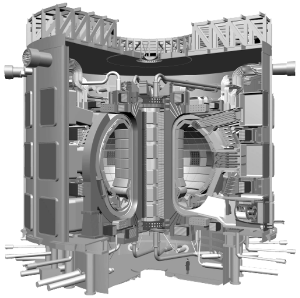


International Thermonuclear Experimental Reactor

ITER DIAGNOSTICS

presented by

A E Costley

ITER JWS, Naka, Japan

2^{nd} Meeting of the ITPA Topical Group on Diagnostics,
General Atomics, San Diego, 4 - 8 March, 2002

OUTLINE

ITER-FEAT

- Programmatic Objectives
- Key Machine Parameters
- Operation Scenarios and Phases

Requirements for Plasma and First Wall Measurements

- Requirements for Different Operating Scenarios

Diagnostic System Selection and Design

Assessment of Measurement Capability

- Plasma current, position and shape, electron temperature

Necessary Next Steps

Concluding Remarks

ITER-FEAT: PROGRAMMATIC OBJECTIVES

Plasma Performance

- achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power of at least 10
- for a range of operating scenarios
- with duration sufficient to achieve stationary conditions on the time scales characteristic of plasma processes.
- aim at demonstrating steady-state operation using non-inductive current drive with the ratio of fusion to current drive power of at least 5
- the possibility of controlled ignition should not be precluded

Technology

- **demonstration of integrated operation of technologies essential for a fusion reactor**
- **testing of key components for a fusion reactor**
- **testing of concepts for a tritium breeding module**

Key Design Requirements

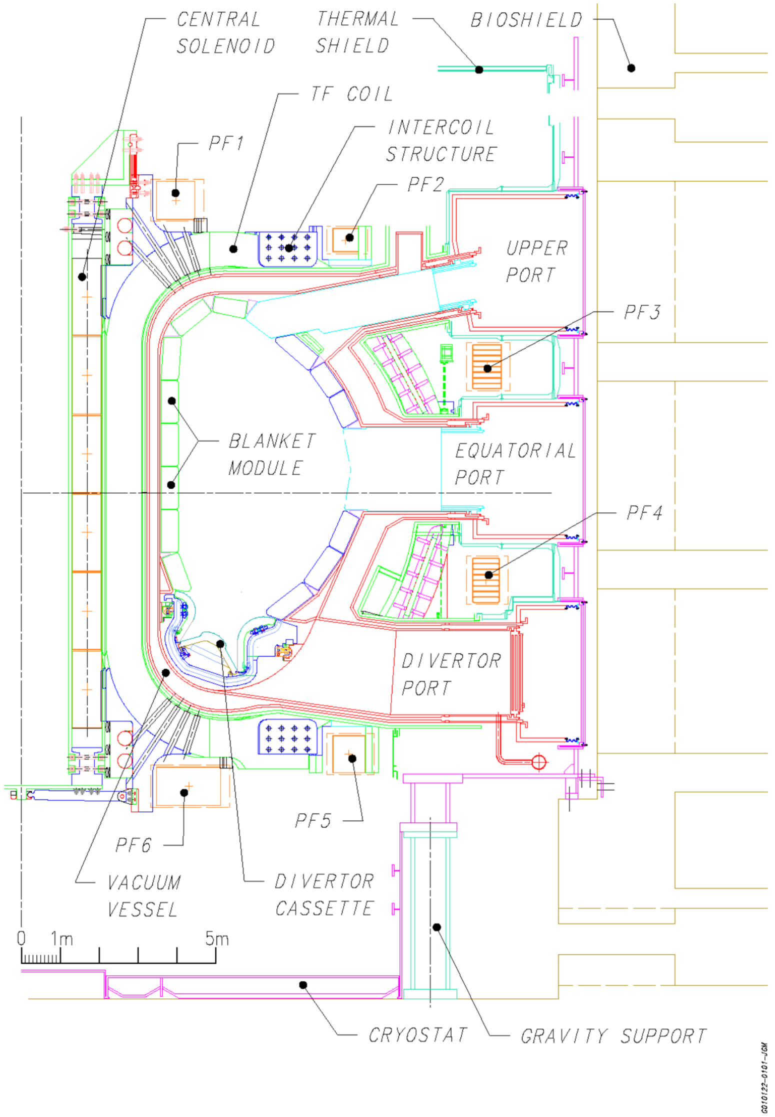
- **An inductive flat-top capability during burn of 300 to 500 s.**
- **The design should be capable of supporting advanced modes of plasma operation under investigation in existing experiments, and should permit a wide operating parameter space to allow for optimising plasma performance.**
- **For nuclear and high heat flux component testing**
- **Average neutron flux $\geq$ 0.5 MW/m^2**
- **Average neutron fluence $\geq$ 0.3 MWa/m^2**

Main Device and Plasma Parameters

Total Fusion Power	**500 MW *(700 MW)***
Q — fusion power/additional heating power	**≥ 10**
Average 14MeV neutron wall loading	**0.57 MW/m^2 *(0.8 MW/m^2)***
Plasma inductive burn time	**≥ 400 s**
Plasma major/minor radius (R)/(a)	**6.2/2.0 m**
Plasma current (I_p)	**15 MA *(17 MA (1))***
Vertical elongation @95% flux surface/separatrix (κ_{95})	**1.70/1.85**
Safety factor @95% flux surface (q_{95})	**3.0**
Toroidal field @6.2 m radius (B_T)	**5.3 T**
Plasma volume	**837 m^3**
Plasma surface	**678 m^2**
Installed aux. heating/current drive power	**73 MW *(2)***

(1) The machine is capable of a plasma current up to 17MA, with the parameters shown in parentheses) within some limitations over some other parameters (e.g., pulse length).

(2) A total plasma heating power up to 110MW may be installed in subsequent operation phases.



Tokamak Poloidal Cross-Section

Heating and Current Drive Parameters

	NB (1MeV)	EC (170 GHz)	IC (~ 50 MHz)	LH (5 GHz)
Power injected per unit equatorial port (MW)	16.5	20	20	20
Number of units for the first phase	2	1	1	0
Total power (MW) for the first phase	33	20	20	0
The 20 MW of EC module power will be used either i) in up to 3 upper ports to control neoclassical tearing modes at the q = 3/2 and q = 2 magnetic surfaces, or ii) in one equatorial port for H&CD mainly in the plasma centre.				

Operation Scenarios and Phases

The design of ITER allows for operation in various scenarios.

1. **Inductive operation I: 500 MW, Q=10, 15 MA operation with heating during current ramp-up**
2. **Inductive operation II: 400 MW, Q=10, 15 MA operation without heating during current ramp-up**
3. **Hybrid operation**
4. **Non-inductive operation I: weak negative shear operation**
5. **Inductive operation III: 700 MW, 17 MA, with heating during current ramp-up.**
6. **Non-inductive operation II: strong negative shear**

Operation Phases

H Phase

This is a non-nuclear phase, mainly planned for full commissioning of the tokamak system in a non-nuclear environment. Develop full DT phase reference operation. Check partial detached divertor operation

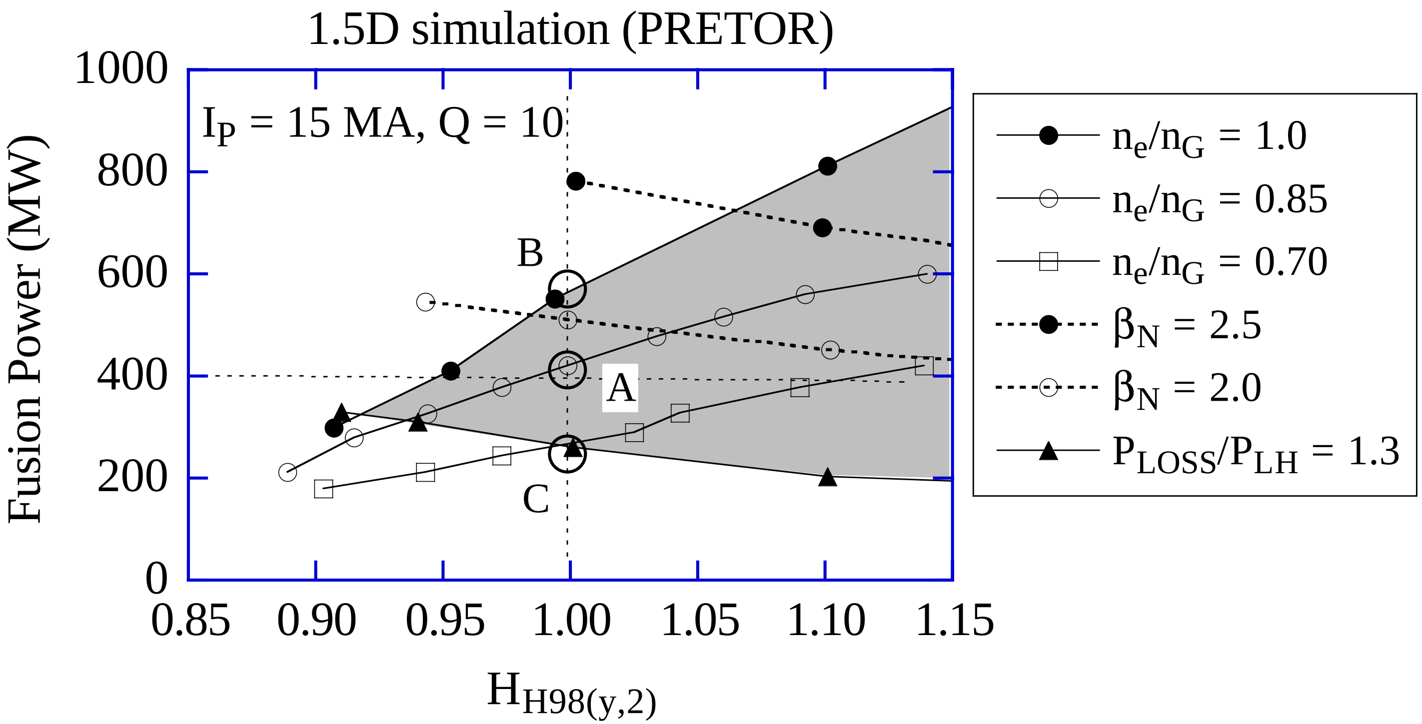
D Phase

Tritium produced from DD reactions → DT reactions. Reference DT operational scenarios can be established.

