ITER:The Future International Burning Plasma Experiment
Present Status

R. Aymar, ITER Director

Burning Plasma Science Workshop II

General Atomics, San Diego

1-3 May 2001

strategic objective

- to establish fusion energy as a real energy option for the future
  - e.g. “creating prototype reactors for power stations to meet the needs of society: operational safety, environmental compatibility, economic viability”
- the ITER experiment will provide the physics understanding and technological proof of principles on which to base a demonstration fusion power station

Reducing the costs of ITER

- ITER EDA 1992-1998 produced
  - FDR design to meet programmatic, technical and cost targets
  - supporting physics and technology R&D results of general application
- Scope to reduce costs with less demanding technical objectives, still satisfying programmatic objective - SWG Task #1
- Review a broader concepts - modular approach - SWG Task #2
- Conclusions:
  - rationale for pursuing an integrated physics/technology next step: reactor relevant phenomena can be explored only in experiments with α-particles dominating
  - need for a burning plasma experiment
  - revised technical objectives targetting 50% cost reductions

System Studies

- relate plasma parameters, physics design, engineering constraints and global costs.
- apparent domains of feasible operating space
  - R ~ 6 - 6.5 m
  - characterised by aspect ratio
- no clear optimisation
- analysis of representative options
  - HAM IAM LAM
  - HAM rejected - limited access

revised guidelines for ITER (1998)

- **plasma performance**
  - extended burn in inductively driven plasmas @ Q>10 for a range of scenarios
  - aim at demonstrating steady-state through current drive @ Q>5
  - controlled ignition not precluded
- **engineering performance and testing**
  - demonstrate availability and integration of essential fusion technologies
  - test components for a future reactor
  - test tritium breeding module concepts
- **target cost saving ~ 50% of FDR estimate**
- **new requirements still satisfy ITER programmatic objective — permit integrated "one-step" strategy to demonstration power station**

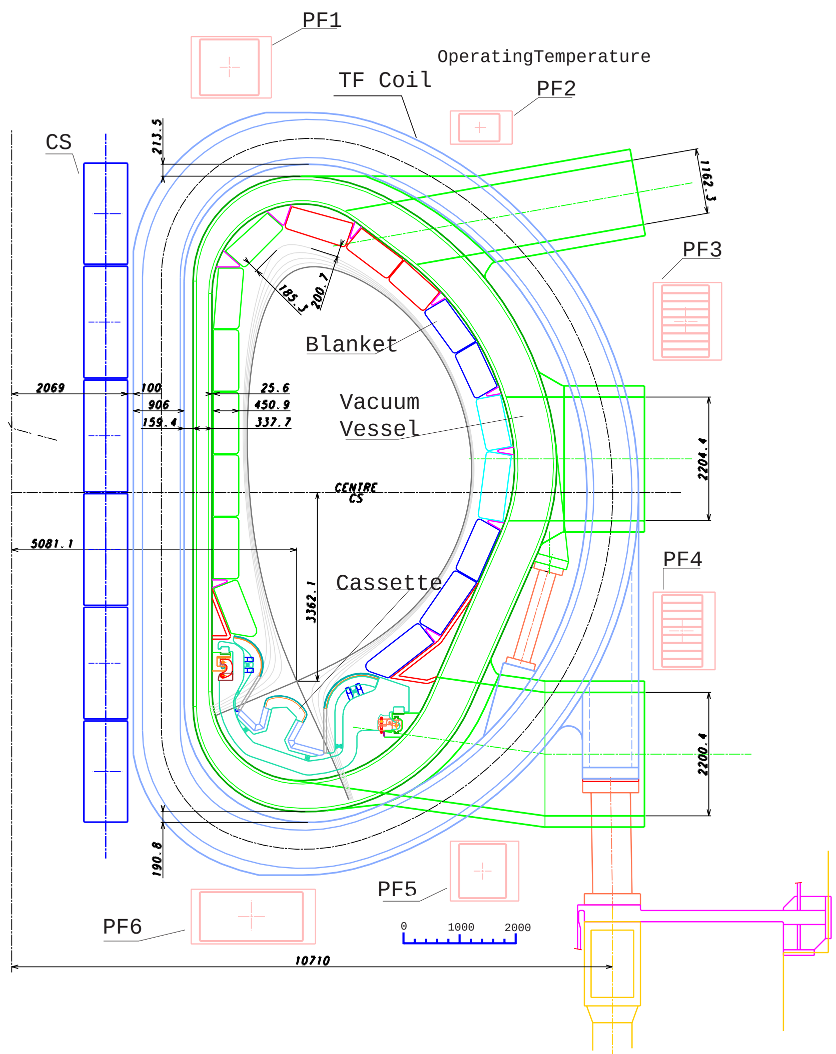
policy for the reduced cost design

- **re-balance** plasma parameters, physics design, engineering constraints under new cost target
- **preserve physics margins** for plasma performance to be able to explore and qualify the full range of physics issues for a future fusion reactor
  - feasible operating space for R ~ 6-6.5 m
- **exploit the technology R&D results** to squeeze technical margins
- **optimise manufacturing processes** to meet cost targets

parameters of the ITER design

Total Fusion Power	500MW (700MW)
Q — Fusion power/aux. heating power	10
Average Neutron Wall loading	0.57MW/ m^2 (0.8MW/ m^2)
Plasma inductive burn time	≥ 1000 s.
Plasma major radius	6.2 metres
Plasma minor radius	2.0 metres
Plasma current (I_p)	15 MA (17.4 MA)
Toroidal field @ 6.2 m radius	5.3 T
Plasma Volume	837 m^3
Plasma Surface	678 m^2
Installed Aux. Heating/Current Drive power	73 MW (100 MW)

