /surface sputtering : It will be nesessary to install TBMs or , at least there ferromgnetic and first wall equvalents from the first days of peration.**
- **Replacement frequency**: As a rule, the TBMs may be changed once per year in accordance with the ITER operational schedule. Some modules may need additional replacements, to be synchronized among the Parties and with machine operation. Target replacement time must be less than 1 month for all 3 ports together**.**
- **Coolant leaks.** The frequent leaks of TBMs must not influence the machine availability. Direct leeks of water or other chemically active coolants into the vacuum vessel must not lead to significant (>2.5 kg) generation of hydrogen.
- **Li** The upper limits for the quantity of liquid lithium within a TBM system should be 0.035 m^3 and for $Pb_{17}Li_{83}$ 0.28m^3.

As the port cells form a containment volume, overpressure events (such as the rupture of cooling and other pipes) have to be considered

Each TBM must have a pressure suppression system.

The water cooled modules must have separate pressure suppression tanks.

He-cooled modules may release helium into the vacuum vessel if the total amount of released gas is small enough, and if the vessel will not be presurized higher than 1 bar.

Modules must be designed to be able to remove residual heat in the case of loss of coolant or loss of flow by thermal radiation to the vacuum vessel and to the shielding .

Acceptance criteria .

At the moment we do not expect any special licensing requirements for TBMs.

However , to guaranty safety and reliability of operations we do expect to apply the same quality assurance and quality control programs to TBMs as for ITER itself at all stages of design , production and testing.

To-day we have established general requirements tfor TBMs (operational conditions, compatibility with ITER design and services , compatibility with ITER operation and safety).

During the design process we expect frequent and detailed reviews of the TBM design and material specifications, validations of the structural integrity of the designed TBMs, establishment of requirements to quality control during design and production.

Full nondestructive examination and proof tests must be performed during factory production .

Acceptance tests will be done at the ITER site after transportation, and additional testing will be done after installation .

Conclusions

ITER will not provide all the conditions for qualification of the breeding blanket design for DEMO .

A special module design will be necessary to amplify and study specific effects.

However, ITER will be the only facility available for testing of DEMO blanket modules in a real fusion environment.

The best effort has to be made to coordinate parties' activities in this area and to achieve the maximum use of space and time available for blanket testing in ITER.

The current organization of the testing program considers ITER as a provider, and ITER Parties as independent customers enjoying the right of testing as a compensation for their participation in the ITER construction.

Taking into account strong interaction between ITER operation and TBMs, both at technical and at organizational levels, such an organization does not look rational, at least, for the first 10 years of operation when ITER regimes must be established and TBM design must be improved by sub module testing.

Strong interest of ITER in TBM reliability and safety is forcing the ITER International Team to take a greater responsibility in TBM design and testing organization .

The necessity to test many specific effects with specialized sub modules will push Parties to integrate their testing programs.

It is becoming clear that the breeding blanket development is a part of the fusion development which can be done in accordance with the ITER model of international development .