DT Phases

During the first phase of DT operation the fusion power and burn pulse length will be gradually increased until the inductive operational goal is reached. Non-inductive, steady-state operation developed.

Second phase of full DT operation, beginning after a total of about ten years. Emphasize improvement of the overall performance and the testing of components and materials with higher neutron fluences



Operation Domain in H_H-Factor and Fusion Power Space when I_P = 15 MA and Q = 10

PLASMA AND FIRST WALL MEASUREMENTS REQUIRED FOR ITER

(Control, Evaluation and Anticipated Physics Studies)

GROUP 1a Measurements For Machine Protection and Basic Control	GROUP 1b Measurements for Advanced Control	GROUP 2 Additional Measurements for Performance Eval. and Physics
Plasma shape and position, separatrix- wall gaps, gap between separatrixes Plasma current, q(a), q(95%) Loop voltage Fusion power $\beta_N = \beta_{tor}(aB/I)$ Line-averaged electron density Impurity and D,T influx (divertor, & main plasma) Surface temp. (div. & upper plates) Surface temperature (first wall) Runaway electrons 'Halo' currents Radiated power (main pla, X-pt & div). Divertor detachment indicator (J_{sat}, n_e, T_e at divertor plate) Disruption precursors (locked modes, m=2) H/L mode indicator Z_{eff} (line-averaged) n_T/n_D in plasma core ELMs Gas pressure (divertor & duct) Gas composition (divertor & duct) Dust	Neutron and α-source profile Helium density profile (core) Plasma rotation (toroidal and poloidal) Current density profile (q-profile) Electron temperature profile (core) Electron density profile (core and edge) Ion temperature profile (core) Radiation power profile (core, X-point & divertor) Z_{eff} profile Helium density (divertor) Heat deposition profile (divertor) Ionization front position in divertor Impurity density profiles Neutral density between plasma and first wall n_e, T_e of divertor plasma Alpha-particle loss Low m/n MHD activity Sawteeth Net erosion (divertor plate) Neutron fluence	Confined α-particles TAE Modes, fishbones T_e profile (edge) n_e, T_e profiles (X-point) T_i in divertor Plasma flow (divertor) $n_T/n_D/n_H$ (edge) $n_T/n_D/n_H$ (divertor) T_e fluctuations n_e fluctuations Radial electric field and field fluctuations Edge turbulence MHD activity in plasma core Pellet ablation

Measurement Requirements According to Operating Scenario For Control and Evaluation (not for Physics studies)

Operating Scenario	**Required Measurements**	
	Control	**Evaluation**
H phase. Inductive. Ohmic L- mode. Limited H-Mode	Plasma shape and position, vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall	q(a), halo current, impurity identification and influx, $n_e(r)$ and $T_e(r)$ in core, T_i in core, P_{rad} from core, line-averaged Z_{eff}, H/L mode indicator, gas pressure and composition (divertor and duct), ELM occurrence and type
D phase. Inductive. ELMy H-mode	Plasma shape and position, vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall**, H/L mode indicator, ELM occurrence and type,** divertor detachment	q(a), halo current impurity identification and influx, $n_e(r)$ and $T_e(r)$ in core, T_i in core, P_{rad} from core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), shape and position (500 s), β, q(95%), $n_e(r)$ and $T_e(r)$ at edge, P_{fus}, $P_{rad}(r)$, heat deposition profile in divertor

New entry in red. Promotion from evaluation to control in bold

Operating Scenario	**Required Measurements**	
	Control	**Evaluation**
High power D/T phase. Inductive. ELMY H Mode	Plasma shape and position (500 s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, **T_e(r) in core, T_I in core, P_{rad} from core, P_{fus}** , n_{He}(r), n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, v_{tor}(r) and v_{pol}(r)	q(a), halo current impurity identification and influx, n_e(r) in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), β, q(95%), n_e(r) and T_e(r) at edge, P_{rad}(r), heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, T_i(r) in core, Z_{eff}(r), D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles
D/T Phase. Inductive ELMY H mode. High β	Plasma shape and position (500 s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, T_e(r) in core, T_I in core, P_{rad} from core, P_{fus} n_{He}(r), n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, v_{tor}(r) and v_{pol}(r), β, localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs.	q(a), halo current impurity identification and influx, n_e(r) in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), n_e(r) and T_e(r) at edge, P_{rad}(r), heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, T_i(r) in core, Z_{eff}(r), D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

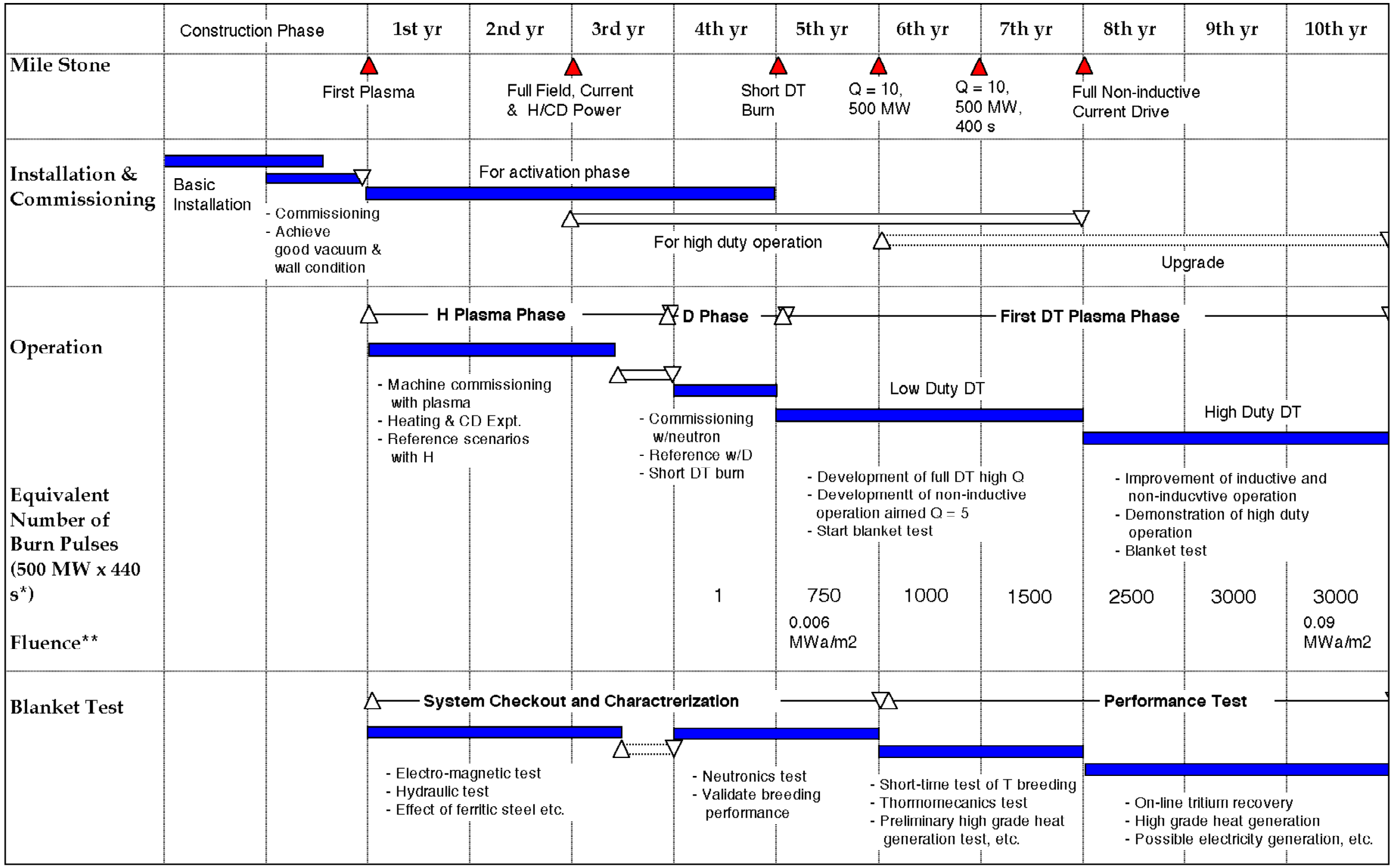
Operating Scenario	**Required Measurements**	
	Control	**Evaluation**
Hybrid operation	Plasma shape and position (1000s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, $T_e(r)$ in core, T_I in core, P_{rad} from core, P_{fus} $n_{He}(r)$, n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, $v_{tor}(r)$ and $v_{pol}(r)$, β,localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs.	q(a), halo current impurity identification and influx, $n_e(r)$ in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), $n_e(r)$ and $T_e(r)$ at edge, $P_{rad}(r)$, heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, $T_i(r)$ in core, $Z_{eff}(r)$, D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