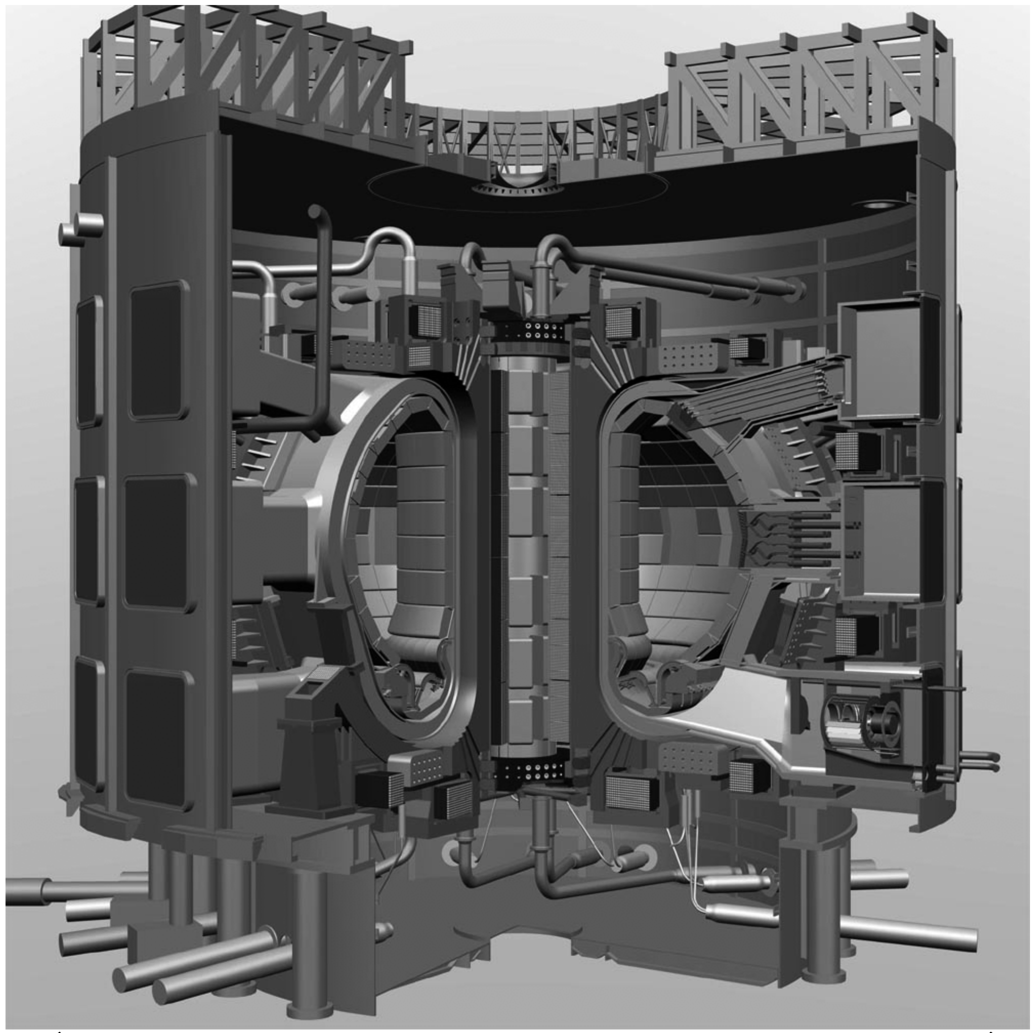
cross section of ITER tokamak



5/8/01

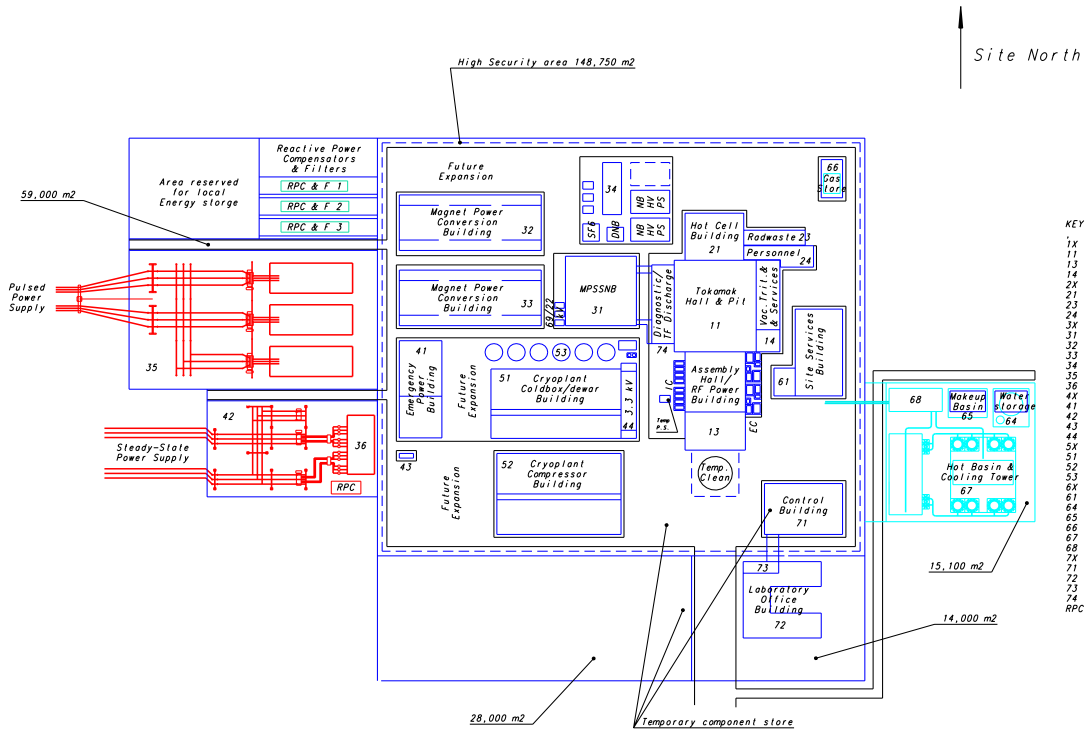
ITER

cutaway of the ITER tokamak



~30 m

ITER site layout



Compulsory area = 249,750 m2

Total area = 264,850 m2

ITER-FEAT Site Layout

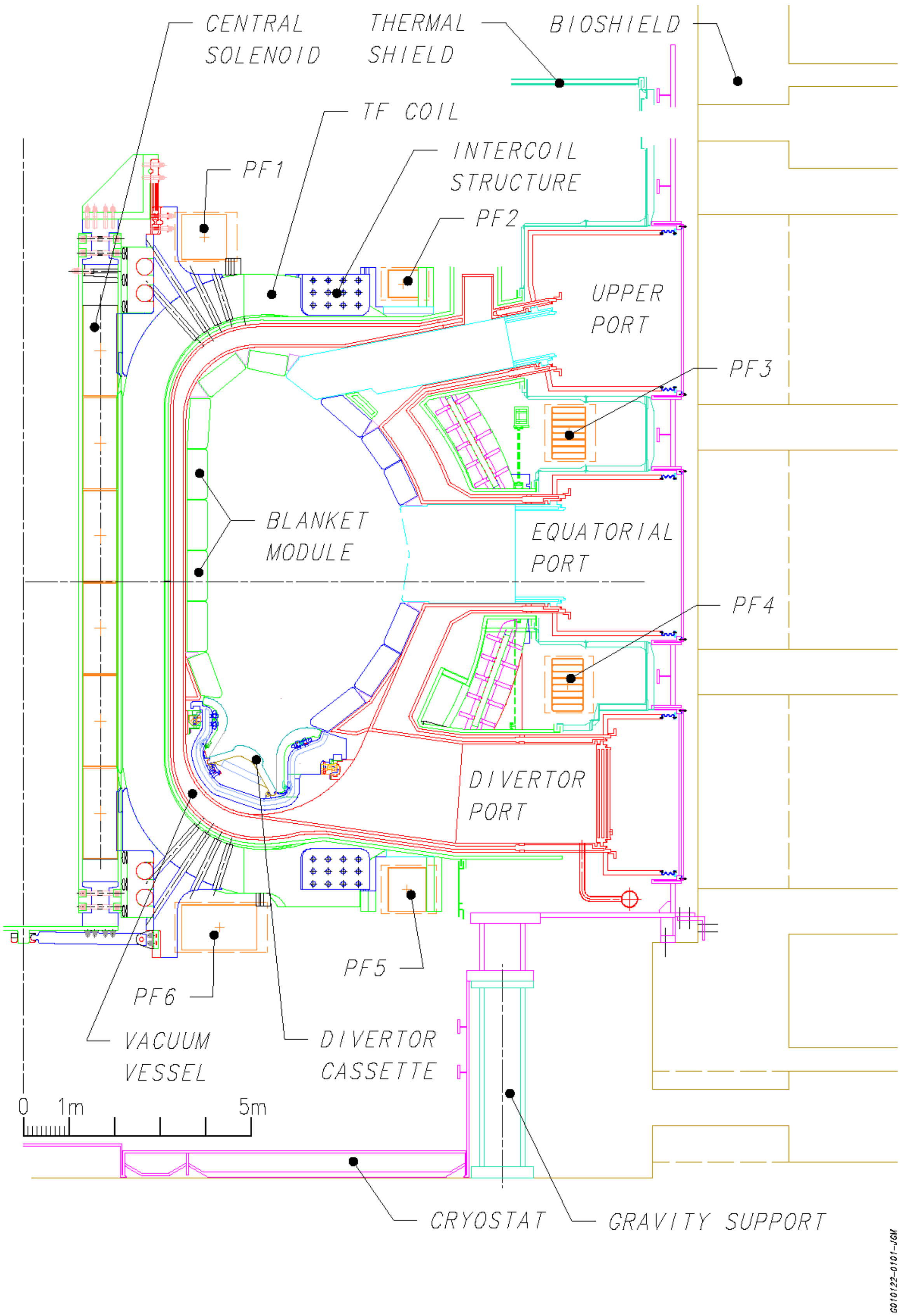
KEY TO TOKAMAK BUILDINGS, AREAS & ACRONYMS

1X Tokamak Core Buildings
11 Tokamak Building
13 Laydown, Assembly & RF Heating Building
14 Tritium, Vacuum, Fueling & Services Building
2X Hot Cell & Radwaste Complex
21 Hot Cell Building
23 Low level Radwaste Building
24 Personnel Building & Access Control
3X Pulsed Power Supply Complex
31 Magnet Power Supply Switching Network Building
32 North Magnet Power Conversion Building
33 South Magnet Power Conversion Building
34 NB Injection Power Supply Switchgear Building
35 Pulsed Power Switchyard, Local Energy Storage & RPC Area
36 Alternating Current Distribution Building
4X Steady State Power Supply Complex
41 Emergency Power Supply Building
42 Steady State Swtchyard & Switchgear Area
43 Emergency Power Supply Fuel Tank
44 3.3 kV Power Transformer
5X Cryoplant Complex
51 Cryoplant Coldbox & Dewar Building and PF Coil Fabrication
52 Cryoplant Compressor Building and PF Coil Fabrication Builc
53 Cryogenic Storage Tanks
6X Laboratory Support Complex
61 Site Services Building
64 Water Storage
65 Make up Basin
66 Gas Storage Area
67 Hot Basin & Cooling Tower
68 Cooling Water Pumping Station
7X Control Complex
71 Control Building
72 Laboratory/Office Building
73 Personnel and Vehicle Access Control Gatehouse
74 Diagnostic Hall & TF Fast Discharge Resistors & Capacitors
RPC = Reactive Power Compensators

Convergence to a New Design (1)

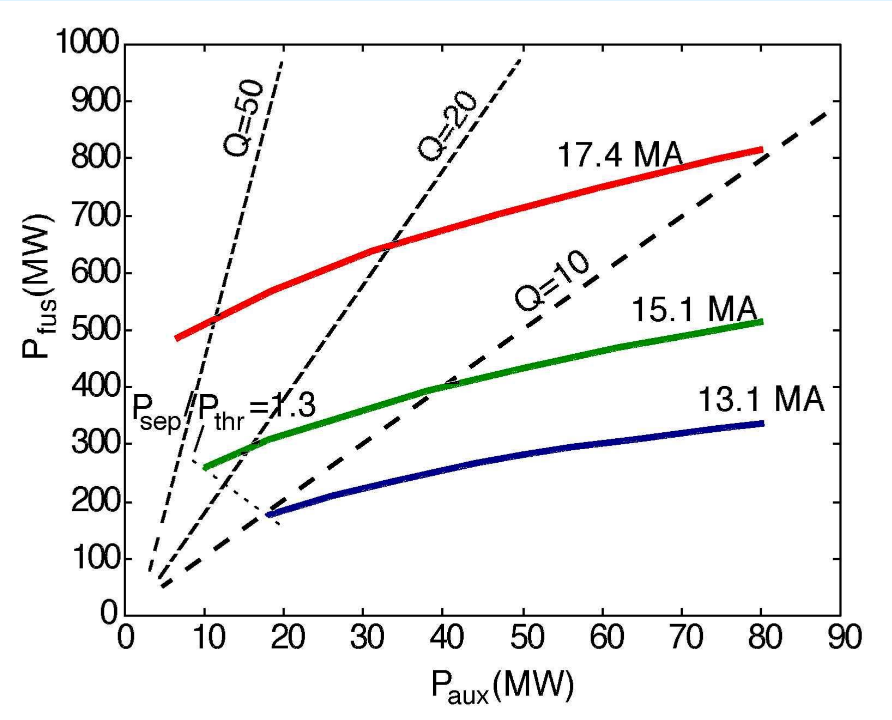
- JCT/HT task forces met in 1998 and 1999 to analyse and compare a range of options;
- System codes used to consistently relate plasma parameters, physics design constraints, engineering features, and costs;
- Representative options that span an appropriate range of aspect ratio were selected for more detailed studies of engineering and physics aspects;
-

	HAM	IAM	LAM
Aspect Ratio	3.5	3.25	2.8
Plasma Current I_p (MA)	12.7	13.0	17.0
Major Radius R (m)	6.30	6.20	6.45
Plasma volume (m^3)	635	725	1180


CENTRAL
SOLENOID
THERMAL
SHIELD
BIOSHIELD
TF COIL
INTERCOIL
STRUCTURE
PF1
PF2
UPPER
PORT
PF3
BLANKET
MODULE
EQUATORIAL
PORT
PF4
DIVERTOR
PORT
PF5
PF6
VACUUM
VESSEL
DIVERTOR
CASSETTE
0
1m
5m
CRYOSTAT
GRAVITY SUPPORT

Inductive Operation (2)

Range of performance

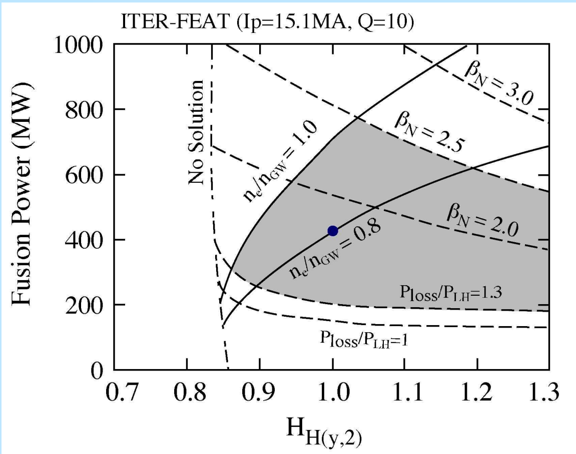


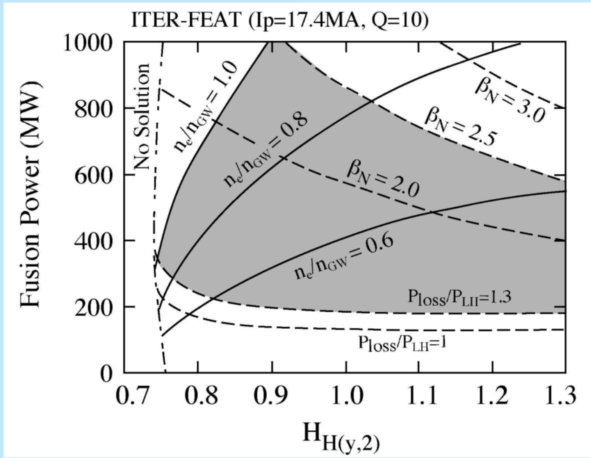
Fusion power (P_{fus}) versus auxiliary power (P_{aux}) for a range of currents and for H_H = 1 and $n_e/n_{Greenwald}$ = 0.85. Minimum fusion power is limited to a factor 1.3 above the expected power at which transition to L-mode would occur, namely:

$$P_{LH} = 0.75\ M^{-1}\ B_T^{0.82}\ n_e^{0.58}\ R^{1.00}\ a^{0.81}$$

Inductive Operation (3)

Flexibility in reaching Q=10





The combination of a range of plasma parameters will allow Q=10 to be obtained. The figures show the operational domain in terms of fusion power and H_H, plus the various limiting boundaries that are thought to apply.

Q=10 is maintained within the shaded region by adjusting auxiliary power and density.

Inductive Operation (4)

The results

- show the flexibility of the design,
- show its capacity to respond to factors that degrade confinement,
- show its ability to maintain the goal of extended burn Q=10 operation,
- imply the ability to explore higher Q operation,

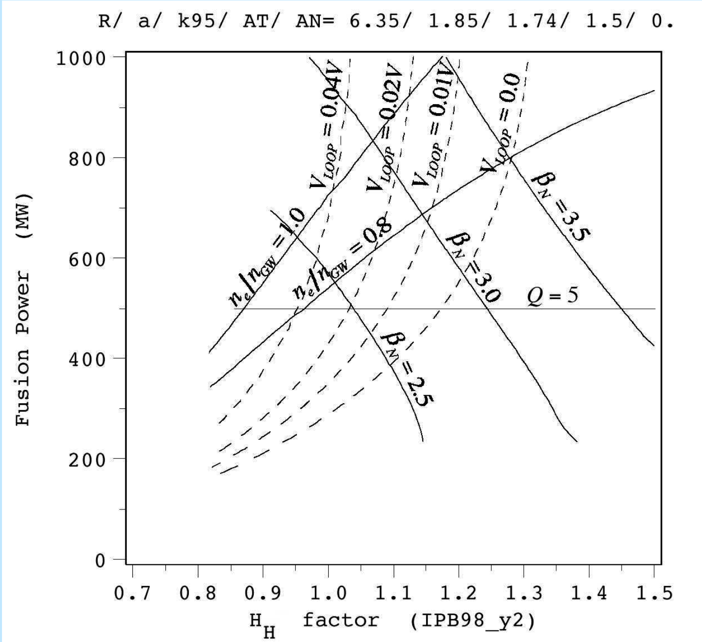
provided energy confinement times consistent with the confinement scaling are maintained.

Steady State/Hybrid Operation

Hybrid modes of operation are being evaluated as a promising route towards establishing true steady-state modes of operation. There, in addition to inductively driven current, a substantial fraction of the plasma current is driven by external heating and the bootstrap effect, leading to extension of the burn duration. This form of operation would be well suited to systems engineering tests.

For a given value of fusion power (and hence Q), as the confinement enhancement factor, H_H, increases (simultaneously decreasing plasma density and increasing β_N), the plasma loop voltage falls towards zero.

For example, operation with V_{loop} = 0.02 V and I_p = 12 MA, which corresponds to a flat-top length of 2500 s, is expected at H_H = 1, Q = 5, $n_e/n_{Greenwald}$ = 0.7, and β_N = 2.5. True steady-state operation at Q = 5 can be achieved with H_H = 1.2 and β_N = 2.8.



Operation space for I_p = 12 MA and P_{CD}= 100 MW, in terms of fusion power versus confinement enhancement factor, and the transition from hybrid to true steady-state operation.

Technical Characteristics

Performance

- Q > 10 with inductive current drive (ignition not precluded).
- Q > 5 using non-inductive current drive.
- Typical fusion power level ~ 500 MW

Design

- Use existing technology and physics database to give confidence but be able to access advanced operational modes.
    - Operation equivalent to a few 10000 inductive pulses of 300-500 s.
- Average neutron flux ≥ 0.5 MW/m^2
- Average end-of-life fluence ≥ 0.3 MWa/m^2

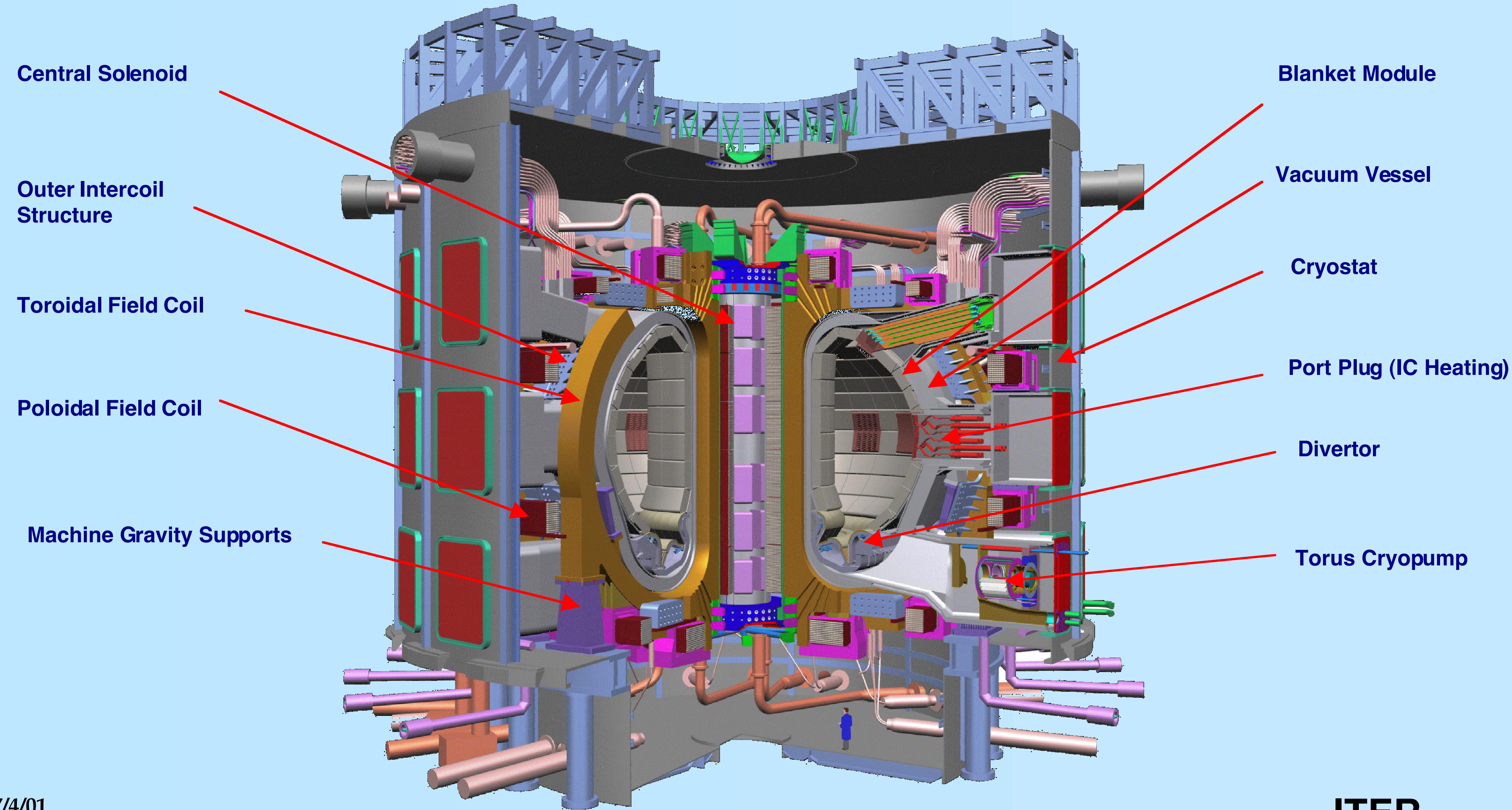
Operation

- Address all aspects of plasma dominated by α–particle heating through burning plasma experiments.
- Make low fluence functional tests of DEMO-relevant blanket modules early, and high reliability tests later.
- Device operation ~20 years. Tritium supplied from external sources.