New entry in red. Promotion from evaluation to control in bold

Operating Scenario	**Required Measurements**	
	Control	**Evaluation**
Steady state operation	Plasma shape and position (1000s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, T_e(r) in core, T_I in core, P_{rad} from core, P_{fus} n_{He}(r), n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, v_{tor}(r) and v_{pol}(r), β,localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs, **T_i(r) in core**, q(r) (in particular localisation and position of q_{min}), high resolution measurements of the gradient of T_e and T_i, measurement of RWMs	q(a), halo current impurity identification and influx, n_e(r) in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), n_e(r) and T_e(r) at edge, P_{rad}(r), heat deposition profile in divertor, shape and position (500 s), neutron and alpha source profiles, impurity profile, Z_{eff}(r), D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

New entry in red. **Promotion from evaluation to control in bold**

Initial Operation Plan



Construction Phase | 1st yr | 2nd yr | 3rd yr | 4th yr | 5th yr | 6th yr | 7th yr | 8th yr | 9th yr | 10th yr

Mile Stone

First Plasma

Full Field, Current & H/CD Power

Short DT Burn

Q = 10, 500 MW

Q = 10, 500 MW, 400 s

Full Non-inductive Current Drive

Installation & Commissioning

Basic Installation

- Commissioning
- Achieve good vacuum & wall condition

For activation phase

For high duty operation

Upgrade

Operation

H Plasma Phase

- Machine commissioning with plasma
- Heating & CD Expt.
- Reference scenarios with H

D Phase

- Commissioning w/neutron
- Reference w/D
- Short DT burn

First DT Plasma Phase

Low Duty DT

- Development of full DT high Q
- Developmentt of non-inductive operation aimed Q = 5
- Start blanket test

High Duty DT

- Improvement of inductive and non-inducvtive operation
- Demonstration of high duty operation
- Blanket test

Equivalent Number of Burn Pulses (500 MW x 440 s*)

1 | 750 | 1000 | 1500 | 2500 | 3000 | 3000

Fluence**

0.006 MWa/m2 | 0.09 MWa/m2

Blanket Test

System Checkout and Characterization

- Electro-magnetic test
- Hydraulic test
- Effect of ferritic steel etc.

- Neutronics test
- Validate breeding performance

Performance Test

- Short-time test of T breeding
- Thermomecanics test
- Preliminary high grade heat generation test, etc.

- On-line tritium recovery
- High grade heat generation
- Possible electricity generation, etc.

*** The burn time of 440 s includes 400 s flat top plus 40 s of full power neutron flux to allow for contributions during ramp-up and ramp-down**

**** Average fluence at first wall (neutron wall load is 0.56 MW/m^2 on average and 0.77 MW/m^2 at outboard equator)**

ITER Diagnostic Systems and Planned Status at the Startup of the H Phase and of the DT Phase

Diagnostic	Status at Startup (H phase)	Status at Start of DT Phase
Magnetic Diagnostics		
Vessel Wall Sensors, Divertor Magnetics, Continuous Rogowski Coils, Diamagnetic Loop	Complete	
Neutron Diagnostics		
Radial Neutron Camera, Vertical Neutron Camera	Interfaces complete	Complete
Micro-fission Chambers (In-Vessel) (N/C)	In-vessel components and interfaces complete	Complete
Neutron Flux Monitors (Ex-Vessel)	Interfaces complete	Complete
Gamma-Ray Spectrometer		Complete
Activation System (In-Vessel), Lost Alpha Detectors	In-vessel components and interfaces complete	Complete
Knock-on Tail Neutron Spectrometer (N/C)		Complete
Optical/IR(Infra-Red) Systems		
Core Thomson Scattering	Complete except for two lasers and one power supply system	Complete

Diagnostic	**Status at Startup (H phase)**	**Start of DT Phase**
Edge Thomson Scattering , X-Point Thomson Scattering	Complete except for some spares	Complete
Divertor Thomson Scattering	Penetrations, in-vessel optics and interfaces complete	Complete
Toroidal Interferometer/ Polarimeter, Polarimeter (Poloidal Field Measurement)	Complete	
Collective Scattering System (N/C)	Penetrations, in-vessel optics and interfaces complete	Complete
Bolometric Systems		
Arrays for Main Plasma, Arrays for Divertor	Complete	
Spectroscopic and Neutral Particle Analyzer Systems		
H Alpha Spectroscopy, Visible Continuum Array	Complete	
Main Plasma and Divertor Impurity Monitors, X-Ray Crystal Spectrometers,	Penetrations, in-vessel optics and interfaces complete. Partial operation	Complete
Charge eXchange Recombination Spectroscopy (CXRS) based on DNB, Motional Stark Effect (MSE) based on heating beam, Soft X-Ray Array, Neutral Particle Analyzers (NPA), Laser Induced Fluorescence (N/C)	Penetrations, in-vessel optics/sensors and interfaces complete	Complete

Diagnostic	**Status at Startup (H phase)**	**Start of DT Phase**
Microwave Diagnostics		
Electron Cyclotron Emission (ECE)	Complete except for one spectrometer	Complete
Main Plasma Reflectometer	One LFS (low field side) X-mode and one LFS O-mode complete	Complete
Plasma Position Reflectometer, Divertor Reflectometer, Divertor EC absorption (ECA), Main Plasma Microwave Scattering, Fast Wave Reflectometry (N/C)	In-vessel components, interfaces	Complete
Plasma-Facing Components and Operational Diagnostics		
IR/Visible Cameras, Thermocouples, Pressure Gauges, Residual Gas Analyzers, IR Thermography (Divertor), Langmuir Probes	Complete	
Diagnostic Neutral Beam		
Diagnostic Neutral Beam (DNB)	Interfaces and main source components complete	Complete

Diagnostic Design

Diagnostic design has to take into account the specific measurement requirements, the environment for the diagnostic components and the available access. In addition, diagnostic designs must satisfy the ITER general design requirements:

- **vacuum compatibility and integrity**
- **containment of tritium**
- **neutron shielding**
- **maintainability with remote handling tools**
- **reliability**
- **lifetime**
- **etc**

Key Environmental and Radiation Effects

- Radiation-induced conductivity (RIC)
- Radiation induced electrical degradation (RIED)
- Radiation-induced electromotive force (RIEMF)
- Erosion
- Deposition
- Radiation induced absorption
- Radioluminescence
- Nuclear heating
- Dimensional changes (swelling)

ASSESSMENT OF MEASUREMENT CAPABILITY

Plasma Current, Plasma Position and Shape, Loop Voltage, Plasma Energy, Locked Modes

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION		ACCURACY
				Time or Freq.	Spatial or Wave No.	
1. Plasma Current	I_p	Default	0 – 1 MA	1 ms	Integral	10 kA
			1 – 17.5 MA	1 ms	Integral	1 %
		I_p Quench	20 – 0 MA	0.1 ms	Integral	30 % + 10 kA
2. Plasma Position and Shape	Main plasma gaps, Δ_{sep}	I_p > 2 MA, full bore	-	10 ms	-	1 cm
		I_p Quench	-	10 ms	-	2 cm
	Divertor channel location (r dir.)	Default	-	10 ms	-	1 cm
		I_p Quench	-	10 ms	-	2 cm
	dZ/dt of current centroid	Default	0 – 5 m/s	1 ms	-	0.05 m/s (noise) + TBD % (absolute)
3. Loop Voltage	V_{loop}	Default	0 – 30 V	1 ms	4 locations	5 mV
		I_p Quench	0 – 500 V	1 ms	4 locations	10 % + 5 mV
4. Plasma Energy	β_p	Default	0.01 – 3	1 ms	Integral	5 % at β_p=1
		I_p Quench	0.01 – 3	1 ms	Integral	~ 30%
8. Locked Modes	Br(mode)/Bp		10^{-4} – 10^{-2}	1 ms	(m,n) = (2,1)	30 %
9. Low (m,n) MHD Modes, Sawteeth, Disruption Precursors	Mode complex amplitude at wall		TBD	DC – 3 kHz	(0,0) < (m,n) < (10,2)	10 %
	Mode – induced temperature fluctuation		TBD	DC – 3 kHz	(0,0) < (m,n) < (10,2) Δr = a /30	10 %
	Other mode parameters TBD					