ITER Parameters

Total fusion power	500 MW (700MW)
Q = fusion power/auxiliary heating power	≥10 (inductive)
Average neutron wall loading	0.57 MW/m^2 (0.8 MW/m^2)
Plasma inductive burn time	≥ 300 s
Plasma major radius	6.2 m
Plasma minor radius	2.0 m
Plasma current (inductive, I_p)	15 MA (17.4 MA)
Vertical elongation @95% flux surface/separatrix	1.70/1.85
Triangularity @95% flux surface/separatrix	0.33/0.49
Safety factor @95% flux surface	3.0
Toroidal field @ 6.2 m radius	5.3 T
Plasma volume	837 m^3
Plasma surface	678 m^2
Installed auxiliary heating/current drive power	73 MW (100 MW)

Design - Main Features



	ITER-FEAT	ITER-FDR
κ_{95} / κ_x	1.7 / 1.85	1.60 / 1.70
δ_{95}/ δ_x	0.35 / 0.33	0.24 / 0.31
R (m)	6.20	8.14
a (m)	2.0	2.8
R/a	3.1	2.9
B (T)	5.3	5.68
I_P (MA)	15.1	21.0
τ_{burn} (s)	≥ 400	≥ 1000
n / n_{GR}	0.8	1.15
<n> ($10^{20}m^{-3}$)	0.97	0.98
<T > (keV)	9.0	12.8
β_N	1.6	2.25
β (%)	2.3	2.97
P_{FUS} (MW)	400	1500
L_{wall} (MW/m^2)	0.57	1.0
$Q = P_{FUS} /(P_{NBI}+P_{RF})$	10	∞

Comparison of principal parameters for the ITER-FEAT design and the 1998 ITER design at nominal operating points

Design - Main Features (2)

Superconducting toroidal field coils (18 coils)	
Superconductor	Nb_3Sn in circular stainless steel (SS) jacket in grooved radial plates
Structure	Pancake wound, in welded SS case, wind, react and transfer technology
Superconducting Central Solenoid (CS)	
Superconductor	Nb_3Sn in square Incoloy jacket, or in circular Ti/SS jacket inside SS U-channels
Structure	Pancake wound, 3 double or 1 hexa-pancake, wind react and transfer technology
Superconducting poloidal field coils (PF 1-6)	
Superconductor	NbTi in square SS conduit
Structure	Double pancakes
Vacuum Vessel (9 sectors)	
Structure	Double-wall welded ribbed shell, with internal shield plates and ferromagnetic inserts
Material	SS 316 LN structure, SS 304 with 2% boron

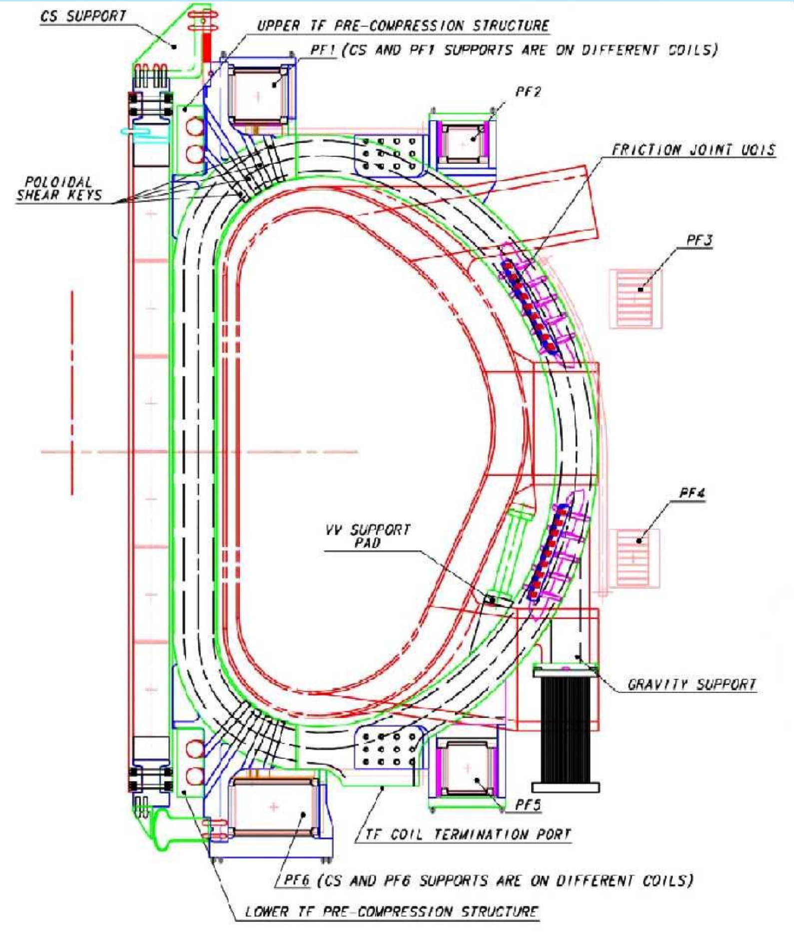
Design - Main Features (3)

First Wall/Blanket (421 modules)		**(Initial DT Phase)**
	Structure	**Single curvature faceted separate FW attached to shielding block which is fixed to vessel**
	Materials	**Be armour, Cu-alloy heat sink, SS 316 LN structure**
Divertor (54 cassettes)		
	Configuration	**Single null, cast or welded plates, cassettes**
	Materials	**W alloy and C plasma facing components, copper alloy heat sink, SS 316 LN structure**
Cryostat		
	Structure	**Ribbed cylinder with flat ends**
	Maximum inner dimensions	**28 m diameter, 24 m height**
	Material	**SS 304L**
Heat Transfer Systems (water-cooled)		
	Heat released in the tokamak duringnominal pulsed op.	**750 MW at 3 and 4.2 MPa water pressure, ~120°C**

Design - Main Features (4)

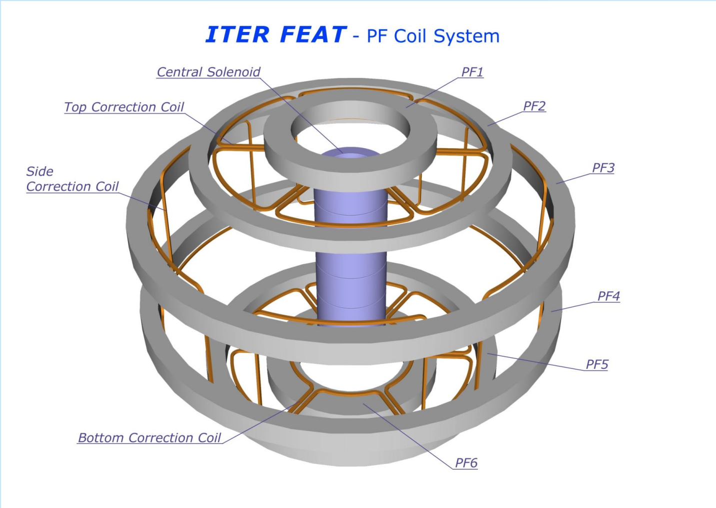
Cryoplant		
	Nominal average He refrig. /liquefac. rate for magnets & divertor cryopumps (4.5K)	55 kW / 0.13 kg⁄s
	Nominal cooling capacity of the thermal shields at 80 K	660 kW
Additional Heating and Current Drive		
	Candidate systems	Electron Cyclotron, Ion Cyclotron, Lower Hybrid , Negative Ion Neutral Beam
Electrical Power Supply		
	Pulsed Power supply from grid: total active/reactive power demand	500 MW / 400 MVAr
	Steady-State Power Supply from grid: total active/reactive power demand	110 MW/ 78 MVAr

Design - Magnets and Structures (1)

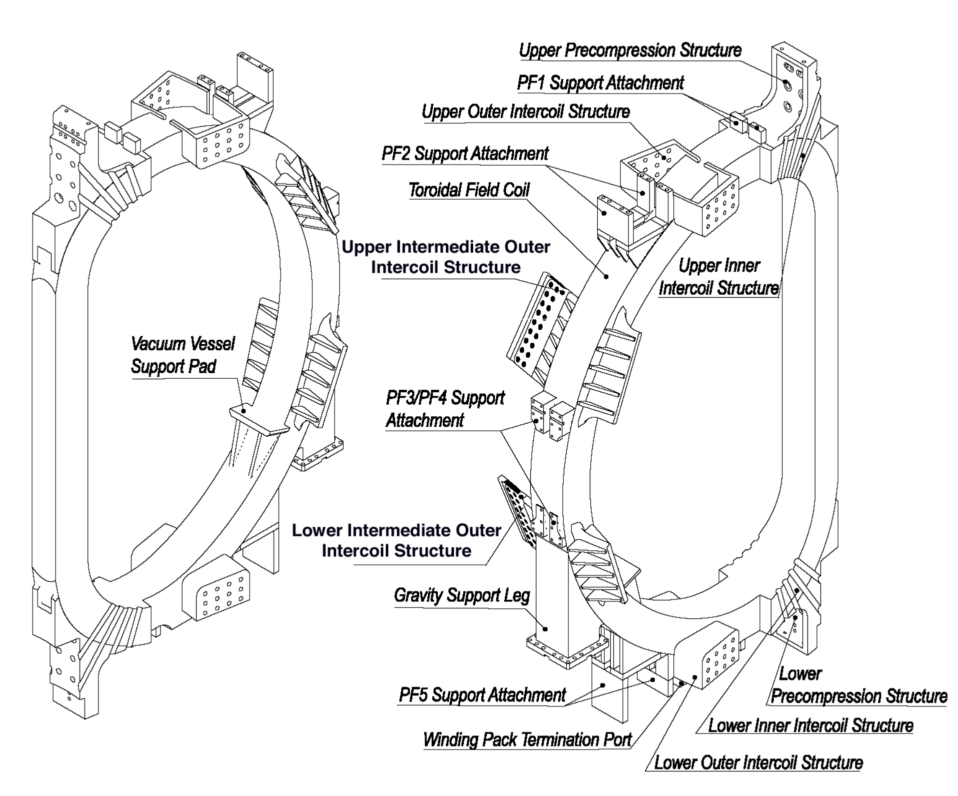


Superconducting. 4 main subsystems:

- **18 toroidal field (TF) coils produce confining/stabilizing toroidal field;**
- **6 poloidal field (PF) coils position and shape plasma;**
- **a central solenoid (CS) coil induces current in the plasma.**
- **correction coils (CC) correct error fields due to manufacturing/assembly imperfections, and stabilize the plasma against resistive wall modes.**

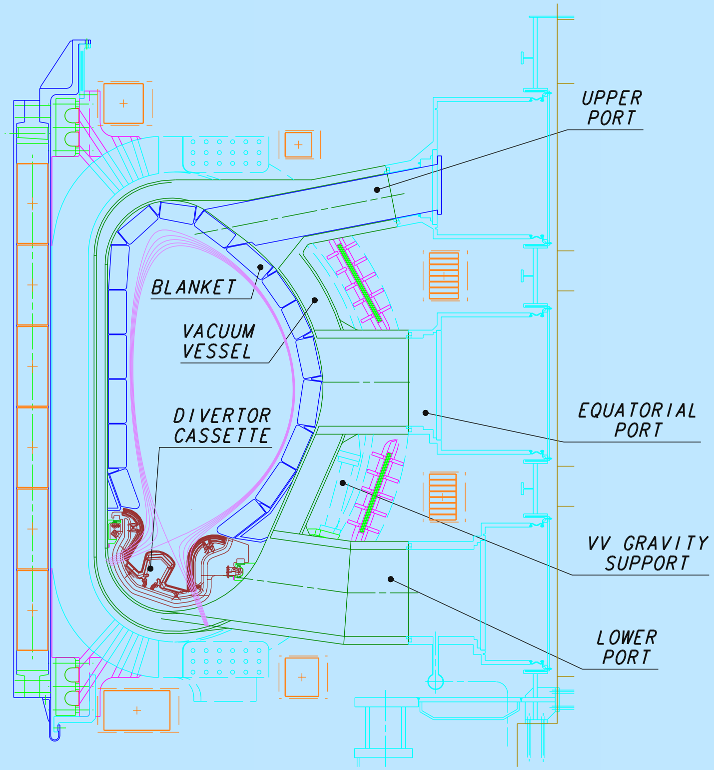


Design - Magnets and Structures (2)



- **TF coil case provides main structure of the magnet system and the machine core. PF coils and vacuum vessel are linked to it. All interaction forces are resisted internally in the system.**
- **TF coil inboard legs are wedged together along their side walls and linked at top and bottom by two strong coaxial rings which provide toroidal compression and resist the local de-wedging of those legs under load.**
- **On the outboard leg, the out-of-plane support is provided by intercoil structures integrated with the TF coil cases.**
- **The magnet system weighs ~ 8,700 t.**

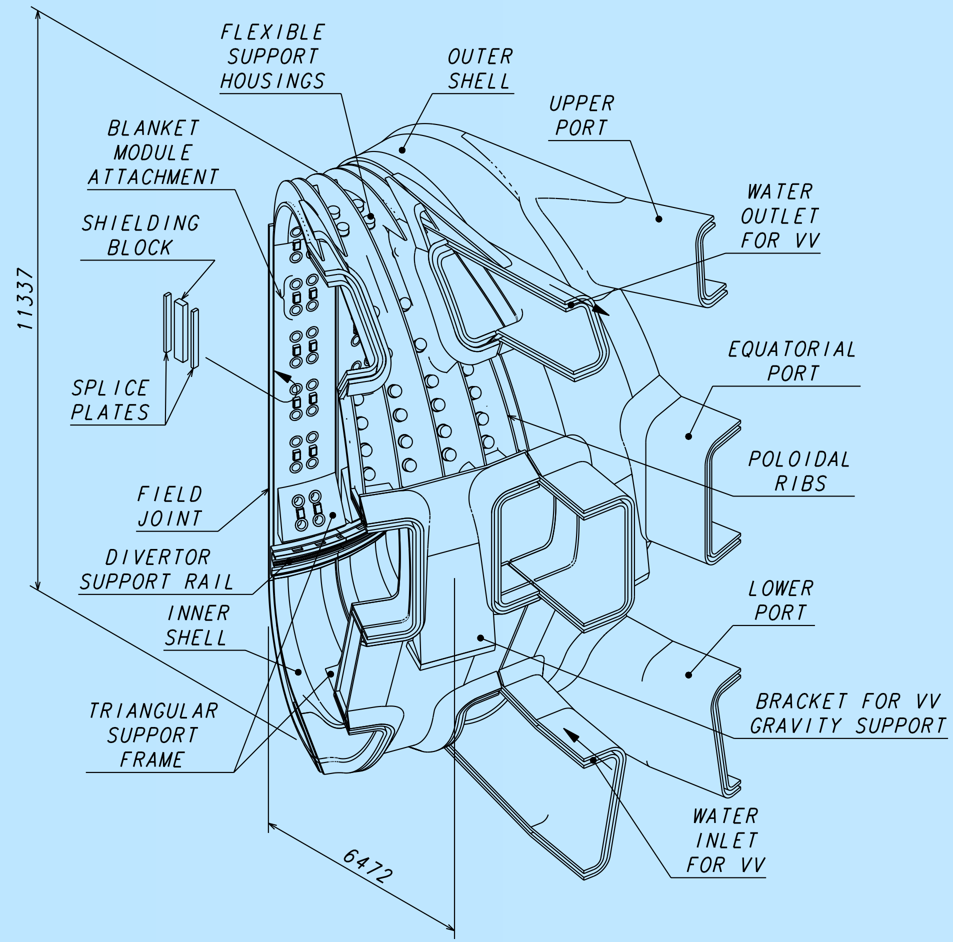
Design - Vessel, Blanket & Divertor (1)



The double-walled vacuum vessel is lined by modular removable components, including blanket modules, divertor cassettes, and diagnostics sensors, as well as port plugs for limiters, heating antennae, diagnostics and test blanket modules. All these removable components are mechanically attached to the VV. The total vessel/in-vessel mass is ~10,000 t.

These components absorb most of the radiated heat from the plasma and protect the magnet coils from excessive nuclear radiation. The shielding is steel and water, the latter removing heat from absorbed neutrons. A tight fitting configuration of the VV to the plasma aids passive plasma vertical stability, and ferromagnetic material "inserts" in the VV located in the shadow of the TF coils reduce toroidal field ripple and its associated particle losses.

Design - Vessel, Blanket & Divertor (2)

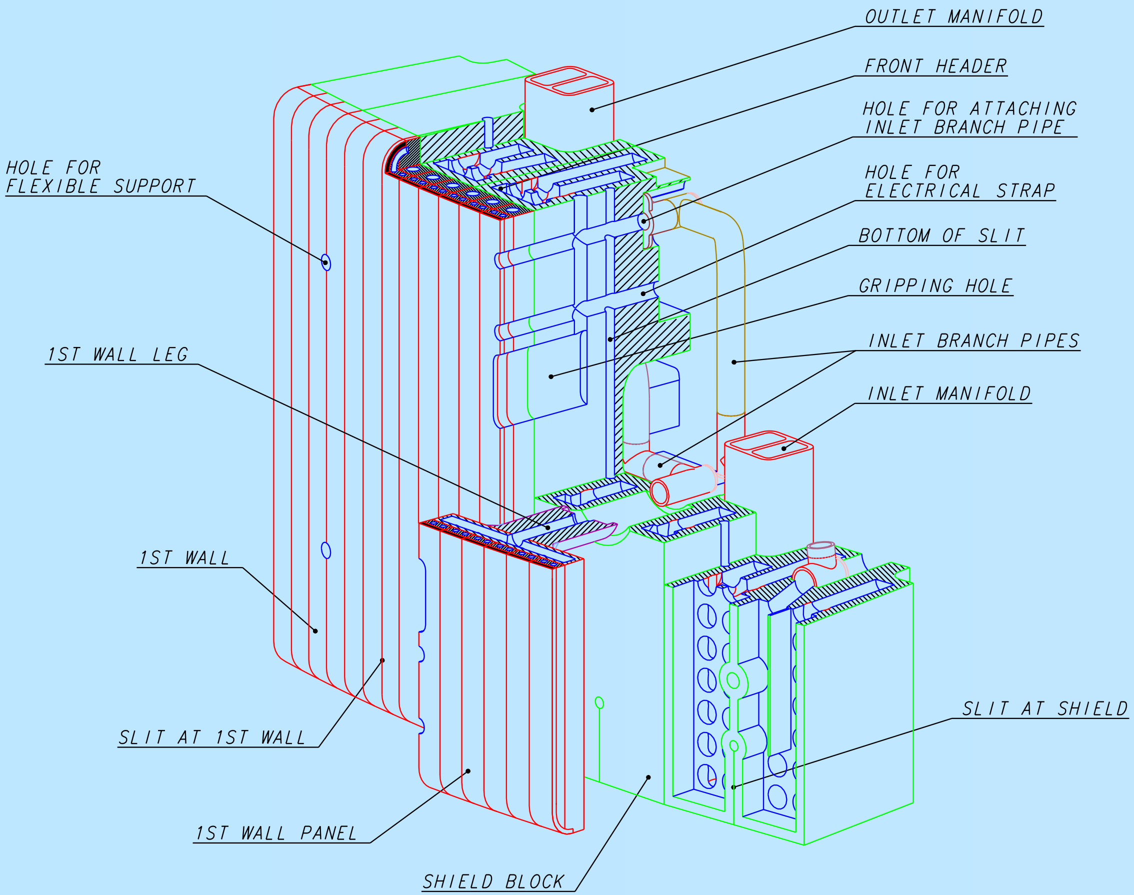


The primary functions of the vacuum vessel (VV) are to provide a high quality vacuum for the plasma, as well as the first confinement barrier to radioactive materials and a second barrier (after the cryostat) for the separation of air from potential sources of in-vessel hydrogen generation.