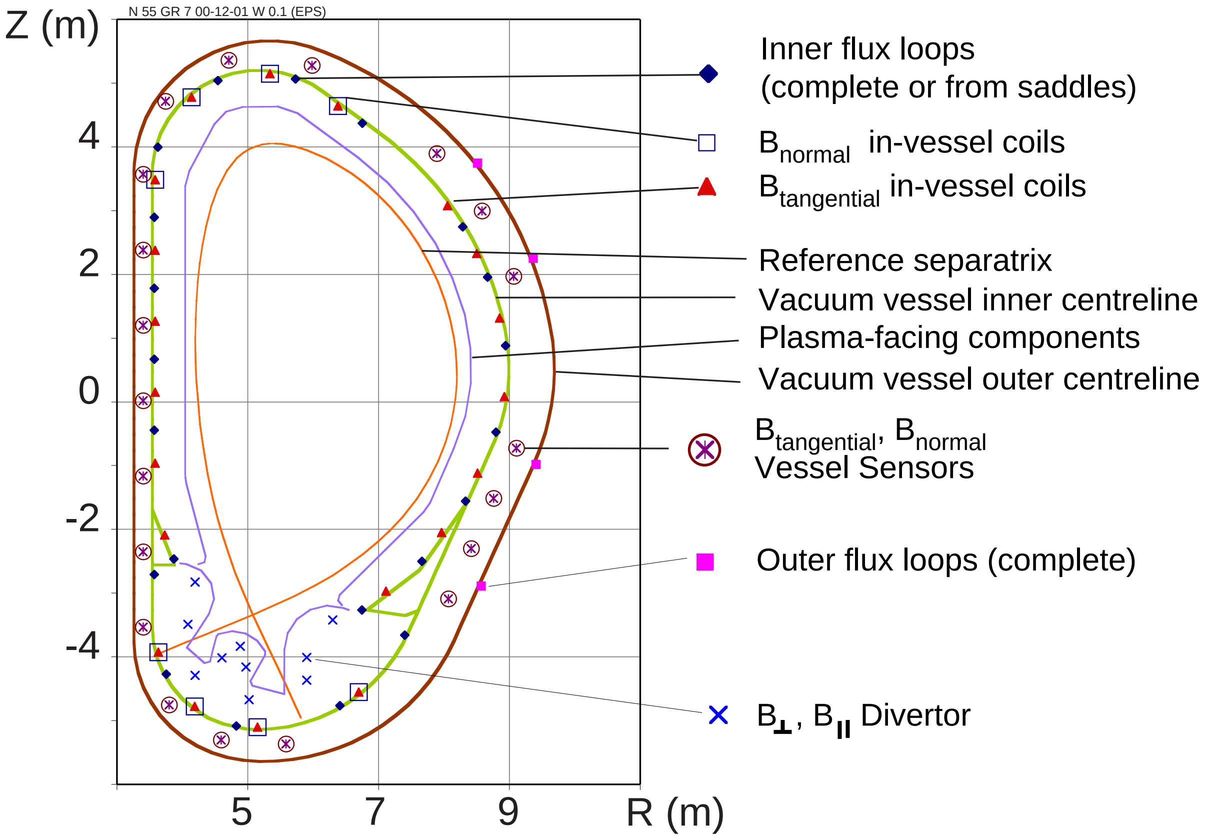
Principal diagnostic: Magnetics

Magnetics

The system is divided partly by location and partly by function into 6 sub-systems as follows:

- **In-Vessel Sensors: Sensors mounted on the plasma side of the vacuum vessel**
- **Vessel Sensors: Sensors mounted outside the vacuum vessel, or in-between the two vacuum vessel skins**
- **Divertor Coils**
- **External Rogowski Coils**
- **Diamagnetic Loop**
- **Internal Rogowskis**

The required magnetic fluxes and fields are obtained after analogue integration with compensation for long term drift. Non-inductive methods and hybrid magnetic sensors are envisaged as a backup system for long pulses.

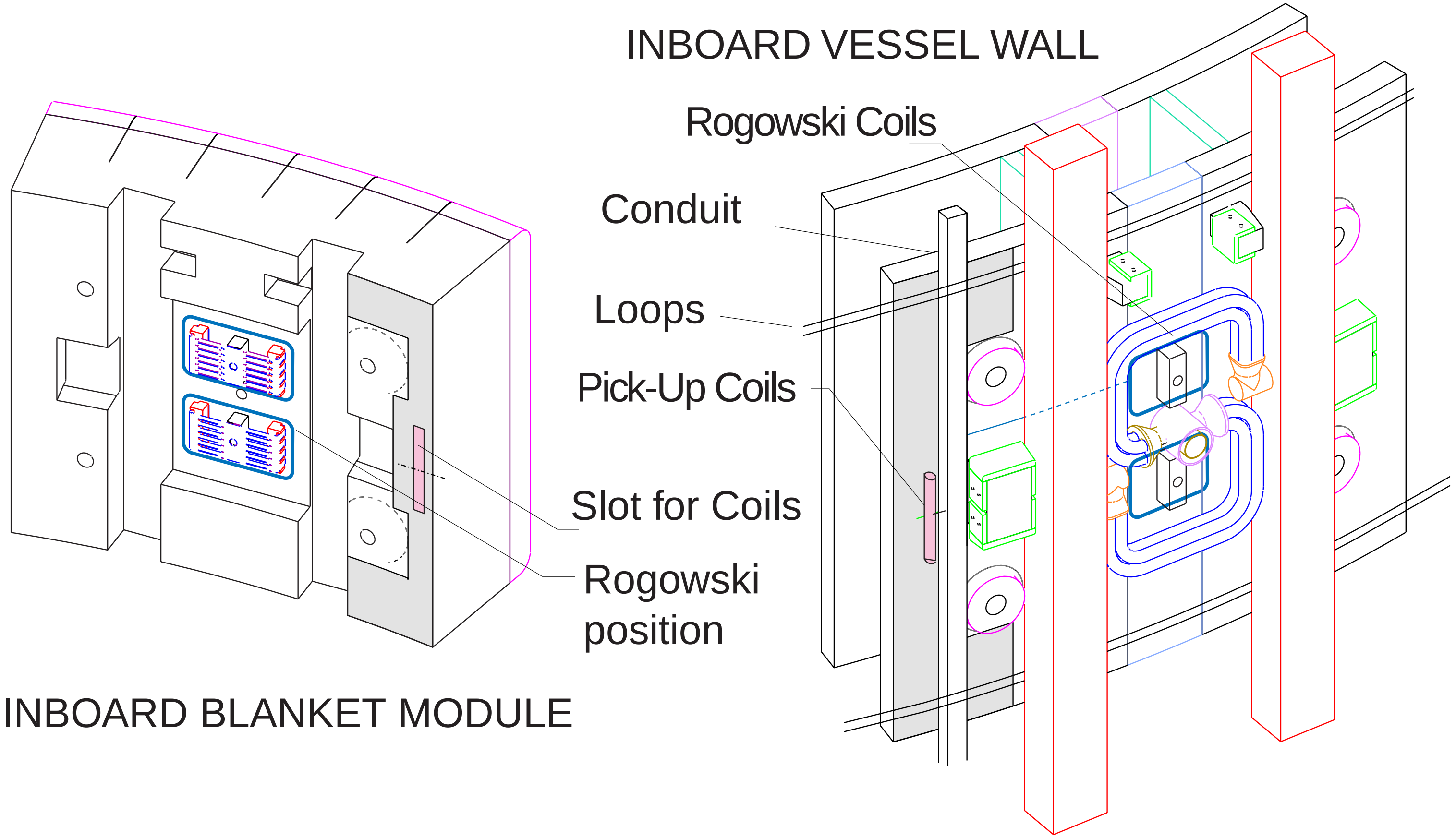


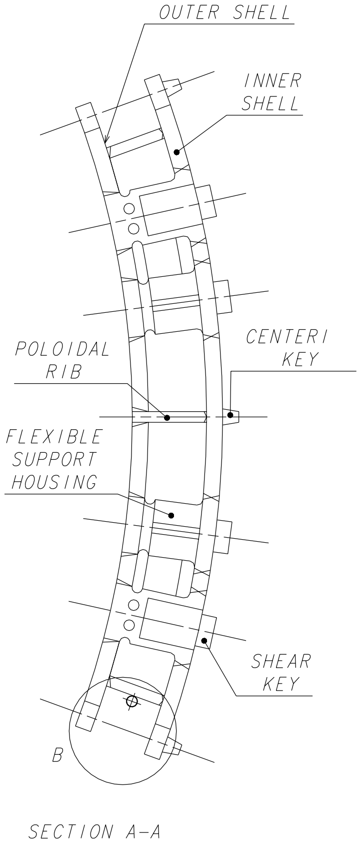
Poloidal Distribution of Magnetic Sensors.
The diamagnetic loops and external Rogowski coils are not shown.
[K Ebisawa et al, Rev. Sci. Instrum, vol. 72 No 1, 545, (2001)].

In-Vessel Sensors

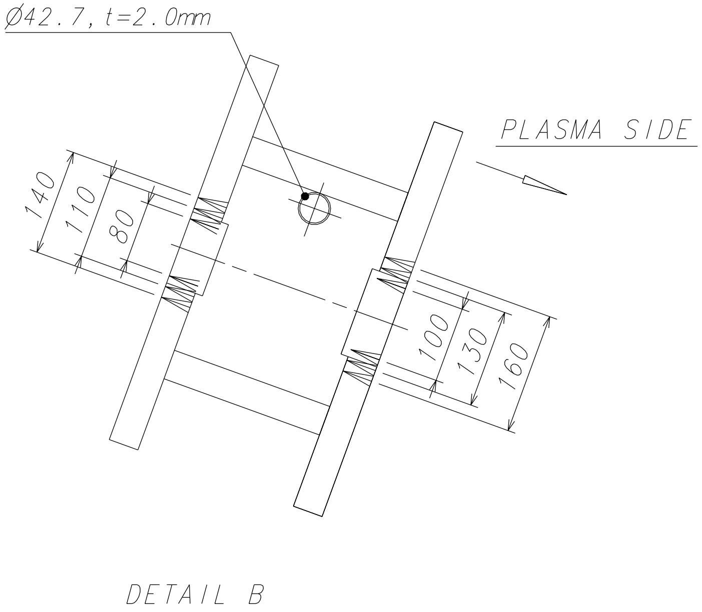
This system is comprised of

- Tangential, normal and toroidal equilibrium coils mounted on the inner surface of the vacuum vessel
- Tangential HF coils mounted on the inner surface of the vacuum vessel
- Complete and partial flux loops mounted on the inner surface of the vacuum vessel
- Dedicated saddle loops mounted on the inner surface of the vacuum vessel


INBOARD VESSEL WALL
Rogowski Coils
Conduit
Loops
Pick-Up Coils
Slot for Coils
Rogowski position
INBOARD BLANKET MODULE



Location of a magnetics tube at the field joint next to the poloidal rib.