The decay heat of all the in-vessel components can be removed by the water in the VV primary heat transfer system (PHTS) system, even in conditions when the other PHTSs are not functioning.

There are 9 x 40° vessel sectors.

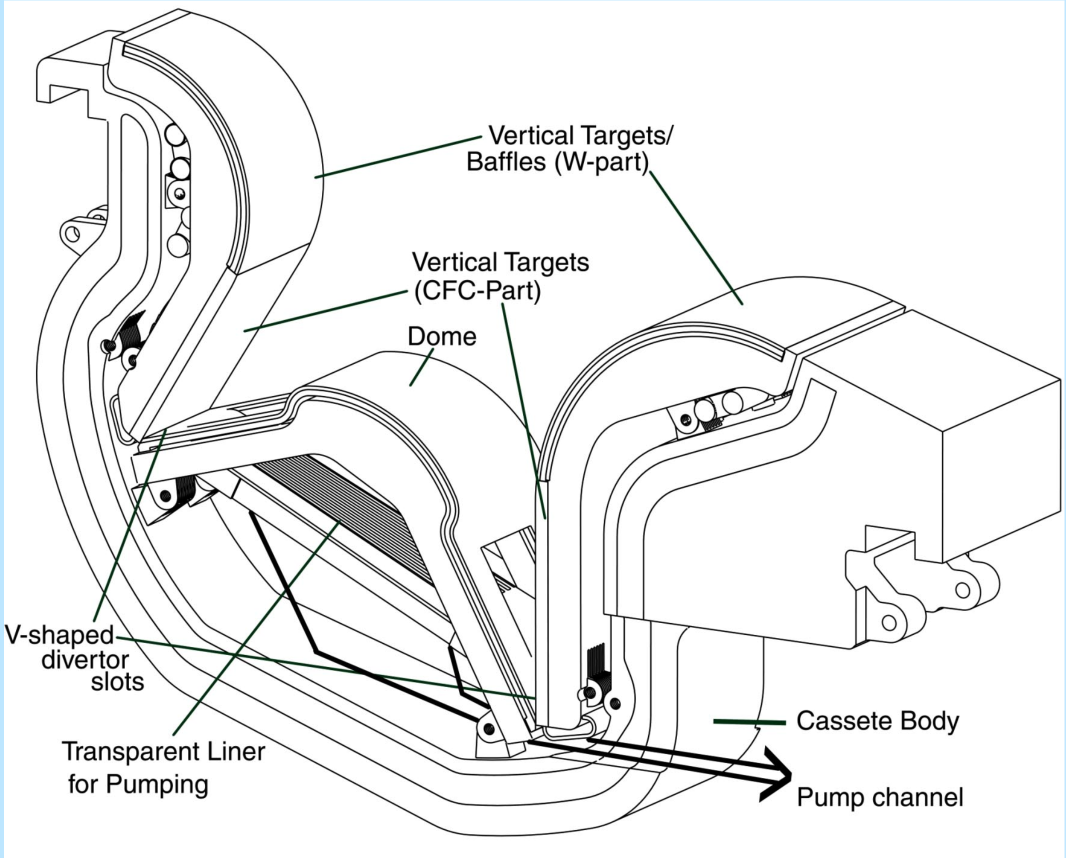
Design - Vessel, Blanket & Divertor (3)



The ~ 420 blanket modules consist of a detachable faceted first wall (FW) built with Be armour and a water-cooled copper heat sink attached to a SS shielding block. This minimises radioactive waste and simplifies manufacture. The blanket cooling channels are mounted on the vessel.

The initial blanket acts solely as a neutron shield, and tritium breeding experiments are confined to the test blanket modules which can be inserted and withdrawn at radial equatorial ports.

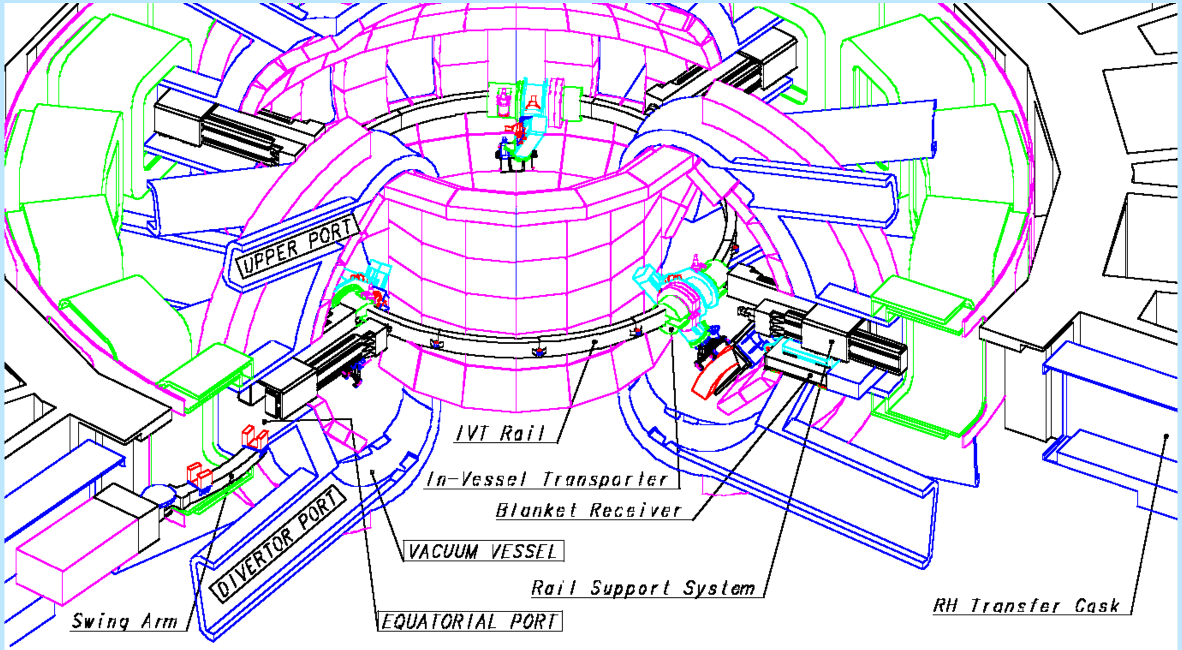
Design - Vessel, Blanket & Divertor (4)



The divertor is made up of 54 cassettes. The target and divertor floor form a V which traps neutral particles protecting the target plates, without adversely affecting helium removal. The large opening between the inner and outer divertor balances heat loads in the inboard and outboard channels.

The design uses C at the vertical target strike points. W is the backup, and both materials have their advantages and disadvantages. C is best able to withstand large power density pulses (ELMs, disruptions), but gives rise to tritiated dust and T codeposited with C which has to be periodically removed. The best judgement of the relative merits can be made at the time of procurement.

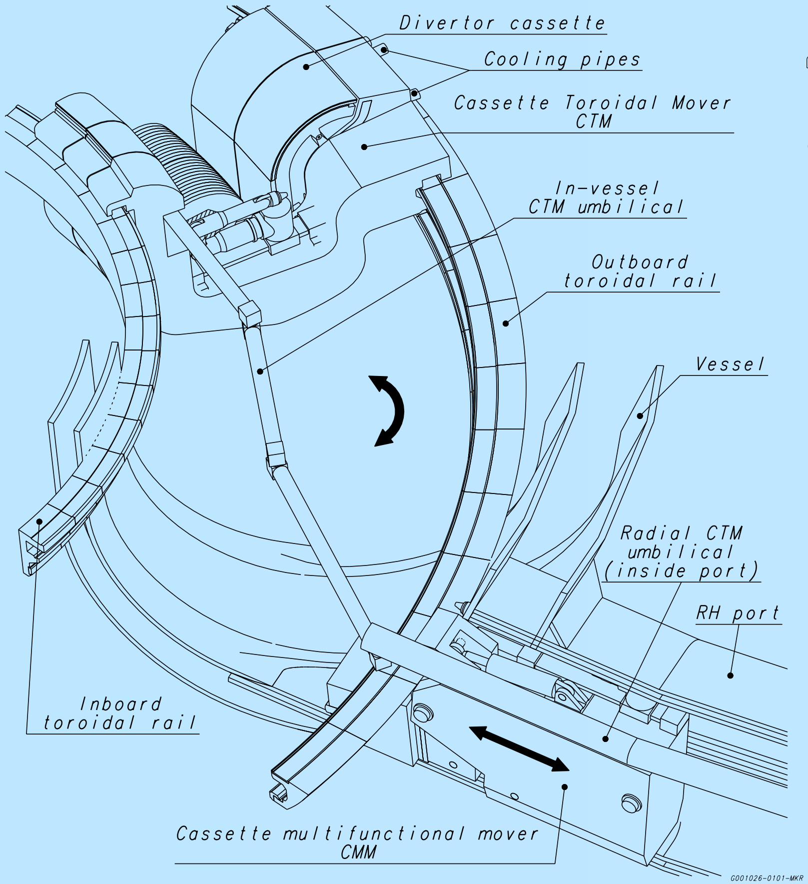
Design - In-vessel Remote Handling (1)

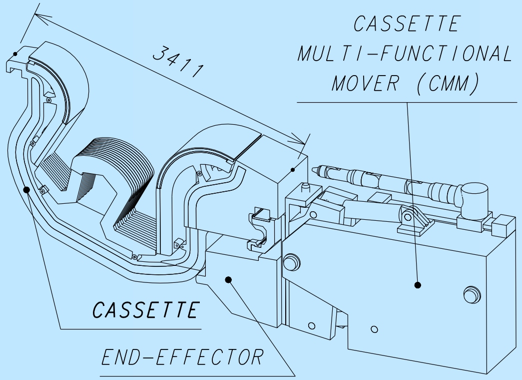


Systems near the plasma will become radioactive and will require remote maintenance, with special remote handling equipment. In-vessel transporters are used to remove and reinstall blanket modules.

Unshielded casks, which dock to the access ports of the vacuum vessel, house such equipment and transport radioactive items from the tokamak to the hot-cell where refurbishment or waste disposal can be carried out. Docking is tight, to avoid spread of contamination. Hands-on assisted maintenance is used wherever justifiable, following ALARA principles.

Design - In-vessel Remote Handling (2)



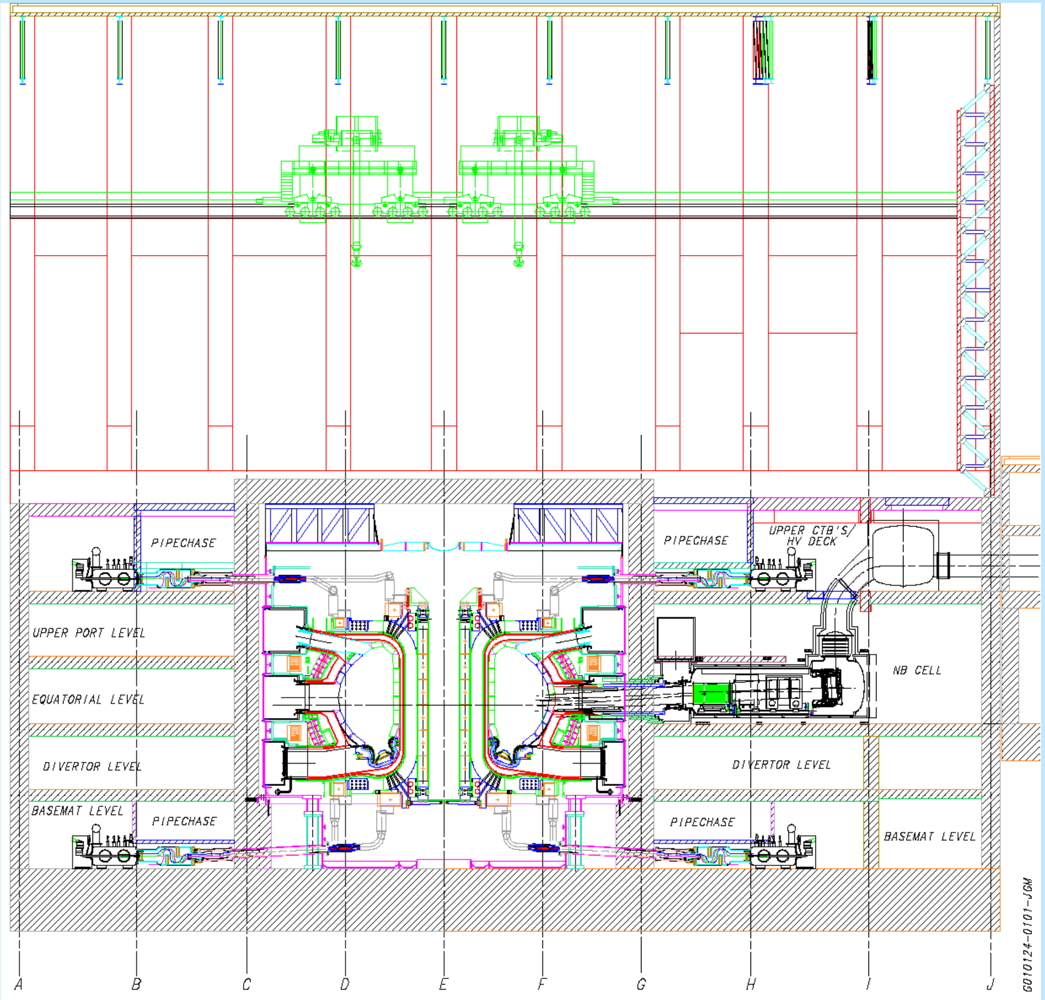


Multifunction manipulators are used for divertor cassette removal and to handle vacuum vessel port plugs. A toroidal mover slides the divertor cassettes along rails into their final position.

Comprehensive R&D has successfully demonstrated that key maintenance operations can be achieved using common remote handling technology.

Crucial issues such as vacuum vessel remote cutting and re-welding, viewing, materials and components radiation hardness have been addressed and demonstrated.

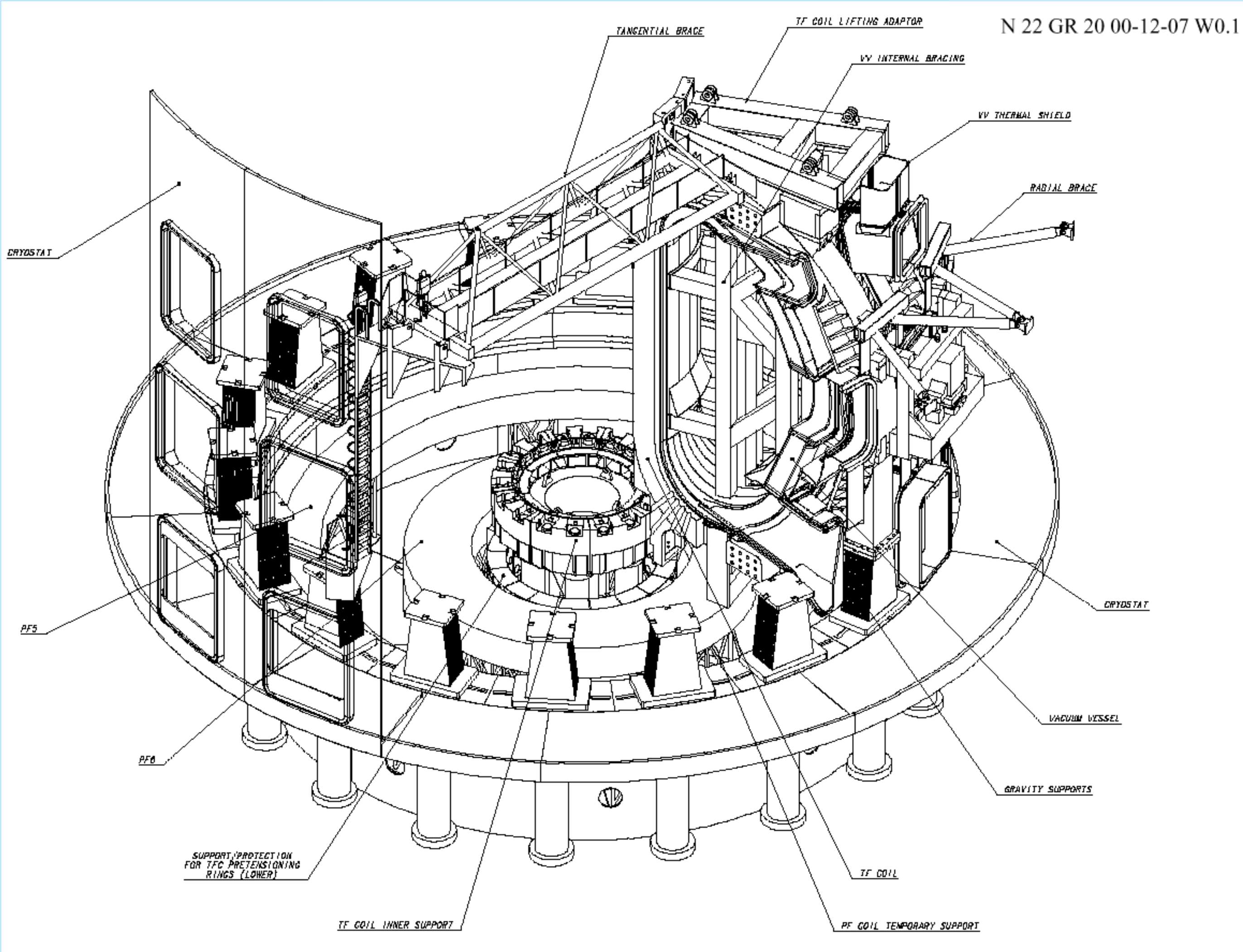
Design - Tokamak Building



- Provides a biological shield around cryostat to minimise activation and permit human access.
- Additional confinement barrier.
- Allows (with HVAC) contamination spread to be controlled.
- Provides shielding during remote handling cask transport.
- Can be seismically isolated.