Sketch showing the tube parameters and relation to the field joint.

Performance Analysis

Plasma Current

Specification for the plasma current measurement

MEASUREMENT	**PARAMETER**	**CONDITION**	**RANGE or COVERAGE**	**RESOLUTION**		**ACCURACY**
				Time or Freq.	**Spatial or Wave No.**	
1. Plasma Current	I_p	Default	0 – 1 MA	1 ms	Integral	10 kA
			1 – 17.5 MA	1 ms	Integral	1 %
		I_p Quench	20 – 0 MA	0.1 ms	Integral	30 % + 10 kA

Recent studies by JCT and the EU HT have considered main sources of errors:

- **Discretisation error on Ip**
- **Integrator drift**
- **Residual error due to angular misalignment**
- **Coil-to-coil error due to relative calibration**
- **Errors due to the eddy currents in the blanket modules**

Find the total expected "random" error is 0.3 % (± 2 s). This leaves a margin for errors due to radiation effects, contact effects and errors due to the miscalculation of the integration contour (which relies on knowing the exact position of the hot vessel). In addition to this random error there will be a permanent calibration error (estimated ~0.6 %).

In addition

- Backup measurement I (vessel set)

These are similar in number and location, and therefore will suffer similar errors, with the significant exception of radiation induced errors.

- Backup measurement II (External Rogowski)

A separate backup system for this measurement is formed by the external Rogowski. This system is still slower than the vessel system and is sensitive to the total vessel current. However, it does not suffer

from discretisation error since it forms a continuous loop, and it suffers least from the effects of integrator drift

Overall assessment

The measurement of the plasma current to the required specification appears feasible.

Plasma Position and Shape

Specification for the plasma current measurement:-

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION		ACCURACY
				Time or Freq.	Spatial or Wave No.	
2. Plasma Position and Shape	Main plasma gaps, Δ_{sep}	I_p > 2 MA, full bore	-	10 ms	-	1 cm
		I_p Quench	-	10 ms	-	2 cm
	Divertor channel location (r dir.)	Default	-	10 ms	-	1 cm
		I_p Quench	-	10 ms	-	2 cm
	dZ/dt of current centroid	Default	0 – 5 m/s	1 ms	-	0.05 m/s (noise) + TBD % (absolute)

Principal diagnostic: Magnetics

Recent studies by JCT and the EU HT have considered main sources of errors:

- **Reconstruction Accuracy**
- **Blanket Eddy Currents**
- **Plasma Noise**
- **Integrator Drift**
- **Integrator Board Noise and Cross-talk**
- **Parasitic Signals and Distortions**

In addition

- **Backup measurement I (vessel set)**

Reconstruction Accuracy

For the present machine the main plasma gaps (3-6) and divertor channel location gaps (1 and 2) used for control are shown.

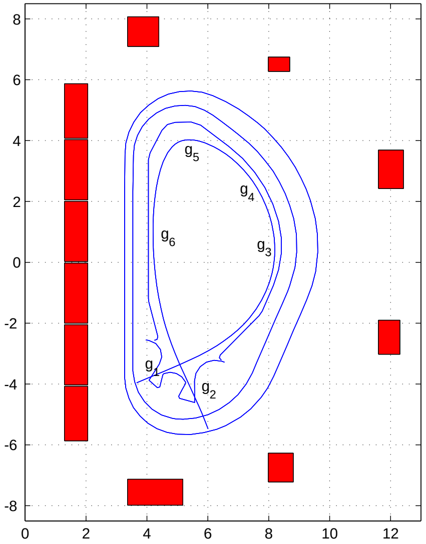


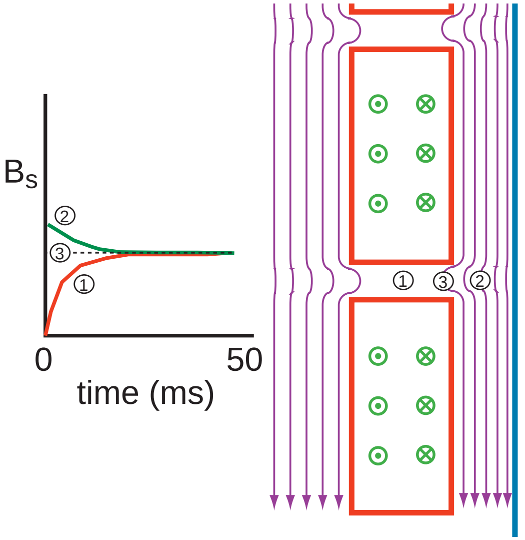
Figure: Location of the plasma control gaps g1-g6 in ITER

Static (neglecting eddies) analysis has shown that the typical reconstruction error of the gaps depends on the injected noise.

For an average gap error (± 2σ) of 1 cm or less, the measurement noise level has to be ± 0.7 % (± 2σ) on average. Therefore, a reasonable target for the measurement error on individual coils or loops of less than 0.7 % (± 2σ).

Blanket Eddy Currents: Effect on Measurement of Fast Vertical Position

- **Initial concern because of the possible instability of the fast vertical position control loop in the presence of blanket eddy currents**
- **JCT and EU HT studies showed that control is affected for delays of 5 ms or more, and unstable at 20 ms**
- **Typical blanket eddy modes are 3 ms and 20 ms (tangential and horizontal axis of magnetic moment)**
- **Paper calculations showed that, at the coil location, a lead rather than a lag is expected**



Right:Toroidal view of the blanket modules shortly after a change in tangential flux (Bs). Left: Time evolution of the tangential field.

Radiation Induced Electromagnetic Force (RIEMF)

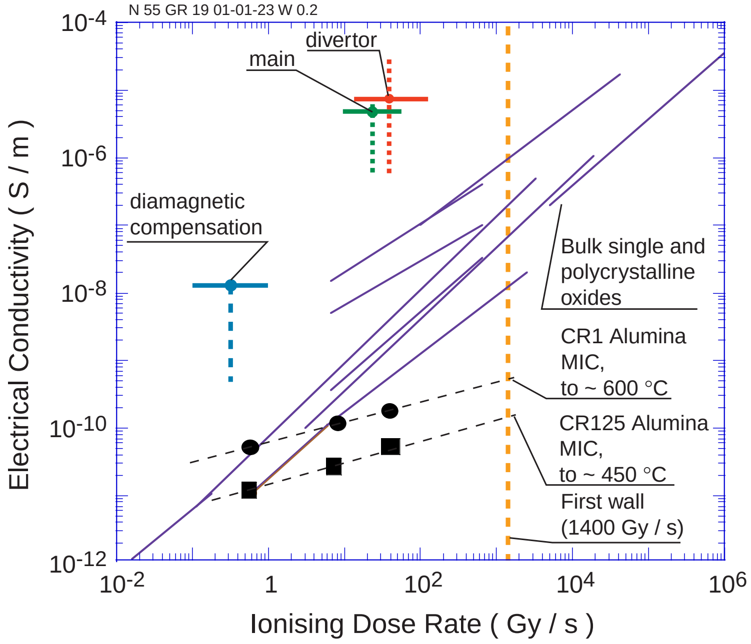
RIEMF is a phenomenon which has been observed to occur in experiments in which mineral insulated (MI) cable has been irradiated in test reactors. A substantial data base has been established and theoretical models of the phenomenon developed.

For the expected values of RIEMF voltage and current, for typical coil resistance, the effects on differential signals are expected to be low enough to be negligible. This is because the available current is typically below 1 µA, at high impedance and the resistance of order 1 Ohm. So even for grossly asymmetric situations, voltages less than 1 µV are expected; for coils that are symmetric at the 10% level, voltages below 100 nV are expected, and for carefully constructed coils placed in a symmetric environment the effect should be negligible.

However in a limited serious of experiments with prototype coils voltages up to ~ 50 µV were either observed or implied from the integrator drift. But in the tests there were potentially other sources of voltage, eg thermoelectroc effects, RF pick-up, and so the tests need to be repeated and more detailed measurements made.

Radiation Induced Conductivity (RIC)

Significant errors in the measurement could appear from the presence of additional loads on the coil, the main such effect being radiation-induced conductivity (RIC).



Measured Radiation-Induced Conductivity in bulk single and polycrystalline oxides, and MI Cables as a function of Ionising Dose Rate combined with magnetic diagnostic requirements at selected locations. The vertical bars represent the range of design values of RIC that can be tolerated for each coil; the horizontal bars represent the uncertainty on the flux

Overall assessment

The measurement of the plasma shape and position to the required specification appears feasible but there maybe a pulse length limitation due to possible parasitic signal (RIEMF).

Similarly we expect to meet the target specifications for the other parameters:

- **Loop Voltage,**
- **Plasma Energy,**
- **Locked Modes,**
- **'Halo' currents,**
- **Toroidal magnetic field,**
- **Low m/n MHD activity**

Critical Areas/Outstanding Tasks

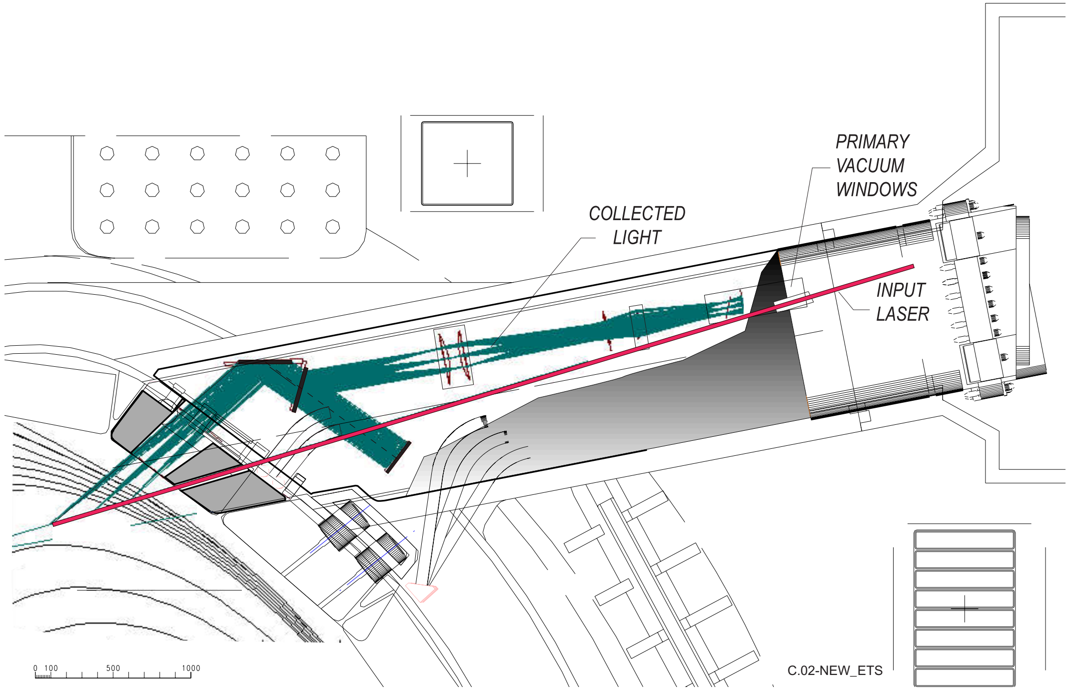
- **Complete design, in particular detail many interfaces**
- **Complete development of steady state magnetic sensor**
- **Resolve RIEMF issues**
- **Complete performance analysis**

Electron Temperature

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION		ACCURACY
23. Electron Temperature Profile	Core T_e	$r/a < 0.9$	0.5 – 40 keV	10 ms	a/30	10 %
	Edge T_e	$r/a > 0.9$	0.05 – 10 keV	10 ms	0.5 cm	10 %
41. Divertor Electron Parameters	n_e		$10^{19} – 10^{22}$ /m^3	1 ms	5 cm along leg, 3 mm across leg	20 %
	T_e		0.3 –200 eV	1 ms	5 cm along leg, 3 mm across leg	20 %

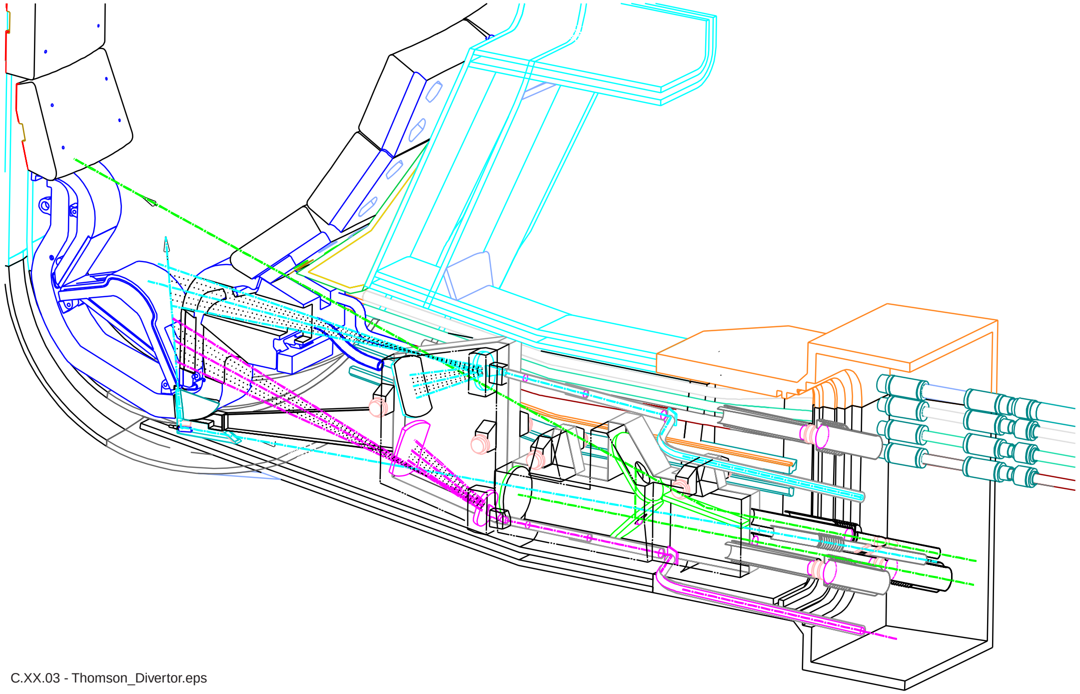
Principal Diagnostics: Thomson scattering (Core, Edge, X-pt and Divertor Systems), ECE

Edge Thomson Scattering System



The front end of the edge Thomson scattering system showing the laser line (orange) to the last element supported from the diagnostic block, and collection optics / raytrace (green) to the first vacuum boundary (courtesy P Nielsen, Consorzio RFX, Padova).

X –Point and Divertor Systems



C.XX.03 - Thomson_Divertor.eps

X-point LIDAR and Divertor Leg conventional Th Sc system installed in a port at the divertor level (courtesy G Razdobarin. Ioffe Institute, St Petersburg)

Main threats to system performance and main sources of error

Thomson scattering

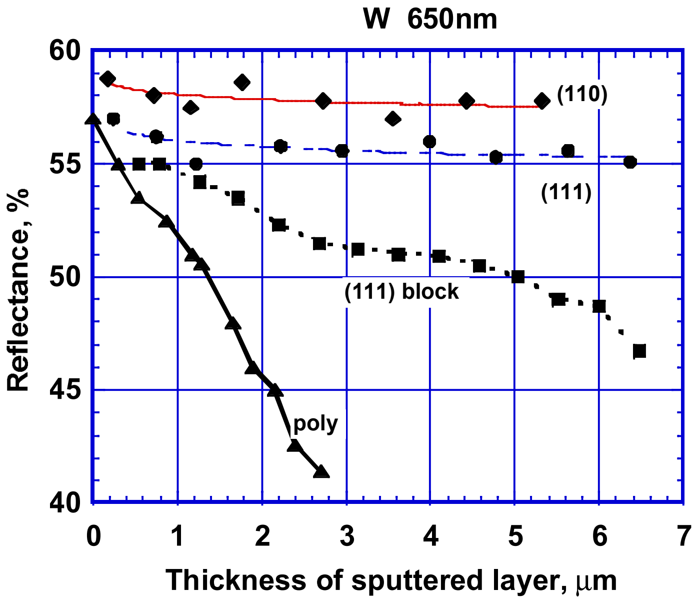
- Lifetime of First Mirrors (erosion, deposition, laser damage)
- Alignment and Calibration
- Radiation induced absorption and windows
- Radiation induced absorption in fibres

ECE

- Relativistic and doppler broadening
- Distortions of the velocity distribution
- Calibration

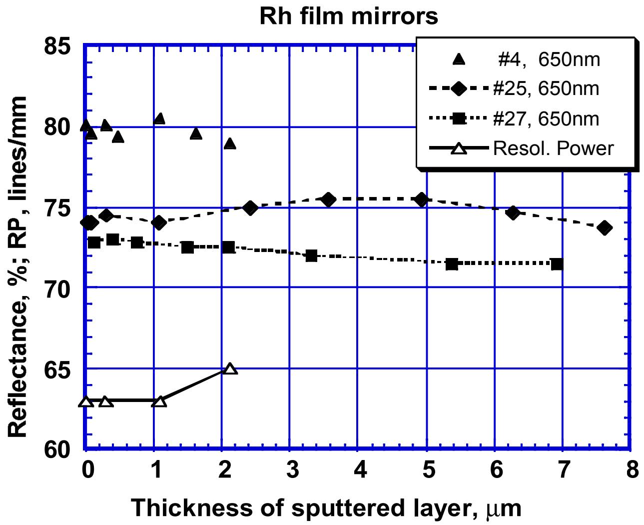
Mirrors and reflectors.

Tests have shown single crystal mirrors (Mo, W) have a high mirror quality even after erosion by sputtering of a layer several μms thick.



Reflectance of W mirrors (polycrystal, block monocrystal and real monocrystals with two planes of orientation) at λ = 650 nm depending on the sputtered layer thickness. [V S Voitsenya et al, Rev. Sci. Instrum, vol. 72 No 1, 475, (2001)]

Suitably chosen metal film mirrors mounted on a metal substrate can have a good resistance to the CXA flux. For example, Rh film mirrors of thickness ~10 μm mounted on Cu can be used in locations where the CXA flux onto the mirror surface will not exceed 2x10^{18}atom/m^{2}s (~1/10 of the CXA flux to the first wall)



Dependences on sputtered layer thickness of reflectance at λ = 650 nm and resolving power versus thickness of sputtered layer for Rh film on copper substrate mirrors. [V S Voitsenya et al, Rev. Sci. Instrum, vol. 72 No 1, 475, (2001)]

Overall assessment

The measurement of the Electron Temperature appears feasible at the specified level in the Core and Edge regions but in the divertor region the specified requirements will probably not be met.

Critical Areas/Outstanding Tasks

Thomson scattering

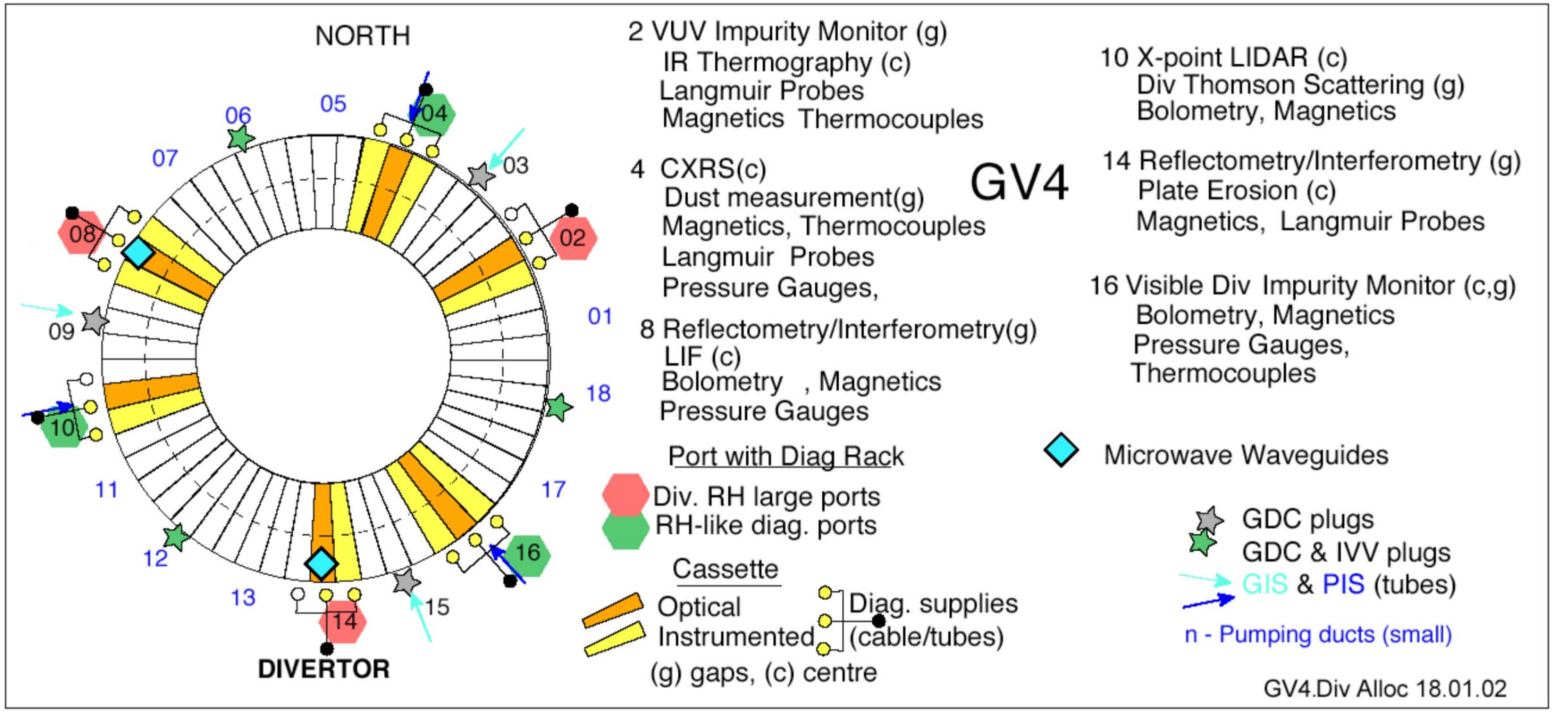
- Lifetime of mirrors due to many pules of high power laser radiation
- Deposition on first mirrors
- Radiation induced absorption in windows transmitting the high power laser radiation

ECE

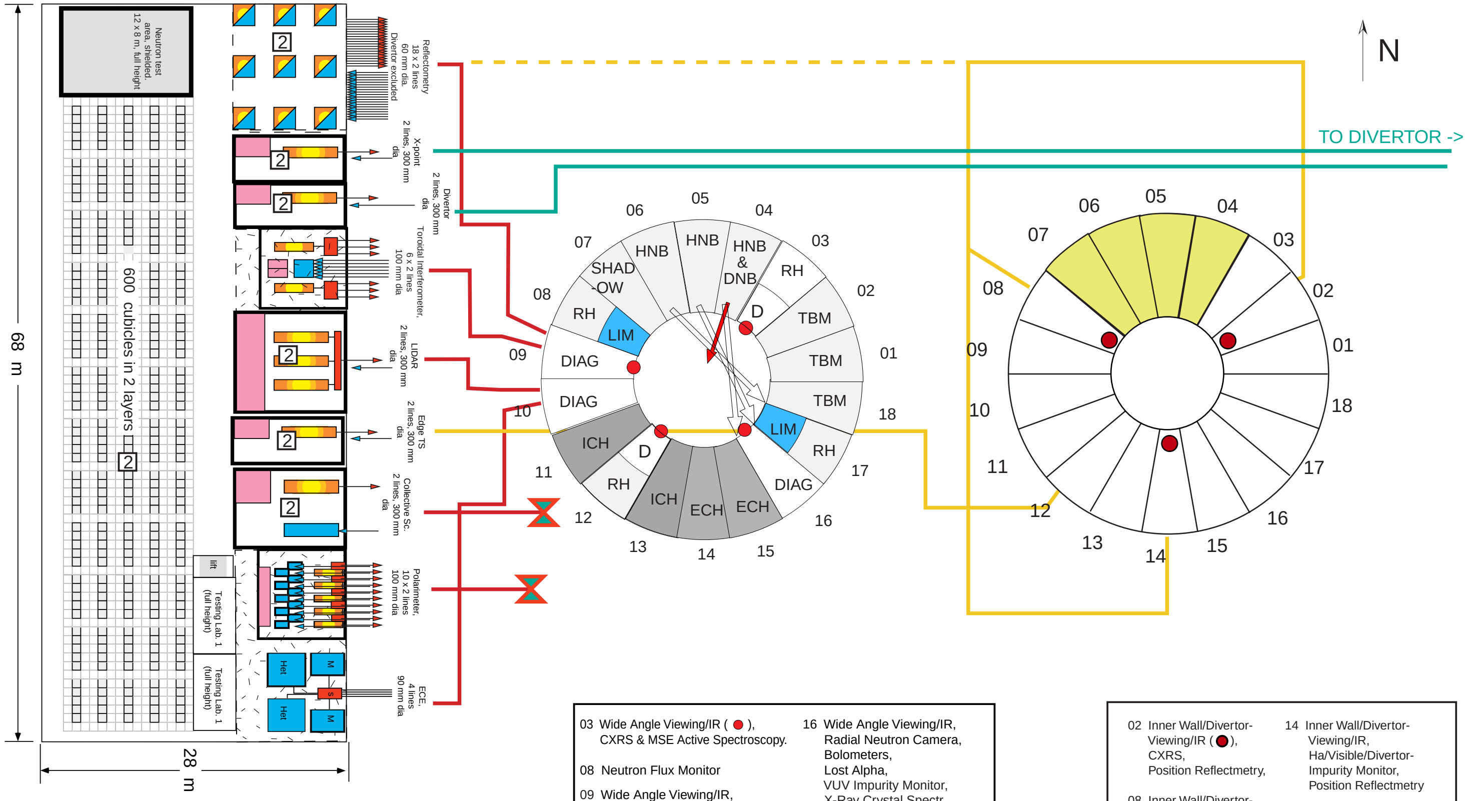
- Calibration

The integration of diagnostics with other tokamak systems is also important.

DIAGNOSTIC DISTRIBUTION IN DIVERTOR



Note the number of systems that have to be accommodated in addition to diagnostics (RH, pumping, GDC, IVV, etc)



Transmitter

Receiver

2 System on 2 levels

Combiner/ Separator

Power supply

Spectrometers on upper level

Unassigned system

03 Wide Angle Viewing/IR (●), CXRS & MSE Active Spectroscopy.

08 Neutron Flux Monitor

09 Wide Angle Viewing/IR, CXRS Active Spectroscopy, Toroidal Interf/Polarimeter, Visible Continuum Array, Reflectometry.

10 LIDAR Thomson Scat., Wide Angle Viewing/IR, Hard X-ray Monitors, ECE.

12 Wide Angle Viewing/IR H-Alpha Spectroscopy.

16 Wide Angle Viewing/IR, Radial Neutron Camera, Bolometers, Lost Alpha, VUV Impurity Monitor, X-Ray Crystal Spectr, Neutral Particle Analysers.

17 Neutron Flux Monitor, Neutron Activation.

Unassigned:
Visible/UV Impurity Monitor,
Collective Scattering,
Two Photon Ly-Alpha Fluor,
Laser Induced Fluor,
Soft X-Ray Array,
Polarimetry.

ODR_EQPORT_V12

02 Inner Wall/Divertor-Viewing/IR (●), CXRS, Position Reflectmetry,

08 Inner Wall/Divertor-Viewing/IR, Ha/Visible/Inner Edge-Impurity Monitor, Position Reflectmetry,

10 Ha/Visible/Outer Edge-Impurity Monitor, Ha/Visible/Upper Edge-Impurity Monitor, Bolometry,

12 Edge Thomson Scat.

14 Inner Wall/Divertor-Viewing/IR, Ha/Visible/Divertor-Impurity Monitor, Position Reflectmetry

15 Neutron Camera (Vertical View)

16 ECCD?

17 Neutron Camera (Vertical View)

18 VUV, X-Ray Spectroscopy

Obscured

Layout to diagnostic Hall (note port 11 is now a diagnostic port and houses the direct coupled systems).

ASSESSMENT OF MEASUREMENT CAPABILITY

GROUP 1a Measurements For Machine Protection and Basic Control	GROUP 1b Measurements for Advanced Control	GROUP 2 Additional Measurements for Performance Eval. and Physics
Plasma shape and position, separatrix- wall gaps, gap between separatrixes Plasma current, q(a), q(95%) Loop voltage Fusion power $\beta_N = \beta_{tor}(aB/I)$ Line-averaged electron density Impurity and D,T influx (divertor, & main plasma) Surface temp. (div. & upper plates) Surface temperature (first wall) Runaway electrons 'Halo' currents Radiated power (main pla, X-pt & div). Divertor detachment indicator (J_{sat}, n_e, T_e at divertor plate) Disruption precursors (locked modes,m=2) H/L mode indicator Z_{eff} (line-averaged) n_T/n_D in plasma core ELMs Gas pressure (divertor & duct) Gas composition (divertor & duct) Dust	Neutron and α-source profile Helium density profile (core) Plasma rot. (tor and pol) Current density profile (q-profile) Electron temperature profile (core) Electron den profile (core and edge) Ion temperature profile (core) Radiation power profile (core, X-point & divertor) Z_{eff} profile Helium density (divertor) Heat deposition profile (divertor) Ionization front position in divertor Impurity density profiles Neutral density between plasma and first wall n_e of divertor plasma T_e of divertor plasma Alpha-particle loss Low m/n MHD activity Sawteeth Net erosion (divertor plate) Neutron fluence	Confined α-particles TAE Modes, fishbones T_e profile (edge) n_e, T_e profiles (X-point) T_i in divertor Plasma flow (divertor) $n_T/n_D/n_H$ (edge) $n_T/n_D/n_H$ (divertor) T_e fluctuations n_e fluctuations Radial electric field and field fluctuations Edge turbulence MHD activity in plasma core

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

ASSESSMENT OF MEASUREMENT CAPABILITY (Contd)
(Control and Evaluation not Physics studies)

Operating Scenario	Required Measurements	
	Control	Evaluation
H phase. Inductive. Ohmic L-mode. Limited H-Mode	Plasma shape and position, vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall	q(a), halo current, impurity identification and influx, $n_e(r)$ and $T_e(r)$ in core, T_i in core, P_{rad} from core, line-averaged Z_{eff}, H/L mode indicator, gas pressure and composition (divertor and duct), ELM occurrence and type
D phase. Inductive. ELMy H-mode	Plasma shape and position, vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment	q(a), halo current, impurity identification and influx, $n_e(r)$ and $T_e(r)$ in core, T_i in core, P_{rad} from core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), shape and position (500 s), β, q(95%), $n_e(r)$ and $T_e(r)$ at edge, P_{fus}, $P_{rad}(r)$, heat deposition profile in divertor

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

ASSESSMENT OF MEASUREMENT CAPABILITY (Contd)
(Control and Evaluation not Physics studies)

Operating Scenario	Required Measurements	
	Control	Evaluation
High power D/T phase. Inductive. ELMY H Mode	Plasma shape and position (500 s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, $T_e(r)$ in core, T_I in core, P_{rad} from core, P_{fus}, $n_{He}(r)$, n_{He} in divertor, n_T/n_D in core, div ionisation front position, $v_{tor}(r)$ and $v_{pol}(r)$	q(a), halo current, impurity identification and influx, $n_e(r)$ in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), β, q(95%), $n_e(r)$ and $T_e(r)$ at edge, $P_{rad}(r)$, heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, $T_i(r)$ in core, $Z_{eff}(r)$, D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

ASSESSMENT OF MEASUREMENT CAPABILITY (Contd)
(Control and Evaluation not Physics studies)

Operating Scenario	Required Measurements	
	Control	Evaluation
D/T Phase. Inductive ELMY H mode. High β	Plasma shape and position (500 s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, $T_e(r)$ in core, T_I in core, P_{rad} from core, P_{fus} $n_{He}(r)$, n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, $v_{tor}(r)$ and $v_{pol}(r)$, β, localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs.	q(a), halo current, impurity identification and influx, $n_e(r)$ in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), $n_e(r)$ and $T_e(r)$ at edge, $P_{rad}(r)$, heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, $T_i(r)$ in core, $Z_{eff}(r)$, D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

ASSESSMENT OF MEASUREMENT CAPABILITY (Contd)
(Control and Evaluation not Physics studies)

Operating Scenario	**Required Measurements**	
	Control	**Evaluation**
Hybrid operation	Plasma shape and position (1000s), vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, $T_e(r)$ in core, T_I in core, P_{rad} from core, P_{fus} $n_{He}(r)$, n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, $v_{tor}(r)$ and $v_{pol}(r)$, β,localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs.	q(a), halo current, impurity identification and influx, $n_e(r)$ in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), $n_e(r)$ and $T_e(r)$ at edge, $P_{rad}(r)$, heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, $T_i(r)$ in core, $Z_{eff}(r)$, D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

ASSESSMENT OF MEASUREMENT CAPABILITY (Contd)
(Control and Evaluation not Physics studies)

Operating Scenario	Required Measurements	
	Control	Evaluation
Steady state operation	Plasma shape and position for 1000s, vertical speed, B_{tor}, I_p, V_{loop}, locked modes, m = 2 modes, low m/n MHD modes, line-averaged density, runaway electrons, surface temperature of divertor plates and first wall, H/L mode indicator, ELM occurrence and type, divertor detachment, $T_e(r)$ in core, T_I in core, P_{rad} from core, P_{fus} $n_{He}(r)$, n_{He} in divertor, n_T/n_D in core, divertor ionisation front position, $v_{tor}(r)$ and $v_{pol}(r)$, β,localisation of q = 1.5 and q = 2 surfaces, high sensitivity measurements of n_e and T_e, detection and measurement of NTMs. $T_i(r)$ in core, q(r) (in particular localisation and position of q_{min}), high resolution measurements of the gradient of T_e and T_i, measurement of RWMs	q(a), halo current, impurity identification and influx, $n_e(r)$ in core, line-averaged Z_{eff}, gas pressure and composition (divertor and duct), q(95%), $n_e(r)$ and $T_e(r)$ at edge, $P_{rad}(r)$, heat deposition profile in divertor, neutron and alpha source profiles, impurity profile, $Z_{eff}(r)$, D and T influx, neutral density (near wall), n_e and T_e in divertor, impurity and DT influxes in divertor with spatial resolution, alpha loss, neutron fluence, erosion of divertor tiles

Expect to meet meas. reqs; maybe/mabe not; expect not to meet meas reqs.

NECESSARY NEXT STEPS

On order to proceed we need

- Further design of individual diagnostic systems

- More R&D

  - Radiation and environmental effects on materials used in diagnostic construction (cables, fibres, windows etc)

  - Radiation and enviromental tests on prototype components (magnetic coils, bolometers, etc)

  - Development of new components (eg steady state magnetic sensors, radiation hard soft x-ray detectors)

  - **Key tests and developments of specific techniques (eg refelcometry from the hfs on the lower cut-off)**

  - **Development of new techniques (eg Fast Wave Reflectometry, techniques for measuring erosion)**

- **Integration of the diagnostic systems and components into the tokamak and buildings**

CONCLUDING REMARKS

The requirements for plasma and first wall measurements must be derived from a careful consideration of the programmatic objectives

Detailed specification is necessary to guide the diagnostic selection and act as a target for the design

The design has to be a careful interplay between the needs of the diagnostic and the generic engineering requirements and must be closely connected with the validating R&D

The assessment of performance has to be related back to the consequences of the measurement capability for the operating scenarios and ultimately the programmatic objectives

For ITER, even though there are considerable uncertainties and much development needs to be done, we believe that the measurements necessary to support the basic programmatic objectives can be made at the specified level

More design and R&D is needed before the final measurement capability can be determined. The design needs to be done at both the system level and for the integration of the systems together and with other tokamak systems.

The R&D is needed on a range of topics from the impact of the environment on components to development to the development of new diagnostic techniques. The design and the R&D work must be closely connected.

The development of new techniques, specifically BPX relevant, would also be beneficial and shoule be encouraged.