ITER Site Layout

Assembly

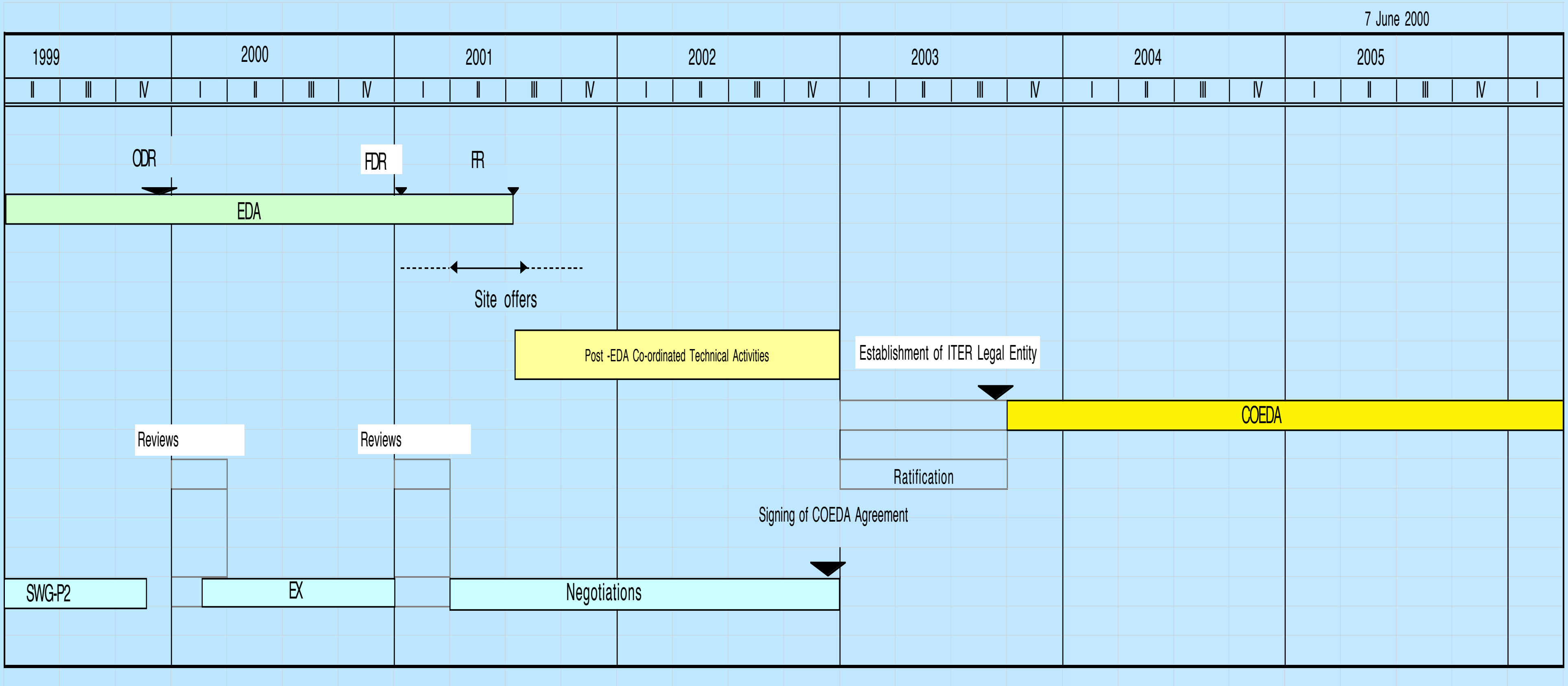


N 22 GR 20 00-12-07 W0.1
TANGENTIAL BRACE
TF COIL LIFTING ADAPTOR
VV INTERNAL BRACING
VV THERMAL SHIELD
RADIAL BRACE
CRYOSTAT
CRYOSTAT
PF5
PF6
VACUUM VESSEL
GRAVITY SUPPORTS
SUPPORT/PROTECTION FOR TFC PRETENSIONING RINGS (LOWER)
TF COIL
TF COIL INNER SUPPORT
PF COIL TEMPORARY SUPPORT

Decommissioning

PHASE	ACTIVITY	DESCRIPTION	DURATION
1	De-activation	a) Remove mobilizable tritium and dust from the machine using available techniques and equipment. Remove and de-activate coolants. b) Classify and package active, contaminated and toxic material. c) Remove all the in-vessel components. OPTION 1: remove ex-vessel components (if not done in phase 2).	~ 5 years
	The ITER facility is handed over to an organization inside the host country		
Radioactivity decay period		a) The vacuum vessel radioactivity is left to decay to a level which allows the extraction of vessel sectors into the tokamak building (during phase 2) for size reduction and disposal. b) No site activities are required except security and monitoring.	As required (at least ~ 25 years)
2	Final Dismantling and Disposal	a) Remove vacuum vessel sectors and reduce their size by remote/semi-remote operations. OPTION 2: remove ex-vessel components (if not done in phase 1) b) Classify and package active, contaminated and toxic material	~ 6 years

Immediate Timetable



The design policy against possible physics issues is to provide tools to mitigate or suppress their consequences

- Confinement: choice of the more conservative (-20%) extrapolation IPB 98 (y,2)
- He content: large pumping throughput (200 Pa m^3/s)
- Sawteeth:
  - Suppression of q=1 surface by early heating during current increase
  - If present and stabilized by high energy particles, reduction of their period by ECCD
- β:
  - stabilisation of islands at limited amplitude by ECCD on q=3/2, 2 surfaces
  - control of RWM by corrections coils
- ELMs: (a possible factor limiting the life time of divertor targets)
  - possibility to obtain grassy ELMs (type II)
  - at least, increase frequency of type I ELMs
- Disruption, VDE: large influx of D, H (or Be) (R/D needed)
- Density: pellet injection from high field side (R/D ongoing)

Direct Capital Cost

Components/Systems	Direct Cost (kIUA*)	% of Total
Magnet Systems	762	28
Vessel, Blanket, Divertor, Pumping & Fuelling	505	18
Cryostat & Thermal Shield	105	4
Assembly	93	3
Auxiliaries	586	21
Buildings	380	14
Heating & Current Drive (73 MW)	206	7
Diagnostics (start-up set)	118	4
Total Direct Capital Costs	**2755**	**100**

*1 kIUA = $\$_{1989}$1 M ≈ $\$_{2000}$1.392M ≈ $€_{2000}$1.279M ≈ $¥_{2000}$148M

Lifetime Cost

	kIUA*
Construction Costs	
Direct capital	2755
Management & Support	477
R&D During Construction	~70
Operation Costs (average per year)	
Permanent personnel	60
Energy	~30
Fuel	~8
Maintenance /improvements	~90
Decommissioning	335

*1 kIUA = $\$_{1989}$1 M ≈ $\$_{2000}$1.392M ≈ $€_{2000}$1.279M ≈ $¥_{2000}$148M

Procurement Strategy

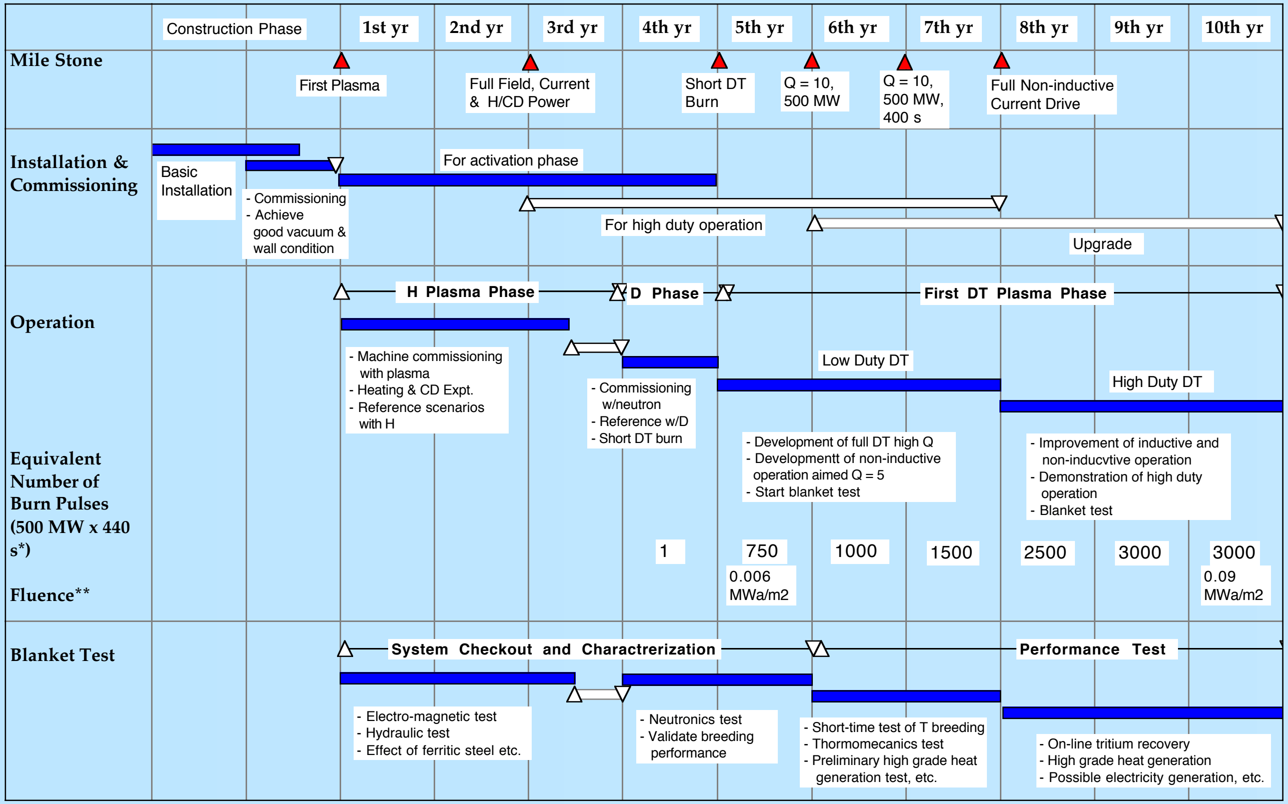
- **Presume that the Parties:**
  - Contribute either "in kind" or/and in "Funds"
  - Request involvement in all new fusion specific technologies
  - limit the total cost
- **Consequences:**
  - Easiest (necessary ?): procurement in fusion specific technologies should be by contributions in kind only
  - Procurements in conventional technologies (or low Party interest) might be by contributions in funds (international call for tenders, without "juste retour" enforced) - even easier with "in kind" contributions
  - Therefore, define procurement packages and their "values"

Conclusions on the Approach to the ITER Construction Cost

- The approach provides fair and consistent relative costs for the different ITER systems and components. The Parties can, jointly, appreciate in advance the relative contributions (in percentage) that each might make to building ITER and,

- Individually, estimate from the underlying physical data the absolute costs (in their own currency) that each might expect to incur in providing specific components, inside their contribution “in kind”, by applying their own appropriate conversion factor to IUA.

Indicative Operation Schedule



* The burn time of 440 s includes 400 s flat top and equivalent time which additional flux is counted during ramp-up and ramp-down.
** Average Fluence at First Wall (Neutron wall load is 0.56 MW/m2 in average and 0.77MW/m2 at outboard midplane.)

A further 10 year DT phase will improve overall performance and test components. The programme should be decided following a review of the preceding results.

Whether to incorporate tritium breeding during this phase will be decided on the basis of the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance.