Project Integration Document
PID

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ITER Organisation

Editor: J.How

	Name	Affiliation
Author/Editor	J.How	Project Office
Checked by **Site heads**	D.Bora V.Chuyanov G.Johnson Y-H Kim C.Alejaldre E.Tada	ITER Senior Management
Approved by	N.Holtkamp K.Ikeda	Principle Deputy Director General (PDDG) Director General

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0 Introduction

0.1 Purpose

The purpose of the PID, together with all its annexes, is to define the technical basis of the ITER Project. Following on from the Plant Design Specification (PDS), which describes the externally imposed, essentially design-independent requirements, the Project Integration Document is the highest level ITER technical document (the Management and QA documentation is based on the ITER Management and Quality Program documents). It originates primarily from the Design Requirements and Guidelines-1 (DRG-1) and the Plant Description Document (PDD) as originally prepared for the Final Design Report of 2001, and also now includes the PSR (Plant Safety Requirements).

The ITER Project *Management System (defined in the ITER Management and Quality Program)* assures:

- Definition of all documentation required for ITER Project systems design, fabrication, construction, installation, test, and performance.
- Correct and complete descriptions of the approved configuration; including specifications, drawings, parts lists, test procedures, and operating manuals.
- Traceability of the resultant product and its parts to their descriptions.
- Accurate and complete identification of each material, part, subassembly, and assembly that goes into the machine and all its ancillary systems.
- Systematic evaluation of proposed changes to the approved configuration and control of implementation of these changes.
- Accurate and complete accounting of all design changes to ITER Plant, as well as the "As-Built" configuration.

0.2 Document Organization

- Section 1 covers the main manuals, standards and design criteria used by the project. Above all, it references the appropriate annexes for the correct level of detail of each subject.
- Section 2 covers the definition of the overall machine configuration, including basic parameters, design configuration tables and operation states.
- Section 3 covers the definition of the general requirements, parameters and interfaces which can be grouped by broad subjects rather than by systems. These include broad safety requirements

(incorporating the Plant Safety Requirements (PSR) annex to the DRG1), plasma scenarios, magnetic fields, allocation of ports, heat loads, mechanical loads, radiation, and machine displacements.

- Section 4 covers, system by system, the main configuration of that system and many of the applicable interfaces.

- Section 5 shows the interaction matrix between design parameters (in tabular form) and the WBS elements. In this fashion it will be easy to immediately identify what other systems are affected by potential changes to these parameters, and the responsible design officers.

In several instances this document refers to its annexes which deal with a particular subject in more detail.

Whenever possible this document will contain numeric values of parameters in tabular format.

In some instances a parameter may be specified for more than one condition:

H value indicates the specification for nominal operation (unless otherwise stated) with hydrogen plasmas in the initial operation/commissioning phase;

DT value, indicates the nominal design value (unless otherwise stated) foreseen for operation during the DT Phase.

TBA value, "To Be Assessed" This is not a design requirement, but a value whose consequences on the design need to be assessed by the designers in view of possible machine operation with or without upgrades – (upgrades that are already foreseen are specified elsewhere as design requirements).

0.3 Parent Document

This document is the "root" of the *internal* ITER documentation structure. The parent document of the PID is the ITER **Plant Design Specification** (PDS).

0.4 Revisions to this document

This document can be modified, with the required management approvals, in response to design decisions through the processes defined in the Project Management Plan annex to this document.
The Revisions to this document are:

Version 1 was released in Sept 2004. It was constructed from the former DRG-1, plus the PDD and PSR (Plant Safety Requirements).
It can be found on
https://users.iter.org/users/idm/get_document?document_id=ITER_D_2234RH&version=v1.0
The source code is available on
https://users.iter.org/users/idm/folderTree/Current_Top_Level_Documents/PID/users/idm_data/BTree/ITER_D_2234RH/versions/v1_0/secondary_binaries/Project_Integration_Document.doc

In addition, there is a secondary binary with the complete document changes recorded:
https://users.iter.org/users/idm/folderTree/Current_Top_Level_Documents/PID/users/idm_data/BTree/ITER_D_2234RH/versions/v1_0/secondary_binaries/PID_Draft_to_Version_1.doc

Version 2 – Sept 2005 represents a major upgrade from most sections of the project.
NOTE HOWEVER some sections have not yet been updated :

- Safety – A full review will take place in Sept 2005
- 4.12 Thermal Shields (late 2005)
- 4.17 Control, alarm and interlocks (review in 2006)

The following IDM link gives all available PID versions
https://users.iter.org/users/idm?document_id=ITER_D_2234RH

Note that in the "secondary binaries" section there is both the source code (ms-word) and the "draft" version of PID-V2.0 including all tracked changes from version 1.

Table 0.4-1 Revision to this document

Changes Between Version 2 and Version 3

Date/Revision	Author	Note
Sept 2005	ITER Team	Release V2.0
Oct 2005	J.How	Editorial corrections
21 Nov 2005	Sugihara/Shimada	editorial error in PID "*3.7Fast transients – Heat Loads*" Tables of VDE are missing and Table 3.7-1 should be Table 3.7-2 Heat Load Conditions during Major Disruptions"
Nov 2005	Walker	Many "comments' from CWR included in magenta for later resolution.
July 2006	J.Mann	Update abbrviations table
Aug 2006	Shimada	Table 3.6-2 Static heat Loads
Aug 2006	Maruyama	Fuelling Table 4.5-4,
Aug 2006	Wykes	Table 3.10-2 Torus pump RH class
Aug 2006	Lister	Complete revision Section 4.17 CODAC and Interlock
Aug 2006	Ioki	Update VV Chapter 4.2
Aug 2006	Ioki	Update Blanket Chapter 4.3
Aug 2006	Ioki	Substantial update to Thermal Shields 4.12
Aug 2006	Kataoka	VVPSS Review 4.10
Aug 2006	Kataoka	Cooling Water 4.11 Reviewed
July/Aug	Saibene/Tanga et al	Complete revision of EC Chapter 4.19

Aug 2006	Wykes	Major revision update to Chapter 4.13 – Vacuum Pumping
Sept 2006	Mitchell	Revision of magnets 4.1, as well as suggestions for plasma, cryo and power supplies
Sept 2006	Gribov	Update of Plasma Control (RWM etc) section 3.2
05 Oct	Kalinin	Major update to 4:15 cryogenics. However still some points to be resolved.
5 Oct	A.Costley	Minor revision 4:22 Diagnostics, A.Costley
13 Oct	J-P Girard	Major Update of Safety Sections 3.1
13 Oct	W.Spears	Construction and Operation Schedule in Chapter 1
16 Oct	M.Glugla	Partial update for Tritium section 4.14 (fuller update in 2007)
27 Oct	Shaw	4.6 Assembly and Tooling – update
Dec	Tanga/Kobayashi	EC simplification rewrite
Jan	Girard/Tada	Major simplification Safety Sections 3.1
Jan	Tesini	Remote Handling
Jan	Kobayashi	More EC
Jan	Shaw	assembly

Version 3 – Oct/Jan 2006/2007 - is this current document.

The updates fall into four major catagories
1. Site adaptation specific to the Cadarache site (especially 3.1 safety)
2. new sections such as CODAC and Safety and Interlock
3. update of the sections that were missed in version 2.
4. Adjustments consistent with the detailed design development.

Nevertheless there are some sections which still require update. Notably the Site and Buildings which are in the process of detailed site adaptation; and likewise the services such as power supply and liquids and gases.

The following IDM link gives all available PID versions
https://users.iter.org/users/idm?document_id=ITER_D_2234RH

Note that in the "secondary binaries" section there is both the source code (ms-word) and the "draft" version of PID-V3.0 including all tracked changes from version 2 (to follow)

Abbreviations used in this document

AACT	Automated Air Cushion Transporter
AC	Alternating Current
ACB	Auxiliary Cold Box
ADS	Atmosphere Detritiation Systems
AH	Additional Heating
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BE	Beryllium
BLKT	Blanket
BM	Blanket Module
BOS	Baking Operation State
BPP	Basic Performance Phase (obsolete)
BSM	Blanket Shield Module
CAD	Computer Aided Design
CC	Correction Coils
CCB	Cryostat Cold Box
CCD	Charge-Coupled Detector
CCWS	Component Cooling Water System
CD	Current Drive
CFC	Carbon-Fibre Composite
CHWS	Chilled Water System
CMM	Cassette Multifunctional Mover
CODAC	COntrol, Data Access and Communication
COS	Common Operating State
CS	Central Solenoid
CSD	CSD Control System Design and Assessment (document)
CSVS	Cyrogenic Service vacuum System
CTB	Cold Termination Box
CTCB	Cryoplant Termination Cold Box
CTL	Circular Transmission Line
CTM	Cassette Toroidal Mover
CTS	Cryostat Thermal Shield
CVB	Cold Valve Box
CVCS	Chemical & Volume Control Systems
CW	Contiuous Wave
CWS	Cooling Water System
CXRS	Charge Exchange Recombination Spectroscopy
DAC	Digital to Analogue Convertor
DAC	Derived Air Concentration
DC	Direct Current
DC	Divertor Cassette
DCR	Design Change Request (form)
DCS	Discharge Control Subsystem
DD	Deuterium-Deuterium

DDD	Design Description Document
Dia	Diameter
Diag	Diagnostic
Disr.	Disruption
Div	Divertor
DMS	Disruption Mitigation System
DNB	Diagnostic Neutral Beam
DP	Divertor Port
DRG	Design Requirements & Guidelines
Drg	Drawing
DT	Deuterium Tritium
DV	Divertor
DW	Dead Weight
e-b	Electron beam (weld)
EC	Electron Cyclotron
ECCD	Electron Cyclotron Current Drive
EC-DC	Electron Cyclotron resonance Discharge Cleaning
ECE	Electron Cyclotron Emission
ECH	Electron Cyclotron Heating
ECRH	Electron Cyclotron Resonance Heating
EDS	Exhaust Detritiation System
EL	Elevation
ELM	Edge Localised Mode
EM	Electromagentic
EMS	Environment Monitoring System
EOB	End Of Burn
EOC	End Of Cooling
EP	Equatorial Port
EPD	Equatorial Port Duct
EPP	Equatorial Port Plug
FDR	Final Design Report
FMEA	Failure Modes & Effects Analysis
FPSS	Fusion Power Shutdown Sysytem
FW	First Wall
FWCD	Fast Wave Current Drive
GDC	Glow Discharge Cleaning
GDS	Gas Detritiation System
GIS	Gas Introduction System
GL	Ground Level
GSSR	Generic Site Safety Report
H	Hydrogen
H&CD	Heating & Current Drives
HCB	Hot Cell Building
HD	Hydrogen-Deuterium
HEPA	High Efficiency Particulate Air filters
HFS	High Field Side
HH	Hydrogen-Hydrogen
HNB	Heating Neutral Beam
HRS	Heat Rejection System
HT	High Tension
HTO	Hydrogen-Tritium Oxide

HTS	Heat Transfer System
HV	High Voltage
HVAC	Heating, Ventilation and Air Conditioning
HVDC	High Voltage Direct Current
HX	Heat Exchanger
IC	Ion cyclotron
IC-DC	Ion cyclotron resonance discharge cleaning
ICE	Ingress of Coolant Event
ICH	Ion Cyclotron Heating
ICRH	Ion Cyclotron Resonance Heating
IEC	International Electrotechnical Commission
IIS	Inner Intercoil Structure
IM	Initial Premagnetisation
INIRC	International Non-Ionising Radiation Committee
IR/TV	Infra-red Television Camera
ISO	International Organization for Standardization
ISS	Isotope Separation System
IVT	In-Vessel Transporter
IVVS	In-Vessel Viewing System
JET	Joint European Torus
LAC	Local Air Coolers
LFS	Low Field Side
LH	Lower Hybrid
LHCD	Lower Hybrid Current Drive
LHe	Liquid Helium
LIDAR	Light Detection and Ranging). (Light equivalent to Radar)
LIM	Limiter
LLPS	Low-Level waste Processing System
LN2	Liquid Nitrogen
LP	Lower Port
LPD	Lower Port Duct
LS	Load Specifications & Combination
LTM	Long Term Maintenance
LV	Low Voltage
MAM	Manipulator Arm
MAMuG	Multi Aperture Multi Grid
MARFE	multifaceted asymmetric radiation from the edge
MD	Maintenance Detritiation
MGS	Magnet Gravity Support
MHD	MagnetoHydroDynamic
MIC	Mineral Insulated Cable
MPCB	Magnet Power Conversion Building
MPSSN	Magnet Power Supply Switching Network
MS	Magnet System

MSE	Motional Stark Effect
MSLD	Mass Spectroscopy Leak Detection
MTL	Main transmission line
NAG	Neutron Analysis Group
NB	Neutral Beam
NBI	Neutral Beam Injection
NBPS	Neutral Beam Power Supply
Nb3Sn	Niobium-Tin
NbTi	Niobium Titanium
NPA	Neutral Particle Analyser
NRW	Non-Radiation Worker
NTM	Neo-classical tearing mode
N-VDS	Normal Vent Detritiation System
OCS	Operation Control System
OD	Outer Diameter
OLC	Operating Limits and Conditions
OIS	Outer Intercoil Structures
PAM	Passive-active multi-junction
PC	Poloidal Coil
PCP	Primary Closure Plate
PCR	Pre-compression Ring
PCS	Plasma Control System
PDD	Plant Description Document (now obsolete)
PDS	Plant Design Specitications
PF	Poloidal Field
PFC	Plasma-Facing Component
PFW	Primary First Wall (now obsolete)
PG	Plasma Grid
PHTS	Primary Heat Transfer Systems
PID	Project Integration Document
PID	Proportional, Differential and Integral Control
PIE	Postulated Initiating Event
PMS	Passive Magnetic Shield
Pol	poloidal
POS	Plasma Operation State
Pos	Position
PP	Port Plug
PPEN	Pulsed Power Electrical Network
PRM & PS	Personal Radiation Monitoring & Protection System
PS	Power Supply
PSM	Pulse Step Modulator
PSR	Plant Safety Requirement (obsolete Document)
QA	Quality Assurance
QC	Quality Control
QMB	Quartz Micro Balance
R&D	Research & Development
RDL	Resonant Double Loop
Refl	Reflectometry
RF	Radio Fraquency

RGA	Residual Gas Analyser
RH	Remote Handling
Rh	Rhodium
RID	Residual Ion Dump
RN	Recombiner Network
RPC	Reactive Power Compensation
RPrS	Raport Préliminaire de Surité
RRR	Relative Resistivity Ratio
RT	Room Temperature
RW	Radiation Worker
RWM	Resistive Wall Mode
SC	SuperConducting
SCADA	Supervisory Control And Data Acquisition
SCHe	Super-Conducting Helium
SCS	Supervisory Control System
SF6	Sulfur Hexa-fluoride
SFC	Sequential Function Chart
SGVS	Service Guard Vacuum System
SHe	Supercritical helium
SHEX	Supercritical Helium flow heat EXchanger
SIC	Safety Importance Component
SL	Support Loads
SLDS	Service Leak Detection System
SN	Splitting Network
SOB	Start of Burn
SO-ECH	Start Of ECH
SOF	Start of Flat top
SOF/B	Start of Flat top/Burn
SOH	Start of Heating
SOL	Scape-Off Layer
SRS	Service Roughing System
SS	Safety System
SSEPN	Steady State Electrical Power Network
SSPSS	Steady State Power Supply System
STM	Short Term Maintenance
STS	Short Term Standby
STS	Support Thermal Shield
ST-VS	Suppression Tank Vent System
SU	Start Up
SUS	Type of austenitic stainless steel
S-VDS	Standby Vent Detritiation System
SVS	Service Vacuum System
TAE	Toroidal Alfven Eigenmode
TBA	To Be Assessed
TBC	To Be Confirmed

TBD	To Be Determined
TBM	Test Blanket Module
TCM	Technical Coordination Meeting
TCS	Test & Conditioning State
TCWS	Tokomak Cooling Water System
TE01	Transvere Electric 01 mode
TEPS	Tokamak Exhaust Processing System
TF	Toroidal Field
TFC	Toroidal Field Coil
Th	Thermal
TIG	Titanium Inert Gas (Spot Weld Method)
TLD	Thermoluminecent Dosimeter (radiation badge)
TS	Thermal Shield
TS	Thomson scattering
TTS	Transition thermal shield
TV	television
TVS	Tokomak Vent System
UHV	Ultra High Vacuum
UNS	Type of Stainless Steel
UP	Upper Port
UPD	Upper port Duct
UPP	Upper Port Plug
UV	Ultra Violet
VDE	Vertical Displacement Event
VFO	Derived Operational Value (fr)
VDS	Vent Detritiation System
VNC	Vertical Neutron Camera
Vert	Vertical
VPR	Vacuum Pumping Room
VS	Vertical Shaft
VSWR	Voltage Standing Wave Ratio
VTL	Vacuum Transmission Line
VUV	Vacuum Ultra Violet
VV	Vacuum Vessel
VVPSS	Vacuum Vessel Pressure Suppression System
VVTS	Vacuum Vessel Thermal Shield
WBS	Work Breakdown Structure
WCS	Water Circulation System
WDS	Water Detritiation System
XPF	X-Point Formation
XRCS	X-ray Crystal Spectrocopy/Spectrometer
XUV	Extreme UV

1 Project Management - Documentation

1.1 Master Documentation Structure - Annexes

1.1.1 Background and evolution since the FDR 2001

The ITER 2001 Final Design Report documentation was based on the existence of two high level documents: the DRG1 and the PDD. The former (DRG1) dealt with the requirements and specifications above the system level. This included not only plant-wide requirements but also interfaces or specifications affecting the design of more than one single system. DRG1 identified the functional and physical interfaces between two systems and refers to any document and drawings defining the interface in more details. More detailed "Design Background" documents were annexed. These annexes, for example, addressed in detail Loads Specifications, Quality Assurance, Design Criteria, Design Manuals and Guidelines, Safety Requirements, etc.

The latter (PDD) was the global plant description. It summarised the design based on the details in the Design Description Documents (DDDs), gave an overview of major plant processes that usually involved more than one system, summarised plant level assessments, and overall planning. More details were explained in "Plant Assessment" document annexes. These annexes described and assessed in more detail the entire Plant and processes involving more than one single system. For example: Plant Control, Plasma Performance, Safety Assessment, Assembly Process, Seismic Analysis, Material Assessments, Nuclear Analysis, etc.

Amongst the two top-level documents only the DRG1 could be practically managed via configuration control. The reason being that the PDD was far too "descriptive".

For the above reason, and to add Project Management issues, this document (the Project Integration Document) is intended to substitute both DRG1 and PDD to form one single top-level technical configuration management item as defined in the configuration management system in the ITER Management and Quality Program. Configuration management procedures are set out in the relevant QA sections of the MQP.

All annexes to the PID are to be considered as an integral part of this document. *Changes to them are covered by the same procedures and level of scrutiny as the main document.*

1.1.2 Current Status

The current status of the ITER Project top documentation structure is shown in the relevant part of the ITER Management and Quality Program. This can be accessed via IDM at the following link: https://users.iter.org/users/idm?document_id=ITER_D_22F4KX.

1.2 Manuals, Handbooks, Design Criteria

The ITER design is to be developed in accordance with the following "Project Background Documents" (click for link to ITER Document Management).

- ITER Structural Design Criteria (including the implied use of materials properties);
- ITER Magnet Electrical and Superconducting Design Criteria;
- ITER Electrical Design Handbook;
- ITER Control System Design Handbook;
- ITER Materials Properties Handbook and Materials Assessment Report;
- ITER Computer Aided Design Handbook;
- ITER Radiation Hardness Design Manual;
- ITER Remote Handling and Standard Components Manual;
- ITER Vacuum Design Handbook;

The above standards should be used wherever possible, and any exceptions have to be approved by the project management.

1.3 Management and Quality Program

The purpose of the ITER Management and Quality Program is to describe and establish the overall framework for the execution of the ITER Project to meet its technical objectives within its budget and schedule constraints.

The goal of the Management and Quality Program is:

- Assure the safe operation of the ITER Facility
- Use ITER resources in a effective way,
- Ensure that the level of quality considered necessary to achieve ITER objectives is specified and implemented,
- Ensure that sufficient documentation is maintained to demonstrate achievement of the required objectives.

It is the goal of ITER management to provide assurances that the activities associated with fusion research and development as well as the design, procurement and construction activities for the ITER Project yield data, designs, procurements and other activities with results for facilities, structures, systems, components, equipment, and materials that: (1) conform to established requirements, (2) are fully traceable to valid data, and (3) are capable of withstanding detailed technical reviews.

This Management and Quality Program:

- Involves all items or activities important to the safety, reliability and performance of the ITER Facility,
- Provides for all those activities that are necessary for verifying that the required quality is achieved and that objective evidence is produced to that effect.

This shall include in particular items whose failure would result in a significant:

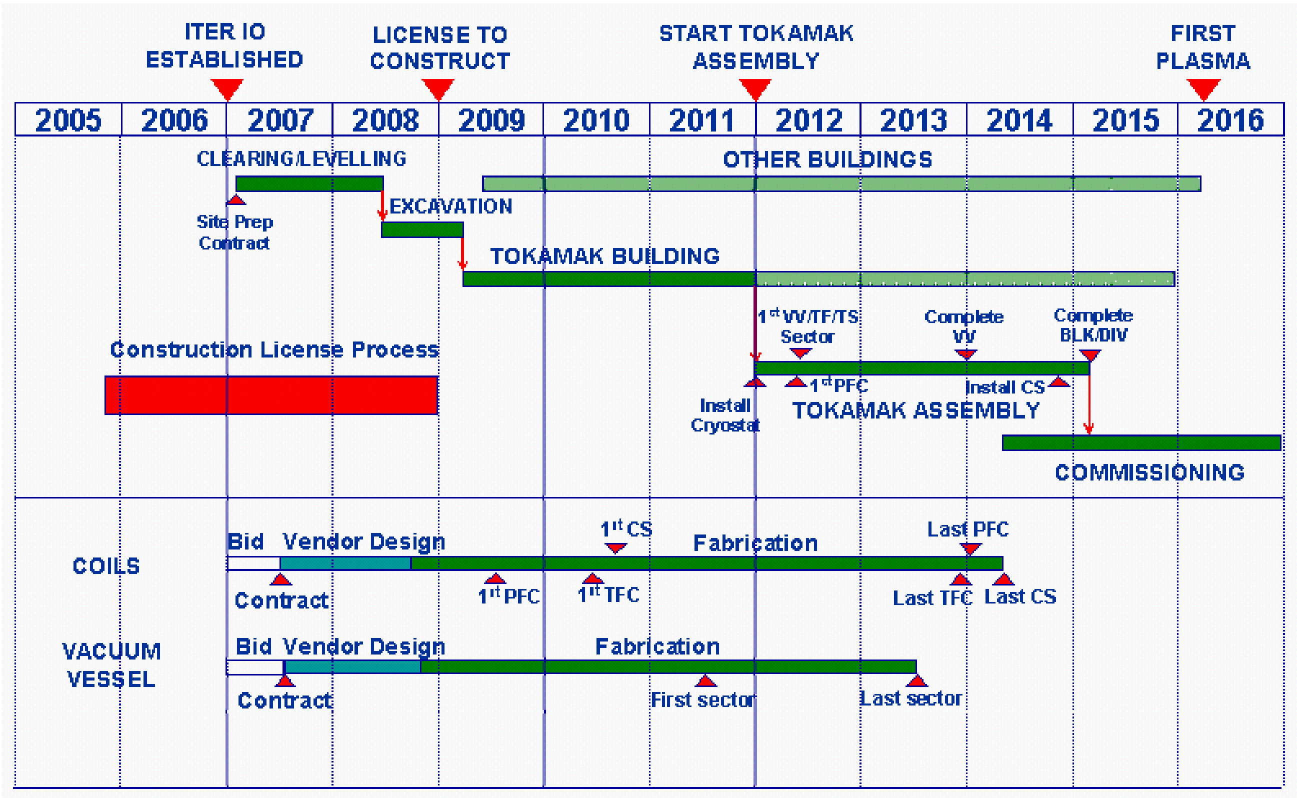
- safety problem to the public and/or workers,
- loss of time and investment,
- unscheduled shutdown of the machine,
- reduction of the quality of data which could be acquired during operation,
- impact on the environment.

1.4 Annexed Plans

1.4.1 Construction, Commissioning and Operation Plans

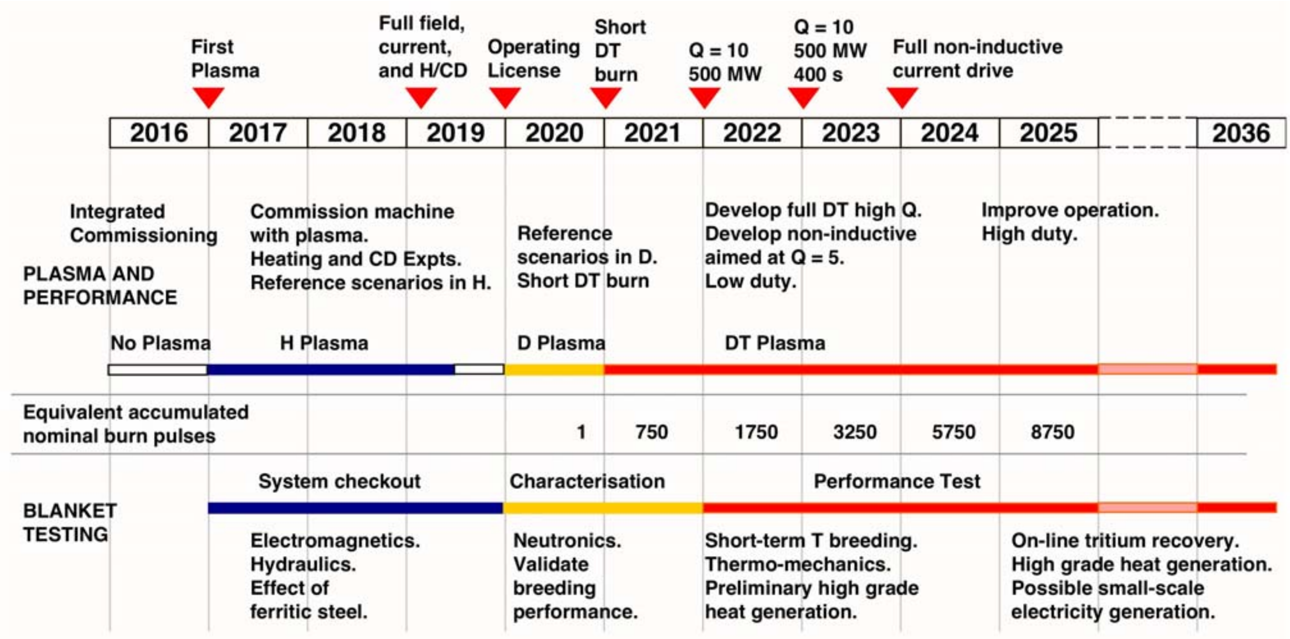
Version 1 of the site adapted reference construction/commissioning integrated project schedule was presented at the December PT-Leaders meeting (ITER_D_254H37 v1.0, ITER_D_2547WB v1.0, and ITER_D_2547VU v1.0). It will be tracked regularly in ITER Technical Coordination and PT Leaders Meetings.

Figure 1.4-1 Construction and Commissioning Schedule



The reference operating schedule is defined below:

Figure 1.4-2 Operating Schedule (normalised to FDR-2001)



The detailed planning for the second 10 years of operation remains to be developed. The gap between the two schedules is intentional, representing a contingency.

1.4.2 *Assembly Plan*

The procedure to be followed for assembly, as well as a detailed time schedule to be followed, are described in the Assembly Procedures annex to this document.

1.4.3 *Remote Maintenance Plan*

The procedure to be followed for remote maintenance of tokamak systems, as well as a detailed time schedule (within overall time constraints as defined in section 3.10) to be followed in each case, are described in the Remote Handling Procedures annex to this document. General maintenance of each plant systems is dealt with under that system in Section 4.

2 Overall Machine Parameters and Configuration

2.1 Overall System Configuration and WBS

The ITER Systems are organised by "Work Breakdown Systems", described in the MQP document, https://users.iter.org/users/idm?document_id=ITER_D_22F49Q, and shown in table 2.1-1 below. A further "Product Breakdown Structure" classification, for drawing office use, is described in the CAD manual. Chapter 4 of this PID document describes individual systems by WBS. An active cross link (cntl. click) is included in table 2.1-1 below. A schematic of the system organisation is shown in figure 2.1-1 below.

Table 2.1-1 ITER Work Breakdown Structure

(right click PID ref for direct access)

WBS	WBS Element	PID Ref.	WBS	WBS Element	PID Ref.
	TOKAMAK BASIC MACHINE			**SITE AND FACILITY SUPPORT**	
1.0	Tokamak Basic Machine – Top Level Drawings		6.0	Site and Facility – Top Level Drawings – Config.Management	4.24
1.1	Magnets	4.1	6.1	Site	
1.5	Vacuum Vessel	4.2	6.2	Reinforced Concrete Buildings	
1.6	Blanket System	4.3	6.3	Steel Frame Buildings	
1.7	Divertor	4.4	6.4	Radiological Protection	4.25
1.8	Fuelling and Wall Conditioning	4.5	6.5	Liquid and Gas Distribution	4.26
1.9	Plasma				
	TOKAMAK ANCILLARY EQUIPMENTAND CRYOSTAT			**SYSTEMS ENGINEERING AND INTEGRATION**	
2.1	Machine Layout (Layout in Pit and Galleries)		7.1	Design requirements	
2.2	Machine Assembly and Tooling	4.6	7.3	System Engineering and Analysis	
2.3	Remote Handling (RH) Equipment including Hot Cell Processing System	4.7 4.8	7.4	Design Standards Handbooks and Manuals	
2.4	Cryostat And VVPSS	4.9 4.10	7.5	Configuration Management	
2.6	Cooling Water System	4.11	7.6	Drawing Management	
2.7	Thermal Shields	4.12			
	TOKAMAK FLUIDS			**SAFETY AND ENVIRONMENT**	3.1
3.1	Vacuum Pumping	4.13	8.1	Fusion Nuclear Safety	
3.2	Tritium Plant	4.14	8.2	Occupational Safety	
3.4	Cryoplant and Cryodistribution	4.15	8.3	Effluent and Waste	
			8.4	Safety Analysis Reports	
	POWER SUPPLIES - COMMAND CONTROL			**PROJECT MANAGEMENT & ADMINISTRATION**	
4.1	Coil Power Supplies (PS) and Distribution	4.16	9.1	Project Management	
4.2	Heating and Current Drive Power Supplies		9.2	Quality Assurance	
4.3	Steady State (SS) Electrical Power Network	4.16	9.3	Cost Estimate	
4.5	Control, Data Access and Communication (CODAC)	4.17	9.4	Construction, Commissioning and Operation	
4.6	Safety and Interlock System		9.5	Decommissioning	
4.7	Poloidal Field Control	3.3	9.6	Information Technology	
	PORT INTERFACING SYSTEMS		9.7	Public Information	
5.1	Ion Cyclotron Heating & Current Drive (IC H&CD) System	4.18			
5.2	Electron Cyclotron Heating & Current Drive (EC H&CD) System	4.19			
5.3	Neutral Beam Heating & Current Drive (NB H&CD) System	4.21			
5.4	Lower Hybrid Heating & Current Drive (LC H&CD) System	4.20			
5.5	Diagnostics	4.22			
5.6	Test Blankets	4.23			

NOTE : A major update to the WBS structure is ongoing (jan 2007) to adapt to the new ITER organization and procurement plans. Hence this table will be updated in the next release.

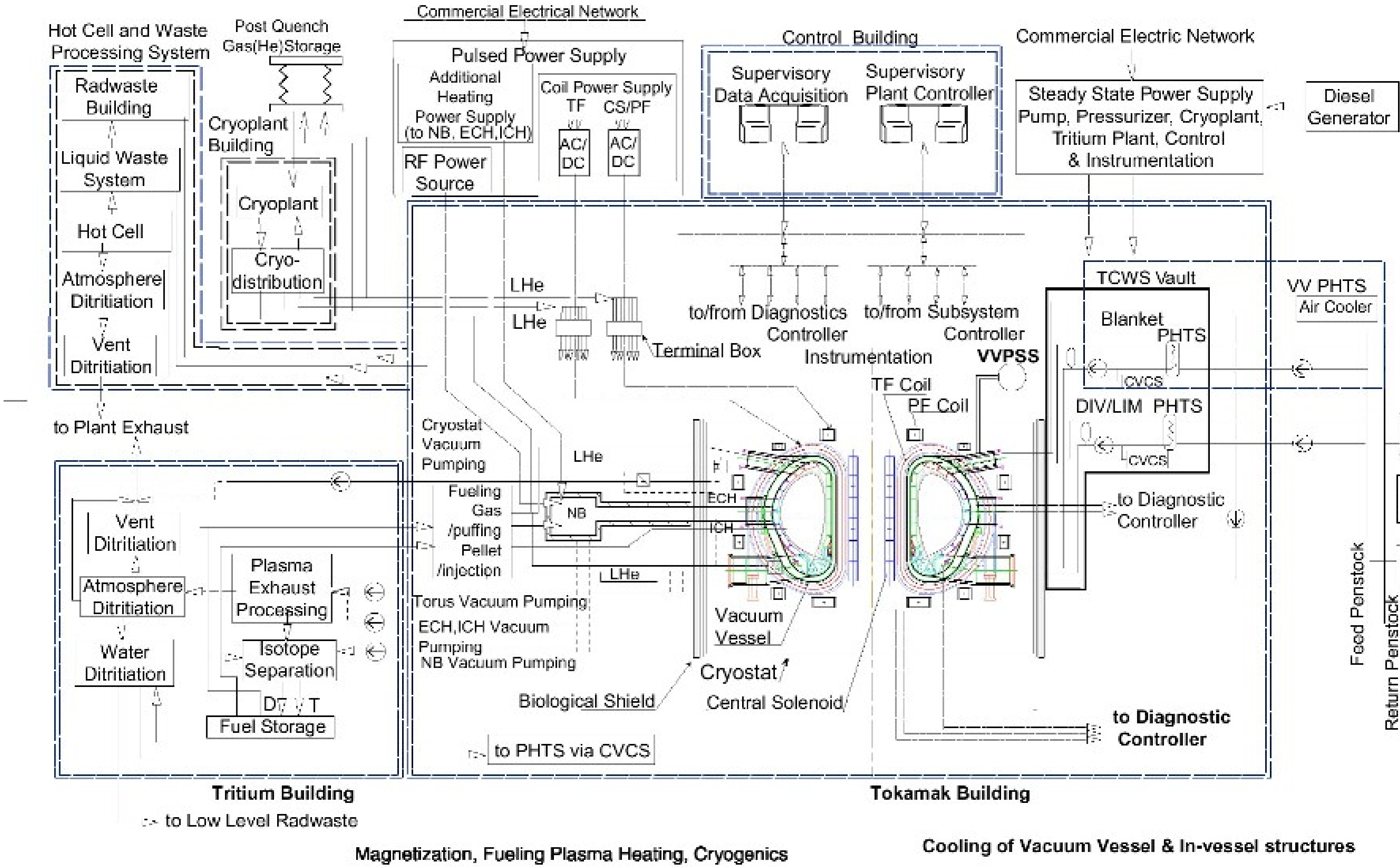


Figure 2.1-1 Overall System Diagram (From 2001 FDR summary)

2.2 Basic Machine Parameters

The plasma size and basic parameters of the ITER device derive from both physics and engineering/technology considerations and represent a global optimisation of the device and plasma performance with a pre-determined cost target. To ensure reliable choice of the basic plasma parameters, the following physics guidelines are introduced so as to determine a nominal pulse mode operation point:

1. a plasma current sufficient to provide adequate plasma energy confinement and MHD stability;
2. adequate in-vessel volume for reliable power exhaust and impurity control;
3. plasma energy confinement sufficient to achieve extended burn in inductively driven plasmas with Q = 10, based on empirical H-mode confinement scaling (IPB98(y,2)) with H_H factor of 1
4. safety factor at the nominal plasma current $q_{95} \approx 3$;
5. normalised beta $\beta_N = \beta aB/I \leq 2.5$ at the nominal plasma current;
6. moderate plasma elongation $\kappa_{95} \leq 1.7$ at the nominal plasma current;
7. single null divertor configuration.

Table 2.2-1 Basic Machine Design Parameters

Parameter	Cross Reference	Unit	H	DT	TBA
Plasma major radius, R		m	→	6.2	
Plasma minor radius, a		m	→	2.0	
Plasma current, I_p (1)		MA	→	15.0	17(2)
Additional H & CD power		MW	→	73(3)	110(3)
Fusion power	Table 3.2-1	MW	0	500(3)	700(4)
Toroidal field at major radius, B_o		T	→	5.3	
Elongation at 95% flux, κ_{95}			→	1.7	
Triangularity at 95% flux, δ_{95}	Table 3.2-1		→	0.33	
Plasma volume		m^3	→	830	
Plasma surface		m^2	→	680	
Nominal Normalised beta, β_N			1.5	2.0	2.5
Plasma nominal thermal energy		GJ	0.27	0.35	0.45
Plasma nominal magnetic energy ($\mu_0 R l_i I_p^2/4$)		GJ	→	0.37	0.5
MHD nominal safety factor at 95% flux, q_{95}			→	3.0	2.6
Average neutron wall load at first wall		MWm^{-2}	0	0.56	0.79
Neutron wall load at outboard FW at midplane	*Table 3.8-1*	MWm^{-2}	0	0.78	1.09
Total average neutron fluence at the first wall		$MWam^{-2}$	0	0.3	0.5
Integrated full power operation time		h	4700		7800
Peak burn duty cycle (5)	§ 3.2	%	→	25	
Maximum number of full power reference pulses	§ 3.2			30000	

Note:
(1) See § 3.4 for the reference direction within machine operation.
(2) This scenario is the most demanding of those to be assessed to see whether they can be accommodated without significant additional costs.
(3) Nominal operation assumes 33MW NB plus 40 MW RF. Various configurations are considered to increase the heating power, see Section 3.1. Upgrading of additional heating power shall be accommodated with the additional investment for auxiliary systems.
(4) This high power operation would be achieved at a reduced pulse length and duty cycle so that costs should not be increased.
(5) Ratio between burn and repetition rate (see Table 3.2-1)

Figure 2.2-1 Elevation View of ITER

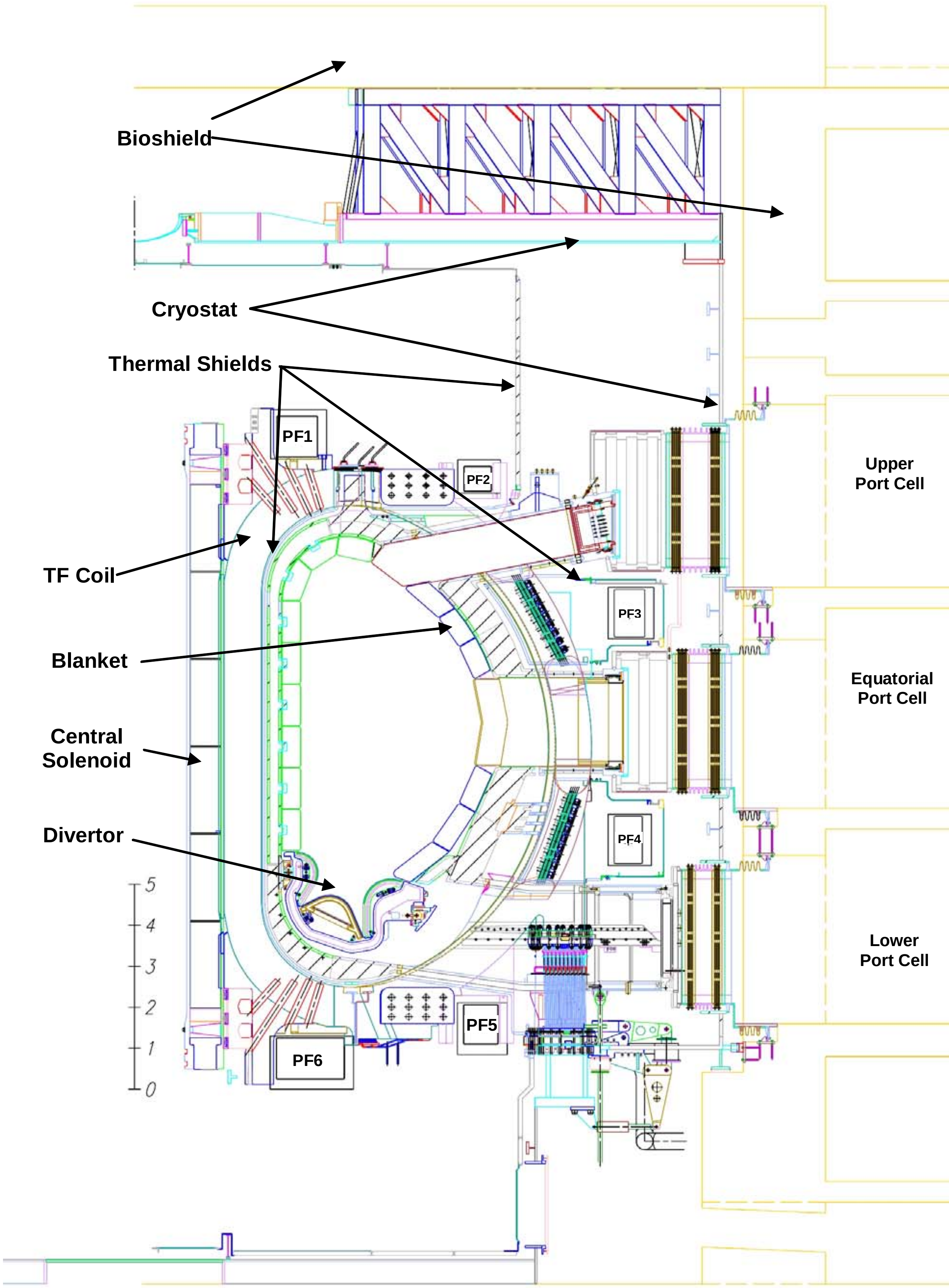


Table 2.2-2 Number of Major Core Components

Component	Cross Reference (cntl.click)	Number
Number of TF coils	§ 4.1.2.4	18
Number of CS modules	§ 4.1.2.5	6
Number of PF coils	§ 4.1.2.6	6
Vacuum vessel segmentation	§ 4.2	9
Divertor segmentation (Cassettes)	§ 4.4	54
Number of limiters	§ 4.3.2.2	2
Shielding Blanket Modules	§ 4.3	440
Ports (Lower, Equatorial, Upper)	§ 3.5	9 + 18 + 18

Note: Table links are "active". Use "click" on PDF, or "cntl click" on MS Word

Table 2.2-3 Size of Major Core Components

Component	Cross Reference (cntl.click)	Approx. Values
Cryostat	- Table 4.9-1	
- Overall dimensions		ϕ 28.5 x 29 m
- Total volume		13945 m^3
- Free Pumpable Volume		8450 m^3
- Inner Surface Area (room temp)		3350 m^2
Cooled Surfaces inside Cryostat	-	
80K Thermal Shield Surface Area	- Table 4.12-1	~5000 m^2
4.5 K Magnet Surface	- Table 4.9-1	~8500 m^2
Vacuum Vessel	-	
- 9 sector dimensions (each)	- Table 4.2-1	13.0 x 8.0 x 7.0 m
- Volume inside VV shell	- -	2054m^3
- Free Pumpable Volume (up to valves)	- -	1330 m^3
- Approx Plasma Volume	- Table 2.2-1	830 m^3
- Wall surface facing plasma (excl divertor)	- -	670 m^2
- Vacuum Surface Area not facing plasma	- -	2000 m^2
- Divertor Surface exposed to plasma	- -	230 m^2

Table 2.2-4 Weight of Major Core Components (approx)

Component	***Cross Reference***	***Approx Weight***
Cryostat Total	Table 4.9-1	3300 t
-Main vessel and Port Stubs -Shielding (440 modules max 4.5 t) Table 4.3-1 -Port structures (excl. Port Plugs) -54 Divertor Cassettes (SS,Cu,W,CFC,water) Table 4.4-2 -Shielding Water -Supports -blanket cooling manifolds and in-port pipes Vacuum Vessel, Shielding, Divertor and Manifolds	3344 t 1610 t 1781 t 678 t 236 t 160 t 140 t 7945 t	7945 t
Plug structures Equatorial plugs (< 14x50t) Upper plugs (< 18 x 20t)		700 t 360 t
Thermal Shields in Cryostat (80K) and all connections	Table 4.12-1	820 t
Blanket and VV cooling pipes (supported by the VV and cryostat) Cryopumps 10 x 5tons		20 t (TBC) 50t
Magnet systems TF Toroidal Field Coils (18) 6300 t PF Poloidal Field Coils (6)2840 t CC Correction Coils 80 t CS Central Solenoid 930 t TOTAL Magnet Systems	Table 4.1-2 ***§ 4.1.2.6*** § ***4.1.2.7*** § ***4.1.2.5***	***9548 t***
Magnet and Structure cooling feeders		TBD
Total: ***Cryostat + VV + Magnets*** (excluding additional pipework and diagnostics)		***> 22,740 t***

Note: Table links are "active". Use "click" on PDF, or "cntl click" on MS Word

2.3 Design Configuration Tables

The overall machine configuration (layout) and space allocation is set through configuration models which belong to the 10.XXXX series up to the pit, and 62.XXXX series for the rest of the plant.

The models that describe the reference machine configuration are listed in the "Configuration Model List", held in the IDM folder (click to follow link):
"Project_Integration__Administration_Services/Design_Integration/Configuration_Management/Configurtion_Model_List"

2.4 Plant Operation States – Transitions

The ITER plant is always in one of a number of well defined states, listed below. Transitions from one state to another are well controlled and require a number of conditions to be satisfied. The ITER Safety Interlock System is conditioned according to the current state. Hence access and work control are dependent on the plant operation state.

The five states can be summarised by:

- Long Term Shutdown (for serious maintenance or upgrade, after a significant nuclear decay period): Vessel Vented, all magnets off; but the cryostat can be under vacuum and cold –
- Short-Term Maintenance (Typical weekends, holidays, short breaks): Vessel evacuated, all magnets at zero current; but magnets cold. Some reduced current magnet testing allowed. Tightly controlled access.
- Test and Conditioning: Most systems ready for test-pulsing (no-plasma) but cooling systems in low-flow. Magnets can be operating. *Very* restricted, tightly controlled access
- Short-Term Standby. All systems ready for plasma. TF ON, PF off but ready, cooling systems to high flow. No access. – Typically the state between pulses.
- Plasma – All systems operational

These states are fully defined in the following table.

Table 2.4-1 ITER Plant Operation State

ITER Operation State Plant subsystem	Construction/Long Term Maintenance (LTM)	Short Term Maintenance (STM)	Test & Conditioning State (TCS)	Short Term Stand-by (STS)	Plasma Operation (POS)
Duration	**>30 days**	**1-30 days**		**<8 hrs**	
Magnet State	Maintenance / [Vacuum] / [Cold]	Cold / [Stand-by / [Idle]	Stand-by / [Idle/Ready for pulse]	Idle / [Ready for pulse]	Ready for pulse
Temp.(K)	RT/[10]	5/[10]	5/[RT]	5	5
TF current	OFF	OFF/[ON(reduced)]	OFF/[ON]	ON	ON
PF current	OFF	OFF	OFF/[ON]	OFF	ON
Vacuum vessel					
Pressure	Atmosphere / [Vacuum]	Vacuum	Vacuum	Vacuum	Vacuum
Fuelling State(Pellet)	Stop	Stop	Gas delivery (Injection)	Gas by-pass	Gas delivery (Injection)
Pellet inj.	OFF	OFF	OFF/[ON]	Ready (no ice pellet)	ON (normal)
State(Gas)	Stop	Stop	Gas delivery (Injection)	Gas by-pass	Gas delivery (Injection)
Gas puff.	OFF	OFF	OFF/[ON]	Ready (valve closed)	ON (normal)
Wall conditioning					
GDC	OFF	OFF	ON/OFF	OFF	OFF
RF cleaning	OFF	OFF	ON/OFF	OFF/[ON]	OFF/[ON]
Baking	OFF	OFF/[ON]	ON/OFF	OFF/[ON]	OFF
Cryostat					
Pressure	Atmosphere / [Vacuum]	Vacuum	Vacuum	Vacuum	Vacuum
Thermal shield					
Temp.(K)	RT	80	80 / RT	80	80
VVPSS State	Vent/Normal	Normal	Normal	Normal	Normal
Pressure	0 MPa	0 MPa	0 MPa	0 MPa	0 MPa
Tokamak pit	Open / [Closed]*	Open / Closed	Closed / [Open]	Closed	Closed
Tokamak cooling water			(see note (4))		
State(VV)	[OFF]/Partial maintenance	Part.maintenance /decay heat	Decay heat/Baking (4)	Decay heat	POS(Normal)
VV PHTS	[stop,drain]/Full,RT	Full,RT	Full,RT~100°C /Full,200°C (4)	Full,~100°C	Full,100°C
State(others)	[OFF] /Partial maintenance	Partial maintenance /Decay heat	Decay heat/baking (4)	Decay heat	POS(Normal)
Blanket PHTS	[stop,drain]/CVCS,RT	CVCS,RT/ Low,RT~100°C	Low,RT~100°C /240°C (4)	Low/Full,~100°C	Low/Full,100°C
Divertor PHTS	[stop,drain]/CVCS,RT	CVCS,RT/ Low,RT~100°C	Low,RT~100°C /240°C (4)	Low/Full,~100°C	Low/Full,100°C

ITER Operation State Plant subsystem	Construction/Long Term Maintenance (LTM)	Short Term Maintenance (STM)	Test & Conditioning State (TCS)	Short Term Stand-by (STS)	Plasma Operation (POS)
Duration	**>30 days**	**1-30 days**		**<8 hrs**	
Additional heating	[stop,drain]/CVCS,RT	CVCS,RT/ Low,RT~100°C	Low,RT~100°C /240°C (4)	Low/Full,~100°C	Low/Full,100°C
Diagnostics and others	[stop,drain]/CVCS,RT	CVCS,RT/ Low,RT~100°C	Low,RT~100°C /240°C (4)	Low/Full,~100°C	Low/Full,100°C
Comp. cooling	[OFF]/Ope.(Full),40°C	Ope.(Full),40°C	Ope.(Full),40°C	Ope.(Full),40°C	Ope.(Full),40°C
Chilled water	[OFF]/Ope.(Full),5°C	Ope.(Full),5°C	Ope.(Full),5°C	Ope.(Full),5°C	Ope.(Full),5°C
HRS	[OFF]/Ope.(Low),35°C	Ope.(Low),35°C	Ope.(Low)/[Ope.(Full)], 35°C	Ope.(Low),35°C	Ope.(Full),35°C
Vacuum pumping					
VV	OFF/[ON]	ON	ON	ON	ON
Cryostat	OFF/[ON]	ON	ON	ON	ON
Roughing	OFF/[ON]	ON	ON	ON	ON
Leak detect.	OFF/[ON]	OFF/[ON]	OFF/[ON]	OFF/[ON]	OFF/[ON]
Tritium plant					
Tritium flow	Storage	Storage	Storage / [ON]	ON	ON
ISS and WDS	OFF/[ON]	OFF/[ON]	OFF/[ON]	On/[OFF]	ON/[OFF]
ISS Temp(K)	amb/20~amb	20	20	20	20
Mainten.-Detrit. (MD)	Stand-by	Stand-by/[ON]	Stand-by/[ON]	Stand-by	Stand-by
ADS	Stand-by/[ON]	Stand-by/[ON]	Stand-by/[ON]	Stand-by/[ON]	Stand-by/[ON]
Cryoplant State (3)	OFF/Maintenance	Low flow stand-by	Zero/Low stand-by	Low LHe stand-by	Normal
Coil power supply					
A.C Dist.	Shutdown	ON / [OFF]	ON	ON	ON
TF P/S	Shutdown	OFF / [Ready/ON]	OFF / [Ready/ON]	ON	ON
CS P/S	Shutdown	OFF	OFF / [Ready/ON]	Ready	ON
PF P/S	Shutdown	OFF	OFF / [Ready/ON]	Ready	ON
CC P/S	Shutdown	OFF	OFF / [Ready/ON]	Ready	ON
AH power supply					
IC,EC,NB P/S	OFF	OFF	OFF/[ON]	ON	ON
SS power supply	ON/[OFF]	ON	ON	ON	ON
AH system					
NBI	OFF/Maintenance	OFF/Maintenance	OFF/[Ready for conditioning]	Ready for Pulse	Pulse(Ready for injection, Dwell)
NB Cryo	OFF/Maintenance	OFF/Maintenance	Normal/Slow PRS	Normal	Normal/Fast PRS
IC H&CD	OFF	OFF/S1	OFF/[Pulse (S2-S5)]	S1	Pulse(S2-S5)
EC H&CD	Stop	Stop	Stop/[Pulse(HV,BM,RF)]	Ready RF	Pulse
Radiation monitoring	ON	ON	ON	ON	ON
Water distribution	ON/[OFF]	ON	ON	ON	ON
Gas distribution(Air)	ON/[OFF]	ON	ON	ON	ON
SCS, Interlock	ON	ON	ON	ON	ON
Diagnostic	OFF	OFF	OFF/[ON]	ON(Ready)	ON
FPSS	OFF	OFF	OFF/[ON(Ready)]	ON(Ready)	ON(Ready)

[ ] is optional status for their maintenance or other system's activity requirement.
"/" is OR (selection) status which depends on the specific actions.
"," is AND which needs all designated conditions.
( ) is additional description of the status.
* : Except during cask transporting.

Note:
(1) For operation states TCS, STS and POS, all applicable loading conditions and combinations as specified in the Load Specification and Combination (LS) annexed to the DRG1 should be considered.
(2) Normal transitional sequences are described in the Control System Design and Assessment (CSD) annexed to the PID.
(3) The Cryoplant operation states are described in “Table 4.15-4 Typical operating modes of the cryoplant”
(4) In the baking mode, the configuration of the cooling systems for the port cell must be changed to a by-passes with ~24 mm (TBD) diameter valve (safety credit TBD) or orifice to limit the maximum pressure in the port cell 0.16 MPa (target value)

The following table gives the minimum bounding number of operational transitions which may be permitted to affect a subsystem's capacity and fatigue life.

Table 2.4-2 Number of Operational Transitions

Baking operation	200(1)
TF magnetisation	1000
VV vacuum pump-down	30(2)
Cryostat vacuum pump-down	15(3)
Magnet cooldown/warm-up	100

Notes:

(1) Assuming 10 cycles of operation per year, ~10 days of wall conditioning operation and ~2 weeks of plasma operation in one cycle.

(2) Twice a year during first 5 years, once a year next 10 years and twice in the remaining years. Including 8 unscheduled maintenance of in-vessel components.

(3) Once a year during first 10 years, then once every 2 years.

Note: July 2005. An absolute valve between the NB sources and the vessel is being investigated under DCR-33 (see ITER Technical Web). Such a valve may be necessary to avoid Torus Vent during minor NB repairs. Addendum Oct 2006 : DCR-33 was dropped and reformulated under DCR-49, which also includes the absolute NB valve. This work is ongoing

2.5 Plant Operation Programme – Shift Pattern

ITER will be designed, constructed and operated fully optimizing the available time, 24hours/day, 365 days per year. A two 8-hour shift with a third low activity shift pattern will be used as a basis for planning during the construction

When "integrated" commissioning and operation commences the aim will be to establish two full operational shifts, with a third "silent hours shift" for necessary cryogenic regeneration, tritium recovery sequences and cask movements.

It is anticipated that operations will be carried out in long periods separated by maintenance periods (e.g. 10 days continuous operation and 1 week break) with annual or bi-annual (two year) major shutdown periods of a few months for maintenance, further installation and commissioning. Table 2.4-2 shows some estimates of the transitions expected.

Substantial use will be made of off-site "remote participation" for project coordination (meetings and seminars), data analysis (data base access), and experimental exploitation (technical and physics real-time applications).

This is further covered by the annex documents "Construction, Commissioning and Operation Plan" and "Assembly Plan" (active link to IDM, click here).

3 General Requirements and Interfaces – Broad Issues

3.1 Safety

(Substantial update including site adaptation July 2006 – Jan 2007)

The Plant Safety Requirements (PSR), previously annexed to the DRG1, are now defined below.

The two reports below cover more details of the tables and requirements presented in this section:
The ITER Generic Site Safety Report (GSSR) is an annex of this document.
The ITER Plant Design Specification (PDS) is the parent (top level) document.

Note that the site safety report (in French RPrS: Rapport Préliminaire de Sureté), now in preparation, will supersede the GSSR, the update of this chapter takes all valuable inputs according to the Host State regulation which ITER will have to comply with. It is expected that this chapter will need further update as the site safety report will be validated.

3.1.1 Environmental Criteria

Limits on doses to the public and staff from radioactivity and related releases shall be met by design, construction, operation and preparation for decommissioning (so-called "end of operation"). The dose limits that must be respected during normal operation, as well as in off-normal events (incidents and accidents) are presented in Table 3.1-1.

Table 3.1-1 Dose limits

	General safety objectives	
	For personnel	*For the public and environment*
Situations in design basis		
Normal situations	***As low as reasonably achievable, and in any case less than:*** ***Maximum individual dose*** *≤ 10 mSv/yr* ***Average individual dose for workers type A and B (see section 3.1.5.1)*** *≤ 2.5 mSv/yr*	***Releases less than the limits authorised for the installation,*** ***Impact as low as reasonably achievable and in case less than:*** *≤ 0.1 mSv/yr*
Incidental situations	***As low as reasonably achievable and in any case less than:*** *10 mSv* ***per incident***	***Release per incident less than the annual limits authorised for the installation.*** *≤ 0.1 mSv*
Accidental situations	***Take into account the constraints related to the management of the accident and post-accident situation***	***No immediate or deferred counter-measures (confinement, evacuation)*** *< 10 mSv* ***No restriction of consumption of animal or vegetable products (tbc, as no limit is defined for tritium contamination)***
Situations beyond design basis		
Hypothetical accidents	***No cliff-edge effect; possible counter-measures limited in time and space***	

An important element of the safety analyses is the assessment of consequences. In analyses performed before the selection of the ITER site, in the absence of site-specific information that determines doses resulting from releases, the project has focussed on physical releases in grams. The guidelines established by the project for the maximum releases to the environment are presented in the following Table 3.1-2. They should be minimised by design improvement in line with the principle of ALARA. According to preliminary dose calculations for the ITER site at Cadarache, releases of these magnitudes are expected to satisfy the dose limits of Table 3.1-1.

Table 3.1-2 Project Release Guidelines

Events or Conditions	Goal	Project Release Guideline
Normal Operation comprising events and plant conditions planned and required for ITER normal operation, including some faults, events or conditions which can occur as a result of the ITER experimental nature.	Reduce releases to levels as low as reasonably achievable but ensure they do not exceed project release guideline for Normal Operation	< 1 g T as HT and 0.1 g T as HTO and 1 g metal as AP and 5 g metal as ACP per year.
Incidents, or deviations from normal operation, comprising event sequences or plant conditions not planned but likely to occur due to failures one or more times during the life of the plant but not including Normal Operation.	Reduce likelihood and magnitude of releases with the aim to prevent releases, but ensure they do not exceed project release guideline for Incidents.	< 1g T as HT or 0.1g T as HTO or 1g metal as AP or 1g metal as ACP or combination of these per event
Accidents, comprising postulated event sequences or conditions not likely to occur during the life of the plant.	Reduce likelihood and magnitude of releases but ensure they do not exceed project release guideline for Accidents	< 50g T as HT or 5g T as HTO or 50g metal as AP or 50g metal as ACP or combination of these per event.

HT: elemental tritium (including DT); HTO: tritium oxide (including DTO); AP: divertor or first-wall activation products; ACP: activated corrosion products.

It remains an ITER project objective to avoid the need for evacuation following any event. IAEA recommends the no-evacuation objective: the generic optimised intervention value for temporary evacuation is 50 mSv of avertable dose in a period of no more than 1 week, and this value is also adopted in French regulations.

3.1.2 Inventory Guidelines

The inventories of tritium and activation products constitute a radiological hazard and hence shall be reduced as much as possible. Guidance for these inventories is provided in the following paragraphs.

3.1.2.1 Tritium

The total site tritium inventory will be ≤ 3 kg. In addition, the design should comply with the guidelines for the maximum individual values in Table 3.1-3. Some additional tritium inventory may be present in removed components held in storage (this needs to be assessed).

Table 3.1-3 Tritium Inventory Guideline

Tritium Inventory	Guideline
Mobilisable inventory within the vacuum vessel and extensions	Assessed[1] Value 1000 g (Including 120g in cryopumps open to the vacuum vessel)
Mobilisable inventory within the fuel cycle	Assessed[1] Value 700 g (note 2)
Long-term storage	≤ 450 g per independent storage area[3]
Hot cell	≤ 250 g Inventory in storage to be checked
Vacuum vessel cooling system water	< 0.0001 g/ m^3 (~37 MBq/kg)
In-vessel components cooling system water	< 0.005 g/ m^3 (~1.8 GBq/kg)

Note 1: GSSR (G84RI3R02) Table III.E-1 Assessment Values for the tritium inventory in ITER
Note 2: A final value of 1000g may be applied for in the licence.
Note 3: A final value of 1000g may be applied for in the licence.

Tritium inventories in the various components should be minimised and segmented to prevent combined releases from multiple components.

To ensure that these guidelines are not exceeded, the tritium inventory shall be tracked, during every plasma campaign, in the vacuum vessel, in the fuel cycle subsystems (pumping, fuelling, tokamak exhaust processing, isotope separation and storage and delivery), and in the long-term storage system.

Records will also be kept of the estimated tritium inventory in the Hot Cell and Radwaste areas.

The calculated tritium inventory should be updated at the start and end of each campaign but after no less than 1 month of plasma operation by measuring inventories in the tritium process, in the long-term storage, and in the primary coolant.

Tritium accountability shall be maintained by keeping records of all tritium transfers to and from the Site, and by performing a physical inventory annually. Tritium accountability for security purposes will be done through only two mass balance areas (MBAs): long term tritium storage (MBA1) and the rest of the facility (MBA2), i.e. VV, T-plant, Hot Cell, and removed component storage.

3.1.2.2 Activation Products

The design should limit activation products in the form of dust or activated corrosion products to the administrative limits in the following table.

Table 3.1-4 Inventory Guidelines for Radioactive Materials

Inventory	Total [kg]
In-vessel heavily activated dust (W, Cu, steel, etc.)	100
In-vessel beryllium dust	100
In-vessel carbon dust	200
Activated corrosion products (metals) in any single primary HTS loop	10

The design shall include provision to ensure that these guidelines are not exceeded, by measurement or reliable means of estimation. In safety analyses it is assumed that these figures include a maximum of 6 kg of each type of dust on surfaces which become sufficiently hot to be reactive with steam or air during incidents.

3.1.3 Confinement of radioactive and hazardous materials

Confinement of radioactive and hazardous materials is the fundamental safety function which limits the mobilization and dispersion of tritium and activation products in the event of an accident. Releases would most significantly occur upon breach of barriers and mobilization of tritium and activated materials. The confinement is protected as needed by ancillary functions such as heat removal, control of energies and monitoring.

3.1.3.1 Confinement objective

The confinement objective is to prevent the dispersion of radioactive or hazardous material and to ensure that the project general safety objectives (Table 3.1-1) are met.

3.1.3.2 Confinement methodology

The confinement methodology is to identify radioactive and hazardous inventories and provide an appropriate level of confinement based on the level of risk, i.e. provide sufficient confinement to meet the project general safety objectives (Table 3.1-1).

3.1.3.3 Confinement execution

For each principal inventory at risk, two confinement systems are provided so that sequential barriers are provided for each inventory. Appropriate treatment of penetrations through confinement barriers ensures that the confinement objective is met.

3.1.3.3.1 First Confinement System – Process boundaries

The first confinement system generally consists of the process equipment and is the first credited barrier against the release of radioactive or hazardous material. The first confinement system is provided to prevent the dispersion of radioactive or hazardous material within the facility during normal facility conditions, e.g.: operation, testing, and maintenance. During those maintenance procedures in which the process equipment does not provide a confinement function, other means shall be provided as necessary to prevent the spread of contamination.

3.1.3.3.2 Second Confinement System – Process areas

The second confinement system envelops the first confinement system and includes components, vaults, cells, and rooms with appropriate depressurization, filtration, and detritiation; and is the second credited barrier against the release of radioacvtive or hazardous material. This system is provided to limit environmental releases in events during which the first confinement system fails to completely contain the inventory at risk. Some events might result in temporary overpressure in a limited area with the provision to come back to a safe state with negative pressure within a specified period.

In limited number of case only one confinement system might be provided according to the risk, e.g. Hot cells or LLRWS.

3.1.3.3.3 Personnel areas

Personnel areas are general access areas that are not credited in the safety analysis. These areas are shielded as necessary to protect personnel from exposure during normal operation. They are normally vented to atmosphere.

3.1.3.4 Confinement barriers

Each confinement sytem includes one or more static or dynamic barriers such that sequential barriers are provided for each inventory at risk.

3.1.3.4.1 Static barriers

Static barriers require no moving parts to fulfil their confinement function once initiated. Examples of static barriers for the in-vessel inventories are:

- Vacuum vessel
- Cryostat
- Process piping
- Vacuum vessel pressure suppression system
- Vacuum vessel drain tank
- Process room walls, building walls, etc.

3.1.3.4.2 Dynamic barriers

Dynamic barriers require moving parts in order to fulfil their confinement function. Examples of dynamic barriers are:

- Depressurization ventilation sytems and filtration systems
- Detritiation systems
- Isolation systems

3.1.3.4.3 Requirements for the confinement barriers

Confinement barriers (including related systems) shall meet the following requirements:

- Leak-tight (less than the leak rate specified in Table 3.1-5 through Table 3.1-8),
- Capable of withstanding pressures and environments resultant from accident sequences
- Capable of returning the contained volume to below atmospheric pressures within a specified period following accidents,
- Provide a detritiated, filtered, controlled and monitored pathway to maintain pressures in the confined volume below atmospheric pressures following an accident, until releases without their operation are acceptable,
- Reliable. Confinement barriers shall be independent (no common parts) and physically separated as far as reasonably achievable to avoid common mode failure that could lead to loss of both barriers.
- The routing/piping of confinement barriers shall be such as to avoid potential damage to confinement barriers e.g. by movement of equipment during maintenance.
- Signals associated with parameters such as pressure, radiation level, etc. shall be provided to actuate safety actions such as isolation of confinement.
- Valves implementing confinement shall operate within specified periods after detection of accidents. The confinement isolation valves shall assume their safe position on loss of power.
- Emergency power shall be provided to TCWS vault coolers, room depression systems, vent detritation systems (VDSs), VVPSS, suppression tank vent system (ST-VS) and safety chillers and safety related instruments and monitors.
- The confinement barriers, including equipment, penetrations, seals, etc. shall be designed and constructed as to allow for initial and periodic leak testing to ascertain their performance as specified by the design.

3.1.3.4.4 Systems implementing confinement functions

First and second confinement systems are specified for the inventories at risk associated with each process area (plus tokamak maintenance):

Tokamak building: Table 3.1-5

Tritium building:

Table 3.1-6

Hot cell building:

Table 3.1-7

Tokamak maintenance: Table 3.1-8

Table 3.1-5 Systems implementing confinement for the tokamak building sources

Tokamak Building Confinement Systems[1]		
	1st Confinement system	**2nd Confinement system**
In-vessel sources	*In analyses, components implementing these confinement barriers are assumed to have:* *Design pressure ≥ 0.2 MPa* *Leak rate ≤ 1volume %/day at 0.2 MPa*	*In analyses, components implementing these confinement barriers are assumed to have:* *TCWS vault and NBI cell design pressure≥ 0.2 Mpa; Leak rate ≤ 100 volume %/day at 0.2 MPa* *Port cell design pressure≥ 0.15 Mpa; Leak rate ≤ 100 volume %/day at 300Pa pressure difference* *VDS detritation efficiency >99% .* *VDS filter efficiency > 99.9%.* *HVAC filter efficiency >99.9%.*
Tritium Activated dust	Vacuum vessel Vacuum vessel extensions Ex-vessel primary piping for in-vessel component cooling Tritium (fuel) process piping	Cryostat & extensions Primary piping for VV CWS Port cells Vaults & depressurization, detritiation, filtration systems
2	VVPSS[2] VV drain tank[2]	Vaults & Depressurization, detritiation, filtration systems[2]
Ex-vessel sources	**1st Confinement system**	**2nd Confinement system**
Activated corrosion products Tritium in cooling water systems	Ex-vessel primary piping for in-vessel coolant loops Vacuum vessel – for off normal conditions in the in-vessel components	Guard pipes where needed Pipe chases Vaults & depressurization, detritiation, filtration systems

[1] *Penetrations through confinement systems require two penetration barriers such as windows and/or automatic valves.*
[2] *Activated in case of vacuum vessel overpressure.*

Table 3.1-6 Systems implementing confinement for the tritium building sources

Tritium Building Confinement Systems[1]		
1st Confinement system		**2nd Confinement system**
In analyses, components implementing these confinement barriers are assumed to have: *Design pressure ≥ 0.2 MPa* *Leak rate ≤ 1 volume%/day at 0.2 MPa*	*In analyses, components implementing these confinement barriers are assumed to have:* Glovebox atmosphere detritiation efficiency > 99% The set point for glove-box isolation > 80 VDO (see Table 3.1-12) The maximum glove-box overpressure 1000 Pa	*In analyses, components implementing these confinement barriers are assumed to have:* VDS detritation efficiency >99% . VDS Filter efficiency >99.9% HVAC Isolation within < 30 s The set point of room isolation and actuation of VDS > 80 VDO (see Table 3.1-12)
Gas injection fuelling process equipment, pellet injector (WBS 1.8)	Secondary container, with N_2 purge, around gas injection fuelling process equipment, pellet injector (WBS 1.8).	The secondary container with N_2 purge is connected to VDS (WBS 3.2).
Isolation valves between gas injection systems (GISs) and fuel supply system (SDS) in the Tritium Plant credited for limiting release from fuel supply lines.		
Fuelling line (WBS 1.8).	Secondary enclosure, with N_2 purge, around fuelling line	The secondary enclosure with N_2 purge is connected to VDS (WBS 3.2); with capacity sufficient to maintain a few kPa below atmospheric pressure.
Cryoforevacuum (WBS 3.1) line up to roughing system secondary enclosure.	Secondary enclosure with N_2 purge around forevacuum line (WBS 3.1) up to roughing system secondary enclosure.	The secondary container with N_2 purge is connected to VDS (WBS 3.2); capacity sufficient to maintain a few kPa below atmospheric pressure.
Isolation of roughing line (WBS 3.1) from vacuum vessel by single valve and Roughing system (WBS 3.1) process equipment.	Roughing systems (WBS 3.1) housed in secondary enclosure connected with VDS (WBS 3.2).	Room in Tritium building with connection to VDS (WBS 3.2); capacity sufficient to maintain 100 Pa relative to the atmospheric pressure.
Tritium plant process piping and equipment (WBS 3.2).	Glovebox or secondary enclosure with GDS (WBS 3.2).	Room in Tritium building with connection to VDS (WBS 3.2); Capacity sufficient to maintain –100 Pa relative to the atmospheric pressure.

[1] *Penetrations through confinement systems require double isolation capability such as automatic valves.*

Table 3.1-7 Systems implementing confinement for the hot cell building

<table>
<tr><th colspan="2">Hot Cell Building Confinement Systems[1]</th></tr>
<tr><th colspan="2">Hot Cell</th></tr>
<tr><th>1st Confinement system</th><th>2nd Confinement system</th></tr>
<tr><td>Receiving/Cleaning Cell (Sub-cell encloses filters and canister; connected to Hot Cell ADS with isolation valves in case of leak and connected to Hot Cell-VDS to maintain pressure differential at –300 Pa inside receiving/dust cleaning & storage cell with isolation if required for isolation of failures.)</td><td rowspan="5">Hot cell building provided with VDS and room air ventilation filter; capacity sufficient to maintain < 1atm (-100 Pa).

In analyses it is assumed:
VDS detritation efficiency >99%.
VDS Filter efficiency > 99.9%
HVAC Isolation within < 30s

The set point of room isolation and actuation of VDS > 80 VDO (see Table 3.1-12)</td></tr>
<tr><td>Storage Cell (leaktight, connected to Hot Cell ADS and connected to Hot Cell-VDS to maintain pressure differential at –300 Pa inside the cell; can be isolated in case of failure)</td></tr>
<tr><td>Hot Cell Repair and Testing (connected to Hot Cell ADS with isolation valves in case of leak, and connected to Hot Cell-VDS to maintain pressure differential at –300 Pa inside cell; with isolation if required for isolation of failures)</td></tr>
<tr><td>Radwaste Process and Storage (leaktight, connected to Hot Cell ADS and connected to Hot Cell-VDS to maintain pressure differential at –300 Pa inside the cell; can be isolated in case of failure), Tritium recovery vacuum furnace will be placed in this area.</td></tr>
<tr><td>HEPA Filter and Decay Heat Removal System Cells (connected to Hot Cell ADS with isolation valves in case of leak, and connected to Hot Cell-VDS to maintain pressure differential at –300 Pa inside cell; with isolation if required for isolation of failures)</td></tr>
<tr><th colspan="2">other rooms</th></tr>
<tr><th colspan="2">Confinement system</th></tr>
<tr><td colspan="2">Cask Docking Area
(leaktight, normal ventilation with HVAC, room depression system to maintain pressure differential at –100 Pa inside room, room depression system can be connected to VDS)</td></tr>
<tr><td colspan="2">Urgent RH Tool Exchange Holding & Repair Room
(leaktight, connected to Hot Cell-VDS to maintain pressure differential at –100 Pa inside the room; can be isolated incase of failure)</td></tr>
<tr><td colspan="2">Dust Cleaning Utility Room
(Air re-circulation blowers with filters are placed in tritium leaktight shielded secondary enclosure with N2 purge connected to HC-VDS; room connected to HVAC, room depression system and Hot Cell-VDS, maintain pressure differential at – 100 Pa, room can be isolated incase of failure).</td></tr>
</table>

[1] *Penetrations through confinement systems require double isolation capability such as automatic valves.*

Table 3.1-8 Systems implementing confinement for tokamak maintenance

<table>
<tr><th colspan="2">Sequential Confinement Barriers</th></tr>
<tr><td colspan="2">To limit source terms from mobilisable in-vessel inventories prior to maintenance:
• In-vessel surfaces shall be 50 oC at maximum
• In-vessel components TCWS shall be depressurised
• In-vessel detritiation of the co-deposited layers should have been carried out.
• In-vessel dust removal should have been carried out.
• In-vessel maintenance and transfer of in-vessel components should be carried out in air and pressures below adjacent room pressures.</td></tr>
<tr><th>1st Confinement system</th><th>2nd Confinement system</th></tr>
<tr><td>Detritiation of vacuum vessel atmosphere (WBS 3.2) in operation prior to opening to maintain tritium concentration in vacuum vessel low.

Remote handling casks (WBS 2.3)
In analyses it is assumed:
Design pressure ≥ 0.11 MPa
Leak rate <1 volume %/d</td><td>Port Cells and Hot Cell provided with VDS (WBS 3.2); capacity sufficient to maintain <1atm (-100 Pa).
In analyses it is assumed:
VDS detritation efficiency >99 %.
VDS Filter efficiency > 99.9%
HVAC Isolation within < 30 s
The set point of room isolation and actuation of VDS > 80 VDO (see Table 3.1-12)</td></tr>
</table>

3.1.3.4.5 Buildings

The following building volumes provide a physical envelope for confinement:

- Port cells
- Rooms containing vacuum vessel pressure suppression system and drain tanks
- Volumes containing piping for fuelling and vacuum pumping between port cells and tritium building
- TCWS Vaults, and connecting pipeshafts (to 0.2 MPa)
- NB Cells
- Tritium building
- Hot cell building

(1) Ventilation and filtration systems, and detritiation systems shall be provided to assure controlled and monitored releases. To accomplish this, and protect site personnel from air-borne contamination:
  (a) pressures within buildings that may receive radioactive leakages shall be kept at negative pressures relative to ambient;
  (b) air flow within the buildings shall be in the direction from lower to higher zones of contamination;
  (c) ventilation exhaust shall be through controlled and monitored release points.

(2) Designated areas within the buildings shall resist the effects of accidents such as pressurisation, explosion, pipe whip, fire, etc. if failure threatens safety equipment or workers.

(3) Designated areas within the building shall have passive and active features that prevent, detect, retard or extinguish fire which threaten or degrade safety-related components or worker safety.
(4) Structural integrity of buildings shall be ensured in case of underpressure, for example due to failure of vacuum boundaries.
(5) The control room shall remain habitable during accidents.

3.1.3.5 Protection of Confinement Barriers

3.1.3.5.1 Heat removal

The design shall provide reliable means to remove the heat generated during normal operation as well as the decay heat of activation products and the heat from potential chemical reactions, if this heat could lead to a challenge to a confinement barrier (including indirectly, for example by elevated temperatures causing a higher rate of a hydrogen-producing chemical reaction). Their reliability shall be commensurate with the subsequent impact on the confinement. Passive means for decay heat removal shall be provided as a last resort.

Passive, long-term, post-accident cooling modes shall be available, such as natural circulation capability of the vacuum vessel cooling system, to remove the maximum power transferred passively to the vacuum vessel under off-normal conditions. The vacuum vessel shall be cooled by two independent loops in such a way that failure of one of the loops will not lead to unacceptable vacuum vessel temperatures or unacceptable thermal stresses (alternating-sectors preferred).

Self-sustaining chemical reactions (such as steam reactions with plasma-facing components) shall be precluded by designing for sufficient heat transfer to colder parts of the machine. Alternating sectors cooled by different HTS loops enhances radiative cooling to non-affected blanket sectors.

3.1.3.5.2 Control of confinement pressure

To ensure confinement is not impaired, the design shall provide means to accommodate the accidental release of coolants, in particular those used for in-vessel components, vacuum vessel and superconducting magnets. For the magnets, due consideration shall be given to the fact that cryogenic fluids can absorb large amounts of energy from the ambient so that phenomena such as underpressure and overpressure can be generated.

A pressure suppression system shall be incorporated to reduce accidental overpressure in the vacuum vessel to an acceptable level in the event of an ingress of coolant into the vacuum vessel. Passive devices (e.g. rupture disks) shall be used in the flow path between the vacuum vessel and the vacuum vessel pressure suppression system tank. A drain line shall be provided to drain water from the vacuum vessel to drain tanks to limit long-term steam formation. The vacuum vessel pressure suppression system and drain tanks shall be located in a room such that the leakage can be vented through the vent detritiation system. The VVPSS shall be provided with a means (Suppression Tank Vent System, ST-VS) to remove non-condensible gases if present in accident sequences.

The Tokamak Cooling Water System (TCWS) vault shall be designed to confine and not to fail unpredictably or catastrophically in case of double-ended guillotine break during baking, with some means to relieve pressure if necessary. The TCWS vault will then also confine any accidents including a double-ended guillotine break during pulsed operation.

Overpressure relief to a closed vessel or process shall be provided for tritiated gases at cryogenic temperatures in the tritium plant.

The following strategy shall be implemented to deal with potential failures in cryogenic components:

- limit amount of cryogenic helium that can be spilt into the vacuum vessel, to assure reliable functioning of the VVPSS;
- limit amount of cryogenic helium that can be spilt into the cryostat;

3.1.3.5.3 Control of chemical energy

The design shall be such that chemical energy inventories are controlled to avoid energy and pressurisation threats to confinement. ITER shall be designed, in particular, to minimise hydrogen production during accidents, to avoid explosive mixtures of hydrogen with air/oxygen and to minimise the release of chemical energy as heat.

(Note Oct 2006: To avoid possible hydrogen/dust explosions a special prevention system is currently being designed and evaluated (DCR-52). This system would inject neutral gas inside the VV, in case of air and water ingress, to reduce the hydrogen and oxygen concentration at which deflagration of hydrogen can occur. If this system is implemented in the design it may be possible to relax the requirements related to hydrogen and dust in the following three paragraphs.)

Temperature and dust inventory limits (Table 3.1-4) shall be determined for specific first wall armour materials to limit the amount of H_2 generation due to heating of cryopumps and to Be dust water/steam reaction, to meet the following criteria:

- Pressure due to H_2 deflagration: < 200 kPa,
- H_2 detonation: no loss of confinement function.

Present analyses show that these requirements are met if the hydrogen mass inside the vacuum vessel remains below 4 kg. The vacuum vessel and other systems forming the first confinement barrier shall be designed to withstand loads from deflagration or detonation of 4 kg hydrogen in the vacuum vessel without loss of confinement function.

Excessive chemical reactions between beryllium and steam will be limited by avoiding elevated temperatures (see also Section 3.1.3). In case of loss of cooling function for the plasma facing components, overheating due to continued plasma burn shall be avoided by terminating fusion power.

A sufficient condition to ensure the 4 kg hydrogen production is not exceeded is to terminate the plasma burn before the FW PFC temperatures exceed 660°C (in the case of loss of cooling function) and to maintain decay heat driven long-term first wall temperatures < 385°C to limit the beryllium-dust reactions with steam.

Except for limited cases that will be identified and approved, the design shall ensure that two barriers exist between air and hydrogen, which may exist during normal operation or may be generated by accidents. The exceptions will be authorised only where there is a strong single barrier and/or hydrogen quantities are small. Isolation shall be provided to prevent air ingress into the vacuum pumping system, fuelling system or tritium plant in the event of air ingress into the vacuum vessel. Hydrogen pressure control shall be provided in case of loss of cryogenic refrigeration. *(Note: This will be reviewed).*

Cryogenic needs at locations where there may be a neutron flux or other ionising radiation should be met by helium, not liquid nitrogen. The use of nitrogen in any region subject to radiation must be justified by an analysis of the possible generation of ozone through radiochemical conversion of trace levels of oxygen.

3.1.3.5.4 Control of magnetic energy

For operational and investment protect purposes, magnet systems are designed and constructed to a very high degree of integrity and reliability. These requirements exceed those of safety considerations, but nevertheless it must be confirmed that failures in magnets shall not damage systems providing safety functions leading to a release of radioactivity exceeding the release guidelines.

3.1.3.5.5 Safety Related Auxiliary Systems

Systems that provide safety functions may need support services in order for them to function. These support services shall be designed and operated such that the intended safety function can be fulfilled when required.

1. The Class I, II and III power systems shall have sufficient generating or stored energy capacity to power SIC (safety importance component - see §3.1.4) loads when necessary, even if one of the emergency generators fails to start, or starts and fails to accept loads.
2. The maximum power interruption times shall be:
    a. Class I No time delay
    b. Class II Full load transfer within one-half cycle of the degraded power-sensing signal.
    c. Class III Full load transfer within specified time of the degraded power-sensing signal.
3. The electric power for all control systems shall be non-interruptible and the design shall follow single failure criteria if failure to provide power to the control system threatens public or workers safety.
4. The steady state power supplies shall provide remote controlled breakers and switchgear such that all major non-safety loads may be disconnected by the plant electrical control centre.
5. Class I, II power systems shall provide power for at least one hour to safety loads.
6. The Class I, II and III power systems shall have a reliability, which exceeds (tentatively) 0.999 per loss of power event.
7. The Class III power supply shall have sufficient on-site fuel to maintain full loads for a specified period.
8. Provisions shall be made to auto/manual-synchronize each emergency/backup power source to its bus for periodic testing.
9. Instrument air loads for equipment needed to maintain the confinement barriers in accidents shall be separated from other air load supply system.
10. The fire protection system has no direct nuclear safety related function but performs a safety role in mitigating event sequences in which fire could contribute to the potential release of radioactive or hazardous material.

3.1.3.5.6 Monitoring

The design shall provide means for monitoring and controlling radioactive or hazardous material releases as well as dose rates to the public around the site and in areas accessible to site staff. The design shall provide systems for assuring reliable information on all operational events and accidents, and for monitoring the performance of the confinement and its protection during accidents.

1. Status of Safety Important Component (SIC, see §3.1.4) systems shall be monitored under normal and off-normal conditions and displayed in the control room as required to ensure safety functions can be (are being) performed.
2. The release monitoring system shall provide measurement of tritium, beryllium and radiological particulate emissions/effluents.
3. The environmental monitoring system shall be capable of monitoring levels of less than a specified fraction of the daily project release guidelines for normal operation.

4. The radiological monitoring system shall provide monitoring and warnings for airborne tritium, airborne particulate and for gamma radiation field.
5. Monitoring shall be provided in rooms/areas under the Access Restriction System and shall confirm or override the logic of the Access Restriction System to allow or deny access based on actual measured conditions.
6. The Fixed Area Gamma Monitors shall have a minimum detectable level less than the specified minimum value.
7. The Fixed Area Tritium Monitors shall have a minimum detectable level less than the specified minimum value.

3.1.3.6 Requirements for Test Blankets

Strictly the only top level safety requirements for the test blanket programme are the ITER release and occupational dose guidelines.

For a consistent strategy to assure these top level requirements, ITER defines additional sets of project guidelines for the Test Blanket Module (TBM) inventories and requirements for TBM components and systems e.g. for the confinements.

For consistency with the general ITER safety approach, the TBM design shall consider the following requirements:

- All ex-vessel parts of cooling and other auxiliary system are part of the first confinement system.
- Decay heat removal should be achieved by thermal radiation to basic machine.
- Chemical reactions between coolant, air and breeder/multiplier material shall be limited so that confinement function is not threatened. Self-sustaining chemical reactions shall be precluded by designing for sufficient heat transfer to colder parts of the machine. The 2.5 kg limit for additional hydrogen production should not be exceeded.
- Special consideration for Li fires in local test module confinement shall be made.
- Intermediate cooling loops are necessary for liquid Li system.
- Helium spills to the vacuum vessel shall be limited to 45 kg (this limit applies to all in-vessel He releases to assure reliable functioning of the VVPSS).

For the safety analysis of the TBM' the following assumptions shall be taken into account:

- limit potential hydrogen production to 2.5 kg for each independent module (no common cause failure) by
    - Liquid Li shall be limited to less than 35 liters
    - PbLi should be limited to 0.28 m3. Alternatively, detailed analysis of water/PbLi interaction should be performed.
    - Beryllium of the first wall of test module should be limited to 10 kg (in addition to breeder multiplier inventory potentially producing hydrogen).
    - Potential beryllium-steam reactions in pebble-bed breeder designs need to be addressed (Hydrogen, Heat of reaction etc.).
- Test blanket port cells shall enhance confinement function for radiological inventories.
- The test blanket assembly cask (which stays in place during operation) should have a confinement function.
- Test blanket module shall be recessed 50 mm from the first wall of basic machine
- *Deviations from these assumptions must be justified by detailed accident analysis and agreed with ITER.*

Normal Operation

- Releases (leakage, permeation, maintenance) for one TBM shall be two orders of magnitude below the ITER annual release guideline guideline
- reduce radioactivity such that effluents are ALARA;
- monitor the effluents.
- The ITER zoning (contamination level, direct radiation) of port cells, TCWS vault and Tritium plant according PID section 3.1.5 should not be affected seriously by the TBM.

To assure that the radiological requirements are met, through the entire life cycle of ITER, a radiation protection program (RPP) shall be developed and implemented.

3.1.4 Component Classification

System and components important for personnel or public safety (Safety Importance Class, SIC) are identified in Table 3.1-9 based on the functional importance to safety of the components. Unless otherwise specified, components are "not safety classified".

Table 3.1-9 Safety Importance Components (SICs) and Systems

Active systems are in **bold** type

System	SIC providing a Safety Function (Rationale for Classification)
WBS 1.1 Toroidal Field (TF) Coils System	Feeders through the cryostat that provide a confinement function and magnet gravity supports
WBS 1.2 Poloidal Field (PF) Coils System	Feeders through the cryostat that provide a confinement function
WBS 1.3 Central Solenoid System (CS)	Feeders through the cryostat that provide a confinement function
WBS 1.5 Vacuum Vessel	Main vessel, ports etc that provide a confinement and/or decay heat removal function. VV Gravity supports.
WBS 1.6 Blanket	No safety function, but as a potential initiator of events, failure frequency should be minimised.
WBS 1.7 Divertor	Otherwise, no safety function but as a potential initiator of events, failure frequency should be minimised.
WBS 1.8 Fuelling and Wall Conditioning	Process components and secondary containers that provide a confinement function. **Components needed to terminate plasma under TCWS abnormal conditions.**
WBS 2.3 Remote Handling Equipment	Casks, cask doors, other confinement barrier components in remote handling equipment that provide a confinement function.
WBS 2.4 Cryostat	Cryostat vessel and penetrations that provide a confinement function, and associated leak, ice and ozone detection equipment.
WBS 2.4.E Vacuum vessel pressure suppression system and drain lines tanks (WBS 2.6)	Components that limit pressure in vacuum vessel under accident conditions and provide confinement of in-vessel source terms during accident conditions
WBS 2.6 Cooling Water Systems	Components providing a confinement function
WBS 2.6.E Vacuum vessel primary heat transport system	Components providing a confinement function and components that provide decay heat removal of in-vessel components by natural circulation under accident conditions.
WBS 2.6.P Chilled water system including associated heat rejection	**Components needed to supply chilled water to safety important TCWS vault coolers, vent detritiation systems and Tokamak Venting System**

System	SIC providing a Safety Function (Rationale for Classification)
WBS 3.1 Vacuum Pumping and Leak Detection Systems	Components providing a confinement function
WBS 3.2 Tritium Plant	Components providing a confinement function
WBS 3.2.F Atmosphere Detritiation Systems	**Components of Normal Vent Detritiation System, Standby Vent Detritiation System, Hot Cell Vent Detritiation System and Tokamak Vent System, providing filtering, detritiating and pressure control functions for confinement envelopes**.
WBS 3.2.N Ventilation and Confinement Systems	Room depression systems composed of fire proof air cooler, HEPA filter and air fans. The system can be **switched** from plant exhaust stack to VDS in case of simultaneous event of fire and tritium contamination.
WBS 3.4 Cryoplant and Cryodistribution	Components providing confinement function (as part of systems supplied) and providing isolation functions to limit releases of cryogens into confinement.
WBS 4.1 Coil Power Supplies	Components to **detect faults and switch off the PF** coil power supplies (Fault detection components may be part of magnet system). *(TF is too small to drive significant current, but PF has higher voltage and energy, and can sustain an arc if not disconnected).*
WBS 4.3 Steady State Electrical Power Network	A Class I, II, III power sources, its connections to the distribution system and its activation sensing equipment shall be a Safety Importance Component if the system to which they provide power is a Safety Important Component.
WBS 4.6 Interlocks System	**Components of Interlock System if required for protection of confinement; Access control for occupational safety; Fusion Power Shutdown System to terminate plasma under TCWS abnormal conditions**
WBS 5.1 Ion Cyclotron H&CD	Components providing a confinement function
WBS 5.2 Electron Cyclotron H&CD	Components providing a confinement function
WBS 5.3 Neutral Beam H&CD	Components providing a confinement function **and interlock system to prevent damage to the VV**
WBS 5.3 Lower Hybrid H&CD	Components providing a confinement function
WBS 5.5 Diagnostics	Components providing a confinement function
WBS 5.6 Test Blanket	Components providing a confinement function. Possible additional safety functions depending on materials employed and any additional hazards introduced in TB design. As potential initiator of events, failure frequency should be minimised (analysis will provide justification if FPTS is required).
WBS 6.2 Buildings	Areas within Tokamak Building, Tritium Building and Hot Cell which form confinement envelopes. TCWS vault cooler to restore pressure in vault sub-atmospheric. Components to avoid excessive over or under-pressure in confinement envelopes.
WBS 6.3 Hot Cell Processing and Waste Treatment	Components providing a confinement function
WBS 6.4 Radiological Protection	Components for Radiological Monitoring for occupational safety; **Final monitoring system of environmental releases via stack.**
WBS 6.5 Liquid	**Fire protection system for protection of confinement barriers.**
WBS 6.6 Gas Distribution	A gas distribution component shall be a Safety Importance Component if the system to which the system provides the service is a Safety Important Component.
WBS 6.7 Plant Sampling Systems	**Measure radioactivity in HRS and provide signal to isolate HRS from HTS on high levels.**

The design of SICs shall include all loading events for which the component performs a safety function (refer to the "Load Specification and Combination"[1]), in particular those relating to earthquakes as indicated in Table 3.1-10. Damage to non-SIC should not lead to failure of SICs or prevent them from performing safety functions.

Table 3.1-10 System Seismic Requirements

WBS	System	Safety Requirements during and following SL-2 Earthquake
1.1-1.3	Magnet Systems	-no damage to vacuum vessel or cryostat confinement barriers.
1.5	Vacuum Vessel	-leakage from vacuum vessel no greater than that assumed in safety analysis (see Table 3.1-5)
1.6 1.7	Blanket Divertor	-in-vessel coolant leakage limited; no loss of structural integrity leading to large leakage.
1.8 3.1 3.2	Fuelling and Wall Conditioning Vacuum Pumping and Leak Detection Tritium Plant	-no significant leakage of activity from system to rooms -vent detritation systems (Normal and Standby VDSs) continue to function; interruption during earthquake acceptable; must be able to be restarted.
2.3	Remote Handling Equipment	-no significant leakage of activity from system to rooms
2.4	Cryostat	-no significant leakage into/from system -Vacuum vessel pressure suppression system functional
2.6	Cooling Water System	-no significant leakage from system -chilled water to VDS continues; interruption during earthquake acceptable; must be able to be restarted.
4.1	Coil Power Supply & Distribution	-ability to switch off coil power supplies remains functional during and after earthquake; detection to switch off power to PF coils and related circuit to be functional during earthquake or triggered by seismic monitoring. *(TF is too small to drive significant current, but PF has higher voltage and energy, and can sustain an arc if not disconnected).*
4.3	Steady State Power Supplies	-ability to provide power to systems providing safety function retained; interruption during earthquake acceptable; must be able to be restarted.
4.5 4.6 4.7	Command Control and Data Acquisition Interlocks Poloidal Field Control	-safety interlocks remain operational
5.1 5.2 5.3 5.4 5.5	Ion Cyclotron H&CD Electron Cyclotron H&CD Neutral Beam H&CD Lower Hybrid H&CD Diagnostics	-no significant leakage of activity from system into rooms
5.6	Test Blankets	-no significant leakage of activity (or lithium, if applicable) from system into rooms
6.2 6.3	Buildings Hot Cell	-vent and clean-up systems are functional
6.4	Radiological Protection	-ability to monitor (estimate) releases from site retained -radiation protection monitoring (possibly portable) available
6.5	Liquid Distribution	-fire protection available following earthquake

Design rules and standards shall be selected for each system or component in consideration of SIC using the guidelines in Table 3.1-11

[1] ITER_D_222QGL

Table 3.1-11 Guidelines Related to Safety Importance Components

Issue	Guideline for SIC
Design (use of codes and standards, degree of conservatism, margins, etc.)	• If an appropriate design code exists, the code requirements for design, construction, testing etc. should be followed. Deviations from code requirements to be documented. • As a baseline for structural components ASME VIII or ANSI B31.3 (lethal service or Service Category M) and related codes may be used. • Where an appropriate design code does not exist, an agreed surrogate developed specifically for ITER may be used • Prototype/non-code items require testing, proven and documented manufacturing process, control of materials, etc. • Standard commercial components acceptable if appropriate to conditions of use.
Materials (restrictions on which materials can be used, extent of testing, sources of data, margins in data, etc.)	• Materials to be specified and compliance ensured. • Materials in standard commercial component may be acceptable if appropriate to conditions of use.
Fabrication and Installation (manufacturing process qualification, weld types, welding procedures and welder qualification, etc.)	• Manufacturing, assembly and installation process/procedures to be specified and compliance ensured. • Compliance with design code requirements (if applicable). • Standard commercial component fabrication may be acceptable.
Inspection and Examination (extent of inspection, third party or owner, non-destructive examination, etc.)	• Inspection and examination to ensure safety function to be specified and compliance ensured. • Compliance with design code requirements (if applicable).
Testing (pressure testing, performance testing,)	• Testing required to demonstrate safety function to be specified and compliance ensured. • Compliance with design code requirements (if applicable).
In-Service or Periodic Inspection (inaugural, frequency and extent of in-service tests)	• In-service inspections, monitoring and/or tests to ensure that the equipment can continue to provide its safety functions with the required level of reliability. • Test records, calibration records, personnel training requirements, etc. to be specified as part of the normal maintenance procedures.
Quality assurance	• ITER Quality Management requirements. • Records and documentation as required for compliance with design codes and to demonstrate compliance with specifications.
Environmental qualification	• Justification to be provided that component can withstand the abnormal environmental conditions that may arise from an accident for which their operation is needed.
Reliability	• System to perform safety function even with single active fault/failure (or alternative system available to provide the safety function). • Use of proven, good industrial quality components may suffice as a justification.
Independence, physical separation	• As required to ensure safety function cannot be undermined by underlying common cause or cascading failures. • Protective I&C for a system should be separate and functionally isolated from normal instrumentation for that system (separate signal channels appropriately de-coupled and shielded), and with physical separation between redundant channels.
Equipment status indication	• Status under normal conditions and functioning of system under emergency use available to operators, possibly at remote location.
Equipment data trail (during operation)	• Maintenance logbooks to record operational history. • Inspection status must be indicated (tag or inspection record).
Equipment outage	• Maintenance and other outages to be covered by operating procedures and a set of 'limiting conditions for operation' to be determined such that performance and reliability requirements can be met.
Response and recovery after malfunction	• Covered by plant procedures. • Principles for resuming operation established (e.g. 'limiting conditions for operation') in advance

Issue	Guideline for SIC
Reporting of malfunctions	• Malfunctions impeding safety function to be reported. • Annual logbook available for audit.
Modification of equipment and retrofit	• Impact on safety functions needs to be analysed.

3.1.5 Zoning

Zoning for radiation hazards is defined according to French regulations as shown in Table 3.1-14.

Radiological zoning is a very important tool for the protection of workers against the danger of ionising radiation. According to the host state regulations :

A supervised zone is defined if, in normal work conditions, the workers could be exposed to a total dose greater than 1 mSv/y or 1/10 of the annual limits set for equivalent dose to skin, hands, feet and eye lens (crystalline).
A controlled zone is defined if, in normal work conditions, the workers could be exposed to a total dose greater than 6 mSv/y or 3/10 of the annual limits set for equivalent dose to hands, feet and eye lens (crystalline). Specially restricted areas may be defined inside controlled zones.

The zoning procedure should refer to a job description. A job description must anticipate the potential dose (irradiation + contamination) for each different working position (for contamination doses the breathing protection is not taken into account for the zoning).

As shown in Table 3.1-12, up to the higher limit of green zone (25 µSv/hr) the zones are defined on a basis of hourly dose rates. These dose rates are not instantaneous rates but apply to the potential dose being delivered in 1 hour of time in normal conditions of work. For the yellow, orange and red zones the instantaneous dose rate should also be considered.

The zoning is usually done according to fixed walls, though a clear fence system can be used, even outside the buildings as long as the surface used for these extension are under the control of the operator. These possibilities are not allowed for in “specially controlled area” that is yellow and orange.

The classification of workers is independent from zoning.

In the principle the dose limits for zoning apply to irradiation doses and to internal doses, the equivalence being derived from inhalation dose factors, breathing rate and so on. Individual protection devices are not taken into account in these calculations.

The atmospheric contamination is expressed as a number of VDO (Derived Operational Value).

One VDO for one radionulide or a mix of radionuclides leads to an internal dose rate of 25 microSv/hr. Note than this concept is different from the former ITER DAC (1 DAC = 20mSv/year or 10µSv/hour).

If the source of radiation is not continuous, in the case of an electrical particle generator for example, the zoning can evolve according to the machine operation. Then a specific signalisation may be set up with lights, sound and eventually automatic locking of accesses.

Table 3.1-12 Radiological zoning according to total doses

RADIOPROTECTION ZONING			External exposure	Internal exposure Atmospheric contamination	
TYPE		COLOUR		Total dose in 1 hr	*Units of VDO (1)*
Non regulated area			Total dose < 80 µSv/month	-	*No atmospheric contamination*
Supervised		Blue	Total dose in 1 hour < 7.5 µSv	< 7.5 µSv	*< 0.3*
Controlled		Green	Total dose in 1 hour < 25 µSv	< 25 µSv	*< 1*
	Specially regulated	Yellow	Total dose in 1 hour < 2 mSv And dose rate < 2 mSv/h	< 2 mSv	*< 80*
	Specially regulated	Orange	Total dose in 1 hour < 100 mSv and dose rate < 100 mSv/h	< 100 mSv	*< 4000*
	Restricted	Red	Total dose in 1 hour > 100 mSv or dose rate > 100 mSv/h	> 100 mSv	*Greater than 4000*

Note 1 : VDO (Derived Operational Value (fr)). 1 VDO for one radionulide or a mix of radionuclides leads to an internal dose rate of 25 microSv/hr. Note than this concept is different from the former ITER DAC (1 DAC = 20mSv/year, 10µSv/h). 1 VDO of tritium is 0.772 MBq/m^3, T as HTO, including 1/3 skin transfer.

Table 3.1-13 Radiological zoning according to equivalent doses to hands and feet

RADIOPROTECTION ZONING			External exposure to hands and feet
TYPE		COLOUR	
Supervised		blue	Equivalent dose in 1 hour < 0.2 mSv
Controlled		green	Equivalent dose in 1 hour < 0.65 mSv
	Specially regulated	yellow	Equivalent dose in 1 hour < 50 mSv
	Specially regulated	orange	Equivalent dose in 1 hour < 2500 mSv
	Restricted	red	Equivalent dose in 1 hour > 2500 mSv

Table 3.1-13 will be used when the external exposure to hands and feet supersedes the total exposure. In case of exposure of the eye lens (crystalline), values indicated in Table 3.1-13 should be multiplied by 0.3 (150/500).

The marking (signalisation) of the zones from greyish blue to red should follow the norm MF M 60-101. It should clearly posted on all accesses to the zones. The marking of the zones should be modified according to every evolution of the zoning.

Inside the zones the individual sources of radiation should be clearly indicated. This does not apply to orange and red zones, but in this case a document should be given to the workers during the preparation time prior to the intervention.

If it is necessary to wear an individual protection equipment in a regulated zone, This should be clearly indicated.

Food, tobacco, personal cosmetics, handkerchieves are not allowed when a contamination risk may exist.

The changing rooms should be separated in two parts, one for civil clothes, the other one for work clothes. Showers and sinks will be available.

If there is a risk of contamination, the surfaces should be made of materials easy to decontaminate. The maximum level of labile (loose) surface contamination will be determined through a specific study.

3.1.5.1 Workers classification

The classification of workers is in independent from zoning. The workers can be classified in three categories:

A classification: workers whose exposure to ionising radiation is liable lead to a dose greater than 6 mSv a year, or greater than 3/10 of the other annual limits (skin, hands, feet and eye-cristalline). Pregnant women, nursing-mothers and students between 16 and 18 cannot be classified A

B classification: workers expose to ionising radiation and not classified A.

All others are said to be "**non A – non B**", and work in radioactive areas is studied case-by-case. Pregnant women should not work in radioactive areas as soon as pregnancy is declared

In order to be classified A or B, the workers must have an analysis of the risks in their jobs, an adequate training and a medical follow-up.

Table 3.1-14 Exposure limits for workers*

<table>
<tr><th>Part of body exposed</th><th>Dose type</th><th>A</th><th>B</th><th>No A, no B</th><th>Pregnant women</th><th>Nursing Mothers</th><th>People between 16 and 18 *</th></tr>
<tr><td></td><td></td><td colspan="3">For 12 consecutive months</td><td></td><td></td><td></td></tr>
<tr><td>Whole body</td><td>efficient</td><td>20 mSv</td><td>6 mSv</td><td>1 mSv</td><td rowspan="4">Avoid exposure (if not less than 1 mSv to the fœtus.)</td><td rowspan="4">No contamination risk allowed</td><td>6 mSv.</td></tr>
<tr><td>skin</td><td>Equivalent (surface 1 cm²)</td><td>500 mSv</td><td>150 mSv</td><td>50 mSv</td><td>150 mSv</td></tr>
<tr><td>Eye Cristalline</td><td>Equivalent</td><td>150 mSv</td><td>45 mSv</td><td>15 mSv</td><td>50 mSv</td></tr>
<tr><td>Hands, arms, ankles, feet</td><td>Equivalent</td><td>500 mSv</td><td>150 mSv</td><td></td><td>150 mSv</td></tr>
</table>

* Workers means people with full employement contracts; others (training, students, interim) should not have access to orange and red areas

In order to acces the radiologically regulated zones the workers should have ;

- Adequate training, renewed as necessary, and at least every three years,
- Analysis of the risks at their job; this analysis is given to the health at work medical staff, in order to determine the medical follow up,
- Adequate working clothes and if necessary individual protection equipment,
- Adequate dosimeters, both passive (film, TLD…) and active (electronic dosimeters with alarm thresholds).
- For the controlled areas, an analysis of the work to be performed, stating dose objectives, ALARA studies, maximum residence time and other relevant data.

- Temporary workers and students, are not allowed in orange and red zones.

Table 3.1-15 lists the beryllium zones that shall be established as required to control access to locations and work with beryllium contamination.

Table 3.1-15 Beryllium Zoning

Zone	Hazard Level		Access and Control Conditions
	Airborne	Surface	
Uncontrolled Zone	$<0.01 \mu g/m^3$	$< 0.1 \mu g/m^2$	No reasonable possibility for beryllium exposure. Time unlimited access areas with no protective or monitoring devices. Control Rooms, ordinary offices, all other areas not directly or indirectly connected to operations with beryllium.
Controlled Zone	$0.01 \mu g/m^3$ – $0.2 \mu g\text{-}/m^3$	$0.1 \mu g/m^2$ - $10 \mu g/m^2$	Beryllium measured or anticipated. Only designated beryllium workers are allowed access. Access time and protective equipment will be determined by the activities with beryllium bearing equipment and the potential for airborne beryllium.
Respiratory Protection Zone	$> 0.2 \mu g/m^3$	$> 10 \mu g/m^2$	Airborne beryllium is either measured or expected to exceed levels requiring respiratory protection. These areas must be physically enclosed and outfitted with appropriate ventilation to ensure the areas outside this zone do not experience elevated beryllium due to the work within the zone. Workers will require respiratory protection commensurate with the work and the hazards measured.

Regulatory conditions for exposure to magnetic fields are set out in Table 3.1-16 , and derived from these, zones and conditions for worker entry to areas with electromagnetic fields are defined in Table 3.1-18.

Table 3.1-16 Static magnetic field limits

	Regulatory or recommended thresholds	Conditions of exposure
Workers	200 mT (2,000 Gauss)	8 hours non-stop/ day Entire body
	2 T (20,000 Gauss) $(dB/dt) < 3$ T/s for $t > 10$ ms $(dB/dt)^2.t < 0.09$ T^2/s for $t < 10$ ms	Temporary Entire body Comply with the conditions opposite, including temporal variations of the magnetic field (e.g. during power interruption or device switch-on)
	5 T (50,000 Gauss)	Duration < 5 min/h Body extremities
	> 10 mT (100 Gauss)	The medical care office is to be informed in the case of regular exposure
General public	40 mT (400 Gauss)	Non-stop
	> 40 mT	Access occasionally when authorised under controlled conditions and without exceeding the related “worker” threshold

Table 3.1-17 Thresholds for other areas that should be properly signed

Informing people of possible exposure	0.5 mT (5 Gauss)
Hazard for people with active implants (e.g. pacemakers)	0.5 mT
Hazards represented by airborne metal objects attracted by magnetisation (missile effect)	3 mT (30 Gauss)
Risk of deterioration of magnetic supports (analogue watches, credit cards, magnetic tape, floppy disks, etc.)	From 1 to 10 mT (10 to 100 Gauss)

Table 3.1-18 Electromagnetic Field Zones and Conditions

Zone Name	B [mT]	Access and Control Condition
Uncontrolled Zone	< 10	Unlimited access.
Controlled Zone	> 10,< 200	Access is limited to 8 hours per day
Prohibited Zone	> 200	Entry is prohibited, aside from exceptional circumstances where temporary access may be allowed with a high level of approval

The radio-frequency exposure for personnel working in areas adjacent to sources of hazard should comply with the limits recommended by the International Non-Ionising Radiation Committee (INIRC). The table below summarizes the exposure limits for workers expressed as Equivalent Power density for plane waves.

Table 3.1-19 Guidelines Related to Radio-Frequency Powers

Power density for ECRH and LHCD waves	< 5 mW/cm^2
Power density for ICRH waves	< 1.0 mW/cm^2

3.1.6 Normal Operation

As part of efforts to ensure effluents are reduced to levels as low as reasonably achievable (ALARA):

(1) Radioactive gaseous effluents shall be released through the vent detritiation system (efficiency > 99 % for tritium) to control releases.
(2) Tritiated liquid effluents shall be released through the water detritiation system when needed to limit effluents.
(3) Means shall be provided to detect a leakage of coolant from a primary heat transfer system loop into the vault, so that the vault can be isolated to avoid exceeding release guidelines for Incidents:
leakage > 10 kg/s should be detected within 1 hour
leakage > 10 g/s should be detected within 24 hours.

3.1.7 *Radioactive Waste*

Systems and components design as well as materials selection should aim at reducing the volumes and radiotoxicity of materials which may remain as long-term waste after decommissioning.

This should be achieved in particular by modular design of components and by limitation of impurities in the materials with regard to their relevance for clearance.

3.1.8 *Occupational Safety*

3.1.8.1 ALARA in design

The radiological safety of workers must be defined after an optimisation procedure. The key issue is to demonstrate that studies have been made to maintain the doses As Low As Reasonably Achievable (ALARA), considering technical, social and economical factors. The ALARA procedure is an iterative and continuous process, described in Figure 3.1-1 hereafter.

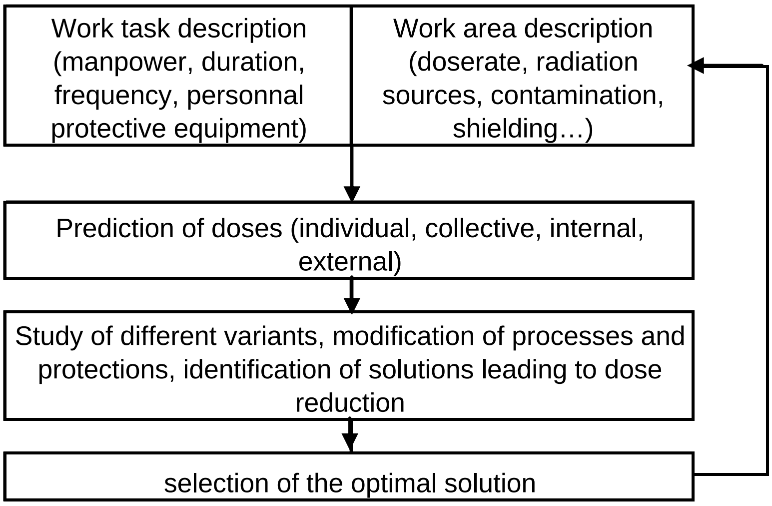


Figure 3.1-1 Outline of the ALARA procedure

It is important to state that all doses should be justified, and therefore the personnel should enter a regulated area only if their job requires it. The Access Restriction System shall provide physical interlocks for personnel protection.

The risk analysis and ALARA procedure have to be conducted both at a collective level, for example in the design of shielding and ventilation, and at individual level for each exposed work task.

At the start of the ALARA process, it is not mandatory but useful to define a general dose target, which is more or less arbitrary and acts as a constraint, and a description of the work areas.

For ITER the collective annual worker dose, averaged over the operational life time should not exceed an annual target of 0.5 person-Sv, and no individual doses should be greater than 10 mSv/year. This is in the range of the annual doses generated by operating a recent fission nuclear power reactor.

For the work areas description, the present zoning plans both for radiation and contamination may be a good starting point. The necessary adjustment of the zoning to the French regulation (see chapter 3.1.5 Zoning), will have to be linked to the global ALARA process.

The ALARA process should be implemented in priority for the most “dosing” operations. Therefore the first step of all should be to list, room by room, the tasks needing hands on operation and to make a preliminary dose assessment. Human factors considerations (such as visibility, ergonomics, etc.) shall be applied to this work task description.

A screening level for a thorough ALARA process could then be a collective dose greater than 5 % of the annual target, and an individual dose greater than 1 mSv.

The protection against the other risks (beryllium, cryogens, magnetic) may benefit from a similar optimisation procedure.

3.1.8.2 ALARA in operation

Before any work in a facility an analysis of the risks must be performed for each work task description, and the adequate protective measures have to be decided upon. If repetitive operations are concerned, this analysis has to be done at least once a year. If a one time work is concerned, the analysis has to be performed before the beginning of work.

In this analysis, an ALARA procedure should be performed in order to optimise the doses with some screening threshold to be decided. In this risk analysis it is necessary to consider also all the other risks and hazards, fire, lack of oxygen, mechanical, chemical, fall, electrical, cryogenic, electromagnetic …

A special emphasis will be put upon the operations needing the intervention of more than one team, (co-activity), possibly belonging to different organisations. Then the interaction and responsibilities of each team should be defined as well as the general coordination and the communication procedures between the teams.

3.1.9 Assessment

Safety assessments shall be an integral part of the design process and the results will be available to assist in the preparation of the safety documentation for regulatory approval. These analyses shall comprise normal operation, incidents and accidents, and waste characterization.

3.1.9.1 Effluents

An assessment shall be made of potential effluents from the ITER site throughout its lifetime. All effluents (airborne and waterborne) shall be identified and their quantity and characteristics estimated. Effluent assessment shall address normal operation and maintenance and shall include, as a minimum, radioactive materials, hazardous materials, direct radiation, magnetic fields, and thermal emissions. Releases of radioactive materials shall be assessed against the guidelines in PDS as part of a demonstration that such releases are ALARA.

3.1.9.2 Occupational Safety

An assessment shall be made of the work to be performed during operation, maintenance, and repair. Specifically, an assessment shall be made of both the accessibility and the estimated exposures for activities, against the radiological limits and guidelines in PDS, and in section 3.1.8 above, and against recognised limits of exposure to conventional (non-nuclear) hazards.

3.1.9.3 Plant level analysis

The objective of the plant level sequence analysis is to support the choice of the sequences analysed in the reference accident analysis and in the ultimate safety margins analysis, and to demonstrate, for a comprehensive set of event sequences, that the consequences of each sequence will be below the release guidelines established for the category (Incidents, Accidents) to which the sequence belongs.

Further, the plant level analysis should support the identification of Safety Importance Components.

A set of Reference Events are selected using a deterministic approach, based on coverage of all major systems, all major inventories, and all types of event initiator. To approach completeness as far as possible, a systematic identification procedure shall also be applied: Postulated initiating events (PIEs) should be identified by a systematic 'bottom-up' method like the Failure Modes and Effects Analysis (FMEA) as well as by a 'top-down' approach like Global Event Trees or Master Logic Diagrams. It can then be ascertained that all identified PIEs are enveloped by one of the Reference Events.

A combination of top-down and bottom-up approaches to event sequence identification is to be provided by means of a matrix, called the PIE Potential Impacts Table (PIE-PIT), which links all PIEs identified by the FMEA studies with one or more Plant Confinement States. These are damage states in which a series of separate confinement failures lead to the potential for an off-site release. By illustrating the principal radioactive inventories and the sequence of confinement systems that protect them, the top-down part of the PIE-PIT shows how a limited number of Plant Confinement States can represent all conditions in which a release is possible. The bottom-up part shows, for every PIE, what sequence of aggravating failures would need to occur for such a State to be reached. This can then lead to the selection of representative event sequence to characterise every State. Such sequences are categorised incidents, accidents, or hypothetical events.

3.1.9.4 Reference Events

The overall approach for safety analysis shall follow a deterministic framework, where necessary complemented by probabilistic assessment, to analyse a set of 'reference events' (limited in number) which shall encompass the entire spectrum of Incidents and Accidents.

The reference events shall account for both initiating events from inside the facility and from external events such as natural phenomena and man-made hazards.

For internal initiating events, the reference event analysis shall apply the following deterministic rules:

1. conservative analysis to cover uncertainties;
2. combination of the initiating event with loss of electric power supply;
3. consideration of an additional independent failure of one component in a system required to accommodate the consequence of the initiating event;
4. the effect of unavailability due to maintenance;
5. the above combination must be assumed to occur at the most unfavourable time in the course of the event.

The superposition of the aggravating failures is a convenient practice to envelop miscellaneous but similar sequences and to account for uncertainties in the postulated sequences vs. real situations. If results from probabilistic assessments suggest another approach, the deterministic rules do not prevent alternative assumptions.

In general, systems not classified as SIC (§3.1.4) cannot be credited in the analysis of Accidents.

For external initiating events, the reference event analysis shall apply the following deterministic rules:

1. consideration of the natural phenomena or man-induced events which are characteristic for the site with sufficient margin to accommodate limited accuracy;
2. combinations of normal operation and event conditions with the natural phenomena and man-induced events;
3. the combination of an earthquake with the failure of seismically qualified components shall be excluded.

The reference events shall be analysed to determine the detailed transients and consequences and to define the associated load conditions and design requirements for the systems important to safety.

3.1.9.5 Hypothetical sequences

Hypothetical sequences shall be used to investigate the ultimate safety margins. The intent is to demonstrate the robustness of the safety case with regard to the project's objectives and radiological requirements. The investigation shall provide insight into the magnitude of consequences in the hypothetical situations. This insight has the potential for identifying additional facility features that could prevent or reduce severe hypothetical consequences. Key notions in the context of hypothetical sequences are:

1. avoidance of 'cliff edge effects', i.e. sharp increases in consequences of Accidents when additional failures are postulated;
2. 'no-evacuation' to help demonstrating the safety and environmental potential of fusion power;
3. show robustness of defense-in-depth.

The hypothetical sequences shall be investigated by taking into account the intrinsic fusion safety features, the design provisions, best estimate data and measures to mitigate radiological consequences. The uncertainties due to data and tools shall be identified.

3.1.9.6 Waste

An assessment of waste arising during operations and decommissioning shall be made to provide a detailed characterisation. The assessment should provide information on:
1. timing and quantities of expected mass and volume of radioactive and hazardous waste;
2. strategy and schedule for decommissioning;
3. waste characteristics (activity, contact dose rate, decay heat, dominant isotopes, contamination, clearance approach) as a function of time.

3.2 Plasma Operation Scenarios

Variants of the nominal scenario are designed for plasma operation with extended-duration, and/or steady-state modes with a lower plasma current operation, with H, D, DT and He plasmas, potential operating regimes for different confinement modes, and different fuelling and particle control modes. Flexible plasma control should allow for "advanced" tokamak scenarios based on active control of plasma profiles by current drive or other non-inductive means.

Four reference scenarios are identified for design purposes. Three alternative scenarios are specified for assessment purposes where it shall be investigated if and how plasma operations will be possible within the envelope of the machine operational capability with the possibility of a reduction of other concurrent requirements (e.g. pulse length).

Design 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

Assessed scenarios:

(5) Inductive operation III: 700 MW, 17 MA operation, with heating during current ramp-up.
(6) Non-inductive operation II: strong negative shear operation
(7) Non-inductive operation III: weak positive shear operation

All these scenarios are summarized in the following tables.

Table 3.2-1 Design Scenarios and Main Parameters (During Burn)

Parameter	1.Inductive operation I	2.Inductive operation II	3.Hybrid operation	4.Non-inductive operation I
R/a (m/m)	6.2 / 2.0	6.2 / 2.0	6.2 / 2.0	6.35 / 1.85
Volume (m^3)	831	831	831	730
Surface (m^2)	683	683	683	650
Sep. length (m)	18.2	18.2	18.2	16.9
Cross-section (m^2)	21.9	21.9	21.9	18.7
Toroidal field, B_T (T)	5.3	5.3	5.3	5.18
Plasma current, I_P (MA)	15.0	15.0	13.8	9.0
Elongation, κ_x/κ_{95}	1.85 / 1.7	1.85 / 1.7	1.85 / 1.7	2.0 / 1.85
Triangularity, δ_x/δ_{95}	0.48 / 0.33	0.48 / 0.33	0.48 / 0.33	0.6 / 0.4
Confinement time, τ_E (s)	3.4	3.7	2.7	3.1
$H_{H\text{-}IPB98\ (v.2)}$	1.0	1.0	1.0	1.57
Normalised beta, β_N	2.0	1.8	1.9	3.0
Electron density, $\langle n_e \rangle$ ($10^{19}m^{-3}$)	11.3	10.1	9.3	6.7
f_{He} [%]	4.4	4.3	3.5	4.1
Fusion power, P_{fus} (MW)	500	400	400	356
P_{add} (MW)	50	40	73	59
Energy multiplication, Q	10	10	5.4	6
Burn time (s)	400	400	1000(1)	3000(1)
Minimum repetition time (s)	1800	1800	4000	12000
Total heating power, P_{TOT} (MW)	151	121	154	130
Radiated power, P_{rad} (MW)	61	47	55	38
Alpha-particle power, P_α (MW)	100	80	80	71
Loss power, P_{loss} (MW) (conduction)	104	87	114	93
L-H transition power, $P_{L\text{-}H}$ (MW)	51	48	45	36
Plasma thermal energy, W_{th} (MJ)	353	320	310	287

Notes

(1) The extended burn under the hybrid and non-inductive operations may be accomplished with additional investment for auxiliary systems.

Table 3.2-2 Design Scenario 1: Inductive Operation I

Phase	XPF[(1)]	SOH[(1)]	SOF/B[(1)]	EOB[(1)]	EOC[(1)]
t (s)	30	70	100	500	560
I_P (MA)	7.5	13	15	15	12
P_{add} (MW)	0	50	50	50	0
nom. $<n_{e,20}>$	0.25	0.4	1.15	1.15	0.4
q_{95}	5.3	3.2	3.0	3.0	6.4
q_0	1.0	1.0	1.0	1.0	1.0
nom. li	0.85	0.85	0.85	0.85	1.0
min. li	0.7	0.7	0.7	0.7	0.8
max. li	1.2	1.0	1.0	1.0	1.2
nom. β_P	0.1	0.1	0.7	0.7	0.1
max. β_P	0.1	0.1	0.8	0.8	0.1
nom. V_{loop} (V)	–	–	0.075	0.075	–
Notes (1) XPF: X-point formation, SOH: start of heating, SOF/B: start of flat top/burn, EOB: end of burn, EOC: end of cooling					

Table 3.2-3 Design Scenario 2: Inductive Operation II

Phase	XPF	SOF[(1)]	SOB[(1)]	EOB	EOC
t (s)	30	100	130	530	590
I_P (MA)	7.5	15	15	15	12
P_{add} (MW)	0	0	40	40	0
nom. $<n_{e,20}>$	0.2	0.4	1.0	1.0	0.7
q_{95}	5.3	3.0	3.0	3.0	3.1
q_0	1.0	1.0	1.0	1.0	1.0
nom. li	0.85	0.85	0.85	0.85	1.0
min. li	0.7	0.7	0.7	0.7	0.8
max. li	1.2	1.0	1.0	1.0	1.2
nom. β_P	0.1	0.1	0.65	0.65	0.1
max. β_P	0.1	0.1	0.8	0.8	0.1
nom. V_{loop} (V)	–	–	0.075	0.075	–
Notes: (1) SOF: "start of flat top", SOB: "start of burn"					

Table 3.2-4 Design Scenario 3: Hybrid Operation

Phase	XPF	SOH	SOF/B	EOB	EOC
t (s)	30	45	100	1100	1160
I_P (MA)	7.5	9.5	13.8	13.8	11
P_{add} (MW)	0	0	73	73	0
nom. $<n_{e,20}>$	0.23	0.4	0.93	0.93	0.4
q_{95}	5.3	4.3	3.3	3.3	3.6
q_0	1.0	1.0	1.0	1.0	1.0
nom. li	0.85	0.85	0.9	0.9	1.0
min. li	0.7	0.7	0.7	0.7	0.8
max. li	1.2	1.0	1.0	1.0	1.2
nom. β_P	0.1	0.1	0.8	0.8	0.1
max. β_P	0.1	0.1	0.9	0.9	0.1
nom. V_{loop} (V)	–	–	0.056	0.056	–

Table 3.2-5 Design Scenario 4: Non-inductive Operation I

Phase	SO-ECH(1)	XPF	SOF/B	EOB	EOC
t (s)	0.2	16	40	3100	3200
I_P (MA)	0.5	5	9	9	5
R/a (m/m)	7.4 / 0.8	6.2 / 2.0	6.35 / 1.85	6.35 / 1.85	6.2 / 2.0
P_{add} (MW)	6	8	59	59	0
nom. $<n_{e,20}>$	0.1	0.2	0.67	0.67	0.4
q_{95}	4	9	5.0	5.0	9
q_0	5	3	3.4	3.4	3
q_{min}	4	2.5	2.4	2.4	–
nom. li	0.8	0.8	0.6	0.7	0.9
min. li	0.6	0.6	0.6	0.6	0.6
max. li	0.9	0.9	0.9	0.9	1.2
nom. β_P	–	0.3	1.5	1.5	0.2
max. β_P	–	0.3	1.9	1.9	0.2
nom. V_{loop} (V)	–	–	0	0	–
Notes (1) Start of EC heating					

Table 3.2-6 Assessed Scenarios and Main Parameters (During Burn)

Parameter	5.Inductive operation III	6.Non-inductive operation II	7.Non-inductive operation III
R/a (m/m)	6.2 / 2.0	6.35 / 1.85	6.35 / 1.85
Volume (m^3)	831	730	730
Surface (m^2)	683	650	650
Sep. length (m)	18.2	16.9	16.9
Cross-section (m^2)	21.9	18.7	18.7
Toroidal field, B_T (T)	5.3	5.18	5.18
Plasma current, I_P (MA)	17.0	9.0	9.0
Elongation, κ_x/κ_{95}	1.85 / 1.7	2.0 / 1.86	2.0 /1.86
Triangularity, δ_x/δ_{95}	0.48 / 0.33	0.5 / 0.41	0.5 / 0.41
Confinement time, τ_E (s)	3.6	3.1	3.1
$H_{H\text{-}IPB98\ (v.2)}$	1.0	1.61	1.56
Normalised beta, β_N	2.2	2.9	2.9
Electron density, $\langle n_e \rangle$ ($10^{19}m^{-3}$)	12.3	6.5	6.7
f_{He} [%]	5.2	4.0	4.0
Fusion power, P_{fus} (MW)	700	340	352
Heating power, P_{add} (MW)	35	60	57
Energy multiplication, Q	20	5.7	6.2
Burn time (s)	100	3000(1)	3000(1)
Minimum repetition time (s)	–	12000	12000
Total heating power, P_{TOT} (MW)	175	128	127
Radiated power, P_{rad} (MW)	70	36	35
Alpha-particle power, P_α (MW)	140	68	70
Plasma thermal energy, W_{th} (MJ)	434	287	284

Notes
(1) The burn duration of the reference design is 400 s. Extended burn under non-inductive operation may be accomplished with additional investment in auxiliary systems.

Table 3.2-7 Assessed Scenario 5: Inductive Operation III

Phase	XPF	SOH	SOF/B	EOB	EOC
t (s)	30	90	130	230	300
I_P (MA)	7.5	15	17	17	14
P_{add} (MW)	0	0	35	35	0
nom. $<n_{e,20}>$	0.23	0.4	1.23	1.23	0.4
q_{95}	5.3	3.0	2.7	2.7	3.1
q_0	1	1	1	1	1
nom. li	0.85	0.85	0.77	0.77	1.0
min. li	0.7	0.7	0.7	0.7	0.8
max. li	1.2	1.0	1.0	1.0	1.2
nom. β_P	0.1	0.1	0.7	0.7	0.1
max. β_P	0.1	0.1	0.8	0.8	0.1
nom. V_{loop} (V)	–	–	0.085	0.085	–

Table 3.2-8 Assessed Scenario 6: Non-inductive Operation II

Phase	SO-ECH	XPF	SOF/B	EOB	EOC
t (s)	0.1	15	40	3100	3200
I_P (MA)	0.5	5	9	9	5
R/a (m/m)	7.4 / 0.8	6.2 / 2.0	6.35 / 1.85	6.35 / 1.85	6.5 / 1.7
P_{add} (MW)	4	9	60	60	0
nom. $<n_{e,20}>$	0.1	0.2	0.65	0.65	0.4
q_{95}	6	9	5.3	5.4	4
q_0	4	2.5	2.4	5.9	3
q_{min}	4	2.5	2.2	2.3	–
nom. li	0.8	0.7	0.7	0.6	0.9
min. li	0.6	0.6	0.6	0.6	0.6
max. li	0.9	0.9	0.9	0.9	1.2
nom. β_P	–	0.3	1.4	1.5	0.2
max. β_P	–	0.3	1.9	1.9	0.2
nom. V_{loop} (V)	–	–	0	0	–

Table 3.2-9 Assessed Scenario 7: Non-inductive Operation III

Phase	SO-ECH	XPF	SOF/B	EOB	EOC
t (s)	4	16	40	3100	3200
I_P (MA)	2	5	9	9	5
R/a (m/m)	7.4 / 0.8	6.2 / 2.0	6.35 / 1.85	6.35 / 1.85	6.2 / 2.0
P_{add} (MW)	5	15	57	57	0
nom. $<n_{e,20}>$	0.1	0.2	0.67	0.66	0.4
q_{95}	4	8	5.3	5.3	8
q_0	2.5	2.2	2.6	2.7	3
q_{min}	2.5	2.2	2.7	2.2	–
nom. li	0.9	0.8	0.7	0.7	0.9
min. li	0.6	0.6	0.6	0.6	0.6
max. li	0.9	0.9	0.9	0.9	1.2
nom. β_P	–	0.3	1.4	1.5	0.2
max. β_P	–	0.3	1.9	1.9	0.2
nom. V_{loop} (V)	–	–	0	0	–

3.3 Plasma Formation & Poloidal Field Control

3.3.1 *Requirements on PF system for Plasma Formation – Poloidal Flux*

The PF system should provide plasma initiation near the outboard first wall with a toroidal electric field of 0.3 V/m and EC assist power above 2 MW [See §4.19].

The value of magnetic stray field at breakdown should be less that 2 mT in a region with centre located at R = 7.48 m, Z = 0.62 m and minor radius 0.8 m.

Inductive plasma current ramp-up shall be assumed to take place in an expanding-aperture limiter configuration located on the outboard first wall.

Normal plasma current ramp-down (following from EOC) in a similar contracting-aperture limiter configuration shall be assumed to be located on the outboard first wall.

The flux swing capability of the poloidal field system shall satisfy the plasma operation scenario requirements outlined in §3.2.

Table 3.3-1 Assumptions on Resistive Consumption of Poloidal Magnetic Flux

Parameter	Unit	Value
Resistive flux loss at breakdown	Wb	10
Resistive flux loss during the plasma current ramp-up until SOH	Wb	$0.5\ \mu_0 \Delta(R_p I_p)$
Resistive flux loss from SOH to SOB for inductive scenarios	Wb	3
Resistive flux loss from SOH to SOB for hybrid and non-inductive scenarios	Wb	17
Resistive loop voltage during the plasma cooling (till EOC)	V	0.1
Resistive loop voltage during the plasma current ramp-down (after EOC).	V	0.4
(SOH=Start of heating, SOB=Start of Burn, EOC=End of Cooling		

3.3.2 *Requirements for Plasma Position, Current and Shape Control*

At the plasma current flattop, in the absence of fast disturbances, the plasma current shall be controlled to be less than ±2% or ±0.05 MA whichever is less restrictive.
During operation in a limiter configuration, the plasma position and shape control system shall be able to control plasma position and shape. The plasma position shall be controlled with plasma current as low as 0.5 MA.
Diagnostics used for control of plasma current, position and shape, including vertical speed of the plasma centroid, shall meet the requirements as specified in § 4.22)
Dynamic control of the separatrix during the power-producing phase shall be consistent with the recovery time for restoration of the separatrix deviations from their desired quasi-static positions.

Dynamic control should limit transient contact of the 10 mm heat flux SOL with the first wall surface to ≤ 1 s. The 10 mm heat flux SOL is defined as the flux surface that passes through a point 10 mm outside the separatrix at the outboard equator.
The plasma shape control system shall be able to meet the following requirements on the quasi-static plasma control (time scales >10 s).

Table 3.3-2 Quasi-static Shape Control (time scales >10 s)

Parameters	Unit	Value
Maximum downward displacement of the separatrix inner leg relative to the leg of target separatrix	mm	20 at low and reference l_i, 0 at high l_i
Maximum upward displacement of the separatrix inner leg relative to the leg of target separatrix	mm	60
Maximum inward displacement of the separatrix outer leg relative to the leg of target separatrix	mm	20 at low and reference l_i, 0 at high l_i
Maximum outward displacement of the separatrix outer leg relative to the leg of target separatrix	mm	60
Minimum clearance between the separatrix and the inner part of the first wall	mm	100 at low β_p (SOH),
Minimum clearance between the 40 mm flux surface and the first wall (unless otherwise specified) (1)	mm	80
Minimum distance between the inner and outer separatrices at the outboard equator	mm	40

Note (1): The 40 mm flux surface is defined as the magnetic surface that passes through a point 40 mm outside the separatrix at the outboard equator.

The control system shall be able to control the plasma current, position and shape in the presence of perturbations produced by ELMs, sawteeth, minor disruptions etc., specified for scenarios with positive magnetic shear as follows:

Table 3.3-3 Plasma Disturbances for Position and Shape Control

Minor Disruption	An instantaneous l_i drop of $0.2(l_{i0} - 0.5)$ without recovery simultaneous with β_p drop of $0.2\beta_{p0}$ followed by 3 s exponential recovery, or only β_p drop of $0.2\beta_{p0}$ followed by 3 s exponential recovery (without variation of l_i). One minor disruption should be considered during the driven burn and two minor disruptions should be considered during the plasma current ramp-up and ramp-down phases.
Compound ELMs	During the sustained burn, an instantaneous l_i drop of $0.06(l_{i0} - 0.5)$ followed by a 1 s linear recovery simultaneous with β_p drop of $0.03\beta_{p0}$ followed by 0.2 s linear recovery. The repetition time is about 10 s.
Type 1 ELMs	During the burn, an instantaneous β_p drop of $0.03\beta_{p0}$ followed by 0.1 s linear recovery with frequency 3 Hz.
Induced ELMs	During the burn, an instantaneous β_p drop of $0.01\beta_{p0}$ followed by 0.1 s linear recovery with frequency 4 Hz.
Plasma noise in dZ/dt diagnostics	During the burn, a white noise with a cut off frequency 100 Hz, which after filtering has the root mean square value about 0.02 m/s.
l_{i0} and β_{p0} are the initial internal inductance and β poloidal.	

The response time constant of the AC/DC converters shall be:

Table 3.3-4 Response Time Constant of AC/DC Converters

Transfer function for slow control	$e^{-0.015s} \cdot \frac{1}{1+0.015s}$
Transfer function for fast control (plasma vertical stabilization)	$e^{-0.0025s} \cdot \frac{1}{1+0.0075s}$

3.3.3 Requirements for Resistive Wall Mode Control

In steady state scenarios (negative magnetic shear), the control system shall be able to control the resistive wall modes (RWM) in the presence of following disturbances:

Table 3.3-5 Response Disturbances for RWM Control

Large RWM Event	A single event during the burn, a RWM evolving without control till the signal of RWM diagnostic (amplitude of n = 1 mode of the poloidal magnetic field) achieves 2.0 mT for moderately unstable RWMs ($C_\beta \approx 0.5$, original power supply) or 1.5 mT, for highly unstable RWMs ($C_\beta \approx 0.8$, upgraded power supply).
Plasma noise in RWM diagnostics	During the burn, a white noise in the RWM diagnostic with a cut off frequency 500 Hz, which after filtering has the root mean square value about 0.5 mT (n = 1 mode of the poloidal magnetic field).
$C_\beta = (\beta - \beta_{no\ wall})/(\beta_{ideal\ wall} - \beta_{no\ wall})$, where $\beta_{no\ wall}$ is the value of β, when n = 1 kink mode becomes unstable without a stabilizing wall and $\beta_{ideal\ wall}$ is the value of β, when the mode becomes unstable, even in the assumption of ideal conductivity of the inner wall of the vacuum vessel.	

For reduction of the AC losses in Side Correction Coils the RWM control system should be informed by the diagnostic system, when Type 1 ELM starts and the RWM control system will not react during the ELM.

The delay time of AC/DC converters used for RWM control is 2 ms, the time constant of the voltage rate limiter is 7 ms.

3.3.4 Error Fields

To avoid locked modes and associated disruptions, the amplitudes of n = 1, m = 1, 2 and 3 helical components of the error magnetic field shall be limited and satisfy the following criterion:

$$B_{3-mode} = \sqrt{0.2B_{1,1}^2 + B_{2,1}^2 + 0.8B_{3,1}^2} \le 5 \times 10^{-5}\, B_{tor},$$

where $B_{1,1}$, $B_{1,2}$, $B_{1,3}$ are the amplitudes of the normal component of the helical magnetic field on the q = 2 magnetic surface, B_{tor} is the value of toroidal magnetic field in the plasma geometrical centre**.**

3.4 TF and Ip Direction and Ripple

The reference directionality of the toroidal current and field shall be as follows: plasma current in the clockwise direction looking from above with the same direction for the toroidal field, giving a downward (towards divertor X-point) ion grad-B drift direction.

The direction of the toroidal field and plasma current shall be reversible, in such a way that the field line maintains the same pitch angle (i.e. the directions of the toroidal field and the plasma current can only be changed together).

The toroidal field ripple magnitude is defined as $\delta(R,Z) = (B_{max}-B_{min})/(B_{max}+B_{min})$, where B_{max} and B_{min} are maximum and minimum values of the toroidal magnetic field on the circle with coordinates (R,Z).

The ripple magnitude and distribution should cause the peak heat flux on the plasma facing components of the high-energy particles (ripple associated losses) to be less than 0.3 MW/m^2. Anyway the peak TF ripple shall be limited to 1.0% at any point inside the plasma separatrix.

Ferromagnetic inserts shall be used for reduction of the toroidal field ripple. Optimisation of the inserts distribution should be done for the nominal value of the toroidal magnetic field minimising the volume between the surface with δ = 0.1% and the outer part of the reference separatrix.

Permissible deviation of the saturated magnetization of the inserts ferromagnetic material from its averaged value should be less than ±3%.

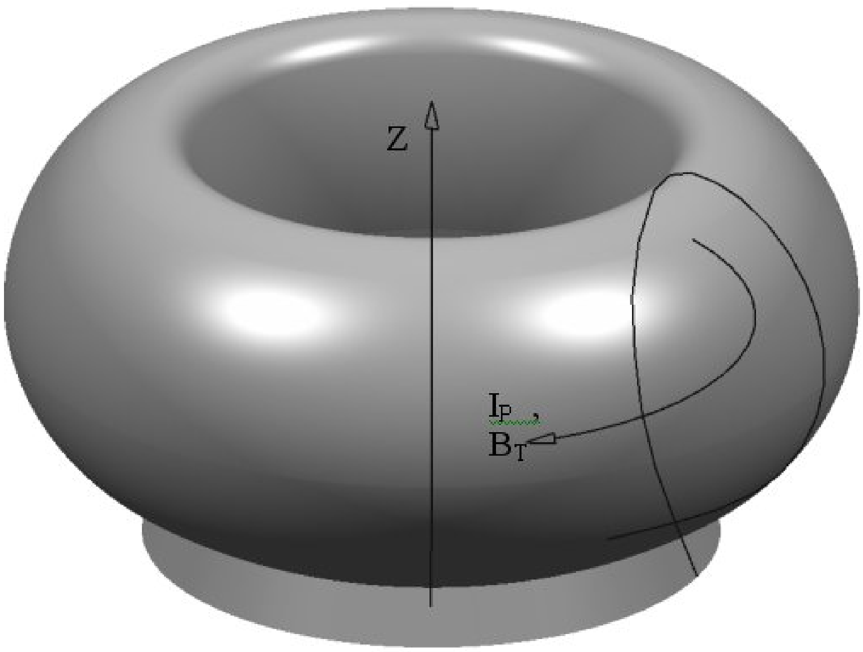


Figure 3.4-1 Plasma Current and Toroidal Field Direction

3.5 Port Allocation

Port Allocation depends on a number of options or upgrades. In particular the equatorial port allocation is heavily dependent on the additional heating upgrade scenarios.

Table 3.5-1 Port Allocation at the Equatorial level : AH Upgrade Scenarios

Port	Startup	Scenario 1(4)	Scenario 2	Scenario 3	Scenario 4
#1	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#2	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)
#3 (RH port)	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#4 (small rad.)	D-NB(3)	D-NB	D-NB	D-NB	D-NB
#4 (tangential)	H-NB(3)	H-NB	H-NB	H-NB	H-NB
#5 (tangential)	H-NB	H-NB	H-NB	H-NB	H-NB
#6 (tangential)		Diagnostics	H-NB	H-NB	H-NB
#7	Closed	Closed	Closed	Closed	Closed
#8 (RH port)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)	sLimiter, Diagnostics(1)	Limiter, Diagnostics(1)
#9	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#10	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#11	Diagnostics(2)	IC	Diagnostics(2)	Diagnostics(2)	Diagnostics(2)
#12 (RH port)	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#13	IC	IC(4)	IC	IC	IC
#14	EC	EC	EC	EC	LH
#15		LH	LH	IC	LH
#16	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)
#17 (RH port)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)	Limiter, Diagnostics(1)
#18	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)

(1)Minimal diagnostic systems with no penetration through the first wall.
(2)No diagnostics necessary for machine protection and basic plasma operation should be located in this port since they may have to be relocated for upgrade Scenario 1.
(3)D-NB: Diagnostics NB, H-NB: Heating NB
(4) Scenario 1 allows 40MW of IC and 20MW of LH. An option of 20/40 is also being considered with an extra LH in port 11
(5) Precise allocation of diagnostics ports is defined in §4.22

Table 3.5-2 AH Possible Maximum Upgrade Scenarios

	Startup		Scenario 1		Scenario 2		Scenario 3		Scenario 4	
	Power [MW]	No. of Equat ports	Power [MW]	No. of Equat ports	Power [MW]	No. of Equat ports	Power [MW]	No. of Equat ports	Power [MW]	No. of Equat ports
NB	33	2	33	2	50	3	50	3	50	3
IC	20	1	40(2)	2	20	1	40	2	20	1
EC	20	1	40	1(1)	40	1(1)	40	1(1)	20	0(1)
LH	0	0	20(2)	1	20	1	0	0	40	2
Total Installed	73	4	133	6	130	6	130	6	130	6

(1) EC H&CD will be able to use 4 allocated top ports for the power upgrade. No additional equatorial ports are therefore foreseen for this system.

(2) Scenario 1 uses IC/LH 40/20MW but 20/40MW is also being investigated (Jan 2004)

Table 3.5-3 Port and Port Cell Allocation at Lower Level

No removable lower port plugs will be available at the odd numbered positions (shaded), but some fixed penetrations are nevertheless welded into those ports (IVVS, GDC).
There are no VV penetrations crossing the field joints between sectors.

Port	Equipment	Diagnostics	Comments
1	Torus Branch Cryopump(1)		Branches from Port 18 pump duct
2	Remote Handling	Diagnostic cassette rack	Divertor Maintenance
3	Glow discharge, In-vessel viewing(3) Cryostat cryopump		Cryostat pump located outside of primary vacuum
4	Torus Direct Cryopump(1)	Diagnostics cassette & cables	
5	Glow discharge In-vessel viewing(3) Torus Branch Cryopump(1)		Branches from Port 4 pump duct
6	Torus Direct Cryopump(1) Pellet injection (4) Gas injection	Diagnostics(2)	One pellet injector at start-up, two pellet injectors after upgrading.
7	Torus Branch Cryopump(1)		Branches from port 6 pump duct
8	Remote handling	Diagnostic cassette/rack/cables	Divertor Maintenance
9	Glow discharge In-vessel viewing(3)		Access Flange to Cryostat
10		Diagnostic cassette rack & cables	RH-like large diagnostics port.
11	Glow discharge In-vessel viewing(3) Cryostat cryopump		Cryostat Cryopump located outside of primary vacuum
12	Torus Direct Cryopump(1) Gas injection Pellet injection (4)	Diagnostics(2)	One pellet injector at start-up, two pellet injectors after upgrading.
13	Torus Branch Cryopump(1)		Branches from prot 12 pump duct
14	Remote handling	Diagnostic cassette rack	Divertor Maintenance
15	Glow discharge In-vessel viewing(3)		Access Flange to Cryostat
16		Diagnostic cassette racks & cables	RH-like large diagnostics port.
17	Glow discharge, In-vessel viewing(3)		Access Flange to Cryostat
18	Torus Direct Cryopump(1) Gas injection, Pellet injection (4)	Diagnostics(2)	No cask (no pellet injector) at start-up, two pellet injectors after upgrading.

Note:

(1) One of the torus cryopump pairs is located in the adjacent cell on a side arm branch duct.
(2) Neutron activation diagnostic pipes, pressure gauges, RGA and hydrogen monitors
(3) Provision is made for 6 IVV locations, but only three IVV units are used.
(4) Pellet injection system cask at each port is capable of accommodating 2 injectors (centrifuges). Gas valve boxes for each pellet injector locates in the adjacent port cell.
(5) Precise allocation of diagnostics ports is defined in §4.22

Table 3.5-4 Port Allocation at Upper Level

The systems assigned to the port plugs are listed here. In addition, diagnostic cabling, waveguides or pipes from the VV exit the vessel at all upper ports. Precise allocation of diagnostics ports is defined in §4.22

Port	Heating and Gas	Diagnostics	Comments
1		Diagnostics[4]	
2		Diagnostics	
3		Diagnostics	
4[5]	Gas Injection[1] DMS[2] if needed		Ports 4,5,6,7 are part of the NB Cell, with higher radiation levels (Zone C) and less hands on access. This area is currently being reviewed (2005)
5[5]		Diagnostics	
6[5]		Diagnostics	
7[5]		Diagnostics	
8		Diagnostics	
9	Gas injection	Diagnostics	
10		Diagnostics	
11		Diagnostics	
12	EC		
13	EC		
14	Gas injection[1]	Diagnostics	
15	EC		
16	EC		
17		Diagnostics	
18	Gas injection DMS[2] if needed		

Note :
(1) Including fusion power shut-down system
(2) DMS = Disruption Mitigation System:
(3) Each GIS system is capable of dynamically mixing gases from 6 common supply lines.
(4) Precise allocation of diagnostics ports is defined in §4.22
(5) Ports 4,5,6,7 are currently under investigation (May 2005) as part of the NB Cell Review. Concerns include the higher radiation level, space considerations for beam components, and floor strength.

(The Safety shut-down system is required only to terminate the fusion power when the coolant capability is missed. The Disruption mitigation may be useful for investment protection but is not required for safety reasons.)

3.6 Slow transients – Heat Loads

Table 3.6-1 Static Plasma Heat Loads Specification

Parameters	Unit	H	DT	TBA[2] (700MW)
Total heat from plasma to in-vessel components	MW		812[1]	1090[5]
of which:				
Nuclear Heating Power	MW	-	630	840
Alpha heating Power	MW	-	105	140
Additional Heating Input	MW	73	77	110
Maximum power excursion	%	20		
Power excursion duration	s	10		
Maximum power to SOL [3]	MW	55	136	← [5]
of which:				
Alpha heating Power	MW	-	79	
Additional Heating Input	MW	55	57	
Maximum radiated power to FW [4]	MW	55	136	← [5]
of which:				
Alpha heating Power	MW	-	79	
Additional Heating Input	MW	55	57	

Notes:

(1) Includes a neutron energy multiplication factor of 1.5 and 5% error for fusion power measurement which gives an additional multiplier for nuclear heating power, alpha heating power as well as additional heating

(2) For 700MW extended scenario (TBA) it includes a neutron energy multiplication factor of 1.5 but NOT the 5% error for fusion power measurement assumed for normal operation and explained in note 1.

(3) For an upper bound, 75 % of the total thermal power is assumed to flow to the SOL. Also in this case an error of 1.05 is assumed in both alpha heating and additional heating (5% error for fusion power measurement and also 5% error for additional heating power)

(4) For an upper bound, 75 % of the total thermal power is radiated to the FW. Also in this case an error of 1.05 is assumed in both alpha heating and additional heating.(5% error for fusion power measurement and also 5% error for additional heating power)

(5) 700 MW operation can be performed while limiting other operation conditions such as burn time etc. Other important conditions to check are the thermal heat loads to the divertor and first wall.

Table 3.6-2 Static Heat Loads Specification for FW from plasma

Parameters	Unit	H	DT	TBA
Maximum power to limiters during start up	MW	15(1)		
Maximum thermal power to divertor (total)	MW	See "Max power to SOL" in Table 3.6-1		
Maximum fraction of thermal power to outboard divertor		2/3		
Maximum fraction of thermal power to inboard divertor		½		
Maximum total heat flux to FW(2)	MW/m^2	0.5		
Maximum number of shots with heat flux higher than 0.25 MW/m^2		10,000 shots with 0.25 - 0.5 MW/m^2		
Maximum total heat flux to FW in the NB shine-through areas	MW/m^2	0.5(3)		
Localised MARFEs radiated heat flux to baffle region FW, peak value	MW/m^2	1.4		
Duration of MARFE	s	~ 10		
Alpha-particle losses peak heat loads in outboard equator	MW/m^2	0	0.1	0.3

Notes:

(1) 15 MW is total power to the limiters. Protruding limiter may receive maximum power of 9 MW.

(2) For the test blanket module (TBM) FW, 0.5 MW/m^2 should be considered, preferably 3,000 cycles per year, and at least 100 cycles per year.

(3) 0.5 MW/m^2 is the requirement, but 1 MW/m^2 is desirable to facilitate H-mode experiments during the HH-phase.

Table 3.6-3 Static Heat Loads Specification for Vessel + In-vessel Cooling Systems

Parameters	Unit	H	DT	TBA
Maximum power to vacuum vessel cooling system(1)	MW	0	10	14
Maximum power to blanket cooling system (2)	MW	55	690	875
of which:				
Nuclear Heating Power		-	554	739
Radiated Power		55	136	136
Maximum power to divertor cooling system including limiter (3)	MW	55	202	223
of which:				
Nuclear Heating Power		-	66	87
Radiated/Transported Power		55	136	136

Notes:

(1) In the ITER geometry, 1.6% of neutron energy is absorbed at the VV. See Table 3.6-1 for nuclear power (no radiated power).

(2) In the ITER geometry, 88% of neutron energy is absorbed at the blankets. See Table 3.6-1 for nuclear power and radiated power.

(3) In the ITER geometry, 9% and 1.4% of neutron energy are absorbed at the divertors and limiters, respectively. See Table 3.6-1 for nuclear power and radiated/transported power

Table 3.6-4 Global Decay Heat Transient

Time after termination	Decay heating power(1)
1 second	11 MW
5 minutes	8.6 MW
30 minutes	5.0 MW
1 hour	4.1 MW
3 hours	2.8 MW
5 hours	2.0 MW
10 hours	1.1 MW
1 day	0.60 MW
3 days	0.35 MW
1 week	0.31 MW
1 month	0.27 MW
3 months	0.22 MW
1 year	0.14 MW
3 years	8.7×10^{-2} MW

(1) Add safety factor of 1.2 for FW, and 1.3 for DV when used for safety relevant assessments.

(2) A longer pulse operation is possible in actual operation as long as the resultant decay heat stays within the transient used in the safety assessments.

Table 3.6-5 Static Heat Loads Specification (Auxiliary)

<table>
<tr><th>Parameters</th><th>Unit</th><th>H</th><th>DT</th></tr>
<tr><td>Max power to EC cooling (1)</td><td>MW</td><td>30</td><td>60</td></tr>
<tr><td>Max power to IC cooling (1)</td><td>MW</td><td>10.8</td><td>21.6</td></tr>
<tr><td>Max power to LH cooling (1)</td><td>MW</td><td>0</td><td>(2)</td></tr>
<tr><td>Max power to NB cooling (1)</td><td>MW</td><td>~ 77(3)
[65.6(LV)]
[11.2(HV)]</td><td>~ 102(4)
[86.9(LV)]
[15.3(HV)]</td></tr>
<tr><td rowspan="2">NB, Power to be removed from high voltage components per one NB injector</td><td rowspan="2">MW</td><td>HNB</td><td>4.1</td></tr>
<tr><td>DNB</td><td>3.0</td></tr>
<tr><td rowspan="2">NB, Power to be removed from low voltage components per one NB injector</td><td rowspan="2">MW</td><td>HNB</td><td>21.3(injection) +
19.6(conditioning)</td></tr>
<tr><td>DNB</td><td>1.8(injection) +
1.75(conditioning)</td></tr>
<tr><td>Integrated nuclear heating to one duct and liner</td><td>kW</td><td>0</td><td>110</td></tr>
<tr><td>Peak nuclear heating to the duct/duct liner</td><td>kW/m</td><td>0</td><td>100</td></tr>
<tr><td>NB, integrated beam power to 1 duct liner</td><td>kW</td><td colspan="2">700</td></tr>
<tr><td>NB, peak beam power density to duct liner</td><td>MW/m^2</td><td colspan="2">0.3</td></tr>
<tr><td>NB, max power density at far wall under normal conditions</td><td>MW/m^2</td><td colspan="2">1</td></tr>
<tr><td>Total power to component cooling water system</td><td>MW</td><td>~ 80</td><td>~ 132</td></tr>
<tr><td>Total power to chilled water system</td><td>MW</td><td>~ 39</td><td>~ 50</td></tr>
</table>

Notes

(1) Cooling water system relating to the H&CD system should be designed to be capable of removing the heat from each system at upgraded performance.

(2) Included in EC and IC cooling capacity.

(3) Two HNB and one DNB injectors are foreseen in the hydrogen phase at the maximum, according to Note (1) above.

(4) Installation of the third HNB injector foreseen sometime in the DT phase, and hence, 3 HNB and one DNB are counted as the maximum.

Table 3.6-6 Reference Nuclear Heating in the TF Coils (kW)

	Inboard Leg	Behind Divertor	^{16}N	Around Ports	Total
Coil Case	4.8	0.9	0.4	1.7	7.8
Winding Pack (first turn total)	6.1 (1.9)				6.1
Total	10.9	0.9	0.4	2.4	13.9

- For the 15MA plasma with a Total Fusion Power of 500MW
- See Magnets DDD-11 Table 1.1.4-1

Table 3.6-7 Nuclear Heating in the PF Coils

(due to Neutrons and Prompt Gamma Rays)

		Ground Insulator	**Winding Packs**	**Clamps**	**Sum (W)**
PF 1	Facing towards central solenoid	0.1	1.3	0.6	2
PF 2	Under upper port	0.6	12	1.4	14*
PF 3	Between upper and horizontal ports	1	41	3	45
PF 4	Under divertor port	5	131	11	147
PF 5	Below divertor port	15	200	35	250
PF 6	Facing towards central solenoid	1	15	4	20
	Total (W)				480

* add 100W for N^{16} gamma radiation (The values in the table include contributions of all energy neutron (not only fast neutron) and prompt photons (~90%) but does not include that of decay gamma rays. Decay gamma contribution is usually very small, but it can be significant when source can move like N-16 in the cooling water). (*From DDD1.1 Table 1.1.4-3 – 2006 version)*

Table 3.6-8 Heat Loads in the Magnets and Feeders

(15 MA Reference Scenario with a Total Fusion Power of 500 MW, 400s Burn, 1800s Repetition Time)

Overall Total average 21.2kW *(From DDD1.1, Table 1.1.4-14)*

Heat loads	*TF structures (+PF supports)*	*TF conductor*	*CS structures*	*CS conductor*	*PF conductor*	*CC conductor*	*Total (average over 1800s)***
Nuclear Rad	*7.8kW in burn*	*6.1kW in burn*	*0*	*0*	*0.6kW in burn*	*0*	*3.2kW*
*AC losses **		*1.72MJ*		*8.46MJ*	*1.60MJ*	*0.03MJ*	*6.56kW*
Eddy currents	*5.34MJ*		*1.41MJ*				*3.74kW*
Joints	*0.0*	*1.00 kW*		*0.01kW average*	*0.07kW average*		*1.08kW*
Thermal Rad Cryostat Vacuum vessel	*0.76kW 0.46kW*	*0 0*			*0.1kW*	*0.01kW*	*1.33kW*
*Thermal Cond Gravity supports Thermal shield*****	*2.3kW 0.2kW*						*2.5kW*
*He feeders, SC bus bars and CTBs****	*0.77kW*			*0.37kW*	*0.66kW*		*1.8kW*
He Cryolines from ACBs and CTBs	*0.05kW*	*0.45kW*		*0.2kW*	*0.3kW*		*1.0kW*

* AC loss assumptions are conservative, based on conductor parameters above the design targets:

- TF conductor based on jc 1000A/mm2, hys 1000mJ/cc, ntau 200ms
- PF conductor based on jc 2900A/mm2, eff fil 6mm, ntau 150ms
- CS conductor based on jc 800A/mm2, hys 600mJ/cc, ntau 150ms
- CC base losses on maximum removable heat (although not used for RWM control in inductive scenario)

**The average values are for a 400 s burn with a 1800 s pulse repetition.

***based on 50W per CTB (thermal radiation, cold end heat load of current lead from DDD1.1 Table 1.2.6-7 and thermal conduction) and 0.5W/m of feeder

****Source for thermal loads, DDD 2.7 (August 2004 version)

- Gravity support conduction section 1.2.6
- VVTS support conduction table 1.1.2-1
- Radiation loads Table 1.1.2-1 and section 2.5 (with correction for removal of VV ropes)

These values are totalled and summarised in "Table 4.15-1 Requirements for cryopumps at 4.5K"

Calculations of heat loads to structures cooled at cryogenic temperature shall be considered as the maximum value for the heat removal systems (e.g. the cryoplant) and as a maximum allowable value for the shielding (e.g. thermal and nuclear shields) systems. The assumption of regular pulsing implicit in the heat loads in Table 3.6-8 applies only for normal operation when the primary heat exchanger bath is at or above 4.15K

Table 3.6-9 shows the range of He consumption expected for the HTS current leads. After allowing some margin, it appears that a cryoplant design for a maximum supply rate of 175g/s of He at 50K would be appropriate, with an expected average consumption of about 140g/s. The He is returned at room temperature

Table 3.6-9 Average 50K helium mass flow rate required for HTS-CL cooling

(from DDD 1.1 Table 1.2.6-8) (based on 0.07g/s/kA for all leads at full current, with the no-load mass flow 50% of the full load value, using load factors from DDD 1.1 Table 1.1.4-18)

	average mass flow rate g/s	*peak mass flow rate g/s*	*no-current mass flow rate g/s*
TF	***86***	***86***	***43***
CS	***25***	***34***	***17***
PF and CC	***25***	***27***	***23***
Total	***136***	***147***	***83***

Table 3.6-10 Mass Flow Rates in Coils for the Reference 500MW Plasma

(400s burn, 1800s total pulse duration) (from DDD1.1 Table 1.1.5-1)

Coil	Total Winding pack mass flow rate (kg/s)	Conductor average mass flow rate (g/s)	Helium transit time in coil (s)	Conductor Flow path length (m)	conductor Pressure drop (bar)**
TF	2.0	7.94	2940	380	1.05
CS	2.0	8.33	760	152	1.05
PF1	0.437	13.66	880	199	0.55
PF2	0.217	10.87	1350	288	0.55
PF3	0.268	8.37	2770	454	0.55
PF4	0.282	8.83	2400	415	0.55
PF5	0.296	9.25	2130	370	0.55
PF6	0.309	9.66	2280	363	0.55
Upper CC	0.017	1.3	950	130	0.55
Side CC	0.091	3.8	940	200	0.55
Lower CC	0.017	1.1	1380	160	0.55

** includes He inlets and outlets, see 1.1.3.1.7

An extra pressure drop contribution in the coil circuits from cryolines, manifolds and valves (including the heat exchanger) of 0.01MPa should be included in addition to the pressure drop values in Table 3.6-10.

Table 3.6-11 Mass Flow Rates in the Structures

(from DDD1.1 Table 1.1.5-2)

Circuit	Number of Loops	No. parallel circuits/coil	Length (m)	Pipe ID (mm)	Flow total (kg/s)	Pressure drop total (kPa)
Inside inner wall	18	34	107	7.82	1.67	3.248
Inside outer 3 walls	18	62	107	7.82	0.83	2.358
Outside outer leg (PF supports) **	18	3	118	7.82	0.06	1.686
CS Structure	9	4	106	7.82	0.06	0.908

** mainly for cooldown, minor contribution to operational cooling

Table 3.6-12 Thermal Shields (TS) Heat Loads Specifications

Parameters	Unit	
VV TS, average heat load on 80 K cryoplant during 500 MW/ 400s/ 1800s operation	kW	140
VV TS, average heat load on 80K cryoplant during VV baking	kW	290
Cryostat and support TS, average heat load on 80K cryoplant	kW	25
Cryostat and support TS, average heat load on 80K cryoplant during VV baking	KW	75
Transition TS, average heat load on 80K cryoplant	kW	55
Transition TS, average heat load on 80K cryoplant during VV baking	kW	110
Thermal anchor in magnet gravity support: average heat load to 80K plant	kW	20

3.7 Fast transients – Heat Loads

The plasma facing components must allow transient heat loads during ELM's, disruptions and VDEs. Their specification from the mechanical loads standpoint can be found in the Load Specification and Combination (LS), which is annexed to this document**.**

Table 3.7-1 Heat Load Conditions during VDEs

Parameters	Unit	H	DT	TBA
Plasma thermal energy	GJ	0.35		
Magnetic energy	GJ	0.37		
Energy partition to first wall as conduction	%	50 – 80		
Energy partition to first wall as radiation	%	50 – 80		
Direction of movement	–	Up or down		
Peak energy deposition to first wall	MJ/m^2	60		
Duration of the contact with first wall	s	0.2		
Number of events	–	See LS document		

Table 3.7-2 Heat Load Conditions during Major Disruptions

Parameters	Unit	H	DT	TBA
Thermal Quench				
Thermal energy during thermal quench phase	GJ	0.175-0.35		
Expansion factor for width of scrape-off layer $f_e = \lambda/\lambda_{ss}$	-	5-10		
Energy quench time in the core plasma	ms	1		
Peak energy deposition on first wall (parallel to the field line) at 12 cm from the first separatrix	MJ/m^2	9 – 18 (f_e =10) 1.65 – 3.3 (f_e =5)		
Energy deposition time on first wall	ms	1-1.5		
Peak energy deposition on the arbitrary position of the limiter (parallel to the field line) *	MJ/m^2	$\frac{28.6 E_{disr}}{\lambda} e^{-(\frac{L_0+x}{\lambda})}$ *		
Energy deposition time on the limiter	ms	1-1.5		
Peak energy deposition on divertor	MJ/m^2	6 – 12.5 (f_e =10) 12.5 - 25 (f_e =5)		
Energy deposition time on divertor	ms	1.5-3		
Radiation energy on first wall (average)	MJ/m^2	≤ 0.08-0.16		
Radiation energy deposition time on first wall	ms	1.5-3		
Peaking factor for radiation to divertor targets	–	3		
Peaking factor for radiation to first wall	–	3		
Number of events	–	See LS document		
Current Quench				
Magnetic energy (converted to thermal energy) during current quench phase	GJ	0.37		
Energy quench time	ms	10		
Peak energy deposition to first wall	MJ/m^2	0.72		
Energy fraction on first wall as radiation	%	≤ 100		
Peaking factor for radiation to first wall	~	1.4		

Number of events		**See LS document**
Runaway Electron		
Predicted runaway current	MA	10
Energy spectrum of electrons (E_0 for $\exp(-E/E_0)$)	MeV	12.5
Inclined angle	degree	1 – 1.5
Total energy deposition due to runaway current	MJ	20
Average energy density deposition	MJ/m^2	1.5
Duration of the average energy density deposition	ms	100
Maximum energy density deposition (end of the plasma termination)	MJ/m^2	25
Duration of the maximum energy deposition	ms	10
Number of event		Every major disruption

Notes:

*Energy loads parallel to the field line on the arbitrary point x of the limiter can be evaluated by the following equations:

$$\frac{28.6E_{disr}}{\lambda}e^{-(\frac{L_0+x}{\lambda})} \quad (MJ/m^2)$$

E_{disr} : total energy loss during disruptions (MJ)
L_0 : distance between first separatrix and limiter head (mm)
x : distance from limiter head (mm)
$\lambda = f_e \times \lambda_{ss}$
λ_{ss} : steady state heat flux width = 5 mm
f_e : expansion factor during disruption (5 or 10)

Table 3.7-3 Heat Load Conditions during ELMs

Parameters	Unit	Controlled*	Uncontrolled
Thermal energy release during ELMs	MJ	3-4	15-20
ELM frequency	Hz	5	1
Peak energy deposition on first wall (parallel to the field line) at 12 cm from the first separatrix	MJ/m^2	0.14-0.19	0.7 -0.93
Energy deposition time on first wall	μs	200 (rising phase) 200 (decay phase)	⇐
Peak energy deposition on the arbitrary position of the limiter (parallel to the field line) **	MJ/m^2	$(0.54-0.72)\times e^{-(\frac{L_0+x}{90})}$	$(2.7-3.6)\times e^{-(\frac{L_0+x}{90})}$
Energy deposition time on the limiter	μs	200 (rising phase) 200 (decay phase)	⇐
Peak energy deposition on divertor	MJ/m^2	0.4-0.5	2-2.7
Energy deposition time on divertor	μs	500 (rising phase) 500 (decay phase)	⇐
Expansion factor for width of scrape-off layer	–	1.5	1.5

* Pace making by pellet injection is assumed.

**Energy loads parallel to the field line on the arbitrary point x of the limiter can be evaluated by the following equations:

$$0.179E_{elm}^{T}e^{-(\frac{L_0+x}{90})} \quad (MJ/m^2)$$

E_{elm}^{T} : total energy loss during ELMs (MJ)
L_0 : distance between first separatrix and limiter head (mm)
x : distance from limiter head (mm)

Table 3.7-4 Heat Load Conditions during Alpha Particle Burst

Parameters	Unit	H	DT	**TBA**
Energy loss per burst	MJ	20		
Peak load due to burst (axisymmetric wall)	MJ/m^2	0.2		

The fast alpha particles are released with the burst and the burst itself will disappear after a number of bursts because of cool-down of the plasma. The design specification for the plasma facing components is 10 bursts with a frequency of 1 Hz.

Table 3.7-5 Accidental Heat Load Conditions at First Wall due to Neutral Beam

Parameters	Unit	H	DT	**TBA**
Maximum power density	MW/m^2	50		
Duration	ms	100		

3.8 Neutron & Radiation Loads – Shielding

Table 3.8-1 Basic Design Parameters - Neutronics

Parameter	Unit	H	DT	**TBA**
Average neutron wall load at first wall	MWm^{-2}	0	0.56	0.79
Neutron wall load at outboard FW at midplane	MWm^{-2}	0	0.78	1.09
Total average neutron fluence at the first wall	$MWam^{-2}$	0	0.3	0.5
14 MeV peak n flux on First Wall (outboard)	$n/s/cm^2$		$4.4\ 10^{13}$	$6.1\ 10^{13}$
Collided n flux on First Wall (outboard)	$n/s/cm^2$		$2.4\ 10^{14}$	$3.4\ 10^{14}$

The main vessel and the in-vessel components together shall provide sufficient nuclear shielding to protect the superconducting coils, and to reduce activation inside the cryostat and at port areas.

Table 3.8-2 Maximum Nuclear Load Limits to the Magnet

Parameters	Unit	H	DT
Local nuclear heat in the conductor	kW/m^3	0	1
Local nuclear heat in the case and structures	kW/m^3	0	2
Peak radiation dose to coil insulator	Gray	0	10×10^6
Total neutron flux to coil insulator	N/m^2	0	10^{22}
Total nuclear heat in the magnets	kW	See Table 3.6-6 and Table 3.6-7	

(For first wall fluxes and fluences see Table 2.2-1)

All field welds to vessel and in-vessel RH class 3 components shall be reweldable up to a fluence of 0.5 Mwa/m^2 at the first wall. The limit for field welds is provided by the allowable levels for the production of He (< 1 appm for thick plate welding and <3 appm for thin plate or tube welding).

ITER shall incorporate shielding design provisions to reduce dose rates in the port regions as low as reasonably achievable to facilitate hands-on maintenance in the port areas. The dose rate shall not be significantly different from those achieved by the shielding capabilities of the bulk shielding (blanket + VV). ITER shall incorporate radiation shielding to permit personnel access in the annular space outside the bioshield. Shielding shall be designed to minimise the number of components located outside the bioshield that require remote maintenance. Shielding cells will be built around dedicated ports to allow parallel hands on maintenance in adjacent volumes when an activated component is in the cell. The level of ionising radiation outside the biological shield (with the exception of the NB cell) immediately surrounding the tokamak shall be limited to 10 µSv/hour 24 hours after shutdown, to allow radiation workers unlimited (zone B) access to those areas. The same level of shielding shall be provided to the hot cell facility.

Areas with limited access requirements dedicated for specific maintenance, such as the NB cell and the areas inside the bioshield of the port maintenance areas, shall meet the requirements for Access

Zone C, 10^6 seconds after shutdown, and should be limited to 100 μSv/h, the ALARA guideline for allowing radiation workers hands-on access. Areas where the guideline of 100 μSv/h is not met shall be reviewed for acceptability on an individual basis. See also § 3.1.5.

ITER shall also incorporate shielding design provisions to reduce dose rates for emergency hands-on repair operation inside the cryostat, such as by reducing fast neutron streaming through gaps. Here the target dose rate is less than 100 μSv/hour 10^6 s (~12 days) after shutdown. Wherever the guideline of 100 μSv/hour is not met shall be reviewed for acceptability on an individual basis.

3.9 Grounding

The ITER grounding scheme is detailed in the ITER Electrical Design Handbook (EDH) (click for direct IDM link).

All surfaces, including bus and cooling lines, which are exposed to the cryostat vacuum, shall be at ground potential.

All in-vessel components shall be electrically connected to the VV.

The VV shall be electrically insulated from the magnet system to avoid shunting of designated grounding paths for TF coils and to break local eddy current loops.

The VV is connected to the cryostat via the cryostat connecting ducts.

All magnet structures (each toroidal segment) are hard grounded through the feeder ducts to the cryostat.

(Note, July 2005: There is an identified ITER Issue 7.5-2 to review and update the ITER Grounding Scheme – Ongoing Oct 2006, including “Site Adaptation”)

3.10 Maintainability

(Updated A.Tesini 20 Oct 2006)

ITER's components design shall include maintainability features, which will allow scheduled maintenance to be performed reliably and in a timely manner thus maximizing machine availability. The possibility and optimisation of hands-on maintenance shall always be considered first; hands-on assistance shall be considered in remote handling procedures as far as practical within the general application of ALARA guidelines.

In-vessel interventions will generally be preceded by in-vessel inspection to obtain information on the extent of damage and maintenance activities required.

Access to the components within the primary vacuum boundary shall be possible from the outside without the need to break the cryostat vacuum.

Maintenance of in-vessel components will generally consist of the replacement of components. The removed, activated and contaminated components will be transported to the hot cell for eventual repair and refurbishment, or, alternatively for preparations for disposal as waste.

RH equipment will be introduced into the vacuum vessel using transfer casks docked to ports' VV flange. The transfer casks are sealed, but not shielded, hence requiring restricted access of personnel to the pit and gallery areas when casks are traveling to and from the hot cell. The cask transporter system uses an air cushion flotation device. Preparatory activities, prior to initial cask docking, will involve hands-on operations, including bioshield plug-removal, pipes, cables and other plug services, cutting of the vacuum seal.

Rescue procedures shall be available for every RH procedure, i.e., all RH equipment shall be designed for remote recovery.

All components inside the cryostat must be designed to last the lifetime of the ITER machine, hence not requiring maintenance. Should, however, components inside the cryostat require repair, then hands-on repair is the reference procedure, with remote repair as a backup where and if at all possible, due to the cryostat internal severe space constraints. NOTE: the requirements for inspection and repair operations inside the ITER cryostat must be reviewed in consideration of the extremely limited access space inside the cryostat.

Gross failure of components inside the cryostat may require their replacement. The design and layout of components inside the cryostat, as well as the design of the cryostat itself, must not preclude the replacement of large components. NOTE: this requirement must be reviewed to take into account: a) the expected radiation dose levels at the time of the required large component replacement, b) the ability to implement any necessary radiation shielding, c) the ability to carry out the component's replacement within a reasonable period of time.

Access to the interior of the cryostat for repair is provided at three levels:

- top of the cryostat via a central access hatch and 4 perimeter hatches;
- 3 divertor port level hatches;
- 3 hatches at the lower horizontal intermediate floor;
- 4 hatches at the lower level cryostat cylinder basement.

Repairs will be conducted with the magnets off, warmed to room temperature and with the cryostat open. Residual magnetic fields will be negligible. Pressure will be ambient and the atmosphere normal air. Normally, radiation doses are assumed acceptable for human access and there will be no need for plastic suits or breathable air supplies. Two access routes will always be provided, in case of emergency.

Magnets or vacuum vessel sector replacement should be possible provided it occurs before the machine and cryostat become inaccessible due to the high background radiation dose. Even so, the maintenance procedure shall be defined and a feasibility and time vs advantages assessment would need to be performed to establish the opportunity of such an extensive and technically challenging task.

RH operations are classified as folllows:

Table 3.10-1 RH Classification

Class 1	those components that require scheduled remote maintenance or replacement.
Class 2	those components that do not require scheduled but are likely to require unscheduled or very infrequent remote maintenance.
Class 3	those components not expected to require remote maintenance during the life time of ITER. The projected maintenance time in case of failure may be long.

All RH equipment for Class 1 and 2 operations must be designed in detail prior to ITER construction. The feasibility of Class 1 tasks shall be verified prior to ITER construction and may involve the use of mock-ups. The feasibility of Class 2 tasks shall be verified prior to ITER construction where deemed practical and necessary and may involve the use of mock-ups. The procedure of maintenance of Class 3 components shall be defined prior to ITER construction.

Table 3.10-2 Maintenance Operation, Maximum Frequency and Maximum Intervention Time (3)

Component or Operation	RH Class	Frequency	Intervention Time
Divertor cassettes (54) including RH port components	**1**	<3 times/1^{st} 10 years, <5 times/2^{nd} 10 years	< 6 months/all cassettes <8 weeks/single cassette (excluding machine shutdown/start up time)
Test blanket modules (3)		~ once each/year	2 weeks to 1 month
NB filament/cesium oven		2 times/year	7 days
Equatorial RH port limiter plug		~ 10 times each/20years	< 1 month
Equatorial RH port diagnostics plug		~ 10 times each/20years	< 1 month
Blanket modules (all or outboard region)	**2**	Once-only changeover at the end of the first 10 years	< 8 weeks/1 module < 3 months/1 toroidal row < 2 years/all modules
Blanket modules including NB modules		3 modules/year	
Lower blanket modules		6 modules/year	
Torus cryopump		4 pumps total /20 years (not scheduled)	< 1 month
RF antennae		≤ 5 times each/20 years	~ 1 month
ECH & CD system mirror		≤ 5 times each/20 years	~ 1 month
Equatorial / upper port diagnostics		≤ 5 times each/20 years	~ 1 month
NB ion source cesium cleaning (grid)		≤ 1 time/year	17 days (TBC)
NB fast shutter(2)		≤ 1 time/20 years	1 month (TBC)
Cryostat cryopump(1). Port bellows. (incl. pumping duct double bellows) ~~Diagnostics~~. (details 4.22) Thermal shields, Magnet termination and module bypass joints, CS, PF and TF coils, Vacuum vessel sector, ports and other in-cryostat components **Neutral Beams**:…… Drift duct liner, bellows, Neutralizer, Residual Ion Dump, Calorimeter.	**3**	No scheduled maintenance	

Notes
(1) Cryostat CP will only need RH when cryostat interior becomes contaminated so as to preclude hands-on CP handling
(2) DCR-33 is currently under study to make the NB fast shutter into a slow acting absolute valve. (July 2005, Ref TCM-16)
(3) These figures are an upper limit for cost and risk assessment purposes. They do not indicate the reliability of components.

3.11 Mechanical Loads

(Updated G.Sannazzarro Oct 2006)

The design of the ITER systems shall be able to withstand loading conditions (including seismic) and combinations as specified and classified in four different likelihood categories in the Load Specifications and Combination [LS], which is annexed to this document.

The following table indicates the definition of different types of damage limits for components and plant.

Table 3.11-1 Damage Limits in Plant and Component Level

Damage Limits	**Damage Limits to Component Level**	**Damage Limits in Plant Level and Recovery of the Plant (Plant Operational Condition)**
Normal	The component should maintain specified service function.	Within specified operational limit. No special inspection will be required other than routine maintenance and minor adjustment.
Upset	The component must withstand these loadings without significant damage requiring special inspection or repair.	After minor adjustment, or replacement of the faulty component, the plant can be brought back to normal operation. No effect on other components that may call for special inspection or repair.
Emergency	Large deformations in areas of structural discontinuity, such as at nozzles, which may necessitate removal of the component from service for inspection or repair. Insignificant general permanent deformation that may affect safety function of the component concerned. General strains should be within elastic limits. Active components should be functional at least after transient.	The plant may require decontamination, major replacement of damaged component or major repair work. In addition to the damaged component, inspection may reveal localised large deformation in other components, which may call for the repair of the affected components. Nevertheless, the plant maintains the specified minimum safety function during and after the events.
Faulted	Gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of component from service. Nevertheless deformation should not lead to structural collapse which could damage other components. The fluid boundary is maintained but degraded, however the safety function is maintained. Active components may not be functional after transient.	Gross damage to the affected system or component. No loss of safety function which could lead to releases in excess of the guidelines established for Accidents. No design consideration will be given for recovery. The recovery of the plant may be judged from the severity of damage. This level of state is not expected to occur, but is postulated for safety assessment because its consequences would include the potential for the release of significant amounts of radioactive material.

The following table indicates the relationship between Load Combination Category (loads and likelihood categories defined in LS document) and acceptable damage limit as a function of the component safety class (SIC or non-SIC) as specified, for all components, in Table 3.1-9.

Table 3.11-2 Damage Limits for Loading Condition Categories

<table>
<tr><td colspan="2">Loading Event
Category</td><td>Category I:
Operational
Loading</td><td>Category II:
Likely
Loading</td><td>Category III:
Unlikely
Loading</td><td>Category IV:
Extremely
Unlikely
Loading</td><td>Test Loading</td></tr>
<tr><td colspan="2">Plant Level</td><td>Normal</td><td>Normal</td><td>Emergency</td><td>Faulted</td><td>Normal</td></tr>
<tr><td rowspan="2">Component</td><td>SIC</td><td>Normal</td><td>Normal</td><td>Emergency</td><td>Faulted
(note 1)</td><td>Normal</td></tr>
<tr><td>Not-
SIC</td><td>Normal</td><td>Upset</td><td>(note 2)</td><td>Faulted</td><td>Normal</td></tr>
<tr><td colspan="7">Notes
(1) Faulted for passive components with no deformation limits. Emergency for active components and some passive components in which general deformations should be limited.
(2) Events need not be considered from the safety point of view, only for investment protection.</td></tr>
</table>

3.11.1 Loads through Interfaces

In the tables below are the reference interface loads, for the various loading conditions and combination, on the Vacuum Vessel interfaces. These include: supports to ground, to the Magnet, to the Divertor and to the Blanket.

3.11.1.1 Loads on VV supports

The Vacuum Vessel is vertically supported at the 9 lower ports. These vertical supports are attached to the pedestal ring and allow radial displacement and rotations around the toroidal axis.
Horizontal loads (in seismic events and VDEs) are reacted by:

- Radial restraints at the lower ports (9 locations)
- Toroidal restraints at the lower ports

(Note: The Toroidal Support Ropes proposed in FDR-01 were removed after the study effected in DCR-28)

Radial dampers located between the lower ports and the pedestal ring prevent VV horizontal displacements caused by dynamic loads (VDE and seismic loads), but allow thermal expansions. Toroidal displacements of the lower ports are restrained at the level of the vertical supports. These two horizontal support systems (radial dampers and toroidal restraints at the lower ports) act in parallel and shear the horizontal forces applied to the VV.

In addition to these mechanical support systems, the VV is magnetically coupled to the TF coils (see reference in footnote [1]). When the TF coils are energised, the VV is fully immersed in the toroidal magnetic field. As a consequence, any horizontal relative movement of the VV with respect to this magnetic field (and therefore to the TF coils) is reacted by an opposing force that is proportional to the relative displacement. This effect is equivalent to a horizontal stiffness that in this document is called magnetic VV/TFC stiffness. This effect is taken into account only for VDE cases, but not for seismic events.

[1] G_15_RE_14_R_0.1 - Magnetic forces on the ITER vacuum vessel after lateral and vertical displacements - 11 August 2004. (IDM ITER_D_22JYTB v1.0)

The design support loads calculated for different loading category (from ref. footnote [1]) are shown in table Table 3.11-3.

For the analysis of the stress in the lower ports where the supports are connected it has to be considered that:

- Maximum vertical and maximum horizontal force might occur at different toroidal position.
- The stress in the VV lower port depends on the combination of the vertical, radial, and toroidal forces and the orientation of the forces (vertical upward, vertical downward, radial inward or radial outward).

When the maximum upward vertical reaction force occurs on a single lower port, the radial reaction force is directed inward (support in compression, as the vv rocks in the same direction of the displacement). When the maximum outward radial reaction force occurs (radial support in tension), the vertical reaction on the same port is relatively small generating small stress.

Considering the difficulty of calculating, taking into account the dynamic of the system and the reversing of the applied support loads during the transient events, the maximum possible compressive force on the vertical support when the radial support force on the LP is tensile, the maximum tensile force on the LP is combined with the maximum compressive force on the lower support.

For the stress evaluation in the attachment of the VV lower port to the main VV, the load combinations reported in Table 3.11-4 have to be considered.

Table 3.11-3 Maximum VV Support Loads (MN) for different load categories at the lower ports (LP) [I,II]

	Category II		**Category III**		**Category IV**	
	Compressive	Tensile	Compressive	Tensile	Compressive	Tensile
VV toroidal force at LP	-1.2	1.2	-3.4	3.4	-1.3	1.3
VV radial force at LP	-3.7	3.4	-6.1	5.6	-3.2	3.2
VV vertical support load	-16	0	-22	6.1	-22	1.2

Notes:

I) all loads are per 40° sector.

II) Positive values indicate that the forces generate tension in the supports, negative values indicate that the forces generate compression in the supports.

III) *Clarification: The main events in category IV are SL-2 and ICE IV. These events do not generate extremely large loads on the VV supports (for seismic event that is due to the seismic isolation). Large loads on supports that are generated in VDEs events (especially with the new definition of VDE III with halo current larger than before). We have not defined any VDE IV case, and we do not combine VDE III events with any other load condition that generates a load case combination in category IV that can produce larger loads on the VV supports.For these reasons there is no cat IV events that produces loads on the supports larger than the in cat III.*

[1] ITER_D_22HST9 – Design loads for the VV supports in load case combinations

Table 3.11-4 VV support loads (MN) at the lower port for the port stress analysis

	Category II			Category III			Category IV		
Load cases	fvert	frad	ftor	fvert	frad	ftor	fvert	frad	ftor
A1) High vertical compressive load with radial force	-16	-3.7	0	-22	-6.1	0	-22	-3.2	0
A2) High vertical compressive load with toroidal force	-16	0	1.2	-22	0	3.4	-22	0	1.3
B) High radial tensile load (2)	-16	+3.4	0	-22	+5.6	0	-22	+3.2	0

Notes:

(1) Positive values indicate that the forces generate tension in the supports, negative values indicate that the forces generate compression in the supports.

(2) Considering the difficulty of calculating, taking into account the dynamic of the system and the reversing of the applied support loads during the transient event, the maximum possible compressive force on the vertical support when the radial support force on the LP is tensile, the maximum tensile force on the LP is combined with the maximum compressive force on the lower support.

(3) fvert, frad, and ftor are the vertical, radial and toroidal forces at the lower ports.

(4) See clarification III above.

3.11.1.2 Loads on the divertor cassette rails

The loads on the divertor supports are described in details in the reference given in footnote[1]. Table 3.11-5 reports a summary of the maximum loads between a single cassette and the VV at the inboard and outboard supports for single load conditions and the maximum values calculated for different loading conditions. To take into account the uncertainty of some loads (e.g disruptions and VDEs) the support loads in the combined cases are increased by a factor 1.1. A dynamic amplification factor of 1.2 is assumed for VDE loads.

Table 3.11-5 Maximum Reaction Forces (kN) for different loading conditions(*)

(on a single divertor cassette[(1)])

	Inboard support			Outboard support		
	Radial force	Toroidal force	Vertical force	Radial force	Toroidal force	Vertical force
Preload	400	0	-93	-400	0	93
Weight	-10	0	40	10	0	48
Thermal load (2)	91	0	-24	-91	0	24
Magnet fast discharge (3)	370	tbc	480	50	tbc	400
Load case combinations (4)						
Category I	950	tbc	400	-535	tbc	610
Category II	1450	tbc	750	-700	tbc	1000
Category III	1850	tbc	1300	-750	tbc	1450
Category IV	530	tbc	-170	-650	tbc	150

Notes:

(*) The estimate of the divertor support loads in disruptions and VDEs events is planned to be reviewed in mid 2007. Values in the present table refers to old divertor cassette design and old specification of disruptions and VDEs conditions. Divertor support design is also under revision with the implementation of an additional toroidal support for each cassette.

(1) A positive sign of the force value indicates that the reaction force is directed radially outwards or vertically upwards

(2) Max power density due to nuclear heating in the cassette body is about 1.1 W/cm^3.

(3) Assumption: 75% of the induced poloidal current flows into the divertor. The rest flows into the VV.

[1] ITER_D_22HT35v1.0 – Loads on the external supports of the divertor. – This document has been prepared following an old design of the divertor cassette and will be updated after new EM analyses of the cassette will be completed

(4) Values in the table represent an envelope of the maximum values for all the load combinations that have to be considered for the divertor that fall in the same load category. Radial and vertical forces reported in the table do not always occur at the same time.

3.11.1.3 Loads on the blanket supports

Main loads on the blanket modules are:

(1) Poloidal moment
(2) Radial moments
(3) Toroidal moments
(4) Poloidal forces
(5) Radial forces
(6) Toroidal force

Poloidal moments, toroidal moments, and radial forces are reacted by the flexible cartridges (radial forces). Radial moments and poloidal forces are reacted by the keys. Toroidal forces on modules 1 to 10 are reacted by the centering keys and toroidal forces on modules 11 to 18 are reacted by the stub keys.

The design loads for the blanket supports are defined in Table 3.11-6. Data in the table represent the maximum load during the transient events (the dynamic amplification factor is taken into account).

Table 3.11-6 Design loads (kN) for the blanket supports

	Blanket module	Category I and II	**Category III**
Flexible cartridge: radial force (2) side displacement flange rotation	 all all all	 +/-500 kN 300/-416 kN (2) 1 mm 3 mrad	 +/-600 kN +360/-500 1 mm 3 mrad
Inter-modular keys (1)	1 to 9	850 kN	1000 kN
Prismatic keys (1)	10	600 kN	720 kN
Stub keys (1)	11 to 18	500 kN	600 kN
Centering keys	1 to 10	250 kN	300 kN

(1) Maximum net poloidal force on the keys due to the poloidal forces and radial moments on the modules. On the module supported by the inter-modular keys the radial moments generate torsion on the keys.
(2) Support flexible cartridges for modules 11 to 18 have a smaller radius than the flexible cartridges for modules 1 to 10, therefore they have a smaller design load. The limit on the tensile load (positive value in the table) is driven by the strength of the bolt connecting the flexible cartridge to the shield module and the limit on the compressive load (negative value in the table) is driven by the buckling of the cartridge.

To prevent excessive stress in the flexible cartridges that connect radially the VV and the blankets, the maximum allowed relative rotation of the two flexible cartridge ends due to VV and blanket thermal and mechanical deformation is 3 mrad. The maximum relative transverse displacement is 1 mm.

3.12 Machine Displacements

(Updated G.Sannazzarro Oct 2006)

The displacements of the tokamak machine are summarised in the following tables.

3.12.1 Locations where the displacements are calculated

Unless differently specified components relative displacements are reported at the locations defined in Table 3.12-2 shown Figure 3.12-1 and Figure 3.12-2.

Table 3.12-1 Locations where displacements are calculated with respect to the basemat

Component	Position	Location description
TFC	TF-A	Nose of TF casing equatorial plane
	TF-B	Nose of TF coil upper straight leg
	TF-C	Start of upper OIS(1) segment
	TF-D	End of upper OIS segment
	TF-E	Start of upper intermediate OIS segment
	TF-F	End of upper intermediate OIS segment
	TF-G	Equatorial plane outer leg
	TF-H	Start of lower intermediate OIS segment
	TF-I	End of lower intermediate OIS segment
	TF-J	Start of lower OIS segment
	TF-K	End of lower OIS segment
VV	VV-A	VV inboard equatorial plane
	VV-B	VV inboard upper end of cylindrical region
	VV-C	VV top
	VV-C1	VV top hanger (at TFC level)
	VV-D	VV upper port tube flange
	VV-E	VV upper port extension (bottom)
	VV-F	VV equatorial port extension (bottom)
	VV-G	VV lower port extension (bottom)
	VV-H	VV bottom
	VV-I	VV inboard lower end of cylindrical region
VV Connecting ducts	VV-E1	Upper port connecting duct top
	VV-F1	Equatorial port connecting duct top
	VV-G1	Lowerl port connecting duct top
Thermal shield	TS-A	TS inboard equatorial plane
	TS-C	TS top
	TS-G	TS outboard equatorial plane
	TS-J	TS bottom
Poloidal field coils	PC-1	Poloidal coil 1
	PC-2	Poloidal coil 2
	PC-3	Poloidal coil 3
	PC-4	Poloidal coil 4
	PC-5	Poloidal coil 5
	PC-6	Poloidal coil 6
Central solenoid	CS-top	CS top
	CS-mid	CS equatorial plane
	CS-bot	CS bottom
Cryostat	C-UP	Cryostat at the upper port duct
	C-EP	Cryostat at the equatorial port duct
	C-LP	Cryostat at the lower port duct

(1)OIS : Outer Intercoil structure

Table 3.12-2 Locations where relative displacements are calculated

Component	Position	Location description
VV/TFC	VV/TF-A	Inboard wall at the equatorial plane
	VV/TF-B	Inboard wall top of the straight cylindrical VV region
	VV/TF-C	Top region
	VV/TF-E	Outboard wall at the equatorial plane
	VV/TF-J	Bottom region
	VV/TF-K	Inboard wall bottom of the straight cylindrical VV region
	VV/TF-C1	VV lifting plate and TFC
	UP/TF	Upper port and magnets
	EP/TF	Equatorial port and magnet
	LP/TF	Lower port and magnets
VV/cryostat	UD/C	Upper port duct and cryostat
	ED//C	Equatorial port duct and cryostat
	LD/C	Lower port duct and cryostat
Thermal shield	TS/VV-A	TS and VV at the inboard equatorial plane
	TS/VV-C	TS and VV top
	TS/VV-G	TS and VV at the outboard equatorial plane
	TS/VV-J	TS and VV at the bottom
	TS/TF-A	TS and TFC at the inboard equatorial plane
	TS/TF-C	TS and TFC top
	TS/TF-G	TS and TFC at the outboard equatorial plane
	TS/TF-J	TS and TFC at the bottom
Central solenoid	CS/TF-A	CS and TFC at the equatorial plane
	CS/TF-B	CS and TFC at the top
	CS/TF-K	CS and TFC at the bottom
In-vessel components	UPP/VV	Upper port plug (or antenna) and VV
	EPP/VV	Equatorial port plug (or antenna) and VV

Absolute and relative displacements are due to thermal expansion and mechanical loads (mainly seismic and em loads). The seismic analyses have been performed considering the two selected sites for ITER (Cadarache and Rokkasho). For the Cadarache site two types of seismic events are considered characterised by different load levels (SL-1 and SL-2). These two events are classified in category II and IV. For the Rokkasho site only one type of event is considered and is classified in category III. Both sites have seismic isolations.
The most recent seismic analyses are described in the report[1]. Results from that report have been slightly increased (about 20% and 10% for horizontal and vertical displacements respectively) to take

[1] ITER_D_22GP36v1.0 - Seismic Analysis of the ITER Tokamak with wave-propagation Rocking Effect with SSI and the Building Simulator in the Candidate Sites.

into account the uncertainties of some parameters assumed in the analyses. The seismic analyses have been performed assuming that the TF coils are not energised during the seismic event. In this case the magnetic coupling of VV and magnet has no effect.

3.12.2 Absolute displacements (respect to the basemat)

The absolute displacements (respect to the basemat) of the Magnets, Vacuum Vessel and Thermal Shield due to the excursion from normal ambient temperature (20 °C) to operating temperatures are reported in Table 3.12-3 for different operating conditions. The locations where the displacements have been computed are illustrated in Figure 3.12-1 and Figure 3.12-2.

Temperature values of various components in different operating conditions are listed in Table 2.4-1.

The in plane displacements of characteristic points of the TF coil due to the Magnet energisation are summarised in Table 3.12-4, while the toroidal displacement due to the out of plane loads are summarised in Table 3.12-5 (see N 11 DDD 115 01-06-27 R 0.2, chapter 2.2). Absolute displacements due to seismic events are reported in Table 3.12-6.

In the present design the VV supports are independent from the TF coils (in previous design the VV was supported by the TFC by flexible plates). That means that the distortion of the TFC due to the out-of-plane forces caused by the contemporaneous activation of the toroidal and poloidal magnets (displacements shown in table 3.12-5) are not transferred to the VV. In simple words there is no toroidal movement of the VV due to the TFC distortion

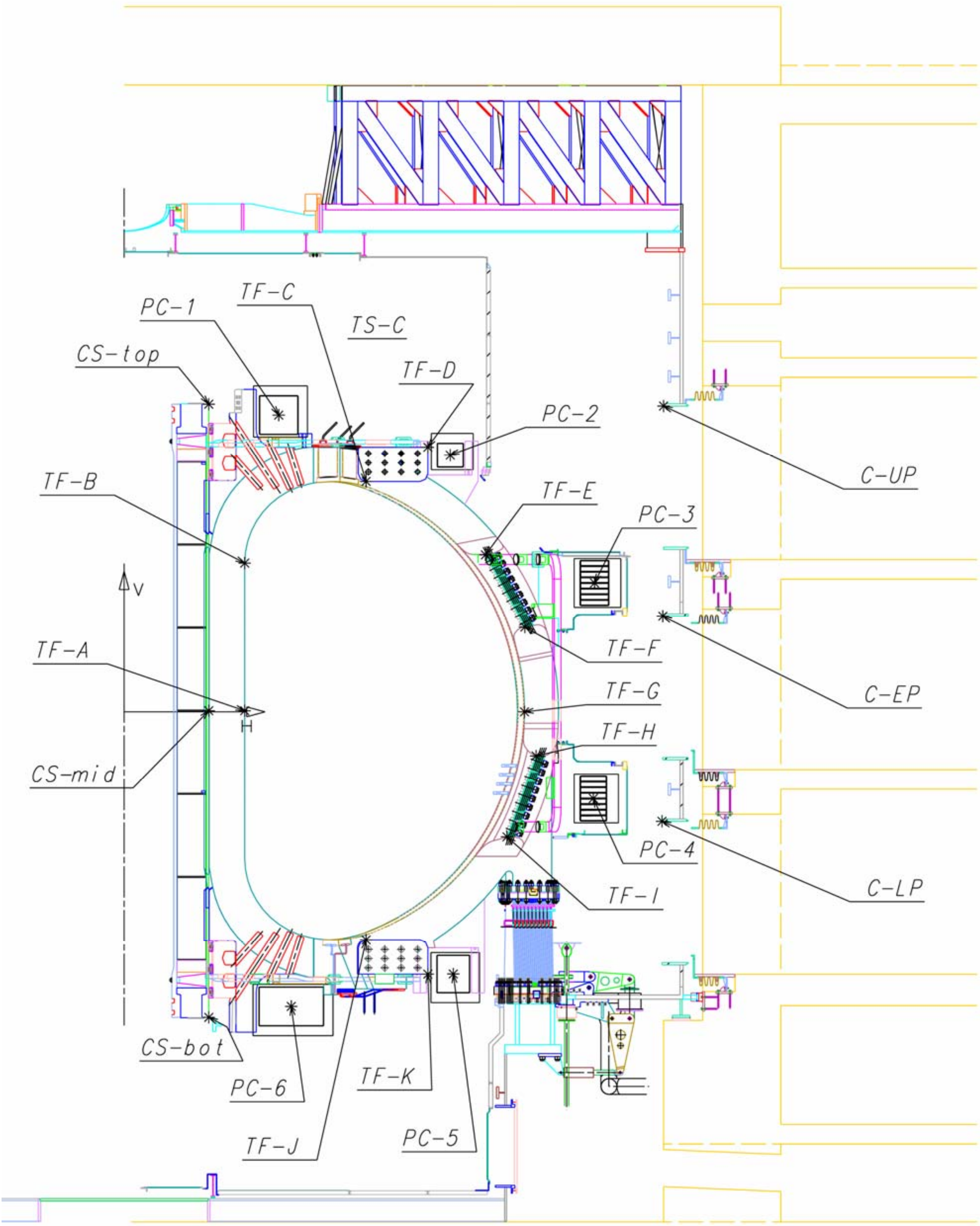


Figure 3.12-1 Identification of characteristic points around the coils

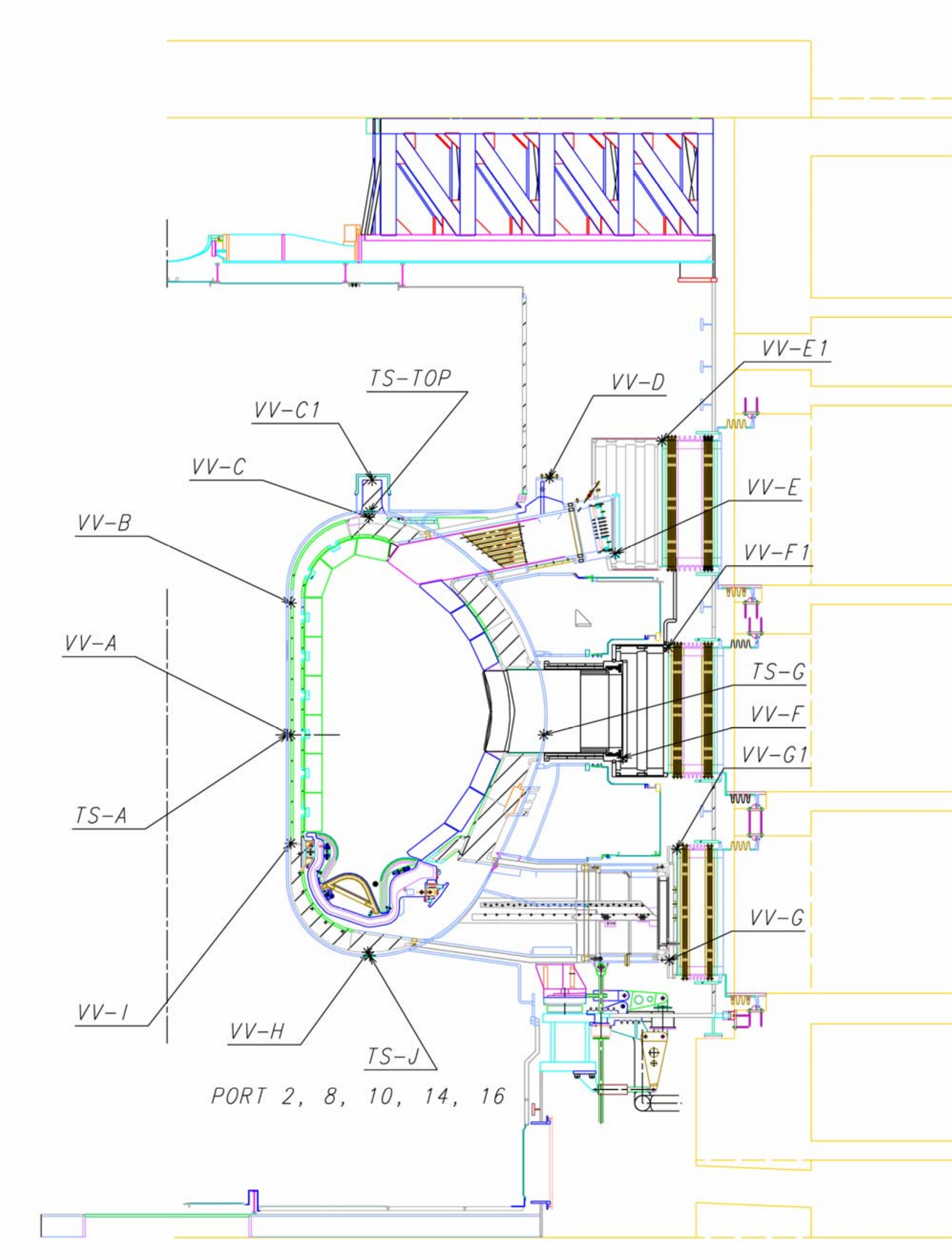


Figure 3.12-2 Identification of characteristic points around the Vacuum Vessel

Table 3.12-3 Thermal Movement of VV , Thermal Shield and Magnets

(from Room Temperature to Operating Temperature)

Position (1)	Displacement (mm)							
	Normal operating temperature		Baking temperature		VV loss of coolant		VV outgassing	
	Radial	Vertical	Radial	Vertical	Radial	Vertical	Radial	Vertical
TF-A	-9.2535	-18.178	-9.2535	-18.178	-9.2535	-18.178	0	0
TF-B	-9.2535	-29.53	-9.2535	-29.53	-9.2535	-29.53	0	0
TF-C	-18.57	-35.803	-18.57	-35.803	-18.57	-35.803	0	0
TF-D	-23.31	-38.44	-23.31	-38.44	-23.31	-38.44	0	0
TF-E	-27.729	-30.163	-27.729	-30.163	-27.729	-30.163	0	0
TF-F	-30.756	-24.604	-30.756	-24.604	-30.756	-24.604	0	0
TF-G	-30.927	-18.091	-30.927	-18.091	-30.927	-18.091	0	0
TF-H	-31.614	-14.704	-31.614	-14.704	-31.614	-14.704	0	0
TF-I	-29.382	-8.236	-29.382	-8.236	-29.382	-8.236	0	0
TF-J	-18.57	-0.553	-18.57	-0.553	-18.57	-0.553	0	0
TF-K	-23.31	2.084	-23.31	2.084	-23.31	2.084	0	0
VV-A	5.330706	10.87274	9.846685	20.05859	13.96387	25.35134	9.846685	20.05888
VV-B	5.330706	16.54363	9.846685	30.53366	13.96387	40.20633	9.846685	30.53395
VV-C	8.694599	20.24276	16.06035	37.36655	22.77564	49.89625	16.06035	37.36684
VV-C1	8.834967	21.83635	16.31963	40.31018	23.14334	54.07071	16.31963	40.31048
VV-D	16.46937	21.93709	30.42162	40.49626	43.14178	54.33458	30.42162	40.49655
VV-E	19.29986	18.67558	35.65	34.47172	50.5563	45.79101	35.65	34.47201
VV-F	19.63014	9.881899	36.26008	18.22835	51.42147	22.75582	36.26008	18.22864
VV-G	21.63328	1.190604	39.96021	2.174107	56.66874	-0.01117	39.96021	2.174399
VV-H	8.646708	1.520883	15.97188	2.784186	22.65019	0.854004	15.97188	2.784479
VV-I	5.330706	6.217452	9.846685	11.45952	13.96387	13.15674	9.846685	11.45981
VV-E1	21.31952	23.52243	39.38064	43.42464	55.84682	58.48741	39.38064	43.42493
VV-F1	21.60026	14.63792	39.89921	27.0135	56.58222	35.21429	39.89921	27.01379
VV-G1	21.96356	5.968091	40.57029	10.99891	57.53391	12.50354	40.57029	10.9992
TS-A	-8.66088	-18.705	-8.66088	-18.705	-8.66088	-18.705	6.531761	-5.86795
TS-C	-14.5894	-34.7654	-14.5894	-34.7654	-14.5894	-34.7654	11.00283	6.244329
TS-G	-27.1225	-18.705	-27.1225	-18.705	-27.1225	-18.705	20.45494	-5.86795
TS-J	-14.5894	-2.82944	-14.5894	-2.82944	-14.5894	-2.82944	11.00283	-17.8408
PC-1	-11.865	-40.918	-11.865	-40.918	-11.865	-40.918	0	0
PC-2	-25.032	-37.828	-25.032	-37.828	-25.032	-37.828	0	0
PC-3	-36.1005	-28.003	-36.1005	-28.003	-36.1005	-28.003	0	0
PC-4	-36.0105	-11.428	-36.0105	-11.428	-36.0105	-11.428	0	0
PC-5	-25.26	2.072	-25.26	2.072	-25.26	2.072	0	0
PC-6	-12.828	4.562	-12.828	4.562	-12.828	4.562	0	0
CS-top	-6.4965	-41.758	-6.4965	-41.758	-6.4965	-41.758	0	0
CS-mid	-6.4965	-18.178	-6.4965	-18.178	-6.4965	-18.178	0	0
CS-bot	-6.4965	5.402	-6.4965	5.402	-6.4965	5.402	0	0
UP-plug	-9.2535	-18.178	-9.2535	-18.178	-9.2535	-18.178	0	0
EP-plug	-9.2535	-29.53	-9.2535	-29.53	-9.2535	-29.53	0	0
C-UP								
C-EP								
C-LP								

(1)Selected positions are listed in Table 3.12-1.

Table 3.12-4 Radial and Vertical Displacement along TF Coil Perimeter for In-plane Load Cases

Position(1)	Displacement (mm)					
	After preload of precompressing ring		After TF coils cooldown		At TF coils energisation	
	Radial	Vertical	Radial	Vertical	Radial	Vertical
TF-A	2.66	-0.03	-3.96	-16.69	-11.84	-14.30
TF-B	0.75	-0.69	-5.89	-29.98	-9.18	-21.70
TF-C	-1.14	1.69	-19.67	-34.93	-19.06	-25.81
TF-D	-1.18	1.09	-24.08	-33.81	-21.86	-24.16
TF-E	-1.52	0.52	-29.95	-30.10	-27.00	-23.05
TF-F	-1.76	0.21	-34.63	-22.81	-34.64	-19.33
TF-G	-1.95	0.14	-35.79	-16.63	-36.67	-14.50
TF-H	-1.90	0.12	-35.56	-14.11	-36.19	-12.47
TF-I	-1.88	-0.02	-32.43	-6.04	-30.42	-6.66
TF-J	-1.23	-0.88	-24.10	0.53	-21.76	-4.71
TF-K	-1.13	-1.64	-19.67	1.58	-19.04	-2.86

(1)Selected positions are listed in Table 3.12-1.

Table 3.12-5 Toroidal Displacement of characteristics points of the TF Coil

	Displacement (mm)							
Position(1)	Initmag1	Initmag2	Initmag3	Soflat	Eoburn1	Eoburn2	Eoburn3	Eoburn4
TF-A	- 0.19	- 0.05	- 0.02	2.57	2.91	2.46	3.16	2.93
TF-B	- 0.39	0.00	0.00	1.78	2.85	2.37	3.03	2.66
TF-C	- 0.97	- 0.07	- 0.19	10.00	13.20	11.81	14.09	14.30
TF-D	- 0.88	0.00	- 0.13	12.22	16.20	14.24	17.31	17.23
TF-E	- 0.57	- 0.07	- 0.08	12.22	16.80	14.08	18.60	17.14
TF-F	- 0.51	- 0.08	- 0.07	11.92	16.40	13.75	18.37	16.78
TF-G	- 0.20	- 0.06	- 0.03	6.68	8.89	7.79	9.87	9.43
TF-H	- 0.06	- 0.05	- 0.02	4.51	5.85	5.31	6.36	6.35
TF-I	0.00	- 0.01	0.00	1.39	1.80	1.64	1.92	1.95
TF-J	- 0.78	0.63	0.21	- 5.84	- 6.28	- 7.56	- 4.27	- 7.81
TF-K	- 0.53	0.71	0.34	- 4.75	- 5.13	- 6.41	- 3.26	- 6.56

(1)Selected positions are listed in Table 3.12-1.

Table 3.12-6 Absolute Displacements (mm) respect to free soil due to seismic events

	Cadarache SL-2(1)
TFC radial displ (mm)	77
VV radial displ (mm)	78

(1) For SL-1 event scale values by a factor 0.34

Table 3.12-7 Maximum Absolute Displacements (mm) of VV and TFC due to an Asymmetric VDE III [(1)] (TFC displ TBD)

Position [(2)]	Radial	Toroidal	Vertical
VV-A equat. inboard	8.8	8.8	13
VV-B top of straight parts	9.0	9.0	13
VV-C top	11	9.1	9.4
VV-F equat. outboard	10	9.0	6.0
VV-H bottom	9.0	9.0	11
VV-I bottom of straight part	10	9.0	14
VV-E1 Upper port connecting duct	12	10	7
VV-F1 Eq port connecting duct	10	9.0	15
VV-G1 Lower port connecting duct	12	9.0	7

(1) For a VDE II values have to scaled by a factor 0.6
(2) Selected positions are listed in 3.12-1

Table 3.12-8 Maximum Absolute Displacements in the VVTS for Selected Load Combinations

(with respect to Assembly Condition)

Regime	**Direction**	**Inboard VVTS**	**Outboard VVTS**	**Upper port**	**Equator Port**	**Lower port**	**Port support plate**
Dead weight (DW)	radial vertical toroidal	0.7 1.5 -	0.8 2.5 -	0.3 1.5 -	0.7 1.5 1.9	0.3 2.5 1.1	0.5 - -
Plasma Disruption I,II+ DW	radial vertical toroidal	1.0 1.4 -	0.9 2.4 -	0.4 1.6 0.3	0.9 1.7 2.3	0.3 2.4 1.8	2.2 - -
Fast VDE, III + DW	radial vertical toroidal	0.7 1.6 -	1.0 2.7 -	0.6 2.8 1.6	0.9 1.7 2.5	0.4 2.7 2.1	2.8 - -
Slow VDE, III + DW	radial vertical toroidal	0.7 1.6 -	1.1 3.6 -	0.8 2.8 1.6	0.9 2.3 3.1	0.5 3.6 2.8	3.9 - -
TF fast discharge + DW	radial vertical toroidal	0.8 1.4 -	0.5 1.5 -	0.4 1.5 0.5	0.5 0.9 1.0	0.4 1.4 0.6	1.2 - -
TFC out of plane deformation + DW	radial vertical toroidal	0.2 1.0 -	0.9 3.2 -	0.4 3.2 17.4	0.7 3.5 19.2	0.3 2.1 18.3	0.5 - -

3.12.3 Relative displacements (respect to the basemat)

Table 3.12-9 Relative Displacements (mm) between components for seismic events

Position[1]	Cadarache SL-2 [2]		
	Radial	**Toroidal**	**Vertical**
VV/TF-A equat inboard	1.9	1.8	10.2
VV/TF-B top of straight parts	2.1	2.0	10.1
VV/TF-C top	2.3	2.0	7.7
VV/TFC upper port up	2.2	2.0	5.6
VV/TFC upper port bottom	2.2	2.0	4.3
VV/TF-E Outboard wall at eq. plane	2.2	1.9	3.9
VV/TF-J bottom	2.0	1.6	7.5
VV/TF-K bottom of straight part	1.7	1.7	10.2
VV/TF-C1 VV lifting plate and TFC top	3.4	2.1	7.7
UD/C VV/Cryo upper duct	3.4	2.8	2.5
ED/C VV/Cryo eq duct	2.7	2.3	7.2
LD/C VV/Cryo low duct	3.1	1.7	2.8
UPP/VV VV Upper Port/ Upper Port Plug	0.6	0.2	0.2
EPP/VV VV Equat Port/ Equat Port Plug	0.6	0.1	0.3
CS/TFC-A equatorial plane	0.6	0.6	1.0

(1) Selected positions are listed in 3.12-2
(2) For SL-1 event scale values by a factor 0.34

Table 3.12-10 Relative Displacements (mm) between VV and TFC due to VDE III[1]

Position[2]	**Radial**	**Toroidal**	**Vertical**
VV/TF-A equat. inboard	9.0	9.0	14
VV/TF-B top of straight parts	12	11	12
VV/TF-C top	12	11	10
VV/TF-E equat. outboard	11	10	6
VV/TF-J bottom	10	9.5	11
VV/TF-K bottom of straight part	10	9.1	11
VV/TF-C1 VV lifting plate and TFC	13	12	10

(1) For a VDE II values have to scaled by a factor 0.6
(2) Selected positions are listed in 3.12-2

Table 3.12-11 Displacements (mm) of VV and TFC due to seismic event respect to the basemat

Position	Cadarache SL-2 (1)	
	radial/toroidal	**vertical**
VV- Equat. plane inboard	4.4	11
VV – Top	5.1	9.1
VV - Equat. outboard	4.7	6.5
VV – Bottom	3.7	8.8
TF - Equat. inboard	1.9	2.6
TF – Top	2.1	2.3
TF - Equat. plane outboard	2.0	2.1
TF - Bottom	1.6	2.6

(1) For SL-1 event scale values by a factor 0.34

3.13 Machine Accelerations

3.13.1 Locations where the accelerations are calculated

The locations are those shown in the figures in section 12 (displacements)

3.13.2 Accelerations

The accelerations are the following tables.

Table 3.13-1: Maximum accelerations (m/s^2) of VV and magnets due to seismic event

	Cadarache SL-2 (1)	
Position	**radial/toroidal**	**vertical**
VV - Equat. plane inboard	1.2	21
VV - Top	2.1	17
VV - Equat. outboard	2.1	8.6
VV - Bottom	2.3	16
VV - Upper duct	5.2	7.7
VV - Equatorial duct	2.6	14
VV - Lower duct	4.4	7.2
VV - Upper plug	2.3	15
VV - Equatorial plug	2.3	15
TF - Equat. inboard	1.1	8.4
TF - Top	1.4	7.2
TF - Equat. plane outboard	1.7	3.2
TF - Bottom	1.3	7.2
Central Solenoid	2.3	12
PFC 1 and PFC 6	2.0	8.0
PFC 2, 3, 4, and 5	2.0	3.3

(1) For SL-1 event scale values by a factor 0.34

Table 3.13-2: Maximum accelerations (m/s^2) of VV and magnets due to slow VDE III events

Position	**radial/toroidal**	**vertical**
VV- Equat. plane inboard	6.8	4.4
VV - Top	7.6	3.2
VV - Equat. outboard	7.3	1.8
VV - Bottom	6.9	3.5
Magnet average	4	2

(1) For VDE II scale values by a factor 0.6

4 Systems Functions – Configuration – Parameters

The function and principal configuration/parameters of the various systems into which the ITER plant is subdivided is described in the following sections.

The plasma control system function is not described below, because its functional requirements and design are described in the CSD (Control System Design and Assessment) plant-level assessment document. With that exception, the systems described below cover the complete plant.

4.1 Magnets (WBS 1.1)

(Updated N.Mitchell Sept 2006

4.1.1 Functional Requirements

The magnet system provides the following.

- TF (toroidal field) coils for the magnetic field that gives the specified plasma safety factor during the various phases of operation.
- A CS (central solenoid) for the majority of the magnetic flux change needed to initiate the plasma, generate the plasma current and maintain this current during the burn time. It contributes towards the fields needed to and control the plasma.
- PF (poloidal field) coils for the magnetic fields that shape the plasma and control its position during the various phases of operation, including plasma initiation, growth, burn and shut-down. The PF coils also contribute a portion of the magnetic flux change needed to ramp up and maintain the plasma current.
- CC (correction coils) to compensate for field errors due to design asymmetries and geometric tolerances in the as-built machine, and to stabilize MHD instabilities such as resistive wall modes and to provide the field required for the compensation of some helical harmonics of the magnetic field generated by irregularities in the coil current distribution.
- TF coil structures are integrated with the PF coil supports and the CS structures to restrain the electromagnetic loads on the coils under normal operating and fault conditions. The TF coil structures also resist the gravity and seismic loads of the magnet system.

4.1.2 Configuration

4.1.2.1 Overall

4.1.2.1.1 Magnet Content

The magnet system for ITER consists of:

- 18 Toroidal Field (TF) coils,
- Central Solenoid (CS) with 6 segments
- 6 Poloidal Field (PF) coils
- 18 Correction Coils (CCs)

4.1.2.1.2 Superconductor Choice

Both CS and TF coils operate at high field and use Nb_3Sn-type superconductor. The PF coils and CCs use NbTi superconductor.

4.1.2.1.3 Magnet Cooling Conditions

All coils are cooled in normal operation with supercritical helium in the range 4.4-4.7K.

Conductor Configuration: The conductor type for the main coils is a cable-in-conduit conductor with a circular multistage cable containing a central cooling channel.

4.1.2.1.4 Magnet Electrical Insulation System

The coil electrical insulation system is composed of multiple layers of polyimide film-glass impregnated with epoxy resin.

4.1.2.1.5 Magnet Void Filling

Epoxy-glass is used extensively to fill tolerance gaps.

4.1.2.1.6 Winding Configuration

The CS and PF coils are pancake wound with a conductor that has a square outer section. The TF coils use a conductor with a circular outer section that is contained in grooves in so-called “radial plates”. There is one radial plate for each double pancake and the conductor is contained in grooves on each side.

4.1.2.1.7 Magnet Fault Recovery

All TF coils, the CS, the upper PF coils and CCs are designed to be removable from the machine in case of a major fault. The PF coils are designed with extra capacity and temperature margin. Hence individual double pancakes of the PF coils may be disconnected and by-passed in-situ in case of fault, since the PF coils have accessible joints located on their external side, with jumpers already in position (the condition with a double pancake bypassed is referred to as “backup” in the following tables). In addition, the cryostat design allows the lower (trapped) PF coils to be rewound in situ under the machine.

Table 4.1-1 Overall Magnet System Parameters

(DDD11 table 1.1.1-1)

Number of TF coils	18
Magnetic energy in TF coils (GJ)	~41
Maximum field in TF coils (T)	11.8
Centering force per TF coil (MN)	403
Vertical force per half TF coil (MN)	205
CS peak field (T)	13.0

Table 4.1-2 Weight of Magnet System Components

(from DDD1.1 Table 1.2.8-1)

Magnet System	Weight t	No.	Total weight t
TF Coil		18 1	6444 358
Winding Pack	***107.8***	***1***	*105.3*
TF Coil Case	***195***	***1***	*186.2*
Gap between WP and TFCC	***1***		*1*
IIS keys set	***0.05***	***13***	*0.7*
Upper&Lower OIS bolting set	***1.3***	***2***	*2.6*
Friction joints			*11.8*
Pre-compression system			*25.4*
Gravity support	***24.9***		*25*
PF Coils			2071.6
Winding Pack			
PF1			*134.3*
PF2			*117.1*
PF3			*294.5*
PF4			*261.7*
PF5			*234.7*
PF6			*245.5*
Clamping system			
PF1	***4.89***		*44.1*
PF2	***3.45***		*62.1*
PF3	***4.54***		*81.7*
PF4	***4.39***		*79.0*
PF5	***4.9***		*88.2*
PF6	***12.52***		*112.7*
PF3+4 Support	***11.16***		*201*
Protection cover			*32.6*
Spacers blocks			*82.6*
Correction Coils			93.6
Top correction coil	***3.2***	***6***	*19.2*
Side correction coil	***8.7***	***6***	*52.2*
Bottom correction coil	***3.7***	***6***	*22.2*
Central Solenoid			939.1
Module	***104.6 t***	***6***	*627.6*
Pre-load structure	***255.9***		*255.9*
CS support structure			*43.4*
Bolting for structures			*12.2*
TOTAL			*9548t*

Note : This is approx. 500 tons lighter than the 2005 estimate, due to a more thorough evaluation of the coil packing and jacket volumes and densities.

4.1.2.2 Conductor Parameters

Table 4.1-3 TF Conductor

(DDD 11 Table 1.2.1-3)

Total number of turns per coil	134
Jacket type	circular steel
Number of turns per pancake (from side pancake to middle pancake)	3-9-11-11-11-11-11
Type of strand	Nb_3Sn
Operating current (kA)	68.00
Nominal peak field (T)	11.8
Operating temperature at peak field (K)	5.0
Total operating strain (%)	-0.77
Equivalent discharge time constant (s) hot spot	15 (1)
n at operating point	7
Tcs (Current sharing temperature) (K)	5.7
Iop/Ic (Operating current/critical current)	0.78
Cable diameter (mm)	40.5
Central spiral outer x inner diameter (mm)	9 x 7
Conductor outer diameter (mm)	43.7
Jacket material	316LN(modified variant)
SC and Cu strand diameter (mm)	0.82
SC strand cu : non-cu	1.0
Cabling pattern	((2s/c+1Cu)x3x5x5+core)x6
SC strand number	900
Cu core at 4th stage	3x4 Cu wires 0.82mm
Wrap coverage on final substage (0.05mm SS)	50%
Local void fraction (%) in strand bundle	33.2
SC strand weight/m of conductor (kg/m)	4.50 (2)

Notes

(1) note the use of non-linear discharge resistors

(2) $\cos\theta = 0.95$, d_{strand} 9 kg/dm^3

Table 4.1-4 CS Conductor

(DDD 11 Table 1.2.3-2)

Number of turns per module	549
Jacket type	Square steel
Type of strand	Nb_3Sn
Operating current (kA) IM/EOB	40.0/45.0
Nominal peak field (T) IM/EOB	13.0/12.4
Operating temperature (K)	4.7
Operating strain (%)	-0.74
Equivalent discharge time constant (s)	11.5
n at operating condition	7
Tcs (Current sharing temperature) (K) @ 13.0T	5.4
Iop/Ic (Operating current/critical current) IM	0.80
Cable diameter (mm)	32.6
Central spiral outer x inner diameter (mm)	9 x 7
Conductor outer dimensions (mm)	49 x 49
Jacket material	JK2LB
SC strand diameter (mm)	0.83
SC strand cu : non-cu	1.0
Cabling pattern	(2s/c+1Cu)x3x4x4x6
SC strand number	576
Wrap coverage on final substage (0.05mm SS)	80%
Local void fraction (%) in strand bundle	33.2
SC strand weight/m of conductor (kg/m)	2.95 [(1)]

(1) $\cos\theta = 0.95$, d_{strand} 9 kg/dm^3

Table 4.1-5 PF Conductors

(DDD 11 Table 1.2.4-7)

	PF1 & 6	PF2, 3 & 4	PF5
Coolant normal/backup	inlet 4.7K/4.4K	inlet 4.7K	inlet 4.7K
Type of strand	NbTi	NbTi	NbTi
Operating current (kA) normal/ backup(1)	45/52	45 (2) /52	45/52
Nominal peak field (T) normal/backup	6.0/6.4	4.0	5.0
Operating temperature (K) normal / backup	5.0/4.7	5.0	5.0
Equivalent discharge time constant (s) hot spot	18	18	18
Tcs (Current sharing temperature) (K) normal/backup	6.5/6.27	6.65/6.51	6.60/6.51
Iop/Ic (Operating current/critical current) normal/backup	0.127/0.144	0.365/0.422	0.264/0.305
Cable diameter (mm)	38.2	34.5	35.4
Central spiral outer x inner diameter (mm)	12x10	12x10	12x10
Conductor outer dimensions (mm)	53.8x53.8	52.3x53.2	51.9x51.9
Jacket material	316L	316L	316L
SC strand diameter (mm)	0.73	0.73	0.72
SC strand cu:non-cu	1.6	6.9	4.4
Cabling pattern (+ is Cu core)	3x4x4x5x6	((3x3x4+1)x4+1)x 6	((3x3x4+1)x5+1)x6
SC strand number	1440	864	1080
Cu core 2/3/4 stage (mm)	0/0/0	0/1.8/3.5	0/1.2/2.7
Local void fraction (%) in strand bundle	34.5	34.2	34.3
SC strand weight/m of conductor (kg/m)	4.885	2.931	3.564

Notes

(1) PF “backup” is for the case of failure of one pancake. (see also Table 4.16-4)

(2) although the P2 conductor is designed for 45kA, the operating current in normal conditions is 41kA. This is because this coil contains only 5 double pancakes and in back-up mode, with one DP lost, the current cannot exceed 52kA. Therefore, the normal operating current has to be limited to 41kA.

Table 4.1-6 Number and Capacity of Current Leads

(from DDD1.1, Table 1.2.6-6)

Coils	Number of pairs	Maximum Current (kA)	Max. Voltage (kV) normal/fault
TF Coil	9	68	3.5/15
PF Coil	6	52	17/17
Correction Coil	9	10	7.5/15
CS Coil	6	45	20/20

4.1.2.3 Strand Parameters

Table 4.1-7 Performance Assumptions for Nb3Sn Strand

(DDD 11 Table 1.1.2-1)

	CS		TF	
Jc (A/mm^2) requirement (at 10μV/m) ***	260 at 12.7T effective field and 5.2K, -0.74%*		291 at 11.3T effective field and 5.7K, -0.77%*	
Hysteresis loss (design basis/max acceptable) mJ/cc of strand +/-3T cycle	600/1000		1000/1000	
Cu:non-Cu	1.0		1.0	
RRR (after coating and heat treatment)	>100		>100	
Strand diameter	0.83mm		0.82mm	
n value 12T, 4.2K, from 10 to 100 μV/m	20		20	
Cr plating thickness [μm]	1-2		1-2	
Reference heat treatment at which jc is achieved	210^{o}C for 50hrs, 340^{o}C for 25hrs, 450^{o}C for 25hrs, 575^{o}C for 100hrs, 650^{o}C for 200hrs Transition rate 5°C/hr		210^{o}C for 50hrs, 340^{o}C for 25hrs, 450^{o}C for 25hrs, 575^{o}C for 100hrs, 650^{o}C for 200hrs Transition rate 5°C/hr	
Possible scaling interpretations to derive the jc at 12T and 4.2K				
Jc (A/mm^2) 10μV/m 12T, 4.2K, unconstrained isolated strand with assumed strain -0.25% based on Summers scaling**	950	750	800	700
Bc2om** (Summer's scaling) T	28	32	28	32
Tcom** (Summer's scaling) K	18	17	18	17
Co** (Summer's scaling) A/mm^2	1.555x10^4	0.998x10^4	1.303x10^4	0.93x10^4

* these values are the assumed total strain at the design point

** values for use in the Summer's scaling law, see DRG1 Annex, Magnet Superconducting and Electrical Design Criteria and L.T. Summers, M.W. Guinan, J.R. Miller, and P.A. Hahn, A Model for the Prediction of Nb3Sn Critical Current as a Function of Field, Temperature, Strain, and Radiation Damage, IEEE Trans. on Mag., Vol. 27, No27, March 1991

*** The estimated total strand strain in the cable has been derived by interpretation from ITER model coil and short sample test results, using different scaling formulae . The degradation in jc is represented entirely by an extra strain component in this model. This is not entirely satisfactory, particularly as regards the transverse load-rleated degradation, and is discussed further in DDD 1.1 section 1.1.3.1.

Table 4.1-8 Performance Assumptions for NbTi Strand

(DDD 11 Table 1.1.2-2)

	P1/6	P2/3/4	P5	Side CC	T&B CC
Critical NbTi current density (A/ mm^2) @ coil operating point	196 A/mm^2 at 6.5K, 6T peak field	994 A/mm^2 at 6.5K, 4T peak field	559 A/mm^2 at 6.5K, 5T peak field	201 A/mm^2 at 7.6K, 3.4T peak field	193 A/mm^2 at 7.3K, 4.2T peak field
Filament diameter (μm)**	7*	7*	7*	7*	7*
Outer Ni coating thickness (μm)	1-2	1-2	1-2	1-2	1-2
n value @5T, 4.2K, between 10 and 100μV/m	20	20	20	20	20
Strand diameter (mm)	0.73	0.73	0.72	0.73	0.73
Cu : non-Cu	1.6	6.9	4.4	1.4	1.4
RRR	>150	>150	>150	>150	>150
*Possible scaling interpretations to derive the jc at 5T and 4.2K based on Bottura scaling ****					
jc at 5T and 4.2K, A/mm2	2900	2900	2900	2900	2900
Scaling parameters Tco, Bc2o,Co A/mm^2	9.2K, 14.5T, 8.03×10^4	9.2K, 14.5T, 8.03×10^4	9.2K, 14.5T, 8.03×10^4	9.2K, 14.5T, 8.03×10^4	9.2K, 14.5T, 8.03×10^4
Scaling parameters n, alpha, beta, gamma	1.0,0.57, 0.9,2.32	1.0,0.57, 0.9,2.32	1.0,0.57, 0.9,2.32	1.0,0.57, 0.9,2.32	1.0,0.57, 0.9,2.32

* these parameters are compatible with a strand without internal CuNi barriers. For some coils used with high frequency feedback control (side correction coils) it may be appropriate to specify a lower loss strand once the power supply controller design is finalised, section 1.1.4.3

** the filament diameter is set to avoid export licensing problems under the Wassenaar Agreement regarding sensitive technology and this is therefore the minimum value

*** L. Bottura "A practical fit for the critical surface of NbTi," IEEE Trans App Sup, Vol 10, No 1, March 2000

4.1.2.4 TF Configuration

4.1.2.4.1 Magnet Structural Arrangement

The TF coil case encloses the winding pack and is the main structural component of the magnet system.
The TF coil inboard legs are wedged all along their side walls in operation, with friction playing an important role in supporting the out-of-plane magnetic forces.
In the curved regions above and below the inboard leg, the coils are structurally linked by means of two upper and two lower precompression rings formed from unidirectional bonded glass fibre and by four upper and four lower poloidal shear keys arranged normal to the coil centreline.
In the outboard region, the out-of-plane support is provided by four sets of Outer Intercoil Structures (OIS) integrated with the TF coil cases and positioned around the perimeter within the constraints provided by the access ducts to the vacuum vessel.
The OIS form four toroidal rings and act as shear panels in combination with the TF coil cases.

4.1.2.4.2 Avoidance of Toroidal Current Paths in the Magnet

There is low voltage electrical insulation toroidally between TF coils in the inboard leg wedged region, at the poloidal shear keys and between the OIS connecting elements.

Table 4.1-9 Parameters of TF Magnet

(from DDD 1.1 Table 1.1.1-2)

	TF Coil
Maximum coil current (MA)	9.1
Number of turns per TF coil/CS module: Radial Toroidal/Vertical Total	 11 14 134
Conductor unit length (m)	760 (double pancakes)
Turn voltage in normal operation (V)	26.5 (discharge)
Ground/Terminal voltage (kV) in normal operation (including fast discharge)	3.5/3.5 * (2 coils in series)
Ground/Terminal voltage (kV) in faulted operation	15/3.5
Coil DC Ground Test Voltage (kV)	16
Number of current lead pairs	9

* A voltage surge (of a few ms) caused by the jitter of the switches may reach about 5kV.

The TF case is made of ausentitic stainless steel. The local yield stress requirements are graded according to performance, to minimise cost. The peak performance requires a yield stress in excess of 1000MPa at 4.2K and a fracture toughness in excess of 200MPam$^{1/2}$ which requires a narrow material specification, extra material refining (electo-slag refining), and forging of components. The lowest performance is a yield stress in excess of 550MPa and a fracture toughness in excess of 200MPam$^{1/2}$ which can be achieved with a standard ANSI 316LN.

4.1.2.5 Central Solenoid Configuration

4.1.2.5.1 CS Arrangement

The CS consists of a vertical stack of six independent winding pack modules, which is hung from the top of the TF coils through its pre-load structure. At the bottom, there is a spring loaded sliding connection to provide a locating mechanism and support against dynamic horizontal forces.

4.1.2.5.2 CS Preload

The CS assembly is preloaded by a structure, which consists of a set of tie-plates located at the inner and outer diameters of the coil stack, providing axial pressure on the stack.

4.1.2.5.3 CS Modularisation

The number of CS modules (i.e. 6) has been selected to satisfy the plasma shaping requirements.

4.1.2.5.4 CS Force Transfer

The CS stack is self supporting against the coil radial forces and most of the vertical forces, with the support to the TF coils reacting only the weight and net vertical components resulting from up-down asymmetry of the poloidal field configuration.

Table 4.1-10 Parameters of CS

(DDD 11 Table 1.1.1-2)

	CS
	Modules CS 1, 2, 3
Maximum coil current (MA)	21.9 at IM (13.0T)
Number of turns per TF coil/CS module: Radial Toroidal/Vertical **Total**	 14 40 549
Conductor unit length (m)	895 (for 6 pancakes) 594 (for 4 pancakes)
Turn voltage in normal operation (V)	20 (IM)
Ground/Terminal voltage (kV) in normal operation (including fast discharge)	19.5/19.5*
Ground/Terminal voltage (kV) in faulted operation	19.5/19.5*
Coil DC Ground Test Voltage (kV)	41*
Number of current lead pairs	6 **

* These are the CS2U/L values which are higher than the other coils, however all coils are designed to the same conditions

** The current leads for CS modules 1 (upper and lower) are connected in series outside the machine

4.1.2.6 PF Configuration

4.1.2.6.1 PF Force Transfer

The six PF coils (PF1 to PF6) are attached to the TF coil cases through flexible plates or sliding supports allowing radial displacements. The vertical loads are transmitted to the TF Cases.

Table 4.1-11 Parameters for PF coils

(DDD 11 Table 1.1.1-3)

	PF1	PF2	PF3	PF4	PF5	PF6
Max. coil current capacity (MA)*	11.21	4.35	8.33	7.61	9.90	19.13
Number of turns per coil: Radial Vertical Total	 16 16 249	 11 10 106	 12 16 185	 11 16 169	 14 16 217	 27 16 425
Conductor unit length (m) (double pancake, two-in-hand)	387	556	874	799	718	714
Maximum Turn voltage (V) in normal operation**	750	1600	1000	1000	1000	750
Ground/Terminal voltage (kV)** in normal operation	12/12	17/17	17/17	17/17	17/17	12/12
Ground/Terminal voltage (kV)** in faulted operation	12/12	17/17	17/17	17/17	17/17	12/12
Coil DC Ground Test Voltage (kV)	25	35	35	35	35	25

* Current capacity given by maximum conductor current (45kA) x number of turns

**Voltages are for a two in hand winding configuration, double pancakes, in normal operation.

4.1.2.7 CC Configuration

4.1.2.7.1 CC Arrangement

Outside the TF coils are located three independent sets of CCs, each consisting of six coils arranged around the toroidal circumference above, at and below the equator.

4.1.2.7.2 CC Connection

Within each set, pairs of coils on opposite sides of the machine are connected in series.

Table 4.1-12 Parameters for CC coils

(DDD 11 Table 1.1.1-4)

	Top	Side	Bottom
Max. coil current capacity (kA)*	140	280	180
Number of turns per coil: Number of single pancakes	14 2	28 4	18 2
Conductor unit length (m) (double pancake)	246	405	316
Maximum Turn voltage (V)**	3.6	268	3.6
Ground/Terminal voltage (kV)**	0.05/0.1	7.5/15	0.05/0.1
Ground/Terminal voltage (kV) faulted operation	0.1/0.1	15/15	0.1/0.1
Coil DC Ground Test Voltage (kV)	1.5	31	1.5

* Current capacity given by maximum conductor current (10kA) x number of turns

** Voltages are for two coils in series, each with a one in hand winding configuration, in normal operation.

4.1.2.8 Precompression Rings Parameters

The glass fibre precompression rings provide >30MN inward force on top and bottom of each TF coil at 5K.

Table 4.1-13 Parameters for the Glass Fibre Precompression Rings

(DDD 11 Table 1.1.2-6)

Material	Glass fibre (unidirectional)
Peak tensile stress at room temperature (MPa)	496
Ratio of peak to ultimate stress at RT (estimated)	4.6
Total cross-sectional area of three rings (m^2) (each at top and bottom)	0.248
Radial displacement to apply precompression at room temperature (mm)	24 (0.9% strain)

(Note that the recent DCR-49, 2006, introduced a change such that there are now three rings, with three spares stored underneath the machine).

4.1.2.9 Mechanical Interfaces

See also § 3.11 and § 3.6 for mechanical and thermal loads. Displacements of the magnet during operation can be found in § 3.12

4.1.2.9.1 Magnet Gravity Supports

The magnet system gravity supports are composed of pedestals (one under each TF coil), with flexible elements to allow radial displacements. Each TF coil is electrically insulated from its own support (and grounded individually through the feeder ducts). The gravity supports also transmit electromagnetic loads between the VV and coils.

4.1.2.9.2 Principle of Magnet Support Structure

The TF coil structures provides a load path to support the self-equilibrating electromagnetic forces acting between the TF coils, the CS, the PF coils, CCs and the vacuum vessel (VV) during normal operation, plasma disruption and fault events. The magnets must withstand the following;

- Electromagnetic loads acting on the magnets.
- Self weight.
- Seismic loads on the magnets.
- Forces applied during installation and assembly of the magnets.
- Helium coolant pressure loads.

4.1.2.10 Cooling Conditions

See § 3.6 and § 3.7 and § 4.11

4.1.2.11 Electrical Interfaces

See Section 4.16 Pulsed and Steady State Power Supplies (WBS 4.1,4.2,4.3)
See also Section 3.9 Grounding

4.1.2.12 Magnet Discharge Parameters

Table 4.1-14 Magnet Charge, Discharge, Fast Discharge and Quench Parameters

Parameter		Value
TF charge time		>30mins
TF normal "slow" discharge time		30mins
Time for operation after TF fast discharge (2)		4 days
"Fast" Discharge sequencing of PF, CS		Simultaneous
Upset "Fast" Discharge decay time constant (equivalent time for joule heating)	TF(1)	11 s
	CS(1)	11.5 s
	PF(1)	18 s
	CC(1)	18 s
Temperature of TF coils after fast discharge		~ 55 K
Expected number of fast discharges during plant life		50
Expected number of quenches during plant life		10

(1) Due to the use of discharge resistors with a positive temperature coefficient the actual form of the electrical discharge is not exponential. The dI/dt at the start is lower than would be the case with a pure exponential discharge. The delay time for quench detection (~2sec) is not included.

(2) This time is governed by the cryoplant capability, further described in Table 4.15-4 Typical operating modes of the cryoplant

4.1.2.12.1 Coil Discharge Dependence

A fast discharge of a PF coil or the CS will trigger a simultaneous discharge of the CS and of all PF coils, but not of the TF coils. A fast discharge of the TF coils will trigger a simultaneous discharge of the CS and all PF coils. A quench of the CC will require a fast discharge only of the CC. See also § Section 4.16.2.2.

4.2 Vacuum Vessel (WBS 1.5)

(updated by K.Ioki, Aug 2006)

4.2.1 Functional Requirements

The vacuum vessel:

- provides the first confinement barrier and withstands postulated accidents without losing confinement;
- removes the nuclear heating and the surface heat flux within the allowable temperature and stress limits
- removes (via the VV TCWS) the decay heat of all in-vessel components, even in conditions when the other cooling systems are not functioning;
- provides a boundary consistent with the generation and maintenance of a high quality vacuum;
- supports in-vessel components and their loads under normal and off-normal operations;
- together with the in-vessel components, maintains a specified toroidal electrical resistance and contributes to plasma stability by providing a conductive shell tight fitting to the plasma;
- together with the blanket, divertor, and ancillary equipment in ports, provides adequate radiation shielding for the superconducting coils and reduces activation inside the cryostat and at connecting ducts to facilitate remote handling and decommissioning;
- provides access ports or feed-throughs for in-vessel component services and maintenance, maintenance and inspection equipment, fuelling and pumping systems, diagnostics and plasma heating equipment and test blanket modules,
- reduces the toroidal field ripple using ferromagnetic materials inserted in the vessel in the shadow of the TF coils in the outboard area.

4.2.2 Configuration

4.2.2.1 Overall

4.2.2.1.1 Vessel Subcomponents

The main components that make up the VV are the main vessel, the port structures and the VV supporting system.

4.2.2.1.2 Vessel Configuration

The VV is a torus-shaped double wall structure with shielding and cooling water between the shells. The basic vessel design is an all-welded structure. Only the inner shell serves as the first confinement barrier.

4.2.2.1.3 Vessel Code

The VV components need to be designed and manufactured consistent with an accepted code or standard. The ITER VV Structural Code is being developed now (2006) based on the RCC-MR code.

4.2.2.1.4 Vessel Subdivision

The VV is divided into nine toroidal sectors joined by field welding using splice plates at the central vertical plane of alternate ports (of the odd numbers, see chapter 3.5). The sectors are connected to each other with the splice plates.

4.2.2.1.5 Vessel Ports

The VV has upper, equatorial, and lower port structures (including local penetrations located mainly at the lower level of the machine). At the upper level, there are 18 ports of a similar design. At the equatorial level, there are 14 regular equatorial ports and 3 ports for the neutral beam injection (NB ports). At the lower level, there are 5 ports for divertor cassette replacement and/or diagnostics (the divertor RH/diagnostic ports), and 4 ports for vacuum pumping (the cryopump ports). Between these ports, there are local penetrations for the divertor piping, the in-vessel viewing and glow discharge. (See Section 3.5 Port Allocation).

The port structure includes the port stub (integral to the main vessel), the port stub extension, and the port extension (equipped with the connected duct extended to the cryostat). The port components are connected to each other with the splice plates.

4.2.2.1.6 Vessel - Integration of Port Stubs

Each sector includes a full set of port stubs and stubs extensions at the toroidal centre, and a set of half port stubs (split on the port centre) on each side plane for the upper and equatorial ports.

4.2.2.1.7 Vessel Triangular Support

The toroidally continuous outer triangular support for the lower outboard blanket module aids the plasma vertical stability. The triangular support also contributes to the structural integrity of the VV

Table 4.2-1 VV Parameters

Surface temperature in operation	< 110 °C
Size - Toroidal extent of sector - Torus outside diameter - Torus inner diameter - Torus height - Shell thickness - Rib thickness	 40° 19.4 m 6.5 m 11.3 m 60 mm 40, 60 mm
Structure	Double wall
Configuration - Inboard straight region - Inboard top/bottom - Outboard region	 Cylindrical Double curvature 4 facets/Sector
Resistance - Toroidal - Poloidal	 7.9 μΩ 4.1 μΩ
Water inlet temperature (See Table 4.11-1) - at normal operation - during baking	100 ºC 200 ºC
Water inlet pressure (See Table 4.11-1) - at normal operation - during baking	1.1 MPa 2.4 MPa
Surface area / volume - Interior surface area - Interior volume - excluding volume of in-vessel components - including volume of in-vessel components *Summarised in Table 2.2-3*	 ~850 m^2 ~1090 m^3 ~1600 m^3

Mass of the assembled vessel (360 deg) -Main vessel (without shielding) -Shielding -Port structures (excl.connecting ducts) - Connecting ducts **-Total**	 1611 t 1733 t 1487 t 294 t **5124 t**
Main materials - Main vessel and port structures, double wall port components - Primary in-wall shielding (1) - Ferromagnetic insert shielding - Single wall port components and connecting ducts	 SS 316L(N)-IG SS 304B7(1), SS 304B4(2) SS 430 SS 304
Required leak rate (tbc) (3)	3×10^{-7} Pa m^3s^{-1}
Notes: (1) Inboard region: Containing ~2 weight % boron (2) Outboard region: Containing ~1 weight % boron (3) The required leak rate is in discussion (Oct 2006)	

4.2.2.2 Main Vessel

4.2.2.2.1 Main Vessel Arrangement

The main vessel consists of inner and outer shells, ribs and gussets, flexible support housings, triangular supports of the blanket modules, in-wall shielding, splice plates, port stubs and special thick-wall components (such as the divertor support structures).

Stiffening ribs between the shells give the required mechanical strength and separate the shells of the double wall structure.

The inner and outer shells and stiffening ribs are joined by welding. The inner and outer shells are both 60 mm plates and the stiffening ribs mainly 40 mm plate (except 60 mm plate for the central poloidal rib and 80 mm plates in some local areas).

4.2.2.2.2 Sector Sub-Division

Each sector is subdivided into two half-sectors. This sub-division is ensured with a pressure-tight central poloidal rib.

4.2.2.3 Port

4.2.2.3.1 Port Allocation

See Section § 3.5.

4.2.2.3.2 Port Arrangement

Most of the port components are of double wall construction with stiffening ribs between the shells. However, some components of the upper and equatorial ports have a single wall construction as a part of the structure. End parts of the port extensions - the connecting ducts – are mainly composed of thinner shells with reinforcing beams. Local penetrations at the lower level are mostly thin shells (shrouds).

4.2.2.3.3 Vacuum Boundary

At the end portion of the port extension, a closure plate normally provides the primary vacuum boundary. The closure plate can be either integrated with in-port components or attached as an individual component. As a general approach, the vacuum boundary is made with a lip welded joint.

4.2.2.3.4 In-Port Components

a) For the upper and regular equatorial ports, the closure plate is integrated with the in-port components forming the port plug.
b) For the NB ports, there is internal duct (liner) inside each NB port. The liner is equipped with the high-heat-flux panels to withstand the NB power affecting its surface. Additional shielding components (like shield plugs) are also used in these ports

4.2.2.4 In-Wall Shielding

4.2.2.4.1 Main Vessel Shielding

The space between the double wall will be filled with shield structures mainly made of an austenitic stainless steel containing 1-2% weight boron to improve neutron shielding efficiency. The shielding structures occupy 55-60 % of the in-wall space.

4.2.2.4.2 Ferromagnetic Inserts

A ferritic stainless steel is used as the shielding material in the shadow of the TF coils in the outboard area to reduce toroidal field ripple. These plates fill up to 60% of the volume between the shells. This steel has a high saturated magnetization at about 1.5 T.

4.2.2.4.3 Main Shield Fixture/Assembly

The shield blocks are fixed by bolts to the ribs or blanket flexible support housings to withstand the mechanical forces. The gaps between the shield blocks and between the blocks and the ribs are minimized to avoid excessive neutron streaming.

The shielding for the field joints must be assembled on site and removed if replacement of the TF coil and vessel sector is required.

4.2.2.5 Supports and Mechanical Interfaces

4.2.2.5.1 Vessel Supporting System

The VV is vertically supported, at its 9 lower ports, by sliding supports resting directly on the ring pedestal. These sliding supports are radially restrained against fast displacements taking place during seismic events or fast transients. They are however radially free to move during thermal expansion. The VV is also restrained vertically in the upward direction through a set of vertical links, located between the pedestal and the lower port. Additionally, the VV is restrained toroidally for the position centering.

4.2.2.5.2 Lifting Fixtures

The VV sector is equipped with special fixtures for its lifting/transportation at the site. Two main fixtures are integrated: at the sector top and at the upper port.

4.2.2.5.3 Blanket Attachment

The blanket modules are attached directly to the VV by a set of four flexible supports located symmetrically with respect to the module centre and mounted in housings that are recessed into the VV. In addition, the module support structures include toroidal (centering) keys and intermodular keys for the inboard modules, and stub-keys for the outboard ones. Adequate electrical connection of the blanket modules to the VV inner shell is provided by mechanical attachment of an electrical strap.

4.2.2.5.4 Blanket Coolant Manifolds

Blanket coolant manifolds are routed over the vessel plasma-side surface. The manifolds are connected to the cooling circuit with the pipes passing through the upper ports. The inboard and outboard manifolds including the filler shields between the blanket modules are supported on the VV by a set of sockets and end/intermediate supports that distribute the reaction force from the manifolds to a wide surface.

4.2.2.5.5 Divertor Attachment

The divertor cassettes are supported by structures integrated with the inner shell of the VV – these supporting structures are located at the lower level of the inboard and outboard region. Additionally, there are radial rails for the three divertor remote handling ports to support the cassette during remote maintenance.

4.2.2.5.6 Attachment of the In-Port Components

a) The port extensions are equipped with flanges supporting the port plugs as a cantilever. At its end, the port extension is equipped with the fixtures providing interfaces with the RH cask (see chapter 4.7).
b) The port extensions of the NB ports are equipped with flanges supporting the NB liners.

4.2.2.5.7 Port-Cryostat Interfaces

The port structures interface the cryostat with bellows located between the port connecting ducts and the cryostat vessel. The lower local penetrations also interface the cryostat with bellows.

4.2.2.6 Cooling

See also § 3.6 (slow transient heat), § 3.7 (fast transient heat) and § 4.11 (cooling water)

The vacuum vessel primary heat transfer system (VV PHTS) consists of two identical loops so as to have 100% redundancy of cooling capability. The VV PHTS utilises air coolers, with which the heat from the VV and in-vessel components is released to the environment by naturally drafted air.

Two independent loops feed cooling water to each 20° sector of the VV in parallel, to minimize the effect of faults. This limits the maximum possible VV temperature resulting from a coolant leak in one of the VV PHTS loops.

During normal operation, the total heat deposition in the VV is mainly due to nuclear heating and the heat is non-uniformly deposited in the VV.

During off-normal operation, the decay heat of the VV and thermal radiation from the in-vessel components such as blanket and divertor would be removed by the water that is circulated by thermo-gravitational convection due to the heat flux from the vessel wall to the water (i.e., natural convection).

During baking, the water temperature is increased to achieve the required baking conditions of the main vessel and ports.

4.2.2.6.1 VV Coolant Routing

The cooling water is supplied and flows through the lower ports and is routed in an internal supply structure to the bottom of each sector. This structure distributes the water to channels on both the inboard and outboard sides of the vessel to provide a flow in the channels. The water flows up into an internal collector structure at the top of the vessel and is routed through a channel in the upper port wall to an exit point. The cooling channels of the field joints are separated from the main vessel/port components. Individual inlet/outlet pipes for the water flow are provided for each field joint between the VV sectors.

The port components are cooled mainly by the VV cooling water (sometimes separately from the main vessel sectors) – however, the end portions of the upper and regular equatorial ports are cooled by the blanket water connected in series with port plugs.

4.2.2.6.2 VV Coolant versus Blanket Coolant

Performance differences between the VV and blanket require the use of separate cooling water circuits. In order to maintain stresses at acceptable levels for the blanket manifold, the VV cooling water inlet temperature difference with respect to the blanket cooling water inlet temperature has to remain limited (~50˚) for normal and baking operations.

To accommodate the coolant performance difference, the end potions of the upper and regular equatorial port extensions (cooled by the blanket coolant) are separated from the port stubs (cooled by the VV coolant) with non-cooled single wall portions.

4.2.2.6.3 Draining and Drying of VV Coolant

VV coolant water can be drained during maintenance and inspection operations of the VV. To allow the leak testing, the cooling channels can be dried out by hot gas with a drying system.

4.2.2.7 Loading conditions

See also § 3.11 and § 3.6

4.2.2.7.1 Vessel Loads

The loads acting on the VV can be divided into four categories:

- Inertial loads: these are due to accelerations due to gravity and seismic events.
- Electromagnetic (EM) loads: these act upon nearly all conductive structures during fast transients (e.g. plasma disruptions, VDE's, and magnet current fast discharge).
- Pressure loads: these include coolant, and incidental VV internal and external pressure as well as the testing pressure applied to the VV interior, exterior and in-wall space.

- Thermal loads: these are caused by temperature gradients inside the VV structure caused by nuclear heating and thermal radiation, or the temperature difference between the VV coolant and the blanket coolant.

Reference loads through interfaces of the VV can be found in § 3.11.1.

4.2.2.7.2 Vessel Displacements

Displacements of the Vessel during operation can be found in § 3.12

4.2.2.8 VV Wall Deviations

Table 4.2-2 VV Wall Deviations

Parameter	Unit	Value
Fabrication tolerance at factory - Sector overall height - Sector overall width - Surface deviations of a 40-degree sector from the reference geometry after fabrication at factory (both for the plasma- and cryostat-facing surfaces) - Sector wall thickness (distance from inner to outer surface)	mm	±20 ± 20 ± 10 (Same as (1) below) ± 5
Assembly/Positioning tolerances at site - Surface deviations of the torus from the reference geometry after assembly at the pit - Surface deviations of the torus from the reference tokamak geometry after positioning at the pit (Final deviations)	mm mm	± 15 (Sum of (1) and (2)) ± 18 (Sum of (1) to (3))
- **Details**		
(1)Surface tolerances of a 40-degree sector from the reference geometry after fabrication at factory (both for the plasma- and cryostat-facing surfaces)	mm	± 10
(2)Vessel weld distortion due to field/shop welds at the site	mm	± 5
(3)Torus positioning versus ideal location with all support fixtures removed	mm	± 3
(4) Mismatch of the sector surfaces at field joints	mm	± 5

4.2.2.9 VV Manufacturing

4.2.2.9.1 Vessel Fabrication Arrangement

The VV is to be fabricated in the factory as 9 sectors each spanning 40°.

The port stubs on the lateral sides of the sector are not installed in the factory. This allows the TF coils to be installed in the assembly area.

The shield structures are installed at the factory before shipment to the site for all locations except in the area of the field joints. The in-wall shielding for the field joint area is installed at the ITER site.

The blanket cooling manifolds and most of the in-vessel diagnostics are delivered to the VV manufacturer for installation on the VV sectors in his workshop before shipment to the ITER site.

4.2.2.9.2 Vessel Fabrication Scheme

Two concepts have mainly been considered for the sector fabrication scheme.

- One is to complete the inner shell first. Butt weld joints can be fully applied to the inner shell and inspection can be easily performed. Next, all ribs and support housings would be welded to the inner shell. After shield blocks have been installed, the outer shell would be welded with a one-sided weld.

- Another concept is to utilize poloidal segments of a double wall structure, which are fabricated first then welded together to form a sector.

One of the schemes was employed for the full-scale vessel sector fabrication in the L-3 R&D. An intermediate method combining the two concepts can also be used.

4.3 Blanket (WBS 1.6)

(Updated KII and FEO Aug 2006)

4.3.1 Functional Requirements

The blanket system:

- removes the surface heat flux and the nuclear heating within the allowable temperature and stress limits, while minimising the impurity influx to the plasma;
- reduces the nuclear responses in the vacuum vessel structural material for the ITER fluence goal;
- protects the superconducting coils, in combination with the vacuum vessel, from excessive nuclear heating and radiation damage;
- contributes to the passive stabilisation of the plasma;
- keep the plasma-facing surface at a defined distance from the plasma profile;
- via the limiter, defines the plasma boundary during start up and shut down;
- breeds tritium in the case that the outboard shield blanket modules are replaced with tritium breeding blanket modules during the last 10 years of ITER operation;
- provides passage for the plasma diagnostics, for the viewing systems, for the microwave antennas or launchers and for other minor ancillaries.

The first wall and the front parts of the nuclear shield attached to the plugs of the equatorial and of the upper ports are parts of the blanket. Their design is adapted to the device mounted in each port plug while continuing to satisfy the above requirements.

4.3.2 Configuration

4.3.2.1 Overall

4.3.2.1.1 Blanket Module Size

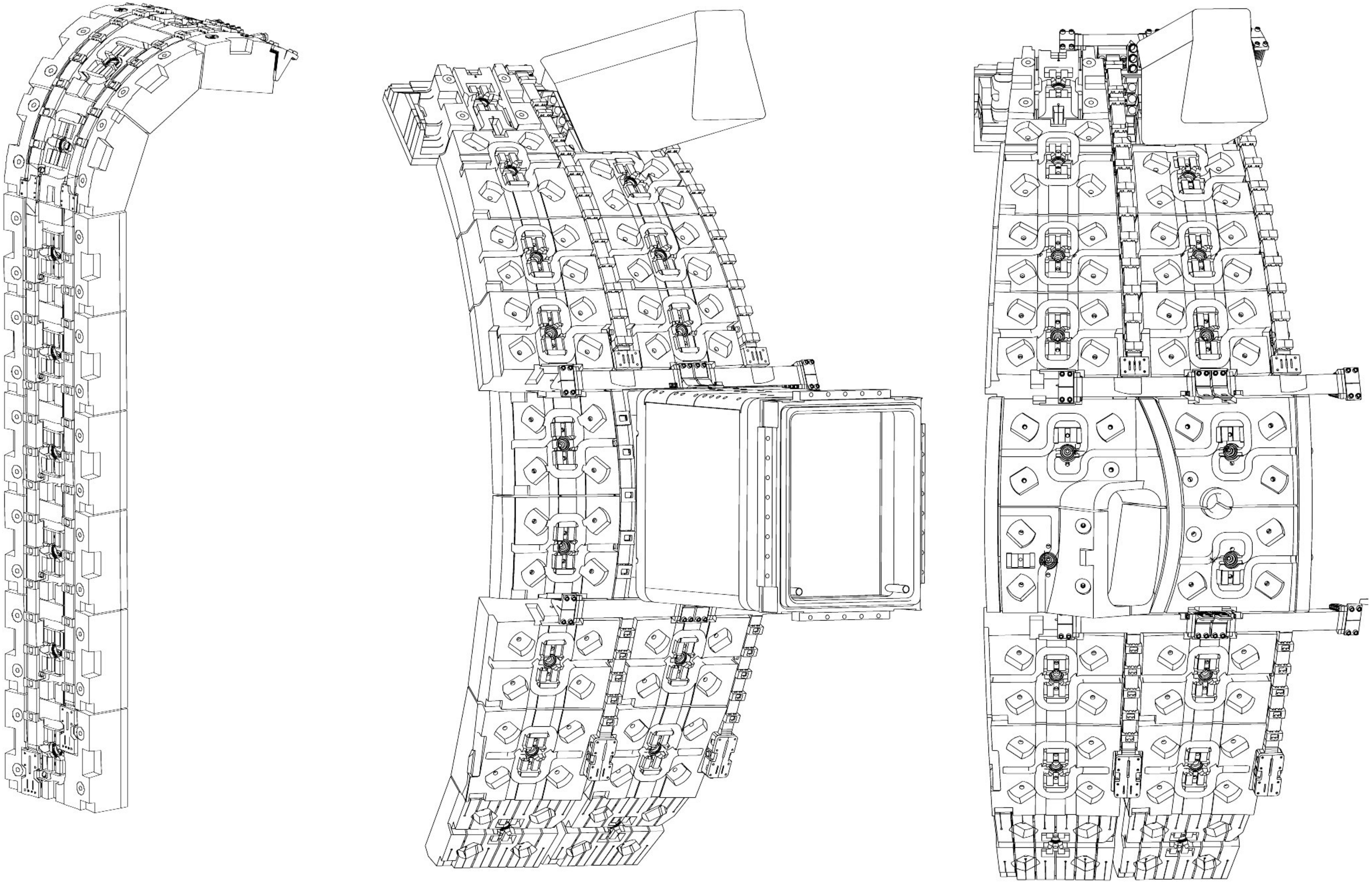
The blanket consists of modules sufficiently small to be introduced and removed through the vessel equatorial ports, and light enough to be installed and transported by the blanket remote handling system (in-vessel vehicle).

4.3.2.1.2 Blanket Module Segmentation

18 rows of modules are arrayed poloidally. The 9 rows inboard of the upper port have 18 modules toroidally, and rows 11-13 and 16-18 outboard have 36 toroidally. Between the upper port (row 10) are 18 modules.
The rows 14 and 15, located in between the equatorial ports, have 13 standard modules and 5 special modules located in between the NB affected ports (3+1 obscured). In addition to that, 4 more toroidal positions are covered by modules to provide the shielding offered elsewhere by the port plugs.
The total number of modules is thus (9x18)+(6x36)+18+2x(13+5+4)=440.
The layout is shown in the following figure.

(DCR-49 is currently (Oct2006) investigating the reconfiguration of the 4 upper port cells in the NB Cell)

Figure 4.3-1 Scheme of the Blanket Module arrangement: Inner, Outer Basic and Special NB sectors



4.3.2.1.3 Blanket Module Attachment/Positioning

The modules are mechanically attached/detached to/from the vessel from the plasma side through 30 mm access holes.

Four tubular cartridge supports with axial slits bolted into recesses in the vacuum vessel wall are provided in each blanket module quadrant to give stiffness under radial tension but flexibility under poloidal or toroidal thermal displacements on the modules. The attachments electrically isolate the module from the vessel.

Torques on the modules (due to electromagnetic forces) are reacted by insulated keys mounted on the vessel (inboard modules) or by insulated stubs concentric with the flexible supports (outboard modules)

A coaxial hydraulic connector attached to the inlet and outlet manifolds by flexible branch pipes supplies and receives the coolant in the centre of the blanket module. This connector halves the vacuum tight welds because the inlet and the outlet streams are separated mechanically. The connector has a main circular weld and a welded plug, closing the access through the shield block

Two electrical connectors, located at a standard distance above and below the hydraulic connector in centre of the module rear side, define the path for plasma halo currents.

There are two standard slots in the front of the module to allow the gripping of the RH transporter, which exploits the threaded holes of the electrical straps to fix temporarily the modules on the vessel wall.

No leading edges are present to avoid too high thermal loads on the FW.

For the breeding blanket modules an additional inlet and an outlet connector for the purge gas are foreseen in the vertical midplane just above and below the electrical connectors. Their 30 mm access holes match the RH gripping slots and do not produce additional discontinuities in the first wall.

4.3.2.1.4 Blanket Module Construction

Modules consist of a shield body of SS and a separable FW, including 4 flat panels of Be-tile-coated Cu-alloy substrate with internal SS coolant pipes. This heat sink is built on a SS plate provided of a central support leg, which is welded to the shield block at its rear side.
The shield body is re-useable after replacing a damaged or eroded FW panel.

The cooling passages of the FW are connected to those of the shield block in the cavity behind the support beam of the FW panel. This cavity is closed on the back by a welded lid and is full of cooling water. The inlet and the outlet streams are separated by a bolted wall/fitting.

The FW panel and the shield block are extensively slitted to reduce the eddy currents and the electro magnetic loads at a level acceptable for the the support beam of the FW, for the keys and for flexible supports of the module, for the wall of the VV.

4.3.2.1.5 Blanket Coolant Manifolds

See also §4.2.2.5.

Manifolds that supply cooling water to the modules are mounted on the vacuum vessel behind the modules (inboard), or in gaps between modules (outboard). While they work as natural vertical filler shields they incorporate also the horizontal filler shields, required above and below the ports to avoid trapping the modules with the plugs. Hydraulic connections between manifolds and modules are made from the plasma side through a dedicated 30 mm access hole in the centre of the module. Each manifold includes multiple channels feeding 2-4 modules. These channels are kept separate as far as the TCWS vault, to improve isolation of leaks. One rank of 8 pipes exits adjacent to each side of each upper port through dedicated chimneys. Additionally 4 couples of inlet/outlet pipes are routed in the equatorial ports/VV-wall for the extra modules required in the NB region.

4.3.2.1.6 Helium Purge Lines

To be able later to accommodate breeder blanket modules in the outboard region, helium purge lines are installed behind the manifolds. The arrangement, routing and connection method is identical to that for the coolant manifolds.

The inlet and outlet lines of each module are tied to the larger coolant branch pipes up to a corresponding connector built in the back side of the module (the shield modules mounted initially need dummy like a recess with electrical insulation).

The pipes are left unplugged at the module end and sealed at the coolant pipe flange during the initial years of ITER operation (TBC).

Table 4.3-1 Shield Blanket Main Parameters

Parameter	Value
Number of modules (see 4.3.2.1.2)	440
First wall surface area	680 m^2
Total Weight of modules (Table 4.2-1)	1,610 t
Weight limit for module	4.5 t/module
Typical blanket module dimension (Inboard equator)	1415x1005x450 mm
Plasma start-up and termination limiters	2
Nominal gap between modules in toroidal direction	14 mm Straight Inboard 20 mm Top and Outboard
Nominal gap between inboard modules in poloidal direction	10mm Straight Inboard
Nominal gap between outboard modules in poloidal direction	20 mm
FW cooling tube inner diameter	10 mm
Approximate area of blanket side and backwalls (for safety analyses)	1956 m2 (~ 3 time area of FW)

Table 4.3-2 Primary Module Materials

Component	Material	Max. temp. (°C) (calculated values)	Peak Neutron damage (dpa)
First wall - plasma-facing material - heat sink - tube and structure	 Be S-65C or equivalent. CuCrZr-IG SS 316L(N)-IG	 <300(1) 220 220	 1.6 5.3 2.7
Shield block	SS 316L(N)-IG	340	2.3
Flexible support - cartridge - bolt/collar	 Ti-6Al-4V with ceramic coating Inconel 718/CuNiBe with ceramic	 200 210	 0.03 0.15
Key structure / pad	SS 316L(N)-IG/bronze with ceramic insulation and MoS_2 coating	260	0.09
Electrical connection - bent sheets/support block - bolt	 CuCrZr-IG/SS 316L(N)-IG Inconel 718	 220 300	 0.05 0.05
Hydraulic connection/ manifold	SS 316L(N)-IG	150(2)	0.05

Abbreviations: SS = stainless steel, IG = ITER Grade
Notes:

(1) Calculated maximum surface temperature. Safety interlock temperatures are given in "3.1.3.5.3 Control of chemical energy"

(2) < 240°C during the baking operation

4.3.2.2 Port Limiters

4.3.2.2.1 Port Limiter Location

Two port limiters are provided on opposite sides of the torus. Each consists of a limiter module, a supporting and alignment system, and a port shield.

4.3.2.2.2 Port Limiter Cooling

The limiter cooling is provided by the divertor cooling system due to similarities in water chemistry and hydraulic parameters as well as due to the absence of significant load on the divertor during plasma start-up.

4.3.2.2.3 Port Limiter Configuration/Operation

The limiter module consists of an assembly of plates individually supported and insulated relatively to reduce electromagnetic loads due to plasma disruptions.

The front of each plate is coated with small Be tiles attached to a Cu-alloy substrate, the rear is a stainless steel plate.

A movement mechanism for the port limiters has been developed to retract the limiter in line with the FW after the start-up of the plasma and to rapidly deploy it before the plasma termination (see DCR-39).

Table 4.3-3 Port Limiter System Materials

Component	Material	Max. temp. (°C)	Peak Neutron damage (dpa)
Limiter module - plasma facing material - heat sink - structure	 Be S-65C. CuCrZr-IG SS 316L(N)-IG	 740 450 160	 1.6 5.3 3.4
Main structural and shield part and alignment system bellows	SS 316L(N)-IG	150	0.02
Front flexible plate support	SS 316L(N)-IG		
Back flexible pivot	Ti-6Al-4V	200	0.02
High strength bolt	Inconel 718	200	0.02

4.3.2.3 Blanket Loads and Cooling Conditions

The heat loads of the blanket are indicated in Table 3.6-1, Table 3.6-2, Table 3.6-3, and the fast transient heat loads in the chapter 3.7. The mechanical loads on the attachment of the Blanket modules to the Vessel are specified in the chapter 3.11. The parameters of the Blanket cooling circuit are specified in Table 4.11-2

4.3.2.4 Blanket FW positioning

Table 4.3-4 Blanket FW Position

Parameter	Unit	Value
Primary modules minimum clearance gap to the flux surface which passes through a point 40 mm outside the separatrix at the outside plasma equator	mm	See section 3.3
Minimum clearance between separatrix and port limiter.	mm	See section 3.3
Maximum limiter pair misalignment	mm	±1
Adjacent module FW alignment	mm	±2
Relative location of test blanket module to FW	mm	-50

4.4 Divertor (WBS 1.7)

(Updated RTY,CLY Oct 2006)

4.4.1 Functional Requirements

The divertor system exhausts the major part of the plasma heating power and He from the plasma, while minimising the impurity influx to the plasma. As the main interface component under normal operation between the plasma and material surfaces, it must tolerate high heat loads while at the same time providing neutron shielding for the vacuum vessel and magnet coils in the vicinity of the divertor.

The divertor is designed to restrict the backflow of neutrals to the main plasma to be compatible with plasma operation.

4.4.2 Configuration

4.4.2.1 Overall

4.4.2.1.1 Divertor Segmentation/Mounting

The divertor consists of 54 cassettes (3/sector) which are inserted radially through three lower level ports and moved toroidally before being locked into position,.

The cassette body is reusable and designed to provide neutron shielding of the vessel and to provide a mechanical support for a variety of plasma-facing components, and diagnostics.

4.4.2.1.2 Divertor Plasma-Facing Components

These include:
- inner and outer channel vertical targets faced with CFC monoblocks for the lower straight section, and W tiles for the upper curved portion to interface with the adjacent blanket modules and to provide a baffle for neutral particles;
- W-tile-coated inner and outer neutral particle reflector plates which together with the targets form a V-shaped channel root;
- a W-tile-coated dome, located below the separatrix, to baffle neutrals and protect the neutral particle reflector plates.

A semi-transparent liner is located below the dome to protect the cassette body from direct sight of the plasma, but to allow He to be pumped away.

4.4.2.1.3 Divertor Support Pads

Support pads are integrated into the cassette body to allow it to be locked and aligned in vessel by attaching them to mating support pads permanently fixed to the vessel.

4.4.2.1.4 Divertor Gas Leakage

The path for gas conductance from the divertor sub-volume to the main chamber is minimised by maintaining close proximity of the divertor cassette to the vessel. Some gas seals may be required between the lower HFC to maintain the divertor pressure, and avoid gas by-passing the liner.

4.4.2.1.5 Divertor Cooling Pipes

Radial cooling pipes connect to the three cassettes in each sector. These are cut and orbitally re-welded from outside the pipe.

4.4.2.1.6 Divertor Diagnostics Integration

Six "diagnostic" cassettes accommodate waveguides and optical diagnostics. A further 10 cassettes are instrumented (e.g. thermocouples), requiring only minor adjustments to the standard design.

The divertor remote maintenance ports accommodate diagnostic blocks supporting waveguides and transmission lines to three of the diagnostic cassettes during operation.

Table 4.4-1 Main Divertor Parameters

Parameter	Value
Size - Toroidal extent of a cassette - Number of cassettes	 6.66° 54
Required Replacement Time	See Table 3.10-2
Coolant type	Water
Materials - Cassette Body - Heat Sink in Targets - Armour in Targets	 SS 316L(N)-IG CuCrZr alloy Pure sintered W or CFC
Max number of foreseeable exchanges in lifetime of machine	8
Minimum lifetime of PFC	>1000 full power pulses
Design load for target in strike region	~ 10 MW/m^2
Design load of target in baffle region	~ 5 MW/m^2
Design peak heat flux for slow transient (max 10s)	20 MW/m^2
Max permissible leak rate	< 10^{-8} Pam^3s^{-1}.
Min critical heat flux margin	1.4
Thickness of carbon fibre composite (CFC) at end of life	> 5 mm
Surface area DV cassette plasma facing surface	~ 210 m^2

Table 4.4-2 Summary of Divertor Component Weights (t)

	Stainless steel	Cu	W	CFC	Water
Outer vertical target Inner vertical target Private region PFC PFC total	87.6 56.1 78.2 221.9	9.2 4.9 15.6 29.7	11.2 5.9 14.8 31.9	2.1 1.6 – 3.7	1.2 0.7 0.7
Cassette body	372.7	–	–	–	15.5
Total (t) *(see Table 4.2-1)* **=678 t**	594.6	29.7	31.9	3.7	18.1

4.4.2.2 Loads

See also § 3.11 and § 3.6
Reference loads from Divertor to Vessel can be found in § 3.11.1.
The transient Heat loads on the divertor FW are defined in § 3.7.

4.4.2.3 FW Divertor Positioning

Table 4.4-3 Divertor FW Position

Parameter	Unit	Value
Circularity	mm	6
Concentricity relative to the magnetic center of machine	mm	7
Vertical long wave tolerance	mm	± 10
Maximum step between adjacent cassette PFCs	mm	4 (1)
Cassette toroidal positioning At toroidal rail level Bottom of the cassette	mm	 ± 2 ± 4
Minimum gap between 2 adjacent cassettes	mm	2
Maximum radial, toroidal & vertical step in-between any adjacent toroidal & radial rail hard cover plates, chamfers and/or rounded edges	mm	1
Positioning accuracy of toroidal & radial racks Maximum step between adjacent rack segments	mm	1.6

Notes (1) The alignment of other divertor sub-components is less critical and normal manufacturing tolerances can be applied.

4.4.2.4 Thermohydraulic, Cooling and Baking

See Table 3.6-1, Table 3.6-3, and Table 4.11-3.

4.4.2.5 Operation & Reference Scenario for Divertor Operational Waste Estimation

For the purpose of estimating operational waste over the lifetime of ITER it is assumed that ITER will consume three sets of divertor targets with CFC strike point armour capable of sustaining 3000 pulses each, followed by an advanced design phase (two sets of targets) with tungsten strike point armour capable of sustaining 8000-1600 pulses each.

Table 4.4-4 Reference Scenario for Divertor Operational Waste Estimation

Armour Material of Divertor Targets	Target Design	Lifetime (Equivalent Number of Pulses)
CFC (3 sets)	Current Design	3000 each
Tungsten (2 sets)	Advanced Design	8000 – 16000 each

4.5 Fuelling – Wall Conditioning (WBS 1.8)

(Updated S.Maruyama Aug 2006)

4.5.1 *Functional Requirements*

The system will provide the following functions:

Fuelling Systems

- provision of the capability for both gas and pellet fuelling,
- provision of supplying hydrogen or deuterium gases to the NB and DNB injectors,
- provision of fuelling into the main plasma at a rate determined by fusion power, density control and SOL flow requirements, with specified response time,
- provision of pellet injection into the periphery plasma to control ELM
- injection of impurity gases into the divertor plasma for radiative cooling, and discharge termination,
- injection of impurity ice pellet(s) using a centrifuge type pellet injector into the plasma for studies of impurity transport and radiative cooling enhancement at the edge,
- injection of impurity gas(es) using dedicated system into the vacuum vessel for disruption mitigation,
- provision of an emergency fusion power shutdown system (FPSS),
- provision of wall conditioning gases using the same system used for hydrogenic gas fuelling of the plasma;

Wall Conditioning Systems

- Provision of wall conditioning systems that reduce and control impurity and hydrogenic fuel outgassing from plasma-facing components for achievement of clean and stable plasma operation.

4.5.2 *Configuration*

4.5.2.1 Fuelling System

4.5.2.1.1 Fuelling System Capability

The fuelling system together with the exhaust pumping system (§4.13) will be capable of providing He removal and plasma core fuelling at a rate which corresponds to the nominal fusion power, as well as the required density ramp-up during the start-up phase (see Table 4.5-1)

Table 4.5-1 Plasma Fuelling Parameters

Parameters	Unit		
Fuelling gas(1)(2)(3)(4)		^3He, ^4He	H_2, D_2, DT, T_2
Average/Peak fuelling rate for H_2, D_2, DT for gas puffing (2)(3)	Pa m^3/s		120/240
Average/Peak fuelling rate for Tritium for pellet injection (2)(3)	Pa m^3/s		50/50
Average/Peak fuelling rate for other hydrogen species for pellet injection (2)(3)	Pa m^3/s		100/100
Average/Peak fuelling rate for ^3He or ^4He	Pa m^3/s	60/120	
Gas purity for each fuelling gas H_2, D_2, and He	mole %		> 99
Allowable H_2 in D_2, DT or Tritium(4)	mole %		< 0.5
Allowable non-hydrogenic gases in H_2, D_2, DT or Tritium(4)	mole %		< 0.05
Allowable He in each fuelling gas H_2, D_2, DT or Tritium(4)	mole %		< 0.03
Duration at peak fuelling rate	s		< 10
Average frequency of peak fuelling (TBC)	Hz	~0.01	~0.01
GIS response time to 63% at 20 Pa m^3/s	s		< 1
Set point control precision of fuelling rate	%		5

Notes
(1)The fuelling isotope mixture should be adjustable, on a pulse-by-pulse basis, between 100%D and 100%H during H/D operation, and between 100%D and 90%T during DT operation.
(2)The plasma fuelling rate during DT operations for isotopic mixtures up to 90%T will not exceed the average fuelling rates defined in Table 4.5-3.
(3)During DT operations the plasma fuelling system will be capable of providing the fuelling scenarios detailed in Table 4.5-3.
(4)Tritium is only supplied as a 90% T 10% D mix, at a maximum of 50 Pam^3/s in total, (although initially 100% T2 may be available).

The Tritium plant supplies eight independent fuel gas streams to the GIS, Pellet and Neutral Beam systems. The following tables details the pressure and use of each line

Table 4.5-2 Supplied Gas Fuelling Lines

Pipe	Gas	GIS Use	Pipe Outer Dia./Thickness (mm)	Tritium Plant Delivery Pressure
1	H_2	Neutral Beam	13.8 / 2.0	0.6 MPa
2	D_2	Neutral Beam	13.8 / 2.0	0.6 MPa
3	H_2/D_2	GIS/Pellet	27.2 / 2.5	0.12 MPa
4	DT/H_2	GIS/Pellet	27.2 / 2.5	0.12 MPa
5	T_2	GIS/Pellet	13.8 / 2.0	0.12 MPa
6	$^4He/^3He$	GIS/Pellet	17.3 / 2.5	0.12 MPa
7	N_2/Ne	GIS/Pellet	17.3 / 2.5	0.12 MPa
8	Ar	GIS/Pellet	17.3 / 2.5	0.12 MPa
9	Evacuation	GIS/Pellet/NB	60.5 / 3.5	-

Note: T_2 is supplied as (90%T,10%D) (100% may be initially available)

Table 4.5-3 Time-averaged Fuelling Flow Rates during DT Discharges as a Function of T/D Ratio

Typical Fuelling Cases	1	2	3	4	5	6	7	8
T/D ratio (-)	25/75	40/60	50/50	50/50	60/40	70/30	75/25	90/10
D_2 flow rate, Pa m^3/s	84	48	0	40	4	0	0	0
DT flow rate, Pa m^3/s	6	42	120	30	66	50	30	0
Tritium flow rate, Pa m^3/s	30	30	0	50	30	50	50	50
Total, Pa m^3/s	120	120	120	120	100	100	80	50

4.5.2.1.2 Fuel Pellet injection

Provision will be made to inject pellets of hydrogen species into the plasma for core fuelling and ELM control.

Table 4.5-4 Pellet Fuelling Parameters

Parameters	Units	Value
Control accuracy of fuelling rate	%	5
Pellet speed (reference/maximum)	m/s	300/500
Typical pellet volume delivered to plasma(1)		
Core fuelling	mm^3	92±20% 50±20%
ELM control	mm^3	33±20% 17±20%
Volume change during pulse (-20 %↔+20 %)	sec	~ 0.2
Volume change between pulses (92 mm^3 ↔ 50 mm^3, 33 mm^3 ↔ 17 mm^3)	Hour	~ 1
Frequency for H_2, D_2, DT and Tritium pellet (nominal/max.)	Hz	4 / 16
Frequency change time (factor of 2)	sec	~ 3
Injection time (target value)	s	400 (3,000)
Overall pellet fuelling efficiency(2)	%	~ 90

Note:
(1) These sizes are one of the examples indicating a variation of sizes during the pulse.
(2) Tentative target value equal to (1-L)/(1-R*L) where L for ablation loss and R for recovery efficiency:- ~90% is based on the assumption with L=35% and R=90%.

4.5.2.1.3 Impurity Pellet Injection

Provision will be made to inject impurity pellets into the plasma for physics studies.

Table 4.5-6 Impurity Pellet Injection for Physics Studies

Parameters	Unit	Value
Impurity pellets		N_2, Ar, Ne
Maximum number of impurity gas species injected per pulse		1
Maximum injection rate	Pa m^3/s	10
Typical pellet size (example: 1 Pa·m^3/s of Ne)	mm^3	6
Frequency (nominal/max.)	Hz	1 / 10
Allowable impurities	mole %	< 3

4.5.2.1.4 Divertor Radiative Cooling Promotion

Provision will be made to inject impurity gases into the private region of the divertor to promote radiative cooling.

Table 4.5-5 Impurity Gas Injection for Radiative Cooling of Divertor

Parameters	Unit	Value
Impurity gases	–	N_2, Ar, Ne
Maximum number of impurity gases to be injected simultaneously	–	2
Average/Peak injection rate for each gas	Pa m^3/s	10/100
Average/Peak simultaneous injection rate all gases	Pa m^3/s	10/100
Allowable impurities	mole %	< 3
Duration at peak fuelling rate	s	< 10
Average frequency of peak fuelling (TBC)	Hz	~ 0.02
Response time to 63% at 5 Pa m^3/s	s	< 1
Set point control of fuelling rate	%	5

4.5.2.1.5 Fusion Power Shutdown System

A fusion power shutdown system (FPSS) will be implemented to terminate the fusion power reaction by the injection of impurity gases.

Table 4.5-6 Fusion Power Shutdown System (FPSS)

Parameters	Unit	Value
Impurity gas	–	Ne
Total quantity of gas to be injected		> 40 Pa m^3
Response time to 63% following initiation of FPSS	s	< 3

4.5.2.1.6 NB Fuelling Parameters

The NB and Diagnostic NB fuelling system will supply the injectors with the required hydrogen isotopes as dictated by HH, HD, DD and DT operations.

Table 4.5-7 NB and Diagnostic NB Fuelling Parameters

Parameters	Unit	H phase	D_2/DT phase
NB Injector, D_2 gas flow for 1 NB injector	Pa m^3/s		18
NB Injector, H_2 gas flow for 1 NB injector	Pa m^3/s	36	
Diagnostic NB Injector, H_2 gas flow to neutraliser, HH operation only.	Pa m^3/s	9	
Diagnostic NB Injector, D_2 gas flow to neutraliser, DD and DT operation only.	Pa m^3/s		6
Diagnostic NB Injector, H_2 gas flow to beam source, HH, DD and DT operation.	Pa m^3/s	8	8
NB Injector and Diagnostic NB Injector maximum allowable gas impurity-for hydrogenic species	atom %		H < 0.5 T<0.02
NB Injector and Diagnostic NB Injector maximum allowable gas impurity for other impurities	ppm	< 10	< 10

4.5.2.1.7 Fuel Pellet Blanket Damage Limits

The blanket will be capable of sustaining without damage a series of impacts on the plasma facing surfaces from pellets delivered by the pellet injector prior to injector shut down following an undemanded termination or disruption of the plasma.

Table 4.5-8 Pellet Impact on Blanket following Plasma Disruption

Typical pellet diameter(1)	mm	3.5	4	5.5
Pellet mass	g	1.0E-02	1.5E-02	4.0E-02
Pellet velocity	m/s	500	500	500
Impact force	N	880	1150	2170
Impact area	mm^2	10	13	24
Pellet delivery frequency	Hz	16	10	4
Pellet injector shut down time after disruption	s	2	2	2
Number of impacts		32	20	8

(1)Cylindrical pellet, diameter = length

4.5.2.1.8 Fuelling System Parameters

Table 4.5-9 Fuelling System Parameters

Mode of fuelling	Gas puffing Pellet injection Disruption mitigation system Fusion power shutdown system
Gas injection system	4 injection points at upper port level 3 injection points at divertor port level See 3.5 Port Allocation
Pellet injection system	Maximum number of injectors Six in total 2 for core fuelling 1 for ELM control 1 for impurity pellet 2 for standby Typical pellet volume and repetition rate: 110 mm^3, 4 Hz 74 mm^3, 6 Hz 60 mm^3, 10 Hz 40 mm^3, 12 Hz Penetration depth: ~15 % of minor radius
Disruption mitigation system	2 injection points at upper port level
Fusion power shutdown system (within 3 seconds)	See Table 4.5-6**.** (> 40 Pa m^3/s for Ne) 2 cylinders (1000 cm^3 at 0.12 MPa) are installed in upper port GIS valve boxes

4.5.2.2 Wall Conditioning System

4.5.2.2.1 Wall Conditioning

Water shall be removed from the surface of in-vessel components to avoid unacceptable levels of oxygen in the plasma.
Outgassing of hydrogen isotopes from in-vessel components shall not be an uncontrolled source of significant particle influx to the plasma.
The VV, all in-vessel components and all surfaces exposed to primary vacuum shall be bakable to the following specifications:

Table 4.5-10 Baking Conditions

Parameters	Unit	Value
Vacuum vessel baking temperature (Table 4.11-1) (1)	°C	200
In-vessel components baking temperature (Table 4.11-3)	°C	240
Baking temperature of surfaces exposed to primary vacuum not included above (1)	°C	*200*
Baking temperature of other pipework on primary torus vacuum *(for example the NB entrance port up the the shutter/valve, and the VVPSS pipes up to the rupture disks. The NB vessel does not need to be baked)*	°C	*200*
Heat-up time from room temperature for VV	h	< 100
Heat-up time from room temperature for in-vessel components	h	< 100

Note 1: This is the nominal value. The operational value is 200 °C +0/-10 °C – See Section 4.11.2.2

Means shall be provided to allow glow discharge cleaning with H and other gases with no toroidal field, and discharge cleaning using auxiliary heating systems.

The specifications do not differ in different operation phases and are summarized as follows:

Table 4.5-11 Glow Discharge Cleaning Requirements

Parameters	Unit	Value
First wall current density	A/m^2	> 0.1
Conditioning gas	–	H_2, D_2 and He
Number of electrodes		6
Operating pressure	Pa	0.1 – 0.5
Pumping speed (molecular flow)	m^3/s	30-40
Throughput (max.)	Pam^3/s	50
All coil currents and fields at zero (TF, PF CS and CC)		0

Table 4.5-12 EC/IC Discharge Cleaning Requirements [1]

Parameters	Unit	Value
Power	MW	~ 1
Distance of resonance layer to inner wall at equator	mm	TBD
Conditioning gas	–	H_2, D_2 and He
Operating pressure	Pa	0.01 – 0.10
Pumping speed	m^3/s	30-40
Throughput (max.)	$Pa\ m^3/s$	4
TF Field		TBD
PF and CS Fields and configurations		TBD

Note (1) EC and IC Cleaning is subject to review (Oct 2006) – See TCM-21.

The following table suggests Reactive Cleaning techniques, which might accompany and supplement glow cleaning or pulse cleaning campaigns (TBD).

Table 4.5-13 Reactive Cleaning Requirements

Parameters	Unit	H / DT
Conditioning gas	–	TBD
Operating temperature	°C	TBD
Operating pressure	Pa	TBD
Throughput	$Pa\ m^3/s$	TBD

4.6 Assembly Plan and Tooling (WBS 2.2)

(Update R.Shaw Oct 2006)

4.6.1 Functional Requirements

The assembly plan:

- defines the sequences of operations developed to complete the assembly of the tokamak, which includes the cryostat and the sub-systems contained therein;
- includes the on-site transport of the components, preparation (pre-assembly) of the components, sub-assembly of the integrated assembly units, and final assembly inside the tokamak pit, and;
- will be the basis for establishing the scope of the detailed assembly procedures, which will be used to assemble the tokamak in accordance with local regulations.

To complete the assembly operations will require both specially designed, purpose-built tools, and standard, commercially available equipment. To support the assembly process, a number of services; metrology, metallurgy, beryllium control, health physics, occupational safety, and site facilities, will be established, and provided at the site.

Although all of the components of the tokamak are designed to last the lifetime of the ITER, there must be provision for repair or replacement, if a failure is conceivable. In particular, the disassembly and reassembly of the magnets or vacuum vessel is foreseen in case of gross failure. The assembly plan and design of the assembly tool interfaces must facilitate the replacement of these large components.

4.6.2 Reference Assembly Process

4.6.2.1 Site Assembly Facilities

4.6.2.1.1 Tokamak and Assembly Halls

The main assembly activities will be performed in the tokamak hall, where the ITER device is installed inside a partially embedded, concrete bioshield, and also in the assembly hall, see Figure 4.6-1. These buildings are connected to form a single, continuous crane hall, approximately 175 m long x 48 m wide. For the duration of the assembly, the building will be operated as a clean area, and an enhanced capacity HVAC system is foreseen for the period. It will also be necessary to maintain a uniform temperature throughout the building, to avoid problems with dimensional changes of the large components. The plant layout is designed to accommodate the flow of materials during assembly, and also during subsequent maintenance operations.

4.6.2.1.2 Crane in the Tokamak Building

The lifting system in the tokamak and assembly buildings will comprise a pair of independent bridge cranes, mounted on rails that run the entire length of the tokamak and assembly halls to form a continuous crane bay. Loads of up to 1,500 t (tonne) can be handled by the synchronized operation of the four 375 t main hoists.

The tallest lift to be made with the cranes is the central solenoid, at approximately 22.5 m, including all of the associated lifting tools, and the necessary operating clearances. This lift drives the height of the cranes, and of the building. The top of the crane rails is currently set at 40.595 m above grade.

An auxiliary, 40 t bridge crane will be located below the two main cranes, to serve the entire length of the Tokamak and Assembly halls. This crane will increase the handling capacity for the majority of the tokamak components, and will be used, primarily, to avoid scheduling conflicts for the main bridge cranes during periods of peak activity, facilitate the efficient use of the main cranes, and provide alternative lifting capacity during crane outages.

Table 4.6-1 Operating Parameters for Tokamak Building Cranes

750 t Overhead Crane	Item	Capacity (Rated Load)	Speed (m/min)			Positioning (mm)	
			Max.	Min.	Control	Resolution	Repeatability
No.1 and No.2 trolley	**Main hoisting device**	**375 t**	**1.4**	**0.05**	**± 0.005**		**Vertical ± 30**
	Aux. Hoisting device	**100 t**	**3.2**	**0.05**	**± 0.005**		**Vertical ± 30**
	Trolley traveling device	-	5	0.05	± 0.005	0.5	Horizontal ± 20
Bridge	Main Bridges	-	5	0.05	± 0.005	0.5	Horizontal ± 20

4.6.2.1.3 Hot Cell Building

For the final preparation of the beryllium containing components, a dedicated beryllium controlled area (BeCA) will have to be established. It is also intended to establish a clean preparation area for the divertor cassettes inside another area of the hot cell.

4.6.2.1.4 On-site Storage Areas

Extensive storage facilities will be required to locate the components between the time they are delivered to the site, and required for assembly. The current generic site layout has a single, paved storage area of 7400 m^2, located to the southeast of the tokamak building, and this will be the main location for the interim storage of the large components.

In addition, further storage space will be made available in the specified temporary construction area, as required. The bonded storage of the small components and materials has yet to be addressed.

4.6.2.2 Basic Assembly Concept

The tokamak is assembled from 9 sectors, each with a toroidal angle of 40°, and comprising a sector of vacuum vessel, two toroidal field coils, the associated VV thermal shield, a pair of intermediate outer intercoil structure (IOIS) friction joints, which connect the outboard regions of the two TF coils. The components are delivered to the site individually, and sub-assembled into sectors using purpose-built jigs and fixtures in the assembly hall.

Prior to installing the sectors in the tokamak pit, the tokamak gravity supports, lower cryostat sections, and the components which cannot be installed following final assembly of the sectors, principally the lower poloidal field coils, lower correction coils, the lower and side correction coil feeders, and the lower precompression rings, are installed, or placed in the pit.

The sectors are transferred to the pit sequentially. The TFCs and VVTS sectors are connected sequentially, immediately following transfer, whereas the VV sectors are joined (welded) according to a plan which aims to minimise deformations, and the associated technical risk.

Following installation of the final sector, the VV is closed toroidally with the welding of three, toroidally equispaced field joints, aligned with the TF magnet, and lowered onto its permanent supports; thereafter, personnel access is via selected horizontal ports at the divertor and mid-plane levels.

The TFC precompression rings are then installed, and the preload is applied to each of the coils. A detailed dimensional survey at this stage provides the geometrical estimate of the magnetic datums for the as-built TF magnet, and these are used as reference for all subsequent installation operations.

Clean conditions are established inside the vessel, and the installation of the in-vessel systems is completed. The completion of the installation of the ex-vessel components proceeds in parallel.

4.6.2.3 Assembly Strategies

A major issue for the assembly of the ITER tokamak is the tight installation tolerances required for the major components. To achieve the specified tolerances, the assembly plan must follow sequences and processes which minimise both deviations and the residual stresses in the components, and which allow for the correction of any deviations as they occur so that alignment errors do not accumulate.

The routing of components through the building complex must be carefully planned to avoid potential conflicts between activities that are carried out in parallel.

To meet vacuum requirements, the assembly of all of the tokamak components must be carried out under clean conditions, and the assembly of the components containing beryllium will require additional precautions.

4.6.2.3.1 Strategy to meet the Tolerance Requirements

With the alignment tolerances close to the limit of what is achievable, the accumulation of deviations is unacceptable. The assembly plan is therefore designed to correct alignment deviations at each step of the assembly sequence. This strategy relies on linking a sophisticated optical metrology system (OMS) to a CAD system, which can generate and analyse complex 3-D models in near-real time, and provide a complete, evolutionary database of the as-built components and of the overall tokamak geometry.

With respect to the major components, it is essential that most geometrical interfaces have the provision for adjustment (probably shimming) included in their design. Additionally, the possibility that components will have to be accepted with some non-conformities, makes the provision for on-site reworking of the large components a sensible precaution.
The strategy for controlling the dimensions of the tokamak build is illustrated in Figure 4.6-2.

4.6.2.3.2 Building Utilisation

The assembly processes in the assembly hall are aimed at obtaining the maximum utility from the available layout, which is constrained by crane coverage limitations for the heavy lifts.

A temporary facility will be annexed to the south end of the building complex for the purpose of maintaining clean conditions in the Tokamak Complex during periods when the large assembly hall doors are open. The facility will also be used for final cleaning of the major components and will be sized to accommodate the main cryostat sections.

4.6.2.3.3 Clean Conditions and Beryllium Control

The correct functioning of the tokamak components necessitates the achievement of a high level of cleanliness prior to operation. To ensure cleanliness of the components, for which a systematic, post-assembly cleaning is unfeasible, the building complex will be operated at *high-degree* clean room conditions.

Special precautions will be necessary when components containing Beryllium (Be) and its compounds are being handled, prepared and assembled. A dedicated Be control area will be established in the Hot Cell, and, prior to installation of the first BM, similar controls will be applied to the VV. The two areas will be maintained under negative pressure via separate, local ventilation systems, which will exhaust via HEPA filters. Access to the Be controlled areas will be restricted to a specific group of registered workers and extensive routine monitoring for surface and airborne Be dust will be required.

4.6.2.3.4 Temperature Control

The temperature *uniformity* of the tokamak and assembly halls will be maintained to ensure that temperature changes do not degrade the assembly accuracy, or prolong the assembly schedule. The temperature specification for the HVAC system of the tokamak building and pit is 25 ± 2° C during summer and 20 ± 2° C during winter, with a gradual transition assumed.

4.6.2.4 Assembly Procedure

The Assembly Plan is defined in the annex "AP".

4.6.2.5 Occupational Safety

The assembly operations present a number of hazards to the personnel involved in the execution of the work. In addition to those inherent to common industrial processes, the assembly of ITER will also subject the workforce to the hazards associated with high level working, confined space working, handling of massive and heavy loads, handling of toxic materials, and ionizing radiations.

To minimize the occupational risk:

- A code of practice will be developed for the ITER project, and all workers, staff members and visitors will be required to comply with the requirements specified in the document.

- All workers, staff members and visitors will, prior to admission to any ITER assembly and/or construction areas, be appropriately trained according to a standardized, formal procedure. A training register will be maintained by the project.

- All workers will be appropriately qualified for the functions they perform. It will be the responsibility of the assembly contractors to confirm that the workers they supply are formally qualified for the trade in which they are employed. ITER will provide directed training for operations which present hazards not covered in the standardized procedure.

- All assembly operations will be documented via assembly procedures. A standardized format will be used, and the procedure will be subject to formal approved prior to the execution of the work.

Working with beryllium components, and ionizing radiations present specific hazards that are not fully addressed by training. To meet the project's statutory obligations, only designated workers may enter the areas occupied by exposed beryllium components, and in which ionizing radiations are present. Worker control and surveillance will be carried out in accordance with the French national regulations.

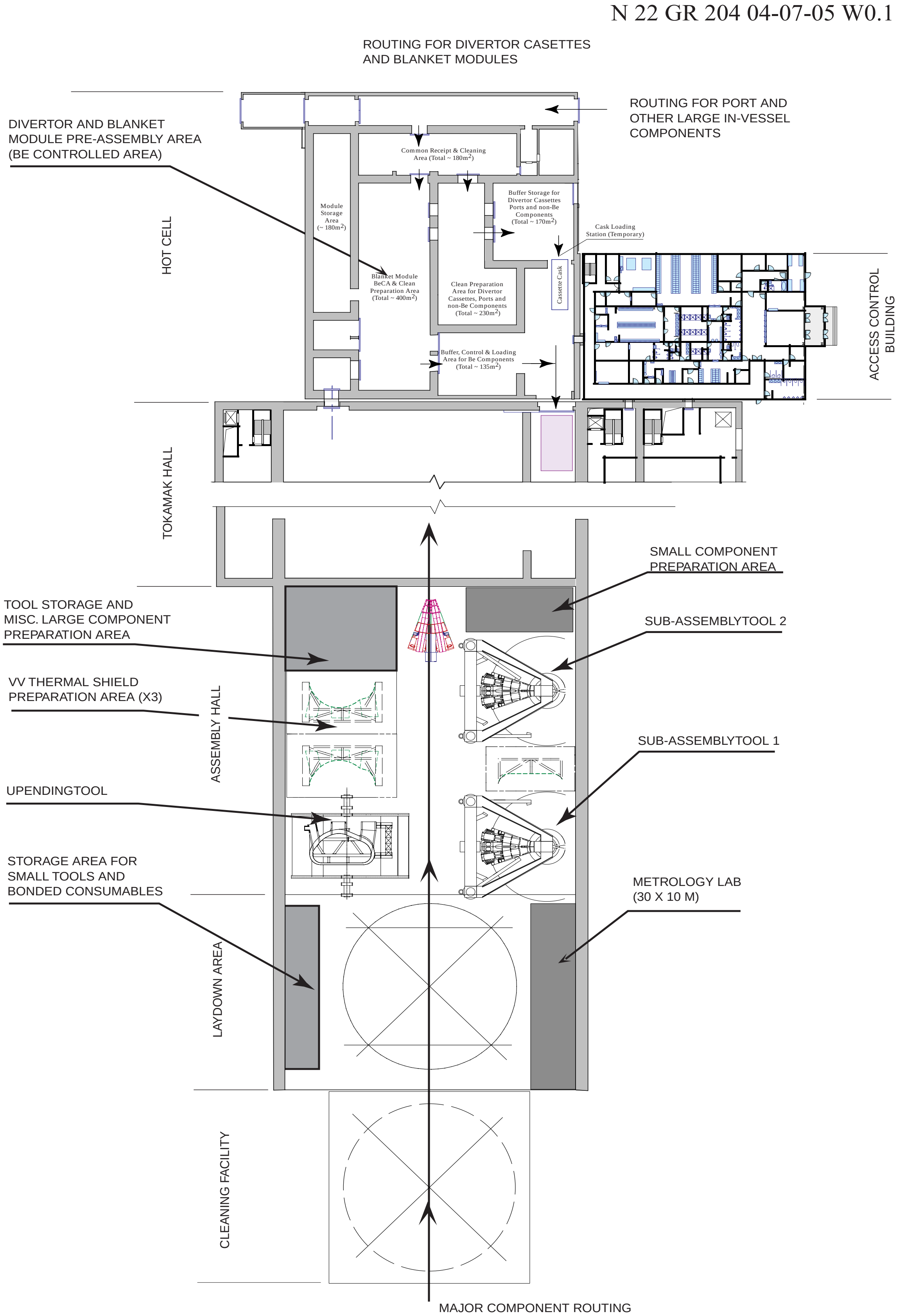


Figure 4.6-1 Utilisation of Tokamak Building during Initial Assembly

((figure will be updated following resolution of DCR35 and DCR-49)

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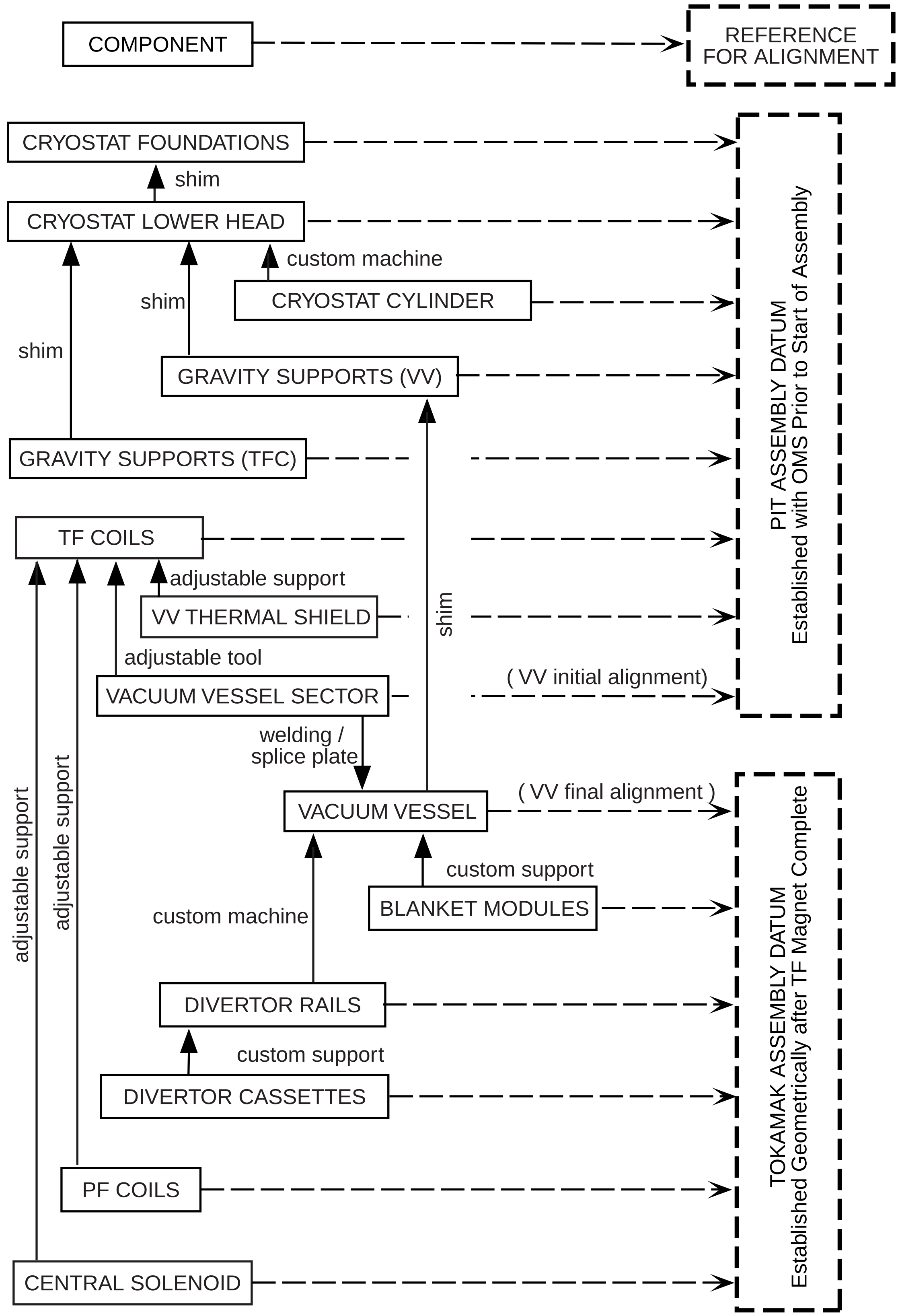


Figure 4.6-2 Tokamak Alignment Strategy

4.6.3 *Assembly Tooling*

4.6.3.1 Assembly Control and Support Tools

This group of "tools" includes the facilities necessary to control the quality of the work (metrology and metallurgy), and to ensure compliance with the Project's statutory obligations in terms of worker safety (beryllium control and health physics and occupational safety). It may be cost effective to sub-contract some of the required services to outside suppliers, and the local engineering infrastructure will be investigated to determine the viability of this alternative approach.

Also included is an on-site workshop (it may also be possible to sub-contract this service), and the specialised access and control equipment required for the VV, cryostat and for the preparation of the beryllium-containing components.

4.6.3.2 Sector Sub-Assembly Tools

The sub-assembly of the 40° sectors is carried out in the assembly hall, with the components in their final, vertical orientation, see Figures 4.6-1 and 4.6-3.

The tooling comprises an upending tool, which rotates the components to the vertical orientation, and a pair of sub-assembly tools. The two sub-assembly tools support the VV, and via a rotary motion about the "machine axis", incorporate first the VVTS, and then the TFCs. The tools then serve as staging for the remainder of the operations that must be completed on the 40° sector, prior to transfer to the cryostat pit.

The sub-assembly tooling also includes a number of handling tools for lifting, and bracing the major components in both horizontal and vertical orientations.

4.6.3.3 Sector Assembly Tools

The principal sector assembly tools comprise an integrated system of supports and braces, see Figure 4.6-4. These support, align, and stabilise the TF coils and VV sectors independently. The VV support tools have the capability of positioning individual sectors, groups of sectors, and, eventually, of positioning the completed vessel.

Also included are the survey tools for the VV field joint splice plates; welding clamps; welding rail and NG-TIG welding tools; and installation tools for handling, and positioning the splice plates.

The maintenance scenario, in which a VV/VVTS/TF 40° sector is removed without removal of PF3 and PF4, must be taken into account in design of these tools.

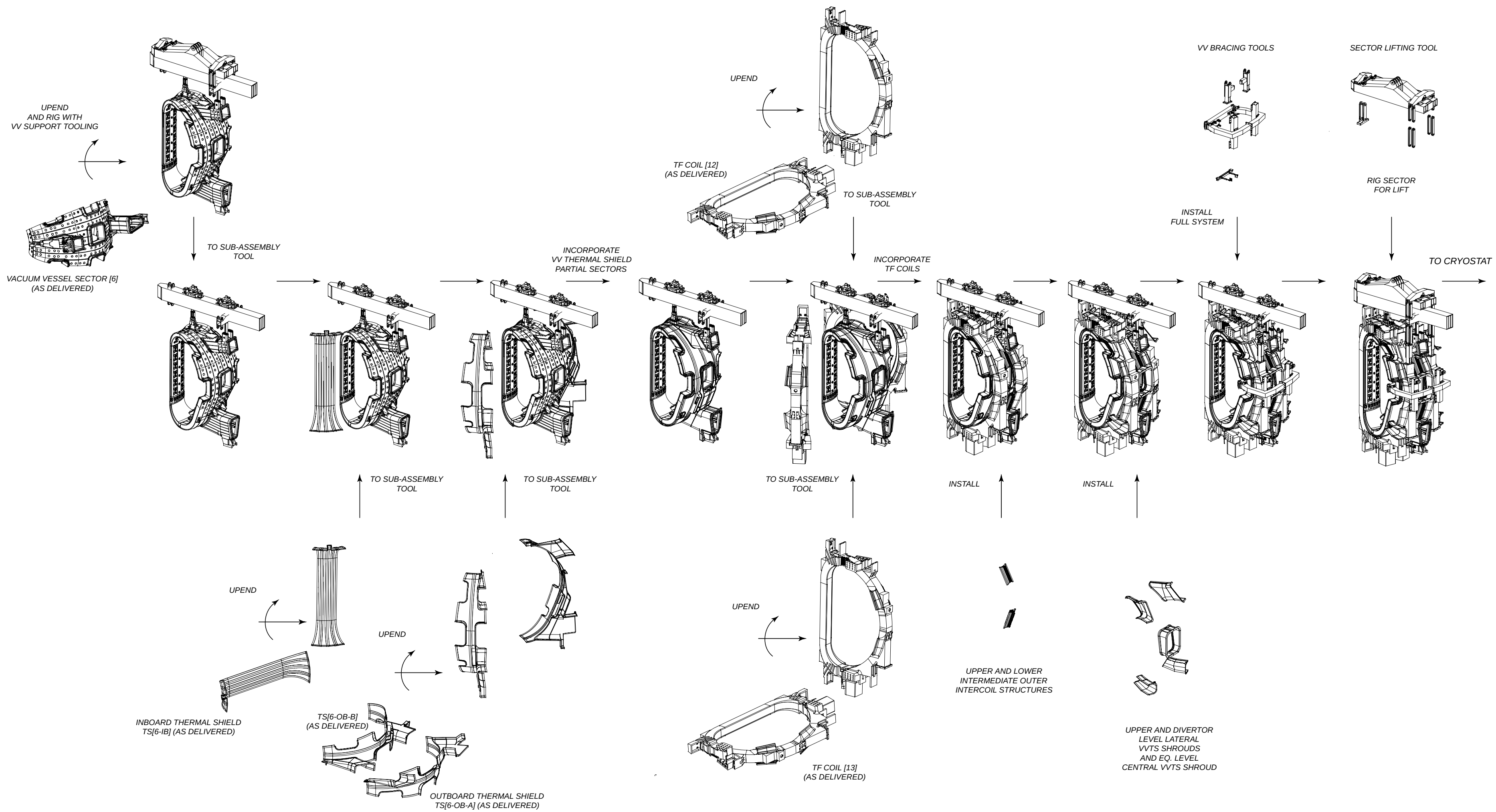


Figure 4.6-3 Sector Sub-Assembly Procedure

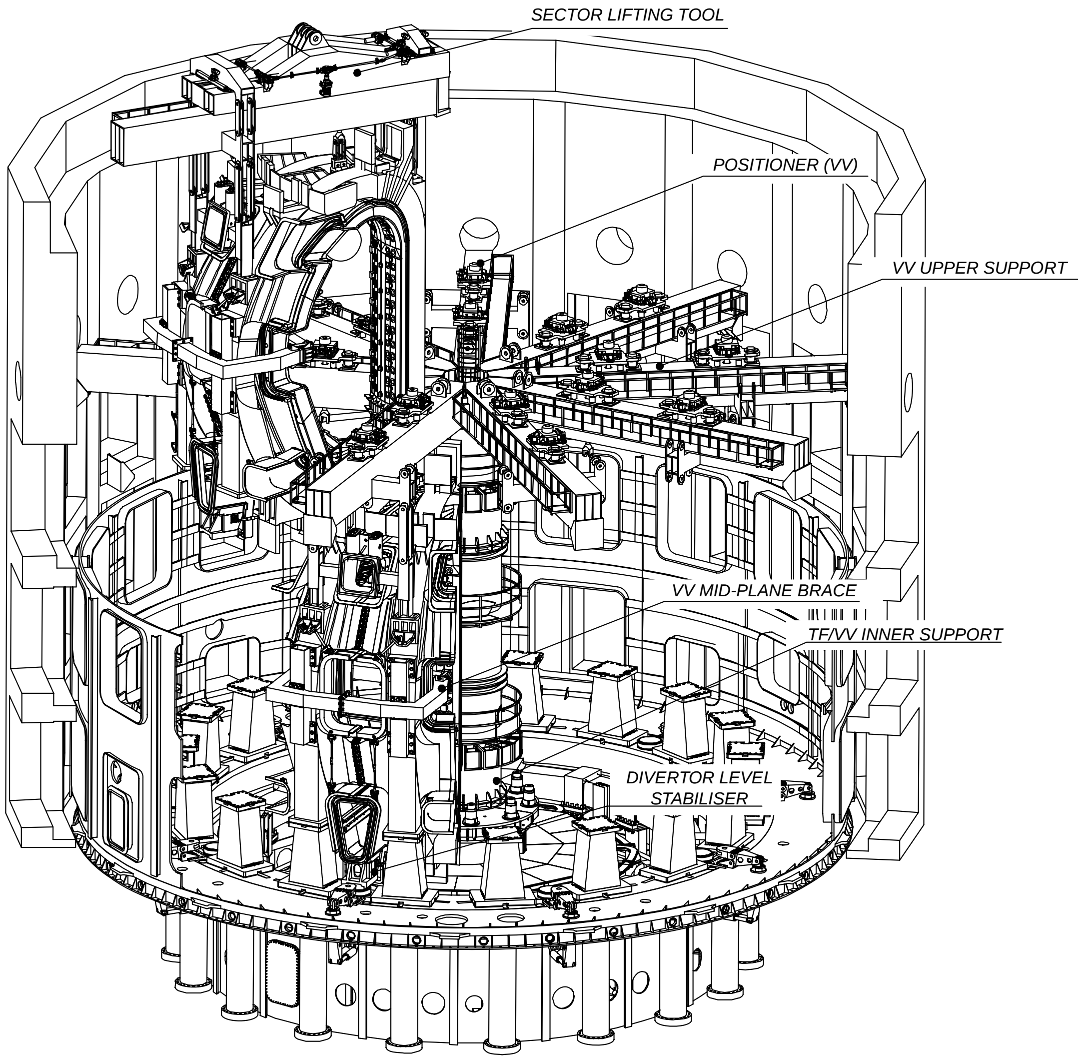


Figure 4.6-4 Layout of Assembly Support and Bracing Tools in the Pit

4.6.3.4 Cryostat and Cryostat Thermal Shield Assembly Tools

The cryostat assembly tools include the installation tools: purpose-built fabrication/transport frames, for transporting the large sections from their sub-assembly area to the assembly hall; adapters that interface with the 4 universal lifting beams, for lifting the sections into the pit with the overhead cranes; and the support and guides which align the field joints as the sections are installed.

Specialised survey tools will be required to control the interfaces between the cryostat base and support pedestals, and also the interfaces between the cryostat base and the bioshield.

The cryostat welding tools comprise the clamps which hold the field joints in alignment during the welding operations, the NG-TIG welding machines on which the welding process is based, and a purpose-built track to support and guide the welding machines.

The assembly tools for the cryostat thermal shields (CTS), transition thermal shields (TTS), and «gravity» support thermal shields (STS), include; the sub-assembly supports for the CTS lower cylinder, and for the cryostat / CTS lid; a large lifting “spider” for the CTS lower cylinder; purpose built handling tools for the different thermal shield elements; temporary supports; survey tools for supports, and panels; and leak checking, and electrical resistance measuring equipment.

4.6.3.5 PF Coil Assembly Tools

This group of tools includes, for each coil: a transportation frame, to support and protect the coil during on-site transit; purpose-built survey tools; lifting adapters to interface with the universal lifting beams; and installation tools, mainly temporary, adjustable supports.

In addition, for the two lower trapped coils (PF5 and PF6): an array of support frames are provided to support the coils from the cryostat base, at the minimum height consistent with providing the required "ground" clearance for the coil feeders, for the duration of the sector assembly stage; and a distributed jacking system, to raise the lower coils to their final, installed elevation.

4.6.3.6 Port and Piping Assembly Tools

The port assembly tools comprise: the handling and positioning equipment necessary to introduce the port extension through the bioshield and align with the port stub/stub extension; a machine tool to correct the perimeter of the port stub to provide the correct welding fit-up for stub/port extension installation (the port handling and positioning equipment will be used to introduce, support and align the machine tool); survey tools for the gaps for the field joint splice plates; welding clamps; welding rail and NG TIG welding tools; and installation tools for handling and positioning the splice plates. There are significant differences in the design of the port components, and the assembly of the NBI and cryopump ports will require separate, dedicated tooling.

To maintain the assembly schedule, the parallel assembly of three ports at each level is foreseen, with a consequent impact on the quantity and cost of the tooling. More productive welding technologies, which could mitigate this are currently under investigation

The piping assembly tools consist of handling fixtures, and standard orbital welding tools.

4.6.3.7 Central Solenoid Assembly Tools

The principal assembly tool for the CS is a lifting tool with the capability of adjusting the verticality of the coil with high precision. The tool must also interface with the upper region of the TFCs, to provide the required support to the CS during installation, without unnecessarily occupying the overhead cranes, which are heavily utilised for assembly.

Survey tools will be required to determine the custom dimensions for the CS flexible supports. In addition, a support and positioning fixture will be required to facilitate installation of the lower centring device.

4.6.3.8 Correction Coil and Feeder Assembly Tools

This group of tools comprises, for each design of CC; a handling strongback, to rigidise and protect the coil during handling; survey tools for the CC supports; and welding fixtures to position and support the coil clamps during installation.

4.6.3.9 In-Vessel Assembly Tools

For the two major in-vessel systems, the blanket and divertor, the approach will be to use the remote handling tools to transport and manipulate the components inside the vessel, possibly operated in a hands-on mode. Also, variants of the remote welding tools, adapted for hands-on operation, may be used if commercially proven alternatives cannot be found for the specialised weld geometries which exist, for example, between the blanket modules and their co-axial feed pipes. Elsewhere, commercially available equipment will be used for the installation of the in-vessel components.

4.6.3.10 Common Handling Tools

The common handling tools include the lifting tools required to interface the overhead cranes with various, purpose-built component lifting tools. Presently these comprise: the dual hoist beam - a device with a single, centrally located, slewing attachment with 750 t capacity; and a set of 4 universal lifting beams, spreader beams that will be connected to each of the four 375 t crane hooks via a central attachment, and to the component via an attachment at each of their extremities.

The other major handling tool is an integrated system of heavy, surface-based transporters based on a diverse range of technologies.

In addition, a number of conservatively sized portable cranes, flat bed trucks and fork lift trucks will be required.

4.6.3.11 Standard Tools

This group includes the standard, commercially available tools required to complete the assembly operations.

4.7 Remote Handling Equipment (WBS 2.3)

(Updated A.Tesini Jan07)

4.7.1 Functional Requirements

The function of ITER remote handling system is to provide dedicated systems for the initial assembly, planned maintenance and unscheduled operations on the following in-vessel components:

- divertor cassettes
- blanket modules
- NB heating system
- port plugs

Also to:

- transfer in-vessel components from/to the Hot Cell
- repair, refurbish and/or process prior to radwaste disposal of the in-vessel components

Further, the ITER remote handling system includes:

- in-vessel viewing/metrology system (partially developed)
- VV cryopumps removal equipment (to be developed)
- dust removal equipment (to be developed)
- leak check probe deployment system (to be developed)
- In-cryostat inspection/repair system (to be developed)

4.7.2 Environment

The radiation environment and operating conditions for the three RH areas : divertor cassettes and blanket modules replacement, NB Maintenance, and Viewing/metrology equipment, are summarized below:

Table 4.7-1 Environment for Typical Remote Handling Operations

<table>
<tr><th></th><th>Inside the VV</th><th>Inside the gallery</th><th>Inside the Hot Cell</th></tr>
<tr><td>Divertor</td><td rowspan="2">~500 Gy/h 12 days after S/D and beyond.

Case 1: all cassettes are inside the VV, RH equipment accessing central cassette cooling pipes near cassette body from duct: ~1 Gy/h

Case 2: some cassettes have been removed, RH equipment accessing second or standard cassettes: ~100 Gy/h</td><td rowspan="2">The expected radiation level inside the tokamak building gallery during a divertor cassette transportation is about ~50 Gy/h. (~120 Gy/h conservative value, i.e. if made equal to the case of two blanket modules in a cask in the gallery).</td><td rowspan="2">~ 300 Gy/h (a - compounded value for HC storage area [~19 cassettes, ~12 blanket modules, ~8 plugs), b - less elsewhere, depending on components distribution and position inside the Hot Cell)

~300 Gy/h max in the radwaste area (Note: to be confirmed after a comprehensive radwaste model of the ITER Hot Cell is assessed)</td></tr>
<tr><td>Blanket</td></tr>
</table>

Table 4.7-2 NB maintenance Operation Environmental Conditions

Description	Design Condition
Pressure	0.1 MPa
Atmosphere	Air
Temperature	Ambient
Radiation($1X10^6$ second after shut down)	-Around ion source:1 Gy/hr -Outside the ion source vacuum boundary:$1x10^{-2}$ Gy/hr -Out side the beam line: $1x10^{-2}$Gy/hr -Around shutter: $1X10^{-1}$ Gy/hr -Outside the magnetic and radiation shield:$1X10^{-4}$ Sv/hr
Magnetic field	None

Table 4.7-3 Viewing/Metrology Equipment – Operation Environment

	SCHEDULED INSPECTION	UNSCHEDULED INSPECTION	MACHINE MAINTENANCE
Postulated operational Scenario	1 day after plasma operations	1 hour after plasma operations	2 weeks after plasma operations
Atmosphere	UHV conditions (>10-5 Pa)		Dry air (approx 1 bar)
Temperature	Normal case <120 ˚C Off-normale case <240 ˚C		<50˚C
Magnetic field (deployed)	5 – 8 Tesla		Approx. zero
Magnetic field (stored)	0.3 Tesla		Approx. zero
Radiation * (dose rate) (deployed)	1,500 Gy/h	15,000 Gy/h	300 Gy/h
Radiation * (accumulated dose) (stored)	Total Neutron Fluence (Basic Perfomance Phase BPP) est. 5.07 E13 n/cm2 Gamma dose (BPP) est. 1.41 E+02 Gy Additional radiation from divertor cooling water ^{16}N (TBD)		
Assumed inspection duration *	12 h/week, twice monthly, over 7.5 years	12 h/week, twice monthly, over 7.5 years	60 h/month over 32 months
Total dose gamma Total dose neutron	2.7 MGy 5.07 E13 n/cm2	27 MGy 5.07 E13 n/cm2	0.6 MGy 5.07 E13 n/cm2

*estimated data

4.7.3 *Configuration*

See Table 3.10-1 and Table 3.10-2 for RH classification, equipment use and expected maintenance times. The following table summarises the expected ITER machine component payloads. The following table summarises the expected payloads for the ITER remote handling equipment design.

Table 4.7-4 ITER Machine Components Payloads

	Component	Mass (est.) (t)	Comments
	DIVERTOR SYSTEM		
1.	Standard cassette	10.7	
2.	Central (or diagnostic) cassette	12.5	
3.	Second cassette	10.7	
4.	Diagnostic rack (with shielding)	25	
5.	Primary closure plate	5	
6.	TBD		
	BLANKET SYSTEM		
7.	Blanket module	4.5	Maximum value for largest module
8.	TBD		
	PORT PLUGS SYSTEM		
9.	Upper port plug	20	
10.	Equator plug	45.6	Max allowable payload based on current cask system design
11.	Test blanket module plug	40	TBD
12.	TBD		
	CASK SYSTEM (see table 4.7-7)		
	OTHER VV MOUNTED COMPONENTS		
13.	Cryopump valve	1 t	
14.	Cryopump body	5	
15.	IVVS probe system	1	
16.	GDC/GIS	1	
17.	NB source	26	
18.	NB fast shutter	6	
19.	TBD		

4.7.3.1 Divertor Maintenance Equipment

4.7.3.1.1 Cassette Multifunctional and Toroidal Movers

CMM (cassette multifunctional mover): This system includes a radial tractor together with a set of specific end-effectors to grip, push, pull, position in a cantilevered manner and lock/assemble the following equipment:
- the primary closure plates (closing the 3 RH ports)
- the diagnostics racks (inside the 3 RH ports)
- the central cassettes (in front of the 3 RH ports)
- the second cassettes (located at the left hand-side of the central cassettes, as viewed from outside the vacuum vessel)
- the third cassettes (located at the right hand-side of the central cassettes, as viewed from outside the vacuum vessel)
- the right hand side and left hand side cassette toroidal movers
- the standard cassettes (all other cassettes)

CMM system mock-up tests are planned in 2007.

CTM (cassette toroidal mover) : It performs the in-vessel toroidal cassette handling operations, the final radial and toroidal positioning, the locking & unlocking of the supports and finally, the inspection of the assembled standard cassettes. A right hand side and a left-hand side CTMs are required to perform the cassette replacement sequence.

4.7.3.1.2 Other systems required for divertor cassette removal

- *Manipulator arm (MAM)*
  The MAM is a dextrous manipulator arm (including associated tools and tools boxes) intended to perform RH operations in-vessel and inside the divertor connecting duct. The MAM must be incorporated into and work in conjunction with, the CTM, the tractor of the cassette multi-functional mover (CMM), possibly the CMM end-effector for second cassette locking (inboard support only), and the cassette toroidal movers. The ability of the MAM to carry out the intended RH operations is yet to be demonstrated. Of particular concern is the ability of a single-robotic arm system to perform complex operations within a very confined space (typical example: divertor cooling pipes orbital tools handling, divertor cassette diagnostics connector handling, etc).
- *Pipe Tool System*
  The divertor pipe tool system implements the cutting, welding and inspection of the divertor cassette water cooling pipes routed along the 18 divertor ports. The ability of such a tool system to successfully carry out the required operations is yet to be demonstrated.
- *Divertor rails*
  The toroidal rails, in conjunction with the cassette supports, implement the alignment and attachment of the cassette, and secure it to withstand the loads imposed during plasma operation. During cassette handling, the toroidal rails guide and support the cassette toroidal mover inside the vessel.
- *Control system*
  Control is implemented both locally using local programmable handheld controllers (commissioning and first training) and remotely from the control-room using workstations.

4.7.3.1.3 Divertor Maintenance Operations Limits

Six months is the maximum (i.e. target) period allocated for cassette refurbishment/replacement (from divertor port primary closure plate opening to PCP closure). Allowance for machine shutdown/startup time should be added to the above period.

The divertor cassettes refurbishment/replacement time is strongly affected by a number of parameters. A study has shown that the combination of:
- 2 divertor transfer casks (each fitted with the necessary divertor cassette RH equipment)
- 2 divertor cassette refurbishment workstation and 2 divertor cassette testing stations inside the Hot Cell
- temporary Hot Cell storage space for up to 19 divertor cassettes
- 1 Hot Cell divertor cask docking port

would allow refurbishment of the 54 divertor cassettes in about six months.

Work is required to further assess the impact of in-vessel divertor handling, RH tools exchange, cask operations, Hot Cell refurbishment & testing processes on the overall cassettes replacement/refurbishment time.

4.7.3.2 Blanket Maintenance Equipment

4.7.3.2.1 RH Features of Blanket Modules

The current blanket RH system is designed to replace blanket modules in small numbers and only occasionally, following the initial requirements of the ITER design. Subject to further, intensive design and development work and to the availability of a full 360° IVT system (currently not in the ITER procurement program) the exchange a complete blanket first wall (shield plus FW) with a new one, of the same design, should take about two years. This is based on the assumption that a set of new, 400+ blanket modules has been procured, tested ready for installation (not the case under the current Project procurement plan).

Each blanket module is equipped with seven remote handling access holes through its plasma-facing first wall. Four allow access for the tool that bolts and unbolts the flexible supports, one gives access for welding, cutting and inspection of the cooling pipes and two for bolting and unbolting of the electrical straps, and these are also used to grasp the module and hold it securely during handing into and out of the VV.

4.7.3.2.2 Blanket Handling System

The blanket handling system has the following features.

- One or more in-vessel transporters (IVTs), with manipulator arm and various attachments, is mounted on a monorail deployed inside the vessel from a dedicated port-mounted cask.
- Module transporters, mounted at intermediate ports, are used to load modules into casks, thereby enabling the modules to be transferred to and from the hot cell.
- Each IVT consists of a vehicle manipulator, a segmented rail, rail-deploying equipment, a rail support device, and cable-handling equipment. The IVT is used for general in-vessel maintenance tasks, and with a variety of end–effectors, depending on the task to be performed.
- The full circumferential rail is made up of two semi-circular rails. For robustness, neither sensors nor actuators are installed on the rails. An important feature of the rail system is that it can function in a reduced configuration. The rail system can be installed and used in the following configurations: 80/100°, 180° and 360°. The rails are held in position by support devices, which are used to install the links that make up the rails.
- Since the vehicle/manipulator moves up to 180° on the rail as it travels around the VV, a trailing, long complex cable with many conductors has to be fed without jamming. To do this, the cable handling equipment is installed in an intermediate cask between the VV port and the RH cask, after the vehicle manipulator and rail are installed in the VV.
- The module receiver portion of the module transporter is moved radially into the VV and the module is transferred to it from the IVT manipulator. Subsequently, the module transporter is moved radially outward and the in-vessel component is transferred into a RH cask docked to the port. Up to three modules can be transported simultaneously to the hot cell. The module receiver pallet can be retracted from the transporter and exchanged inside the RH cask by an in-cask storage rack system.

4.7.3.2.3 Blanket handling equipment configurations

To implement the cask-based IVT system, up to four of the dedicated equatorial RH ports can be used for installation of the IVT system and transportation of the blanket modules. Therefore, both an 80/100° rail and a 180° rail deployment system can be used for blanket maintenance, in addition to the full 360° configuration. The configuration of each system variant is shown in the following table. It is likely that for initial machine operations one 80/100° system will be adequate. Therefore, depending on the location of damaged modules, the 80/100° degree system and 180° degree system can be used.

Table 4.7-5 Possible Configurations

Equipment and device	80/100° system	180° system	360° system (full system)
Rail	100°	180°	360°
Vehicle manipulators	1	2	4
Rail-deploying equipment	1	1	2
Rail support devices	2	3	4
Cable-handling equipment	1	2	4
Intermediate casks	2	3	4
RH casks	2	3	4
Module/tool transporters	1/1	2/1	2/2
In-cask storage racks for modules/tools	1/1	2/1	2/2
Welding/cutting tools	1	1	2
Bolting tools	1	1	2
Inspection tools	1	1	2
Rescue tools	1	1	1

4.7.3.2.4 Rescue Scenario

One failure mode that has serious consequences is a malfunction of the actuator in the vehicle manipulator during blanket module handling. To cater for this eventuality, all the mechanisms of the vehicle manipulator have redundant mechanisms, which incorporate an external drive connection that can be accessed by a rescue tool handled by a second vehicle manipulator. The rescue tool then connects to the redundant, healthy actuator and drives it, thereby bypassing the damaged mechanism of the vehicle manipulator. This scenario has yet to be demonstrated.

4.7.3.2.5 Procurement and Maintenance Time Estimation

A staged procurement is assumed for the IVT system~~s~~. During the first ten years of operation, an 80/100°, or if proven necessary, a 180° system will be used for in-vessel component maintenance with ~~a~~ one or two refurbishment stands and a test stand in the Hot Cell. To speed the replacement of all blanket modules (and as required by the initial blanket modules assembly schedule) an additional 180° system, may be procured.

Table 4.7-6 Time Estimates for Blanket Maintenance

Maintenance [a]	80/100° system	180° system	360° system (full system)
One blanket module replacement	25 days	(same as 80/100° system)	(same as 80/100° system)
One toroidal row replacement	Needs to be deployed from different ports in series (96 days)	60 days	44 days (33 days [b]))
All blanket modules replacement	Needs to be deployed from different ports in series (518 days)	482 days	464 days (244 days [b])

Notes
(a) the estimated times are for the RH operations, VV preparatory and pumpdown operations are not included, and one refurbishment stand and one test stand are assumed in the 360° system column,
(b) two refurbishment stands and two test stands are assumed.

Further note to the above table.:
The above time estimates are preliminary and need to be further validated or adjusted after completion of the following studies:

1) confirmation of FW exchange strategy: FW panels vs whole shield+FW
2) detailed time estimate of FW panels exchange (if the case) process and testing under RH conditions inside the HC (FW panels attachment system design to shield modules to be confirmed)
3) detailed time estimate of whole shield+FW panels exchange (if the case)
4) detailed breakdown and estimate of the IVT operations inside the VV (blanket replacement time)
5) detailed logistics study to establish maximum possible number of transfer casks, HC workstations, buffer storage space, RH tools exchange that can be used without impairing overall process reliability.

4.7.3.3 NB Injector Maintenance

NOTE: a major review is ongoing -2006/7- to redesign the NB system to allow assembly and maintenance using more realistic procedures and equipment ~~systems~~. The expected design changes will include:

- Use of the NB cell as an extension of the Hot Cell, linked to it via a dedicated access corridor
- Use of an overhead crane system for the top assembly and replacement of the NB line components, NB source and front line components (bellows, slow valve, fast shutter valve – components' design yet to be confirmed)
- Combined use of floor mounted transporters and overhead crane mounted tools for the replacement and transport of the NB source from/to the Hot Cell, for the replacement of the ion source filaments, for the replacement of the caesium oven, for the local cleaning of caesium deposits from critical areas

4.7.3.4 Port Maintenance

Port modules are maintained by a dedicated cantilevered system mounted in a cask.

Table 4.7-7 VV Port Operation - Environmental Conditions

Radiation (dose rate inside VV)	Approx. 500 Gy/hr (12 days after shutdown)
Contamination	Activated dust, beryllium, tritium
Magnetic field	Zero
Vacuum vessel port temperature during cask operation	≤ 50°C
Vacuum vessel pressure during cask operation	0.1 MPa (average)
Cask internal atmosphere	Air at ~ -100 Pa w.r.t. ambient

Prior to cask use, some operations (clearing of port cell equipment) will be carried out by personnel.

4.7.3.5 Control Systems

The RH equipment control system allows the operation of RH equipment from the RH control room or hands-on using local programmable handheld control panels (when and where allowed).

The RH equipment control system consists of a supervisor control system, linked to the main ITER CODAC system. Through this, the RH equipment and tools located inside the VV, inside the Hot Cell and inside the Test Stand is controlled.

The main RH control room is located inside the Hot Cell (further design and operational considerations are yet to be made).

4.7.3.6 Casks

The ITER transfer cask system consists of three main sub-systems:

- cask envelope (including the double door system, payload handling systems)
- pallet
- air transfer system (ATS)

The cask enclosure geometry and the cask payload handling equipment vary according to the type of in-VV port interface and to component to be replaced. The pallet and the ATS are of standard design, with the exception of those required for the cryopump replacement (to be designed).

The cask transport and docking safety aspects are covered extensively in section 3.1 (safety), the GSSR, and will be included in the French RPrS: Preliminaire Rapport de Surité.

Table 4.7-8 Summary of Cask Properties

rev. Oct2006

	UPPER PORTS CASK	EQUATOR PORTS CASKS			LOWER PORTS CASK	
Cask type	Upper	Equator	IVT Main	IVT Inter-mediate	Divertor	Cryo-Pump & IVVS
Dimensions						
Envelope Width (mm)	2,040	2,620			2,200	1,920
AACT Transporter Width (mm)	2,160					
Pallet width (mm)	2,620					
Nominal Height (mm) (static)	3,680					2,590 tbd
Nominal Height (mm) (travelling)	3,730					2,640 tbd
Length (mm)	8,500			5,540	8,500	7,500
Weights and Loads						
Estimated payload (t)	20	45	12 (3 blanket modules)	6 (1 bkt module)	13	1
Cask weight (t)	27.8	27.8	tbd	tbd	38.5	<20 tbd
Pallet weight (t)	6.7	6.7	6.7	tbd	6.7	< 6 tbd
AACT weight (t)	8	8	8	tbd	8	8
Tractor weight (t)	5	5.2	5 tbd	2 tbd	9.6 (CMM +SCEE)	< 5 tbd
Nominal loaded cask weight (t) (assume max allowable floor limit 100 t)	67.5	**92.7**	tbd	tbd	75.8	tbd

Table 4.7-9 Summary of Cask Operational Limits

Radiation Resistance	Expected dose inside the cask: 50-200 (max) Gy/h Cask base/Air transfer system shielding factor =10 Radiation hardness: 10 MGy max allowable dose on cask internal mechanisms lubricant (worst case). 30 Mgy max allowable dose for other components
Operating Temperature	to avoid excessive pressurization of the cask internal atmosphere, leading to loss of contamination through seal leakage or other structural damage, the cask inner temperature must be maintained at ≤ 65 °C
Operating pressure	Min 0.90 bar, max 1.05 bar
Leak tightness	< 1 Pa-litres/s (0.01mbl/s)

Recommended Acceptable VV Port Flange Angular Deviation to allow correct cask alignment operations		Upper Port	Equator Port	Divertor Port
	Toroidal	1.2º	1.2º	0.6º
	Poloidal	0.3º	0.3º	0.3º

4.7.3.7 Viewing System

The main function of the ITER in-vessel viewing system (IVVS) is to allow in-vessel inspection to look for possible damage to plasma-facing components. The system can also be used for metrology measurements of the plasma chamber and its components.

Table 4.7-10 Viewing/Metrology Equipment - System Description

Base system	Viewing/metrology probe (IVVS) mounted on deployment arm inserted through the ITER VV port
Number of viewing locations	6 at lower ports positions: 03, 05, 09, 11, 15, 17, not uniformly distributed (See Table 3.5-3 Port and Port Cell Allocation at Lower Level)
Storage system	Within sealed enclosure, inside the lower ports. The storage area accommodates also the probe arm deployment mechanism
Deployment system	Via arm (22 kg payload) propelled and guided through a penetration in the VV wall. Possibility to raise the arm inside the VV IVVS deployment requires prior withdrawal of glow discharge plug (both systems share same VV penetration)
Viewing probe parameters	Field of view: azimuth = 360° /elevation = +70° to -95° Incidence beam angle max: 60° Viewing resolution: 1 mm (target distance: 2 to 7 m) Ranging accuracy: 0.5 mm @ 5 m

Table 4.7-11 Viewing/Metrology Equipment – Performance

Parameter	Performance
Line of sight (nominal case) 90°max incidence beam angle	98.5% of the vessel is in direct “line of sight” of at least 1 probe (based on the use of 6 probes). The missing 1.5% is in the divertor area (represents vessel surface of 13m^2). Each blind spot behind the probe is covered by another probe
Line of sight (actual case) 60°max incidence beam angle	66% of the vessel can be viewed (31% of divertor) (2 m – 7 m range) 72% of the vessel can be viewed (41% of divertor) (1.5 m – 10 m range)
Probe viewing performance No auto focus * 60°max incidence beam angle	Viewable area reduces to - 76% of the blanket and 31% of the divertor (2 – 7 m range) - 81% of the blanket and 41% of the divertor (1.5 – 7 m range)

* The use of autofocus is being investigated

4.7.3.7.1 Viewing System – Radiation Conditions

The expected radiation levels during IVVS usage are (12 days after shutdown):

- ~1000 – 1,500 Gy/h (depending on vertical position of probe inside the VV) 1 day after Shut Down
- ~300 Gy/h 12 days after Shut Down and beyond

4.7.4 RH equipment off-normal loading conditions considerations

With regard to the seismic design and qualification of the ITER RH system, the following should be noted:

The ITER RH equipment design is based on the seismic loads or their combinations caused by earthquakes that are to be assumed during the maintenance operation. The level of earthquakes assumed depends on the host country's regulations. It is based on load values as required by seismic events categorized under SL-1 inclusive of any applicable amplification factor. However, the following conditions/limitations apply:

- any design based on seismic load that leads to dimensional increase of the RH equipment such to prevent its correct function, reduce its clearance with surrounding structures, causes mass increase, will not be accepted even if this will partially or entirely fail to satisfy the SL-1 loading. However, confinement integrity should be maintained. As an ultimate fall back position the vent detritiation system will be available in the tokamak gallery and hot cell areas.

- should the above design approach lead to malfunction or damage to the RH equipment under SL-1 loads, the equipment design shall include solutions that allow retrieval from the VV without contamination release and within a reasonable time

- RH equipment design will be such that in the event of failure damage to high value components will be prevented as far as practical (investment protection)

4.8 Hot Cells and Waste Processing (WBS 2.3bis)

The following section is based on the design evolution since FDR, and new layout proposed at TCM-16, and DCR-35 (See Techical web, TCM-16, and Hot Cell Design Reviews July 2005, Feb 2006, Dec 2006). However this is still subject to discussion within the ITER Project and with the participating parties, with requirements for new elements such as extra Test Blanket Modules and a Port Plug test area, cask storage and repair area, Neutron Test Area, etc.

A discussion is on-going (2007) with the aim of reducing the Hot Cell size. This is possible provided that some Hot Cell functionality and scope is reduced. The impact of such reduction on the machine operations, schedule and availability has yet to be fully quantified.

4.8.1 Functional Requirements

The function of the hot cell component repair system is to process and repair components, tools, and equipment which have become activated by neutron exposure and/or contaminated with tritium or activated dust.

Processing includes examination, preparation of service plan, preparation of samples for material evaluation, evaluation and segregation of parts into those which can be reused and those which must be replaced, disassembly, replacement of parts, re-assembly, and inspection/testing.

Components which enter the system for repair may be diverted to the hot cell waste processing system.

The function of the hot cell waste processing and storage system is to process and store solid radioactive materials which have been removed from the tokamak and which will be discarded. Waste processing includes disassembly.

The hot cell waste processing and storage system provides up to 6 months storage of radioactive waste for an interim period prior to hand-over to the host.

A low-level waste processing system (LLPS) is provided to process solid and/or liquid waste process streams, which have become contaminated with radioactive materials.

4.8.2 Configuration

4.8.2.1.1 Hot Cell Facilities (2005)

The new layout (2005) for the hot cell building and the various refurbishment areas is developed with a view of space optimization and safety consideration, which consist of two operationally independent refurbishment cells: the main refurbishment cell and the port plug refurbishment cell.

The radwaste facility is located between the two. All main gantry cranes are oriented north to south and a floor mounted cell traverse system links the various cells east to west. Components are posted into the hot cell from casks. The refurbishment cell provides dedicated areas for dust cleaning, refurbishment and basic testing of components. Inspection (visual, dimensional and welding inspections) can be undertaken at any workstation. After refurbishment, port plugs can be mounted in a flange with the ex-vessel side in a man-accessible area for testing and commissioning of the internal diagnostics and heating systems (the upper port plugs will require instead full insertion into the hot cell for testing).

The Remote Handling approach within the Hot Cell is based on proven manipulators' technology. Jib cranes and pairs of manipulator arms on short articulated booms, which can drive along and up and down the side walls, have been selected. A vertical telescopic mast mounted on each gantry crane allows for a manipulator to be near the crane hook at all times. This arrangement allows a manipulator to have access to all parts of the hot cell building (below crane gantry level) for ad-hoc tasks and to maintain building services (e.g. lights and cameras). The transport and storage of plant within the hot cell is done using carrier frames, which are also stackable. In-vessel components are generally moved by crane, with the exception of the floor mounted cell traverse system for lateral transport. Hot Cell tools are stored/commissioned/maintained in the HC tool store areas. A palletising system with storage racks for Hot Cell tools and Through-The-Wall-Manipulator maintenance areas are provided in each refurbishment cell. Hot Cell services are situated in green areas as close as possible to the workstation/test tanks where they are deployed. The palletising system delivers tools to the maintenance and test areas.

The above Remote Handling approach is applied consistently throughout the main refurbishment cell and the port plug refurbishment cell. Workstations and test tanks are multi-purpose where possible. The man-in-the-loop principle guarantees maximum flexibility and adaptability. Good lighting and plenty of camera views, both wall mounted and mounted on remote handling equipment, assist the operator together with a real-time Virtual Reality model.

4.9 Cryostat (WBS 2.4)

(This section will be updated in 2007)

4.9.1 Functional Requirements

The cryostat:

- provides a a vacuum environment to avoid excessive thermal loads applied to the components operated at cryogenic temperature, such as superconducting magnet system, by gas conduction and convection;
- forms a physical envelop surrounding the vacuum vessel, which automatically enhances confinement functions and can be a part of the secondary confinement barrier, if required (currently assumed as the secondary confinement barrier);
- allows passive removal of decay heat of vacuum vessel and in-vessel components by gas conduction and convection;
- has penetrations for :
  - equipment connecting elements of systems outside the cryostat to the corresponding elements inside the cryostat (magnet feeders, water cooling pipes, instrumentation feedthroughs, CV pumping systems);
  - access to VV ports;
  - access for maintenance equipment into the cryostat;
  - access to the CS coil for possible direct removal
- transfers all the loads, which derive from the tokamak basic machine and the cryostat itself during the normal and off-normal operational regimes and at specified accidental conditions, to the floor of the tokamak pit through its support structures.
- includes overpressure protection for itself.

4.9.2 Configuration

4.9.2.1.1 Cryostat Configuration

The cryostat is a fully-welded, stainless steel vessel with a large number of horizontal penetrations.

The cryostat is a single wall cylindrical shell with top and bottom lids. The maximum diameter of outer cylindrical part is 28.6m and it is reduced to 19.6m below the VV divertor ports in correspondence with the machine supports.

4.9.2.1.2 Cryostat Support/Relation to Bioshield

The cryostat is supported by the building and surrounded by a concrete bioshield (~ 2 m thick) keeping a radial clearance of approximately 0.5 m and is covered by a concrete shield (~ 1m thick) on the top.

The bioshield includes a ~ 1.2m thick slab above the cryostat, which reduces the nuclear radiation due to sky-shine at the site boundary and is supported in the current design by a truss structure connected to the upper head (top lid) of the cryostat. Another design option is under investigation (DCR-36, June2005) for improving on-site assembly by minimizing the cryostat wall thickness.

In the unlikely event that large components located inside the cryostat need to be replaced, the upper bioshield slab can be removed, and the cryostat head with the support structure can be cut from the cryostat cylinder and also removed.

The upper bioshield slab is designed to be installed and removed in several parts to remain within crane limits. The weight of the cryostat head, including the truss structure, is below 1,000 t.

The upper head for supporting the 1.2m thick slab is a circular flat plate with radial stiffening ribs spaced every 10 degrees and integrated with the bioshield support which consists of a ferritic steel truss structure.

The lower head (bottom lid) is reinforced similarly to the upper head and also connected to the cylindrical shell by welding.

4.9.2.1.3 Cryostat Loads

The main design loads considered are external pressure of 0.1 MPa, for normal operation under vacuum, and 0.2 MPa absolute internal pressure, for the conditions assuming the loss of helium and water from coolant lines routed through the cryostat. The load combination with seismic loads and the pressure loads is taken into account for the structural design.

4.9.2.1.4 Cryostat Ducts and Bellows

Large ducts (~3m height x ~2m width) interconnect the VV ports with corresponding aligned penetrations in the cryostat vessel. Bellows are integrated in the duct to compensate for differential movements. These bellows have a rectangular shape and are made of stainless steel.
The design parameters of the cryostat are summarized in Table 4.9-1 and these will be further optimized in accordance with the site specific conditions such as seismic loads.

Table 4.9-1 Typical Design Parameters for Cryostat

Parameter	Value
Cryostat base section: ϕ 19.6 x 6.0 m	1190 t
Cryostat lower cylinder ϕ 28.6 x 10.2 m	730 t
Cryostat upper cylinder ϕ 28.6 x 8.8 m	680 t
Cryostat lid with structure ϕ 28.6 x 4.2 m	700 t
Total Cryostat ϕ 28.6 x 29m	**~3300 t**
Design temperature of cryostat wall	300K
Structural Material - main shell - truss structure on top of lid	 Type 304 & 304L ASTM A572 Grade 42/50
Operating Vacuum Pressure	< 10^{-4} Pa
Accidental Pressure	0.2 MPa
Typical shell thickness - cylinder - upper head - lower head	 80mm 40 mm 50 mm

4.10 Vacuum Vessel Pressure Suppression System (WBS 2.4 bis)

(Updated Y.Kataoka Aug./Sept. 2006)

4.10.1 Functional Requirements

The Vacuum Vessel Pressure Suppression System (VVPSS) is for overpressure protection to limit the maximum pressure in the Vacuum Vessel (VV) to the value not exceeding to the allowance during in-vessel coolant leak events. This also functions to maintain the VV long term pressure below atmospheric pressure during air or incondensable gases ingress.

A Design Change Request is ongoing (Aug 2006) which will lead to significant changes in the VVPSS system: (DCR-34 Redesign the VVPSS relief pipes to by-pass the NB shutter)

In addition there are some current "Issues" which will also have implications on the design:

- 2.4-4 - Update safety-relevant equipment for the Cryostat and VVPSS
- 8.0-1 – Dust/ Hydrogen hazard
- 8.0-2 – Confinement Strategy

4.10.2 Configuration

4.10.2.1.1 VVPSS Configuration

The VVPSS consists of a large suppression tank with a circular cross section made of ferritic steel, containing enough water at room temperature to condense the steam resulting from the most adverse in-vessel coolant leaks. Typical design parameters of the VVPSS tank are shown in Table 4.10-1.

In the current layout (2005), the VVPSS tank is connected to the vacuum vessel through three relief pipes attached to two of the H&CD neutral beam boxes and the diagnostic neutral beam box. Each relief line (inner diameter ~ 1 m) has two consecutive rupture disks and at the downstream end the three pipes are manifolded into one single duct with flow area of 1.1 m2 and thereafter routed to the VVPSS tank. The rupture disks are located outside the NBI cell (Zone B) for maintenance. A design option (see DCR-34) to route the relief pipes bypassing the NB shutters is foreseen to simplify the NB port structure and rupture disks arrangement, including maintenance of NB shutter and bellows

In the consecutive rupture disk arrangement, the first rupture disk opens at the required pressure difference while the second rupture disk opens at smaller pressure difference so as to avoid backpressure problem from the manifold.

The diameter of relief pipes and flow area of the manifold are chosen to maintain the VV pressure below 0.2 MPa at the largest in-vessel coolant leak assuming multiple break of the FW cooling tubes with total break area of 0.2 m^2. Pressure relief is provided by two of the relief pipes, the third being redundant.
(However it is under discussion (Sept 2006) whether such redundancy is really necessary, hence allowing a smaller pipes or the elimination of one of the three lines pressure relief lines.)

The VVPSS tank is located at level +23.06 m above the cryodistribution cold boxes in the west side of the tokamak building.

4.10.2.1.2 VVPSS Bypass-Bleed lines

The current VVPSS design includes two bleed lines that bypass the rupture disks to avoid their opening at small in-vessel water leaks. Each bleed line with flow area of 0.05 m2 has a valve which opens at lower pressure difference than the rupture disks. Pressure relief is provided by one of the bleed lines, the second being redundant.

4.10.2.1.3 VVPSS Operation

During an in-vessel coolant leak the VVPSS acts in concert with the safety drainage system. Steam produced in the VV flows to the suppression tank where it is condensed. The safety drainage system is brought into play automatically by the opening of the valves which are part of the cooling water system. The drain valves timely opened after the break prevent additional steam generation and long term VV pressurisation due to heat accumulation from the in-vessel components.

4.10.2.1.4 VVPSS Service Interfaces

The VVPSS is connected to the suppression tank vent system (ST-VS) and then to the standby vent detritiation system (S-VDS), the low level waste processing system, the liquid and gas distribution system and the leak detection system.

The VVPSS has provision to maintain the VV long term pressure below atmospheric at air in-leakage and in-vessel ingress of incondensable gases with a concurrent coolant leak in the VV. This can be made by extracting such gaseous exhaust from the VVPSS tank wet-well and transferring it to the normal vent detritiation (N-VDS 1) via suppression tank vent system (ST-VS).Note that the need of the passive recombiner for hydrogen and oxygen, relating the current “Issues 8.0-1 for the case of a torus air leak, is under discussion (2006).

Table 4.10-1 Typical Design Parameters for VVPSS

	Parameter
Overall Diameter	6m
Length	46m
Structural Material (1)	ASME SA-516 Gr.55
Operating Pressure	Vacuum (< 4.2 kPa)
Design Pressure	0.2 MPa
Typical shell thickness	30 mm
Initial max Water temperature	30 C
Initial water mass	~675 t
Tank volume	~1200 m^3

(1) ASME SA-516 Gr.55 is the standard carbon steel that is normally used for pressure vessel plate (moderate and low temperature service). This material has been selected to meet the mechanical strength and low cost (yield strength: 205 MPa, tensile strength: 380 - 515 MPa). Other materials could be envisaged but an assessment would be required (including impact on magnetic field, depending on the relative distance to tokamak.

4.11 Cooling Water (WBS 2.6)

(Reviewed YKA Aug 2006)

4.11.1 Functional Requirements

The main functions of the cooling water system are to:
- remove heat deposited in the in-vessel components, the vessel, additional heating systems, and diagnostics during the burn cycle;
- maintain coolant temperatures, pressures, and flow rates to limit component temperatures and retain thermal margins during the operating campaign as required;
- remove decay heat also during shutdown periods;
- provide baking for the in-vessel components;
- maintain water chemistry as required;
- accommodate draining, and provide refilling and drying for maintenance periods;
- provide confinement of radioactive inventories;
- maintain radioactive inventory as required;
- measure heat removed from the in-vessel components and vacuum vessel;
- remove heat from components of plant auxiliary systems;
- reject all the above heat to the environment.

4.11.2 Configuration

4.11.2.1 Overall

4.11.2.1.1 CWS configuration

The cooling water system (CWS) consists of the tokamak cooling water system (TCWS), the component cooling water system (CCWS), the chilled water system (CHWS), and the heat rejection system (HRS).
The TCWS transfers heat deposited in the in-vessel components and the vacuum vessel to the HRS, and the CCWS and CHWS transfer the heat load from the components of the plant auxiliary systems which are not activated to the HRS depending on the required coolant temperature in the components. HRS is the final heat sink to reject the heat load from the tokamak and components to the environment.
The TCWS consists of the primary heat transfer systems (PHTSs) and their supporting systems, i.e. the chemical and volume control systems (CVCSs), the draining and refilling systems, and the drying system.

4.11.2.1.2 PHTSs

PHTSs (Primary Heat Transfer Systems) are pressurized closed loops to confine activated corrosion products and tritium in the coolant There are two loops for the vacuum vessel (VV) PHTS, three loops for the first wall/blanket (FW/BLKT) PHTS, one loop for the divertor/limiter (DIV/LIM PHTS), and also one loop is designated for the NB injector PHTS.

These PHTSs serve the main vacuum vessel, the field joints on the divertor ports, port extension at the divertor ports and NB injector ports for the VV PHTS, all blanket modules, the in-port components in the upper and the equatorial ports including the port plugs, and duct liner in the NB injector ports for the FW/BLKT PHTS, all divertor cassettes and two port limiters for the DIV/LIM PHTS, and the low voltage and high voltage components in the NB injectors for the NB injector PHTS.

The VV PHTS is designed to provide the ultimate decay heat removal for all in-vessel components should the other PHTSs be unavailable. For this reason, the VV PHTS makes additional use of passive heat removal capability, i.e. by means of natural convection.

The VV PHTS (see also §4.2.2.6) is divided into two loops, and each loop is connected to alternate halves of the VV 40° sectors. This configuration confers redundancy in that decay heat can be adequately removed by one operational cooling loop in the event that the other becomes unavailable.

4.11.2.1.3 CVCS

The CVCS provides purification of the coolant and volume accommodation during the coolant temperature changes of the PHTSs, except the VV PHTS. Three CVCS, one for each PHTS, are included.

The shaft seal water for the main pumps of the PHTSs and spray water for the pressurisers are supplied by the relevant CVCS.

The CVCS can also provide hydrogen addition into the coolant during operation to minimise or suppress corrosion.

4.11.2.1.4 Draining and Refilling System

Drain tanks are provided to accommodate loop coolants during maintenance and inspection operations, and to accommodate expelled coolant from the CVCSs during baking.

Storage is segmented for each PHTS to avoid mixing coolant with different contamination levels or chemistry requirements.

Some of the drain tanks (seeTable 4.11-7) have the function of the safety-related drainage during an in-vessel coolant leak event to limit the amount of steam that the suppression tank has to condense, and the safety related drain tanks are maintained in a partially-evacuated condition, ~ 10 kPa, during operation.

Refilling of the PHTSs loops is performed by the injection pumps through the CVCSs.

The draining and refilling systems, except that for the VV, are located in the drain tank area at the basemat level of the tokamak building. Valves, instrumentation and refilling pumps are also located in a room connected to the detritiation system. The draining and refilling system for the VV PHTS is located on the same floor but outside the room.

4.11.2.1.5 Drying System

A common drying system is provided to remove residual water from in-vessel components after draining to facilitate leak checking and component replacement. Two independent compressors, one compressor blows out most of the residual water with nitrogen gas, and then the other compressor introduces hot nitrogen gas to evaporate and thereby remove the remaining liquid.

The system can blow-out 8 representative blanket modules concurrently. The estimated drying time, following the blow-out, is ~3 days for the blanket modules connected to one PFW/BLKT loop and for all the divertor cassettes.

The major components of the drying system are located in the TCWS vault cooler and dryer room in the tritium building.

4.11.2.1.6 PHTS Arrangement inside Vault/Pipe Chases/Vertical shafts

An upper pipe chase is located above the upper port level and a lower pipe chase below the divertor port level, and these are interconnected by vertical shafts.

Three pairs of inlet/outlet C-shaped manifolds of the FW/BLKT PHTS, two outlet C-shaped manifolds of the VV PHTS and inlet/outlet pipes for the NB injector components inside the NB cell are located in the upper pipe chase, and one pair of the inlet and outlet C-shaped manifolds of the DIV/LIM PHTS and two inlet C-shaped manifolds of the VV PHTS are located in the lower pipe chase.

The feed and return branch pipes of the FW/BLKT PHTSs except the pipes for the duct liner in the NB injector ports and return branch pipes of the VV PHTSs are in form of a bundle that penetrates the bioshield and the cryostat at the upper pipe chase level, and the feed and return branch pipes for the duct liner in the NB injector ports are routed to the NB cell through the vertical pipe shafts.
The feed and return branch lines of the DIV/LIM PHTS are routed between the lower pipe chase and the divertor ports through the vertical shafts, and the feed pipes of the VV PHTSs penetrate the bioshield and the intermediate cryostat floor at the lower pipe chase level.

The TCWS vault of the tokamak building is vertically separated by a intermediate floor. Large openings in the floor allow air (and steam under coolant leak inside the TCWS vault) communication between the upper and lower area.
The east side of the upper pipe chase is integral with the lower area of TCWS vault.

The main loop components of the PHTSs, except for the VV PHTS, are sited in the upper area of the rectangular TCWS vault above the magnet and CVCS levels, and the components of the CVCSs are sited in the lower TCWS vault area at the magnet and CVCS level.

The HRS pipes, which serve PHTSs, are routed below the intermediate floor.

The air-cooled heat exchangers for the VV PHTS are located on the tokamak building roofs on the east and west sides, and the other main components are at the divertor level in the drain tank area and in the crane hall.

4.11.2.1.7 CCWS

The CCWS removes heat from large process components that do not become activated, and transports it to the HRS (see Table 4.11-9). The system provides high quality cooling water where necessary for components with sensitive water chemistry needs.

The maximum temperature of the CCWS feed water is 40°C. The system is partitioned into two zones: inside the site services building for the tokamak and site service zone, inside the cooling water systems building south of the magnet power conversion building for the power supply zone.

It is noted that the additional clients for CCWS such as the diagnostics in the port cells, the vacuum systems in the gallery, and those for the other buildings are under investigation (Aug 2006), and the heat load and flow rate requirements will be up-dated in future though they are not expected to be significant.

4.11.2.1.8 CHWS

The CHWS utilizes industrial chiller units to provide cooling water at 6°C with a return temperature of 12°C, and it provide a flow of the low temperature (~6°C) to those ITER plant systems (seeTable 4.11-12) which require a temperature less than 40°C. The system is partitioned into two zones, one for the tokamak and site services zone and the other for the power supply zone, the same as the CCWS.
Relating the additional clients for the CCWS described above, the heat load for the chilled water system will be up-dated in future if the clients need lower inlet temperature than 40°C.

The CHWS has separate loops for "safety-related" circuits (to the vault, hot cell and tritium plant), which may need to confine radioactivity. This safety related CHWS utilizes air-cooled condensers in the chiller system, therefore no heat rejection systems for the safety related CHWS are needed.
The main components of the safety related CHWS are located on the tokamak building roofs on the east and west sides.

4.11.2.1.9 HRS

The HRS uses cooling towers to reject heat to the environment. The cooling towers with the hot and cold basins are sited at the east boundary of the ITER site.

The HRS includes water circulation systems (WCSs) for the TCWS, the cryoplant warm compressors, the CCWS, and the CHWS.
There are three pump stations included in the WCSs: one near the tritium building, one in the cryoplant compressor building and one station in the cooling water systems building south of the magnet power conversion buildings. The feed and return lines between the basins of the cooling towers and the pump stations are routed in the utility tunnels.

4.11.2.2 Design Parameters of Tokamak Cooling Water System (TCWS)

Table 4.11-1 Vacuum Vessel Cooling

	Normal operation	Baking
inlet temperature	100 ±10 °C (Flat top)	200 +0/-10 °C (from Table 4.5-10)
inlet pressure	1.1 ± 0.2 MPa	2.4 ± 0.2 MPa
maximum cooldown time to operation temp		24h
maximum vessel heat-up time from RT		See Table 4.5-10 Baking Conditions"
max. cooldown time to maintenance temperature (50 °C)	24h	
Thermal power	SeeTable 3.6-3	
In-vessel pressure drop	< 0.05 MPa	
In-vessel water holdup	~ 110 m^3/loop	
Loop number	2	
Flow rate/loop	475 kg/s (during normal operation) 250 kg/s (during baking) 20Kg/s (during off-normal events)	
Loop pipe inside diameter(1)	0.33 m	
Pressure drop(2)	~ 0.6 MPa	
Total water holdup	~ 160 m^3 / loop	~ 175 m^3 / loop
Main pump - Pumping power	1 / loop 450 kW	
Heat exchanger - Heat transfer area	3 / loop 1,591 m^2 /unit	
Heater - Size	1 / loop 1.7 MW	

Pressurizer (pneumatic) - Size	1 / loop 30 m^3
Height difference (at mid-plane) VV - heat exchanger	32 m

(1) Coolant velocity = approx. 6 m/s
(2) Total pressure drop including in-vessel components, heat exchanger and piping

Table 4.11-2 Blanket Cooling

	Normal operation	**Baking**
Thermal power	SeeTable 3.6-3	
Inlet temperature	100 ± 15 °C (Flat top)	240 ± 10 °C (from Table 4.5-10)
Nominal outlet temperature	148°C	
Inlet pressure	3.0 ± 0.2 MPa	4.4 ±0.2 MPa
In-vessel pressure drop	1.0 MPa	~0.01 MPa
In-vessel water holdup	~ 28 m^3/loop	
Loop number	3	
Flow rate	1,130 kg/s / loop	120 kg/s / loop
Loop pipe inside diameter(1)	0.514 m	
Pressure drop(2)	2.0 MPa	0.1 MPa
Total water holdup	~ 130 m^3/loop	
Pump - Pumping power	Main pump 3,660 kW / loop	Low flow pump 25 kW / loop
Heat exchanger size - Heat transfer area	234 MW 2,710 m^2/unit	
Pressurizer volume	16.5 m^3	
Heater	~ 1,400 kW	
Pressuriser relief tank volume	8.1 m^3	
Nominal Cool down rate of intact FW/Blanket cooling loops	to 50 °C in 36 hours	

(1) Coolant velocity = approx. 6 m/s
(2) Total pressure drop including in-vessel components, heat exchanger and piping

Table 4.11-3 Divertor Cooling

	Normal operation	Baking
Thermal power	SeeTable 3.6-3	
Inlet temperature	100° +5/-10 °C (Flat top)	240±10 °C (from Table 4.5-10)
Nominal outlet temperature	150°C	
Inlet pressure	4.2 ±0.2 MPa	4.4 ±0.2 MPa
In-vessel pressure drop	1.6 MPa	~ 0.02 MPa
In-vessel water holdup	~ 23 m^3/loop	
Loop number	1	
Flow rate/loop	1,000 kg/s	100 kg/s
Loop pipe inside diameter(1)	0.514 m	
Pressure drop(2)	2.3 MPa	~ 0.1 MPa
Total water holdup	~ 145 m^3/loop	
Pump - Pumping power	Main pump 3,670 kW / loop	Low flow pump 19 kW / loop
Heat exchanger - Heat transfer area	206 MW 2,410 m^2/loop	
Pressurizer size	23.0 m^3	
Heater	1,600 kW	
Nominal Cool down rate	to 50 °C in 36 hours	

(1) Coolant velocity = approx. 6 m/s
(2) Total pressure drop including in-vessel components, heat exchanger and piping

Table 4.11-4 NB InjectorCooling

	Low voltage components	High voltage components (ion source)	High voltage components (other)
Thermal power	86.9 MW	4.8 MW	10.5 MW
Coolant inlet temp.	75 °C	20 °C	55 °C
Coolant outlet temp.	~110 °C	40 °C(3)	95 °C
Coolant pressure	2.0 MPa	2.0 MPa	
In-Vessel pressure drop	1.0 MPa	0.9 MPa	
Loop number	1		
Flow rate	591.7 kg/s	48.8 kg/s	65.6 kg/s
Loop pipe inside diameter(1)	428.6 mm		
Pressure drop(2)	1.4 MPa		
Total water holdup	87 m^3		
Pumping power	1,460 kW		
Heat exchanger - Exchange heat - Heat transfer surface area	Main heat exchanger 87.7 MW ~2,310 m^2	Chilled cooler 6.7 MW ~ 430 m^2	Pre-cooler 9.2 MW ~ 560 m^2
Pressuriser size (Pneumatic)	4.7 m^3		

(1) Coolant velocity < 6 m/s
(2) Total pressure drop including LV/HV components, Heat exchanger and piping
(3) 40°C of the outlet temperature for the main components with 4.0 MW heat load

Table 4.11-5 Tokamak Cooling Water Chemistry (on CVCS) Main Specifications

Required Operational Parameters(2)	**Feed water**	**Upper limits for action**
Conductivity (at 20°C), µS/cm	<0.1	<0.3
Oxygen, µg/kg	<100	<10 (1)
Chloride and/or Fluoride, µg/kg	<0.5	<5
Sulphate, µg/kg	<20	<5
Copper, µg/kg	<0.5	<5
Iron, µg/kg	<1	<5
Hardness (Ca, Mg, etc.), µg/kg	<5	<5
Oil products, organic, µg/kg	<100	<100
pH at RT	neutral	-
H added to suppress radiolysis (cm^3/kg @ STP)	25	-

(1) The oxygen content for the vacuum vessel cooling system will be limited <100 µg/kg
(2) A plasma pulse will not be initiated if any of the above limits is foreseen to be exceeded during the course of pulse itself.

Table 4.11-6 Main Data for the CVCS of the FW/BLKT PHTS

Operational State or Mode	Plasma Operation State	Baking Operation State	Decontamina-tion Mode
Volumetric Flow Rate (at inlet)	45 m^3/h	45 m^3/h	90 m^3/h
Feed Coolant Temperature	148°C	240°C	100°C
Return Coolant Temperature	120°C	190°C	83°C
Feed Coolant Pressure	1.7 MPa	4.3 MPa	1.7 MPa
Return Coolant Pressure	3.4 MPa	4.6 MPa	3.4 MPa
Number of Inlet Filters in Service	1	1	2
Number of Resin Bed in Service	1	1	2
Number of Re-injection Pumps in Service	2	2	4
Loop Pipe Inside Diameter (1)	97.1 mm		
Recuperative Heat Exchanger - Exchange Heat (Max) - Heat Transfer Surface Area - Size - Number of Units	 6.5 MW 213 m^2 (0.8 m-D x 5.3 m-L) x 3 modules 1		
Letdown Cooler - Exchange Heat (Max) - Heat Transfer Surface Area - Size - Number of Units	 2.6 MW 178 m^2 1.1 m-D x 5.4 m-L 1		
Volume Control Tank - Size - Number of Units	 2.5 m-D x 3.5 m-H 1		
Inlet Filter - Size - Number of Units	 0.5 m-D x 2.3 m-H 2		
Resin Bed Demineraliser - Size - Number of Units	 1.4 m-D x 2.2 m-H 2		
Re-injection Pump - Pumping Power - Number of Units	 70 kW 4		

(1) Coolant velocity < 6 m/s

Table 4.11-7 Draining and Refilling System Tank Volumes

System	Number of drain tanks(1)	Total internal volume (m^3)
PFW/BLKT PHTS	2 (1)	~ 540
DIV/LIM PHTS	2 (1)	~ 220
VV PHTS	2	~ 120
NB injectors PHTS	2 (2)	~ 120

(1&2) number of tanks in parentheses indicates those used for drainage following an in-vessel coolant leak

Table 4.11-8 Main Data for the Drying System

	Blow-out mode	Heat-up mode	Dry-out mode
Fluid	Nitrogen	Nitrogen	Nitrogen + H_2O (steam)
Fluid inlet temp.	40 °C	210 °C	210 °C
Fluid outlet temp.	40 °C	160 °C	160 °C
Fluid inlet pressure	4.0 MPa	2.1 MPa	0.6 MPa
Number of loops	1		
Main piping size	ND 250		

Compressor - Compressor power	Magnetically floating bearing type		
	~500 kW, 1 unit	~300 kW, 1 unit	
Electrical heater - Heater power	Shell and baffle type, 1 unit 690 kW		
Cooler condenser #1 - Exchange heat	Cooled by HRS cooling water, 1 unit 370 kW		
Cooler condenser #2 - Exchange heat	Cooled by chilled water, 1 unit 190 kW		
Drain separator	Cyclone separator type, 2 units		
Mist separator	Vane separator type, 1 unit		
Filter	2 units, 3 μm (Mesh fineness)		

4.11.2.3 Design Parameters of the Component cooling Water System (CCWS)

Table 4.11-9 Heat Loads and Flow Rates for the Component Cooling Water System

System (Loop)	Clients		Heat Load of Clients		Flow Rates		Inlet/Outlet Temp. (°C)	
	Name	Location	MW	Sum.	(kg/s)	Sum[(1)]	In	Out
CCWS in tokamak and site service zone[(2)] [Loop CCW-1A]	PHTSs pumps	Tokamak building	3.4	10.3	162.5	332.8	40	45
	Components in tritium building	Tritium building	6.3		111.4		40	45 - 55
	Components in site services buiding	Site services building	0.6		28.7		40	45
CCWS in tokamak and site service zone [Loop CCW-1B]	IC H&CD components	Assembly hall	21.6	83.1	172.1	725.1	40	70
	EC H&CD gyrotron	Assembly hall	60.0		478.0			
	NB ACCC [(3)]	NB cell	1.5		9.0		40	80
CCWS in tokamak and site service zone [Loop CCW-1C]	IC H&CD power supply	Assembly hall	1.5	5.0	41.1	147.6	30	38.8
	EC H&CD power supply	Assembly hall	3.5		93.1			
CCWS in power supply zone [Loop CCW-2A]	Reactors	Power supply zone	5.8	16.2	69.7	216.3	40	59.7
	Busbars	Power supply zone	9.5		113.3			
	Switching network and fast discharge	Power supply zone	0.9		13.6			
CCWS in power supply zone [Loop CCW-2B]	RPCs [(4)] + coil converters	Power supply zone	7.5	17.1	255.6	642.6	30	37.0
	NB injector power supply	Power supply zone	9.6		328.6			

(1) Included 10% of additional flow rate for resin beds etc.
(2) Not included the heat loads such as a diagnostic components inside the port cells which are not defined
(3) Active Compensation/Correction Coils
(4) Reactive Power Compensating Units

Table 4.11-10 Component Cooling Water System Heat Exchangers

Loop	Heat Duty (MW)	Design Condition							
		Primary Side				Secondary Side			
		Coolant	**Temp. (In) (°C)**	**Temp. (Out) (°C)**	**Design Press. (MPa)**	**Coolant**	**Temp. (In) (°C)**	**Temp. (Out) (°C)**	**Design Press (MPa)**
CCW-1A (tokamak and site service zone)	~ 10.3	pure water	40	~48	~ 1.0	raw water	35	45	< 1.0
CCW-1B (tokamak and site service zone)	~83.1	pure water	40	~68	~ 1.0	raw water	35	60	< 1.0
CCW-1C (tokamak and site service zone)	~5.0	pure water	30	38	~ 1.0	pure water	6	12	~ 1.0
CCW-2A (power supply zone)	~ 16.5	pure water	40	~58	~ 1.0	raw water	35	50	< 1.0
CCW-2B (power supply zone)	~17.1	pure water	30	37	~ 1.0	pure water	6	12	~ 1.0

Table 4.11-11CCWS Supply Circuits

Component Cooling Water System (Loop)	**Pump**				**Main Pipe Size**
	No. of Unit	Flow rate (kg/s/unit)	Head (m)	Power (kW/unit)	
CCW-1A	2	~170	50	~130	ND 350
CCW-1B	2	~365	50	~280	ND 500
CCW-1C	3	~75	50	~ 57	ND 250
CCW-2A	2	~110	50	~85	ND 300
CCW-2B	3	~325	55	~270	ND 500

4.11.2.4 Design Parameters of the Chilled Water System (CHWS)

Table 4.11-12 Heat Loads, Temperatures and Flow Rates for the Chilled Water System

System (Loop)	Clients		Heat Load of Clients		Flow Rate		Inlet/Outlet Temp. (°C)	
	Name	Location	MW	Sum.	(kg/s)	Sum.	In	Out
CHWS in tokamak and site service zone [Loop CHW-1A]	Tokamak building HVAC and LAC	Tritium building	3.8	19.1	149.4	542.3	6	12
	Hot cell/ radwaste/ personnel building HVAC	Hot cell/ Radwaste building	3.0		118.3			
	NB injector chilled HX (for HV)	Tritium building	6.7		266.9			
	Drying system: condenser #2	Tritium building	0.2		7.6			
	Site service zone HVAC	Site service zone	5.0		199.2	219.1		
	Future allowance load	Site service zone	0.5		19.9			
CHWS in tokamak and site service zone [Loop CHW-1B]	CCWS for ICH&CD power supply	Assembly hall	1.5	5.0	59.2	197.5	6	12
	CCWS for ECH&CD power supply	Assembly hall	3.5		138.3			
CHWS in power supply zone [Loop CHW-2]	Buildings HVAC	Power supply zone	2.9	20.0	115.5	796.1	6	12
	CCWS for RPCs(1) + coil converters	Power supply zone	7.5		297.8			
	CCWS for NB injectors power supply	Power supply zone	9.6		382.8			
CHWS for safety-related [Loop CHW-S1]	TCWS vault cooler	Tritium building	1.2	2.8	47.8	122.7(2)	6	12
	Air conditioning in hot cell	Hot cell building	0.38		15.2			
	Tritium plant	Tritium building	0.96		38.4			
	Others	Tritium building	0.26		10.2			
CHWS for safety-related [Loop CHW-S2]	TCWS vault cooler	Tritium building	1.2	2.8	47.8	122.7(2)	6	12
	Air conditioning in hot cell	Hot cell building	0.38		15.2			
	Tritium plant	Tritium building	0.96		38.4			
	Others	Tritium building	0.26		10.2			

(1) Reactive Power Compensating Unit
(2) included 10% of additional flow rate for standby chillers

Table 4.11-13 Chilled Water System Chiller List

Sub-system	Capacity[1] (MW)	Chiller Units	Chiller Location	Remark
CHW-1A	~19.1	5 MW x 4	site services building	
CHW-1B	~5.0	5 MW x 1 or 3 MW x 2	tokamak building (assembly hall)	
CHW-2	~ 20	5 MW x 4	Cooling system building in power supply zone	
CHW-S1	~ 2.8	1 MW x 3	tritium Building	Safety-related (SIC)
CHW-S2	~ 2.8	1 MW x 3	tritium Building	Safety-related (SIC)

(1) Values indicate the heat loads of the clients. The heat load to the HRS is ~ 1.25 times larger.

Table 4.11-14 CHWS Supply Circuits

Chilled Water System (Loop)	Pump				Main Pipe Size
	No. of Units	Flow Rate (kg/s/unit)	Head (m)	Power (kW/unit)	
CHW-1A (1): for tokamak zone)	2	~ 275	40	~ 165	ND 450
CHWS-1A (2): for site service zone HVAC)	2	~ 110	40	~ 67	ND 300
CHW-1B	2	~ 100	40	~ 60	ND 300
CHW-2	2	~ 400	30	~ 190	ND 600
CHW-S1	3	~ 41	30	~ 19	ND 200
CHW-S2	3	~ 41	30	~ 19	ND 200

4.11.2.5 Design Parameters of the Heat Rejection System (HRS)

Table 4.11-15 Summary Water Circulation System Heat Loads and Flows (non-safety related)

System Name	Clients		Design Heat Loads in HRS		Flow Rates		Inlet/Outlet Temp. (°C)	
	Name	Location	MW	Sum.	kg/s	Sum.	In	Out
Primary heat transfer systems WCS [Loop 1]	FW/BLKT heat exchanger #1	TCWS vault	234	1,005	1,400	6,171	35	75
	FW/BLKT heat exchanger #2	TCWS vault	234		1,400		35	75
	FW/BLKT heat exchanger #3	TCWS vault	234		1,400		35	75
	DIV/LIM heat exchanger	TCWS vault	206		1,231		35	75
	NB injector heat exchanger	TCWS vault	88		600		35	70
	Pre-cooler for NB injector (HV)	TCWS vault	9.2		147		35	50
CVCS WCS [Loop 3]	FW/BLKT CVCS-1	TCWS vault	3.5	9.7	55.8	109	35	50
	DIV/LIM CVCS-1	TCWS vault	1.4		22.3		35	50
	NB injectors CVCS-1	TCWS vault	0.3		4.0		35	50
	Test blanket modules PHTSs	TCWS vault	4.5		27.0		35	75
Cryoplant compressor WCS [Loop 4]	Cryoplant compressor	Cryoplant building	30	30	1,024	1,024	35	42
Component cooling water system WCS-1 [Loop 5A]	PHTS pumps	Tokamak building	3.4	96.2	163	1,112	35	40
	Components in tritium building	Tritium building	6.3		112		35	40 - 50
	Components in site services building	Site services building	0.6		20.5		35	42
	Miscellaneous in site service zone	Site service building	0.6		5.7		35	60
	IC H&CD (components outside VV)	Assembly hall	21.7		208		35	60
	EC H&CD gyrotron	Assembly hall	60.2		576		35	60
	NB ACCC[(a)]	Tokamak building	1.5		9		35	75
	Test blanket modules Hxs in port cell	Tokamak building	1.7		10		35	75
	Cooler for cryopump regeneration	Tokamak building	0.2		9.6		35	40
Component cooling water system WCS-2 [Loop 5B]	Reactors	Power supply zone	5.9	24.6	95	590	35	50
	Busbars	Power supply zone	9.6		153		35	50
	Switching network and fast discharge	Power supply zone	0.9		14.4		35	50
	Switching network resisters	Power supply zone	8.2		328.3		35	41

(a) ACCC - Active Compensation/Correction Coils

Table 4.11-16 CS Heat Loads and Flow Rates (non safety-related) (cont'd)

System name	Clients		Design Heat Loads in HRS		Flow Rates		Inlet/Outlet Temp. (°C)	
	Name	Location	MW	Sum.	kg/s	Sum.	In	Out
Chilled water system WCS-1 [Loop 6A]	Chillers for Tokamak/ tritium buildings HVAC	Site services building	4.7	30.1	224	1,439	35	40
	Chillers for Hot cell/ radwaste/ personnel building HVAC	Site services building	3.7		178		35	40
	Chillers for NB injectors (for HV)	Site services building	8.4		400		35	40
	Chillers for IC H&CD power supply	Assembly hall	1.9		89		35	40
	Chillers for EC H&CD power supply	Assembly hall	4.3		207		35	40
	Chillers for Drying system: condenser #2	Site services building	0.24		11.5		35	40
	Chillers for Site service zone HVAC	Site services building	6.3		299		35	40
	Chillers for Future allowance load	Site services building	0.6		30		35	40
Chilled water system WCS-2 [Loop 6B]	Buildings HVAC	Power supply zone	3.6	24.6	174	1,177	35	40
	RPCs[(a)] + coil converters	Power supply zone	9.3		447		35	40
	NB injectors power supply	Power supply zone	11.7		557		35	40
HRS TOTAL	-	-	-	1,221[(b)]	-	11,620	-	-

(a) Reactive Power Compensating Units
(b) Total capacity in clients. Maximum heat load for CTS is ~ 1,140 MW

Table 4.11-17 CTS Thermal Design Parameters

Parameter \ Operation Condition	Reference Plasma		Non-Inductive
	Pulse	Dwell	Burn
Peak heat duty (MW)	~ 1,150	~ 150	~ 900
Duration (s)	~ 400	~ 1,400	3,000
Flow rate (kg/s)	~ 11,620	~ 11,620	~ 11,620
Peak inlet temperature (°C)	~ 60	~ 38	~ 55
Maximum outlet temperature (°C)	35	35	35
Wet bulb temperature (°C)	29	29	29
Hot basin volume (m^3)	12,000		
Cold basin volume (m^3)	20,000		

Table 4.11-18WCS Supply Circuits

Water Circulation System	Pump				Main Pipe Size
	No. of Units	Flow Rates (kg/s/unit)	Head (m)	Power (kW/unit)	
FW/BLKT, DIV/LIM & NB injector	4	~ 1,550	55	~ 1,300	ND 1,500
CVCS	2	~ 55	55	~ 50	ND 250
Cryoplant compressor	2	~ 515	25	~200	ND 600
Component cooling water					
(1) tokamak and site service zone	2	~ 555	40	~ 340	ND 700
(2) power supply zone	2	~ 295	30	~ 135	ND 450
Chilled cooling water system					
(1) tokamak and site service zone	2	~ 720	20	~ 220	ND 750
(2) power supply zone	2	~ 590	20	~ 180	ND 700

4.12 Thermal Shields (WBS 2.7)

(Updated by KII YUN Aug-Dec. 2006)

4.12.1 Functional Requirements

The function of the thermal shield is to limit the heat load to the superconducting coils from warm internal/external sources to levels that can be tolerated by the coils and reasonably removed by the helium cryogenic system.

In particular, the shields intercept thermal radiation from the warm surfaces of the vacuum vessel and ports outer walls, the VV supports, cooling pipes and other warm ducting, and the cryostat inner wall as well as restricting heat loads transferred by conductance through the magnet gravity supports by means of thermal anchors.

4.12.2 Configuration

4.12.2.1 Overall

The thermal shields comprise of (see Figure 4.12-1):

- the upper cryostat thermal shield (upper CTS),
- the lower cryostat thermal shield (lower CTS),
- the central thermal shield (central TS), and
- the support thermal shield (STS) which includes the side support thermal shield (SSTS), and the front support thermal shield (FSTS).

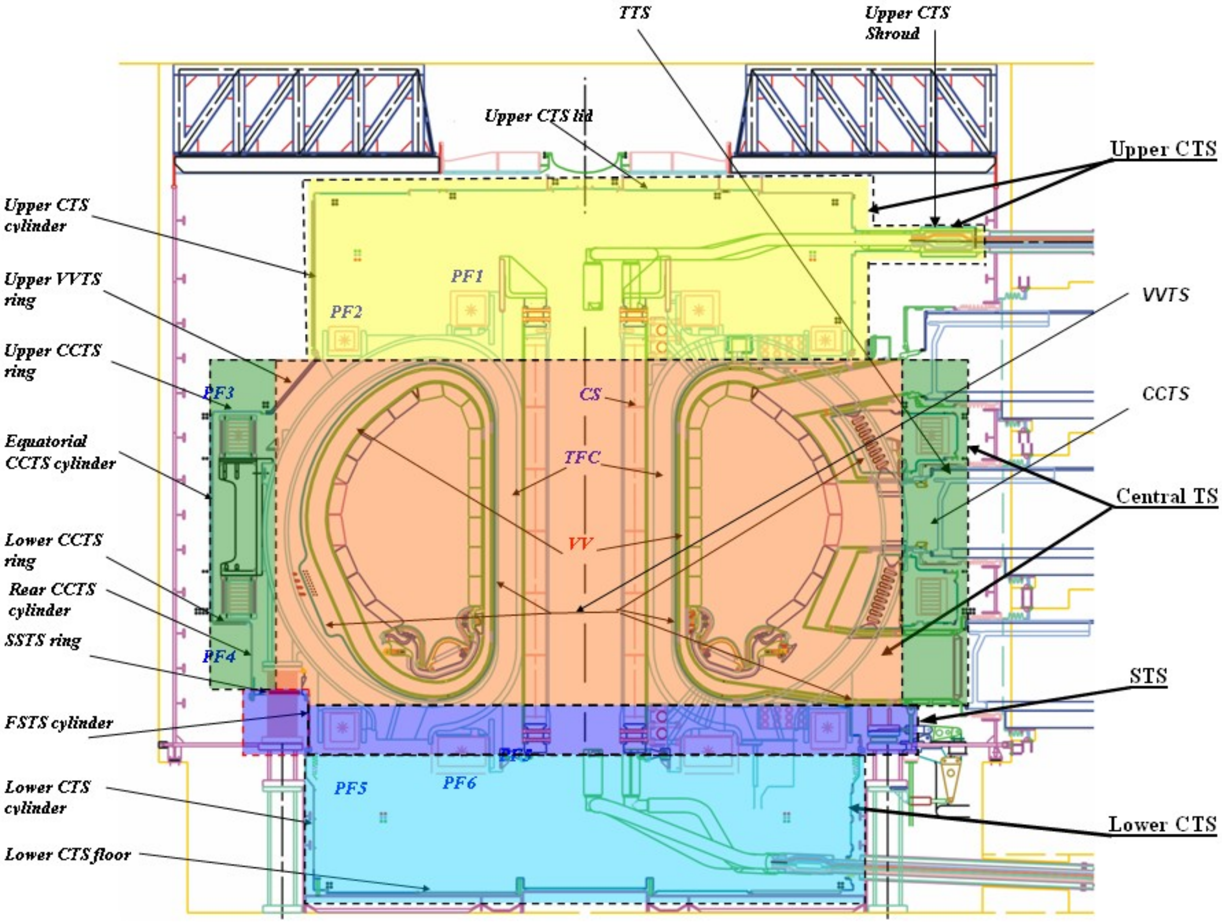


Figure 4.12-1 Main components of the thermal shields

The above components are supported independently and not connected to each other structurally. At the boundary areas of the components, there are labyrinth interfaces which prevent a direct shine-through of the warm surfaces to the cold ones.

The central TS consists of:

- the vacuum vessel thermal shield (VVTS), which interposes between the VV and the cold magnet structures, and
- the central cryostat thermal shield (CCTS), which prevents direct line of sight from the room temperature cryostat walls to the cold structures,

The Central TS is self-standing and is externally supported on the toroidal field coils (TFC's) by inboard and outboard supports. The inboard supports are slender Inconel 718 strips allowing radial and toroidal movements, whereas on the outboard side inclined stainless steel rod-type supports are used to fix the radial and toroidal positions of the Central TS.

The STS prevent radiation to/from the magnet gravity supports (MGS). The SSTS panels are located around the MGS and are supported on the upper flange of the MGS. The FSTS panels are located behind the supports forming a complete ring with openings for the lower port extensions. In addition, the thermal anchors in the magnet supports limit the heat load to cold structures due to conduction through the support structures.

The main material used for the thermal shields, including pipes and ancillaries is stainless steel type 304L (UNS S30403).

4.12.2.2 Central Thermal Shield

The central thermal shield includes the VVTS with special design features and the CCTS with a design typical for the cryostat thermal shield.

4.12.2.2.1 Vacuum Vessel Thermal Shield (VVTS)

The VVTS follows the shape of the VV and the VV sector segmentation. The 40-deg. sector of VVTS consists of inboard and outboard segments connected by electrically insulated joints at the top and bottom (these joints represent poloidal current breaks, which disturb electrical currents flowing in the poloidal direction). Each segment includes additionally two toroidal current breaks, which disturb currents flowing in the toroidal direction (running in the sections avoiding the TS port envelopes). Additionally, the outboard segment includes a joint in central section (without electrical insulation) due to the assembly reasons.

The inboard segment consists of the single wall panels (20 mm thick) whereas the outboard segment consists of the double wall panels (with 10 mm thick inner shell and 5 mm thick outer shell). The cooling pipes are attached to the outer surface of the segments. Two identical circuits of the cooling pipes are envisaged for each component based on the principle of the cooling redundancy.

The VVTS extends towards the CCTS with enclosures around the VV ports. The thermal shields around the lateral upper ports of the VV sector and the lower local penetrations include additional joint to allow the TFC assembly with the VV Sector in the assembly hall. The thermal shields around all equatorial ports also include additional joint to allow access to the back-side of the TFC intercoil structure. At the upper level, the enclosures around the ports are connected with the upper VVTS ring intercepting the upper port connecting ducts.

During the sector assembly into torus, the final joint of the VVTS sectors is made through the narrow splice plate gap of the vacuum vessel for inboard part, top and bottom, and from the outside for other areas. This joint does not include toroidal current break and may be accomplished with the welded splice plate.

Sufficient gaps are to be ensured between the VVTS and the VV/TFC during all modes of operation. For the Category I-III events, the gaps are to be always ensured and for the Category IV events a zero-gap (i.e., the surface touching) is acceptable. To avoid significant damage of the VVTS panels, special bumpers or stoppers may be located in key positions.

4.12.2.2.2 Central Cryostat Thermal Shield (CCTS)

The CCTS includes the following sub-components structurally connected to each other:
- the equatorial CCTS ring;
- the transition thermal shields (TTS) that include enclosures of the equatorial port extensions with connecting ducts,
- the upper CCTS ring above the PF-3
- the lower CCTS ring below the PF-4 and
- the rear CCTS ring around the lower port extensions and local penetrations.

It should be noted that a number of the CCTS sub-components may be decreased depending on the assembly method/sequence selected finally (TBD).

The CCTS comprises of single wall stainless steel panels reinforced where required, and attached to a frame structure. The cooling pipes are attached to the outer surface of the panels. The frames are interconnected by structural joints to form a self standing torus of thermal shield components. The removable panel is designed for the regions where access for inspection and eventual repair to the Magnet is required.

As a part of the CCTS, the TTS is attached in the pit to the end of the VVTS ports by means of the joints allowing compensating of mismatches (with splice plate joints – TBD). Similar type of joints may be used for joining of the CCTS sectors in the pit and the CCTS components to each other (TBD).

The CCTS has toroidal current breaks located in the same sections as the VVTS breaks. All other joints between the CCTS components and the VVTS and CCTS do not require electrical breaks.

4.12.2.3 Upper/Lower Cryostat Thermal Shield and Support Thermal Shield

The design concept for the upper/lower CTS and the STS is similar to that for the CCTS. Also similar, the removable panels are designed for the regions where access for inspection and eventual repair to the Magnet is required.

The upper CTS consists of the upper CTS lid, upper CTS cylinder and five shrouds for magnet feeder-lines (see Figure 4.12-1) . The Lower CTS includes the TS of the cryostat floor and the cryostat lower cylinder. The upper and lower thermal shields are self-standing, and fixed on the cryostat components via low-conductivity titanium alloy supports.

4.12.2.4 TS Overall Thermal-hydraulic Operation

In all cases the thermal shields consist of stainless steel panels that are cooled by 1.8 MPa pressurised helium gas from the main cryoplant in circular tubes with an 80K inlet temperature. To reduce the heat load radiated to the 4K surfaces, as well as the heat absorbed by the TS due to radiation from warmer surfaces, the thermal shield panels are covered on both sides with a thin, low emissivity layer of silver with a minimum 5 μm thickness.

The heat loads for the plasma operation state (POS) and VV baking operation state (BOS), are summarised in Table 4.12-1. The total 80K helium mass flow rate for all thermal shields is approximately 2.7 kg/s, which is determined to control TS outlet coolant temperature approximately at 100 K during POS. The pressure drops of 88 and 98 kPa for POS and BOS respectively are the reference design values of the thermal hydraulic parameters. The total heat load on the magnet system is about 4.1 kW and 5.7 kW at POS and BOS respectively

Table 4.12-1 Thermal Shield Thermo-hydraulic Data

Parameters		Unit	Value
Heat load to magnets	POS	kW	4.1
	BOS	kW	5.7
Heat load to thermal shield	POS	kW	280
	BOS	kW	500
80 K helium flow rate		kg/s	2.7

4.12.2.5 Loading conditions

See also § 3.11

4.12.2.5.1 TS Loads

The loads acting on the TS components can be divided into four categories:

- Inertial loads: these are defined based on accelerations due to gravity and seismic events.
- Electromagnetic (EM) loads: these act upon nearly all conductive structures during fast transients (e.g. plasma disruptions, VDE's, and magnet current fast discharge).
- Pressure loads: these include coolant pressure as well as the testing pressure in the pipes.
- Thermal loads: these are caused by temperature gradients inside the TS structure caused by thermal radiation and conduction.

Reference loads through interfaces of the TS can be found in § 3.11 (some parameters TBD)

4.12.2.5.2 TS Displacements

Displacements of the TS components during operation can be found in § 3.12 (some parameters TBD).

4.12.2.6 TS Deviations

Table 4.12-2 Fabrication and Assembly/Positioning Tolerances

Parameter	Unit	Value
VVTS		
Fabrication tolerance at factory - Sector overall height - Sector overall width - Surface deviations of a 40-degree sector from the reference geometry after fabrication at factory - Flanges of the bolted joints - Port envelopes	mm	±10 ± 10 ± 10 (Same as (1) below) ± 2 ± 6
Assembly/Positioning tolerances at site - Surface deviations of the torus from the reference geometry after positioning at the pit - Surface deviations of the torus from the reference geometry after assembly at the pit (Final deviations)	mm mm	± 13 (Sum of (1) and (2)) ± 16 (Sum of (1) to (3))
- Details		
(1)Surface tolerances of a 40-degree sector from the reference geometry after fabrication at factory	mm	± 10
(2)Sector positioning with all support fixtures removed	mm	± 3
(3)Weld distortion due to field welds at the site	mm	± 3
(4) Mismatch of the sector surfaces at field joints	mm	± 5
CCTS, STS and Upper/Lower CTS		
Fabrication tolerance at factory - Individual panels - Frames	mm mm	± 5 ± 3
Assembly/Positioning tolerances at site - Positioning tolerance - Mutual deviation between frames to be compensated by in-pit joints	mm mm	± 3 ± 12

4.12.2.7 TS Manufacturing

4.12.2.7.1 VVTS

The VVTS is to be fabricated in the factory as 9 sectors each spanning 40°. The sub-components will be formed, welded, machined, silver coated and assembled forming three sub-assemblies (the inboard segment and two halves of outboard segments). A special attention is to be paid to the silver coating – a main coating method (e.g., electroplating) and methods for repair should be studied carefully before the final selection. At this stage, the testing of toroidal current breaks can be performed. After fabrication, the sectors will be pre-assembled to check the assembly procedures and test the poloidal current breaks. The dimensional control and adjustment, as well as the leak and pressure tests can be performed at this stage. Then, the sectors will be disassembled forming three sub-assemblies for shipping to the ITER site.

4.12.2.7.2 CCTS, STS and Upper/Lower CTS

In general, the 40°-segmentation will be kept. The sub-components (like individual panels) will be formed, welded, machined, silver coated and assembled to the frames forming segments of corresponding TS components. At this stage, the testing of toroidal electrical breaks can be performed. After fabrication, the sectors (40°) will be pre-assembled to check the assembly procedures on-site. The dimensional control and adjustment, as well as the leak and pressure tests can be performed at this stage. Then, the sectors will be disassembled for shipping to the ITER site.

4.13 Vacuum Pumping (WBS 3.1)

(Updated M.Wykes, Sept/Nov 2006)

Editors Note: DDD 3.1 has not been updated for some years because of lack of resources. Hence this PID contains the main vacuum pumping system data. When DDD 3.1 is updated, a substantial portion of this PID section will be moved into the DDD

4.13.1 Functional Requirements

The vacuum pumping system provides the following functions:

Roughing System

1. Roughing of the torus, NB, Service Vacuum System (SVS), type 2 diagnostics and cryostat from atmosphere to the cross-over pressure of their respective cryopumps,
2. exhaust of all pumped gases towards Tokamak exhaust processing (tritium plant),
3. evacuation of released gases from the torus, NB, and cryostat cryopumps during partial and total regeneration and from the cryo-cooler (i.e. cryo-refrigerator) pumps of the Service Vacuum System (SVS) and Direct Coupled Diagnostics during regeneration and delivery to tokamak exhaust processing (tritium plant)
4. control of the water vapour partial pressure in the roughing pumps to prevent condensation within the roughing system and tritium plant,
5. allow admission and exhaust of ventilation flow during in-vessel maintenance.
6. sequential pumping/purging to dehydrate the cryostat internals prior to magnet cool-down,
7. pumping of Torus, NB, Service Vacuum System or cryostat during leak testing to allow timely leak testing completion in accordance with the Integrated Project Schedule
8. pumping of volumes subject to water ICE to assist with drying.

Torus Vacuum Cryopumping System

1. Evacuation of the torus from the crossover pressure to the pre-bake base pressures,
2. pump-out of gases released from the vacuum boundary during baking and conditioning (glow discharge cleaning, IC/EC discharge cleaning, reactive cleaning),
3. intra-pulse pumping to remove excess fuelling gas, impurities and helium,
4. pumping between pulses to attain a low enough pressure for pre-fill and breakdown (for some plasma scenarios with assistance from neutral beam cryopumps),
5. enable leak detection of the torus and components within its vacuum boundary,
6. pumping during recovery of tritium from PFC co-deposited layers,
7. provide the required pressure environment for the IC/EC launchers (the designers of these systems are responsible for providing adequate particle conductance between the launchers and the vacuum vessel to handle the total launcher outgassing)

NB Cryopumping system

1 Intra-injection removal of excess NB fuelling gas and impurities,
2 assist the torus cryopumps to attain the terminal dwell pressure,
3 enable leak detection of the neutral beam vacuum boundary and components therein.

Cryostat Cryopumping System

1 Evacuation of the cryostat to high vacuum prior to the cool-down of the magnets,
2 pumping of helium leaks from the magnet and tokamak thermal shields cooling circuits,
3 pumping of protium from long term outgassing from warm in-cryostat metallic surfaces,
4 pump and reveal external air leaks to mitigate the ozone hazard,

5 allow inferential measurement of the quantity of external air inleakage pumped by the 5 K surfaces of the magnets and structures,
6 facilitate leak detection of the cryostat by enabling establishment of appropriate high vacuum conditions for leak testing (and adequately cleaning up the tracer helium that enters the cryostat vacuum through detected leaks) in a timely manner in accordance with the Integrated Project Schedule.
7 pumping of gases generated by irradiation of exposed epoxy (e.g. protium and alkanes)

Service Vacuum System

1 Providing vacuum roughing, leak detection and guard vacuum where needed by clients which are not tritium bearing during normal operation, including
- wave heating systems,
- non-tritium (type 1) diagnostics systems,
- pumping of interspaces between feedthroughs deemed sufficiently fragile to need a second vacuum barrier to enable differential pumping, to allow leak mitigation and continued plasma operation if the reliability of a single feed-through would otherwise not be adequate to meet the overall availability requirements of the tokamak or would result in excessive individual and collective maintenance worker dose during leak localisation and repair,

2 Provision of controlled venting of the torus, cryostat and all other vacuum systems,
3 Conveying all gases from the service vacuum system back to the tritium plant.

Type 2 Diagnostic Cryopumping System (See Section 4.13.3.8)

a. Providing accumulation vacuum pumping where needed by Type 2 diagnostics.

Leak Detection Systems

1. Provision of the leak testing of the vacuum integrity of all major systems which may be exposed to primary, service or cryostat vacuum, both during construction and operation,
2. measurement of the total in-leakage into the vacuum vessel, cryostat and service vacuum clients, (in particular for external air leaks into the vacuum vessel and cryostat to allow reliable plasma start-up and acceptable radiative power loss for the former and conformance with the safety limits for potential ozone formation for the latter) both during construction and operation,
3. measurement of the in-situ leak rates of individual elements, both during construction and operation,
4. provision of the capability to locate, at the component and element level, i.e. blanket module, divertor module, vacuum vessel port etc., any unacceptable leak rate to allow the intervention of remote maintenance activities to substantiate the source of the leak and implement corrective action, both during construction and operation,
5. Provision of the capability for leak testing, in the operational configuration, the various systems of the torus, cryostat and pellet injector during the final construction and commissioning of the machine and after upgrade, repair or maintenance,
6. Periodic confirmation of the integrity of the radiological confinement boundaries to comply with regulatory requirements.

4.13.2 Overview of ITER Pumping Systems

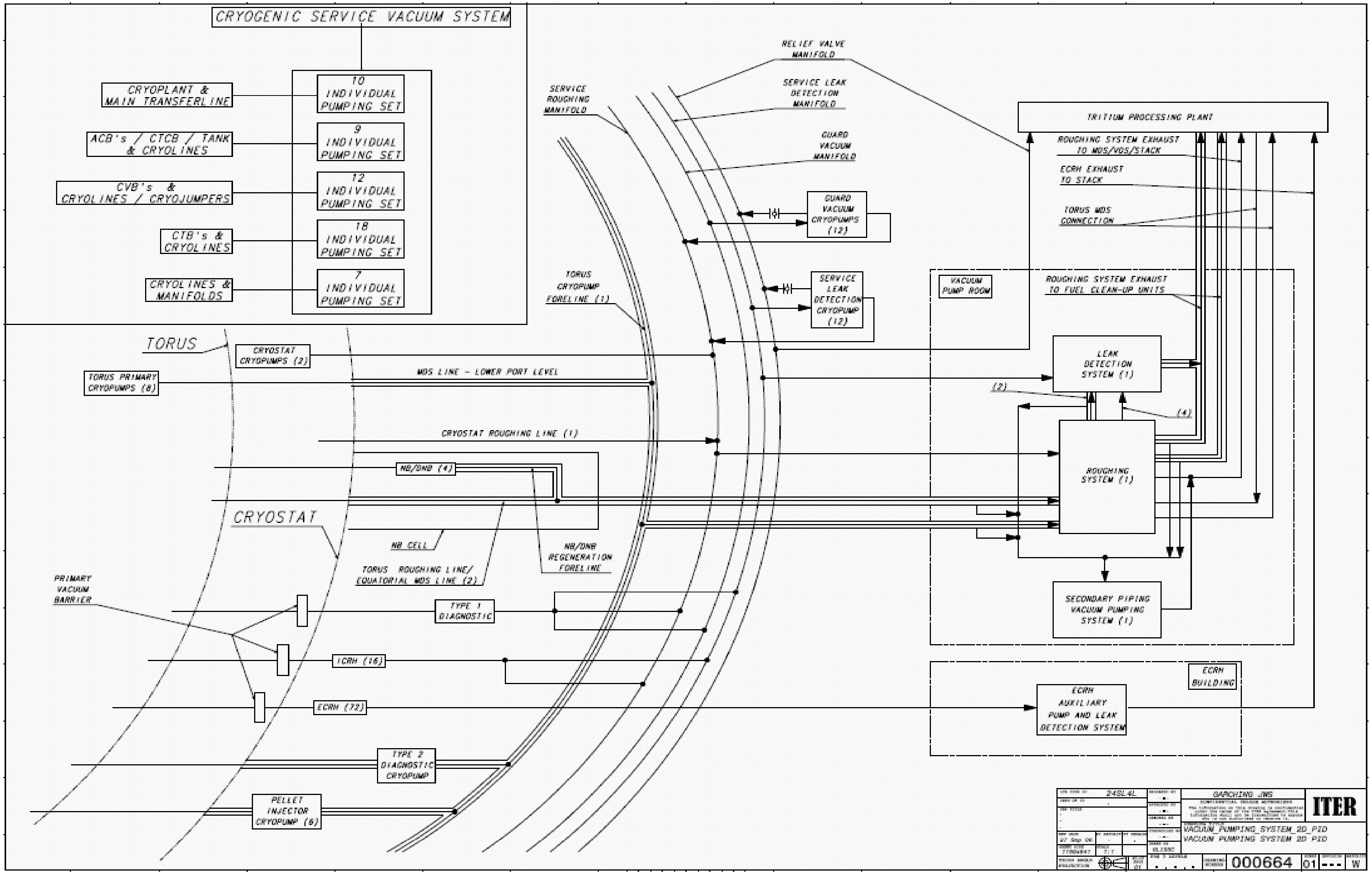


Figure 4.13-1 Overview of ITER Pumping Systems

4.13.3 Configuration

4.13.3.1 Torus Vacuum and Plasma Pumping

The torus is pumped down from atmospheric pressure to the nominal crossover pressure of 10 Pa by a mechanical forepump set located in the vacuum pumping room of the tritium plant. The torus roughing system is designed to evacuate the torus from atmospheric pressure to 10 Pa within < 24 h, in order to allow clean-up of tracer helium from the torus and thereby enabling leak testing on an acceptable timescale for the Integrated Project Schedule. This evacuation time is achieved with the torus vented to dry air or nitrogen as vent gas. In the event that the neutral beam system is provided with a fast shutter rather than an absolute valve (currently being studied as an option) the pump-down has to include the beamline vessel volume.

After bake-out and conditioning, the value of pressures and integrated primary leak rates are shown in the following table:

Table 4.13-1 Torus Vacuum Condition

Parameters	Unit	Value
Base pressure for hydrogen isotopes	Pa	$<10^{-5}$
Base pressure for impurity gases	Pa	$<10^{-7}$
Integrated global air leak rate into the primary vacuum boundary(1)	Pa m^3/s	10^{-6}
Note (1) Based on O_2 limiting in-leak for acceptable plasma start (PDD 2.7.4.1). The full bore plasma radiative power loss versus global air inleakage rate is currently being studied (Oct 2006).		

Table 4.13-2 Plasma Pumping Requirements

Parameters	Unit	Value
Typical divertor pressure during plasma operations	Pa	1-10
Maximum throughput during plasma operations	Pa m^3/s	153(1)
Minimum He pumping speed during plasma operations (molecular flow)	m^3/s	30-40
Base pressure between pulses	Pa	$< 5x10^{-4}$ (2)
Base impurity pressure between pulses(3)	Pa	$< 10^{-5}$
Pumping speed regulation, 0 – 100%	s	< 10
Note (1) The net pumping speed from the divertor volume must be adjustable between 0 and the maximum achievable to provide particle control. Note (2) For sequential 400 s burn pulses at maximum repetition rate, two neutral beam pumps assist the torus cryopumps in achieving the required base pressure at the end of the 1400 s dwell period. Note (3) The sum of partial pressures of impurity gases (e.g. 18, 28, 44)		

The torus roughing pumps will exhaust directly to the tritium plant which will be capable of processing the plasma exhaust at the average composition and flow rates detailed in the table below during the various phases of plasma operations. The outgassing species listed in these tables are the result of plasma wall interactions, outgassing, and other mechanisms e.g. chemical erosion, sputtering, minor water leaks, and the long term release of vent gases etc. that occur.

Table 4.13-3 Maximum Plasma Exhaust Composition and Flow Rates during Discharges

(Time averaged) (6)

Gas Species	He Discharge Pa m^3/s	H_2 Discharge Pa m^3/s	D_2 Discharge(3) Pa m^3/s	DT Discharge(4) Pa m^3/s
H_2	10	120	10	10
D_2			120	0
T_2			0	0
DT			0	120
He	60	10	10	10
$\Sigma(H_2+He)$	70	130	10 (5)	10 (5)
C_xQ_y(1)	5	5	5	5
Q_2O(1)	1	1	1	1
O_2	1	1	1	1
CO_x	5	5	5	5
NQ_3(1)	1	1	1	1
N_2	10	10	10	10
Ar	10	10	10	10
Ne	10	10	10	10
$\Sigma(N_2+Ar+Ne)$(2)	10	10	10	10
Total	93	153	153	153

Note (1) Q is defined as any one or combination of the hydrogen isotopes H, D or T. The composition of hydrocarbons will be based on data from existing machines.

Note (2) The maximum combined exhaust flow rate for N2, Ar and Ne.

Note (3) The table excludes the production of trace amounts of tritium from the DD reaction of < 0.01% of the full throughput.

Note (4) The proportion of T/D may vary over the range 0/100 to 90/10, with the flow rate decreasing progressively from the nominal value above 50/50.

Note (5) The total flux of helium and hydrogen during D and DT discharges does not exceed 10 Pam^3/s

Note (6) During long pulse operation the 8 torus cryopumps operate in a staggered pattern with 4 pumping and 4 regenerating with each one pumping for 600s and then regenerating for 600s prior to its next 600s pumping cycle. The throughput is averaged over the 600s pumping cycle to keep the peak hydrogenic pump inventory below the safety deflagration limit and the administrative tritum inventory limit of 120 g for all open pumps.

4.13.3.2 Cryostat Vacuum

The cryostat is evacuated to crossover pressure by a mechanical forepump set located in the vacuum pumping room of the tritium plant and from crossover pressure to base pressure by two cryostat high vacuum cryopumps located in lower port cells 3 and 11. The pressures and primary leak rates are as follows.

Table 4.13-4 Cryostat Vacuum Condition

Parameters	Unit	Value
Number of forepumps		1
Number of cryopumps		2
Nominal cross over pressure from forepumps	Pa	10
Nominal helium pressure inside cryostat prior to and during operations	Pa	$\geq 10^{-4}$
Maximum global in-leakage into the cryostat including all internally-mounted components *before* initiating operation	Pa m^3/s	$\geq 10^{-4}$
Maximum global helium in-leakage into the cryostat including all internally-mounted components *during* tokamak operations(1)	Pa m^3/s	10^{-2}
Global air in-leakage into the cryostat *during* tokamak operations(2)	Pa m^3/s	10^{-4}
Base pressure of water before cooldown (TBC)	Pa	$\geq 2x10^{-7}$
Base pressure of other impurity gases before cooldown	Pa	$< 10^{-7}$

Note (1) Maximum helium leak to limit helium pressure to 10^{-4} Pa with 2 cryopumps pumping with effective pumping speed of ~100 m^3/s (based on torus cryopump value).
Note (2)) The nominal air leak rate to keep 5 K air condensation heat load low enough to allow maintenance of magnet 5 K standby operation of is ~$4x10^{-2}$ Pa m^3/s.

Provision will be made to vent the cryostat with dry air or N_2 to remove residual water, prior to high vacuum cryopumping of the cryostat. The cryostat forevacuum system shall evacuate the cryostat from atmospheric pressure down to the crossover pressure of 10 Pa in < 24 h (following venting to dry air or nitrogen), in order to allow clean-up of tracer helium from the cryostat vacuum and enabling leak testing on an acceptable timescale for the Integrated Project Schedule.

4.13.3.3 Additional Heating Pumping Systems

Table 4.13-5 Additional Heating System Pumping Requirements

Parameters	Unit	Value
Neutral Beam H and CD injectors (1)		
NB, pumping speed in D_2 for 1 NB Injector	m^3/s	2.6×10^3
NB, pumping speed in H_2 for 1 NB Injector	m^3/s	3.8×10^3
NB, overall regeneration time	s	<1350
Roughing system crossover pressure for H2 (D2) (2)	Pa	30 (20) TBD
Roughing system time to reach crossover pressure (NBs can be roughed down either via their regeneration forelines with the NB gate valves closed, if the gate valve option is adopted, or together with the torus roughing with NB fast shutters open, if the fast shutter option is adopted).	h	≤24 (3)
Note(1) The NB cryopumps are under re-design to comply with top-opening maintenance access requirements. Note(2) For partial regeneration under some plasma scenarios a crossover pressure of 30 Pa for H2 and 20 Pa for D2 is unattainable on account of lack of available pumping time. Maximum crossover pressure for acceptable cool-down SCHe liquefaction power is being studied. Note(3) In order to allow clean-up of tracer helium from the cryostat vacuum and enable leak testing on an acceptable timescale for the Integrated Project Schedule.		
Ion Cyclotron Vacuum Transmission Lines		
Pumping system required for IC VTLs		
IC Vacuum Transmission Line pressure	Pa	$<10^{-2}$
Electron Cyclotron Waveguides		
Roughing and High Vacuum pumping to be provided for waveguides		
Roughing system cross-over pressure	Pa	TBD
Roughing system time to reach cross-over pressure	days	< 1
High vacuum operating pressure	Pa	$< 10^{-3}$ Pa
Time to achieve high-vacuum	days	< 2
Required Component Outgassing rate (cleaning and conditioning of surfaces)	$Pam^3/s/m^2$	$< 10^{-4}$
Lower Hybrid H&CD		
Pumping systems to be provided for transmission line sections in port plugs to prevent RF breakdown		
Base pressure	Pa	10^{-4}
Maximum operating pressure	Pa	10^{-3}
Maximum Outgassing rate	Pam^3/s	10^{-4}

4.13.3.4 Torus Pumping Operation

Table 4.13-6 Torus Cryopumps Configuration and Operation Strategy

Item	Value	Comments
Total Number of Torus Cryopumps	8	2 per port.
Lower Port-direct pumps Branched pumps at adjacent lower ports	4,6,12,18 (5,7,13,1)	Note asymmetry (See **Table 3.5-3** port allocation)
Number of "Direct" (d) pumps	4	Radial pump duct.
Number of "Branched" (b) pumps (on side arm to radial duct)	4	Physically in adjacent lower port cells 5, 7, 13 and 1
Individual D2 Pump speed at pump	m^3/s	55
Divertor dome pressure at reference maximum throughput of 153 Pam^3/s	~1 Pa (TBC)	Depending on on-going assessment of divertor sealing and internal gas flows
Number of "simultaneous" pumps open during staggered operation	4	Pattern in table below. 2 direct and 2 branched open at any one time.
Duration of pumping and regeneration cycles for each pump	600s	During staggered operation.
Incremental time for each isochronous stage of pumping and regeneration	150s	Chosen to satisfy safety related criteria, throughput and minimization of cryogenic heat loads and maximisation of gas release duration

Table 4.13-7 Staggered pumping mode during plasma discharges

Pump No.(1)	Time intervals of 150 s (Incremental Time) from start of plasma discharge																
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
12d	**HR**	**W**	**P**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**
13b	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**
4d	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**
1b	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**
6d	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**
5b	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**
18d	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**
7b	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**	**E**	**C**	**P**	**P**	**P**	**P**	**HR**	**W**

Pump status			
	P	=	Pumping torus
	HR	=	Regeneration step 1, cold helium recovery
	W	=	Regeneration step 2, rapid warm up/gas release
	E	=	Regeneration step 3, evacuation
	C	=	Regeneration step 4, cool down

Note (1) The number denotes the pump lower port location and the character denotes a direct (d) or a branched (b) pump. There are always 2 direct and 2 branched pumps pumping to avoid an oscillation in effective pumping speed at the divertor, since the duct conductances of the branched pumps are somewhat lower than those of the direct pumps.

4.13.3.5 Torus and Cryostat Inlet Valve Pneumatic Actuation System

The positions of the inlet valves of each of the torus and cryostat cryopumps (from fully open to fully closed) are adjusted by means of a double-acting pneumatic cylinder directly mounted at the port cell end of each inlet valve actuator shaft. The working gas is compressed to 1.8 MPa by a dedicated compressor located in the tritium plant (TBD) from where it is conveyed to and from the 10 cryopump actuators by 2 ring manifolds located in the tokamak pit.

4.13.3.6 Torus, Cryostat and Neutral Beam Pumps-Sorbent Regeneration

Table 4.13-8 ITER Sorbtion Cryopump Details

ITEM	Details
Torus Neutral Beam and Cryostat Pump sorbent	500 g/m^2 of activated charcoal bonded to quilted panels by inorganic adhesive. Cooled by SCHe at 4.5 K
Release of hydrogen isotopes and helium from sorption panels	Gas release by heating sorption panel to 100 K (combination of pressure & temperature desorption)
Regeneration frequency for torus cryopumps during DT plasma operation	After 10 min. accumulation of torus exhaust pumping at the reference 50D-50T throughput of 153 Pam^3/s in the pattern in Table 4.13-7 (1)
Regeneration frequency for torus cryopumps during helium plasma operation	After 10 min. pumping at the reference throughput of 60 Pam^3/s with 4 pumps pumping in parallel; for longer pumping duration, staggered operation is used (2)
Regeneration frequency for NB cryopumps during sequential H2 and D2 plasma operation at the maximum repetition rate	In accordance with tables below for the nominal plasma scenarios
Release of “air-like” impurities from sorbent (NB, cryostat and torus cryopumps)	Heating to 300K and pumping (combination of pressure and temperature desorption) (4).
Expected frequency for torus cryopumps	Daily to refresh sorbent from air leaks and accumulated impurities as given in Table 4.13-3
Expected frequency for NB cryopumps	Weekly (TBC) to refresh sorbent from air leaks into NB vessel or torus during dwell pumping
Expected frequency for cryostat cryopumps	Daily as part of the cryostat Ozone hazard mitigation (O_2 accumulation monitor)
Release of water-lik impurities from torus, cryostat(5) and NB cryopumps	Heating to 475K and pumping (combination of pressure and temperature desorption).
Expected frequency for torus cryopump	≤Weekly (6) to refresh sorbent from water in normal torus exhaust and torus water leaks.
Expected frequency for NB cryopumps	≤Weekly (7) (TBC) to refresh sorbent from pumped water leaks in torus and NB

Note (1) The pumping duration for 50D50T operation is limited by safety requirements.
Note (2) The pumping duration for helium plasma operation is limited by sorbent helium capacity
Note (3) The terminal pressures for D2 and H2 are TBD will be decided on the criteria of acceptability of cool-down liquefaction rate (under study). The exact terminal pressure depends on the time evolution of the gas release from the cryo-sorbent.
Note (4) Includes desorption of methane and ethane from irradiated epoxy exposed to cryostat vacuum
Note (5) A high proportion of the outgassing from warm exposed coil epoxy is water which will progressively degrade the cryostat cryopump sorbent which will consequently need periodic regeneration at 475 K during warm evacuation prior to magnet cool-down.
Note (6) To regenerate water in the normal torus exhaust stream and small water leaks issuing from the torus water cooled components following RGA indication of an unacceptable in-vessel water leak. One or more torus cryopumps are used to integrate torus water leaks and will be regenerated nightly at 475K to monitor leaks. Water-likes include alkenes up to C8
Note (7) To regenerate water leaks issuing from the torus during dwell-assist pumping and from beamline water cooled components

Table 4.13-9 Thermal cycle Elevated temperature regeneration of sorbent(1)

For the 8 torus and 2 cryostat cryopumps

Thermal stage	Duration (h)
Heat 8 torus and 2 cryostat cryopumps to from 80 K to300 K	1.33
Heat 8 torus and 2 cryostat cryopumps from 300 K to 475 K(2)	1.33
Cool 8 torus and 2 cryostat cryopumps from 470 K to 300 K	1.33
Cool 8 torus and 2 cryostat cryopumps from 300 K to 80 K	4
Total cycle time	8 h
Note (1) Required for regeneration of air-likes (300 K) and water-likes (475 K) Note (2) The total cycle time is reduced when only heating to 475 K for water regeneration	

Table 4.13-10Neutral beam cryopump operating strategy

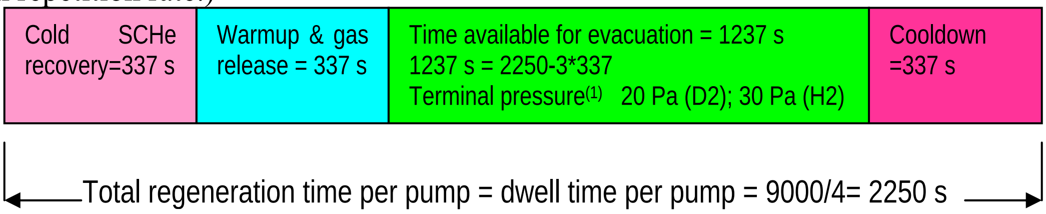
Pulse burn duration (s)	Cryopump regeneration pattern	Peak hydrogenic inventory Mole H_2 (D_2)
3000	4 cryopumps regenerated serially during 9000 s dwell	170(85)
1000	1 cryopump regenerated during each sequential 3000 s dwell	150 (75)
400	1 cryopump regenerated during every other sequential 1350 s dwell	140 (70)

Table 4.13-11 Regeneration pattern for 3000s sequential pulses

Pulse No	1					2				
Pump No.	Burn	Dwell				Burn	Dwell			
	3000 s	9000 s				3000 s	9000 s			
1	A	Reg	P	P	P	A	Reg	P	P	P
2	A	P	Reg	P	P	A	P	Reg	P	P
3	A	P	P	Reg	P	A	P	P	Reg	P
4	A	P	P	P	Reg	A	P	P	P	Reg
Pump status	A	=	Accumulating (pumping during fuelling)							
	P	=	Pumping without fuelling							
	Reg	=	Regenerating							

Figure 4.13-2 Evacuation pattern NB regeneration burn

(Evacuation pattern for NB regeneration during sequential 3000 s burn D2 fuelled pulses at the maximum repetition rate.)



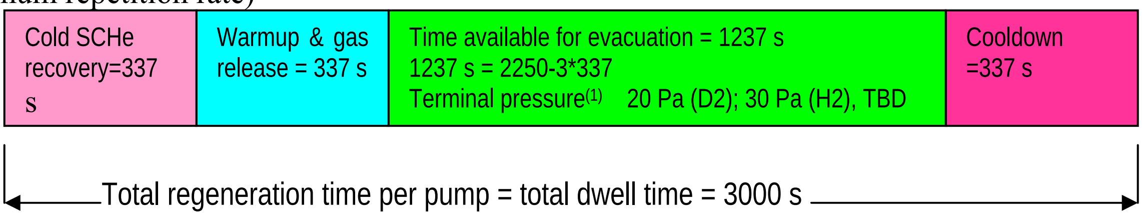
Note(1) The terminal pressures for D2 and H2 are TBD will be decided on the acceptability of cool-down liquefaction rate (under study). The exact terminal pressure depends on the time evolution of the gas release from the cryo-sorbent.

Table 4.13-12 Regeneration pattern for 1000s sequential pulses

Pulse No.	**1**		**2**		**3**		**4**	
Rep. Time	4000 s		4000 s		4000 s		4000 s	
Pump No.	Burn	Dwell	Burn	Dwell	Burn	Dwell	Burn	Dwell
1	A	Reg.	A	P	A	P	A	P
2	A	P	A	Reg.	A	P	A	P
3	A	P	A	P	A	Reg.	A	P
4	A	P	A	P	A	P	A	Reg.
Pump status		A	=	Accumulating (pumping during fuelling)				
		P	=	Pumping without fuelling				
		Reg.	=	Regenerating				

Figure 4.13-3 Evacuation pattern NB regeneration 1000 s burn

(Evacuation pattern for NB regeneration during sequential 1000 s burn D2 fuelled pulses at the maximum repetition rate)



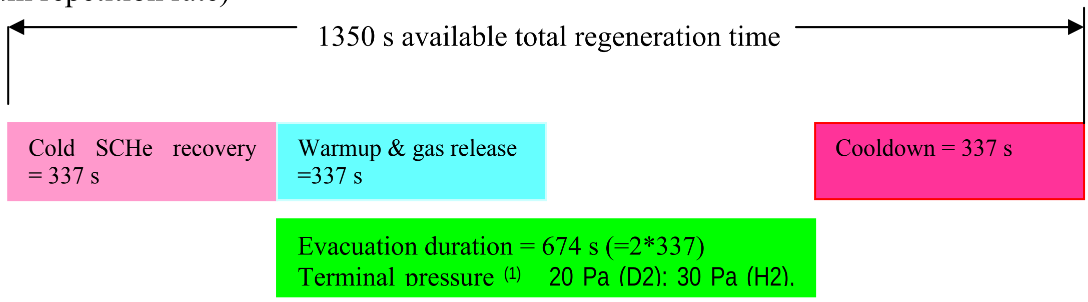
Note(1) The terminal pressures for D2 and H2 are TBD will be decided on the acceptability of cool-down liquefaction rate (under study). The exact terminal pressure depends on the time evolution of the gas release from the cryo-sorbent.

Table 4.13-13 Regeneration pattern for 400 s sequential pulses

Pulse No.	**1**		**2**		**3**		**4**		**5**
Rep. Time	1800 s		1800 s		1800 s		1800 s		
Pump No.	Burn	Dwell	Burn	Dwell	Burn	Dwell	Burn	Dwell	Burn
1	A	Reg.	A	P	A	P	A	P	A
2	A	P	A	Reg.	A	P	A	P	A
3	A	P	A	P	A	Reg.	A	P	A
4	A	P	A	P	A	P	A	Reg.	A
Pump status		A	=	Accumulating (pumping during fuelling)					
		P	=	Pumping without fuelling					
		Reg.	=	Regenerating					

Figure 4.13-4 Evacuation pattern NB regeneration 400 s burn

(Evacuation pattern for NB regeneration during sequential 400 s burn D2 fuelled pulses at the maximum repetition rate)



Note(1) The terminal pressures for D2 and H2 are TBD will be decided on the acceptability of cool-down liquefaction rate (under study). The exact terminal pressure depends on the time evolution of the gas release from the cryo-sorbent.

4.13.3.7 Cryopump heat loads and cooling conditions

Table 4.13-14 Torus cryopump heat loads and at 4.5 K(1)

during reference staggered pumping operation

4.5 K heat load (per pump) (2)	**Value (W)**
Pumping of gas throughput, Qp	72 ~~67~~
Internal radiation heat loads (inlet valve closed), Qrc	29
Internal radiation heat loads (inlet valve open), Qro	34
Radiation transmitted on 4.5 K panels through inlet port, Qrt2	0
Solid conduction, Qs	4
Residual gas conduction, Qg	10
Nuclear heating, Qn	13
Tritium Decay, Qd	10
Two Johnston Couplings	4
Σ 4.5 K heat load per cryopump (watt)(3), (4)	**147**
Recommended design value for cryoplant	**200**

Note (1) During nominal staggered pumping operation at 120 Pam^3/s fuelling with equimolar DT
Note (2) For each of 6 cryopumps. The number 6 relates to the heat load at 4.5K for the four cryopumps that provide the nominal pumping and two cryopumps that are under the regeneration stages of the cold helium recovery and cool-down.
Note (3) Tabulated heat loads include no margin.
Note (4) Excluding 4.5 K heat load of the cryo-jumper connecting each cryopump to its Cold Valve Box (estimated value 10 W per pump)

Table 4.13-15Heat Loads at 80 K (1)

80 K heat load (per pump)$^{(2)}$	Value (kW)
Pumping of gas throughput, Qp	014
Radiation emitted by from outer warm outer enclosure, Qr1	1.29
Radiation emitted from warm valve disc and shaft (inlet valve closed), Qrc	0.2
Radiation emitted from warm valve disc and shaft (inlet valve open), Qro	0.45
Solid conduction, Qs	0.13
Residual gas conduction, Qg	0.08
Nuclear heating, Qn	0.03
Tritium Decay, Qd	0
Σ 80 K heat load per cryopump (nominal pumping)$^{(3)}$	**2.12**
Recommended design value for the cryoplant (nominal pumping)	**2.5**
Note (1) For each of 6 cryopumps. The number 6 relates to the heat load at 4.5K for the four cryopumps that provide the nominal pumping and two cryopumps that are under the regeneration stages of the cold helium recovery and cool-down. Note (2) During nominal staggered pumping operation at 120 Pam^3/s fuelling with equimolar DT. Note (3) Tabulated heat loads include no margin.	

Table 4.13-16Heat loads at 100 K for 2 torus cryopumps(1)

100 K heat load (per pump)	Value (kW)
During regeneration (mainly gas conduction)	19
Note (1) The number 2 relates to the heat load at 100 K for the two cryopumps that are under the regeneration stages of the rapid warm-up/gas release and evacuation.	

Table 4.13-17Heat loads summary per torus cryopump

Pump operation	Heat load into 4.5 K system (W)	Heat load into 80 K system (kW)
Standby	37	1.62
Dwell	42	1.87
Pumping	147	2.12
100 K regeneration	-	19
40 K leak testing	39	1.97

Table 4.13-18 General Cryogenic Requirements for Torus Cryopumps

Parameter	Value
Sorbent Cryopanel	
Cryopanel He supply manifold temperature	4.5K
Cryopanel He supply manifold press.	0.4 MPa
Cryopanel He return manifold temp. (all modes)	T supply + 0.2 K
Cryopanel He supply/return manifold differential press.	0.035 MPa
Leak detection	
Leak checking mode cryopanel He supply temp.	40 K
Leak checking mode cryopanel He supply press.	0.4 MPa
Leak checking mode cryopanel He return temp.	T supply + 2.5 K(1)
Leak checking mode cryopanel He supply/return manifold differential press.	0.035 MPa
Inlet baffle and radiation shield	
80K He supply manifold temp.	80K
80K He supply manifold press.	1.8 MPa
80K He return manifold temp. (all modes except regeneration)	T supply + 10 K nominal
80K He return manifold temp. (regeneration)	100 K
80K He supply/return manifold differential press.	0.2 MPa nominal
Inlet valve cooling	
300K He supply manifold temp.	300K
300K He supply manifold press.	1.8 MPa
Maximum 300K He return manifold temp. (all modes)	T supply + 20K
300K He supply/return manifold differential manifold press.	0.2 MPa

Note (1) The possibility of increasing the temperature increment from 2.5 K to 5 K is being studied (late 2006).

Table 4.13-19 Cryogenic requirements for 8 torus cryopumps

Parameter (units)(8)	
Cryopanel SCHe inlet/outlet temperature (K) for nominal pumping	4.5/4.7
Mass flow - 4.5 K He (g/s) (1)	1100
Heat load-4.5 K He (W) (6 pumps) (2)	882
Mass flow – 80 K He (g/s) (3)	246+TBD (4)
Refrigeration Heat load during pumping (6 pumps)–80K He(kW) (5)	13
Refrigeration Heat load during regeneration (2 pumps)–80K He(kW) (6)	38
Heat load (8 pumps) – 300 K He (kW) (7)	4

Note (1) For the 6 pumps under nominal pumping (185 g/s/pump for 147 W /cryopump at ΔT=0.2 K).
Note (2) 147 W each for 6 pumps under nominal pumping
Note (3) 41(g/s/pump)x6(pumps)= 246 g/s to cool 2.12 kW heat load from 80 K shields of 6 cryopumps (4 during nominal pumping, 1 in "Cooldown" and 1 in "Cold Helium Exhaust"), limiting the ΔT to 10 K.
Note (4) TBD relates to the 1.8 MPa warm gas used to heat the sorbent to 100 K for release of the sorbed gases; thermo-hydraulic code studies are in progress to optimise gas release warming conditions.
Note (5) 2.12 kW per pump for 6 pumps = 13 kW
Note (6) 19 kW per pump for 2 pumps =38 kW
Note (7) 500 W per pump inlet valve for 8 pumps=4 kW
Note (8) Tabulated parameters include no margin.

Table 4.13-20 Regeneration Requirements for 2 Pumps during partial regeneration(1)

(during warm up to 100 K/gas release and evacuation)

Parameter	Value
Mass of 4.5 K, 0.4 MPa SCHe to cool 1 pump from 100 K to 4.5 K	6.68 kg
Total duration of “Cold Helium exhaust”, “Warm-up & Gas release”, “Evacuation” and “Cooldown” regeneration stages	600 s
Energy to be supplied to each sorbent panel array to heat it from 4.5 K to 100 K	3.5 MJ
Mass of 100 K, 1.8 MPa helium to heat 1 sorbent panel array from 4.5 K to 100 K	6.8 kg
Note (1) During warm-up to 100 K for cold helium recovery, gas release, evacuation and cool-down.	

Table 4.13-21 Torus cryopump cryogenic requirements during initial Pump cool-down

Parameter	Value
4.5K Components	
4.5K mass in pump (stainless steel)	330 kg
Total heat to be removed from 300K to 80K	28.4 MJ
Total heat to be removed from 80K to 4.5K	2 MJ
Cooldown time for 8 cryopumps	<1 Hour
80K Components	
80K mass in pump (stainless steel)	525 kg
Total heat to be removed from 300K to 80K	45 MJ
Cooldown duration, 300 K to 80 K, 8 pumps	4 Hours
300K Components	
none	

Table 4.13-22 Heat load at 4.5 K for 1 cryostat cryopump(1)

4.5 K heat load component(6)	Value (W)
Internal radiation heat loads (inlet valve closed) (2)	29
Internal radiation heat loads (inlet valve open)	≤29
Free molecular gas conduction	
Room Temperature pre-cooldown evacuation of N_2 (3)	13
Post-cooldown steady state pumping of helium and H_2 (3)	23.5
Nuclear heating	0
Tritium decay	0
Two Johnston Couplings	4
Cooling of pumped gas throughput	
Room Temperature pre-cooldown evacuation of N_2 (4)	1.6
Post-cooldown steady state pumping of helium and H_2 (5)	2
Solid conduction (2)	4
Total 4.5 K heat loads	
Room Temperature pre-cooldown evacuation of N_2 (28 ~~28.6~~+13+1.6+4~~1~~5)	48
Post-cooldown steady state pumping of helium and H_2 (28.6+23.5+2+4~~15~~)	60

Note (1) All heat loads are pre-design estimations.
Note (2) Based on corresponding torus cryopump value.
Note (3) Scaled from torus cryopump DT value assuming scaling as $(\gamma+1)/\{M^{0.5} \bullet (\gamma-1)\}$, M is mass number, γ the isentropic index, at a pump internal pressure of 10^{-2} Pa (stage 1 cryo-pumpdown).

Note (4) Based on stage 1 cryo-pumpdown with constant N2 throughput of 0.4 Pam^3/s and including concurrent maximum acceptable air leak of $4x10^{-2}$ Pam^3/s. Water not included as mainly pumped by 80 K panels. Note (5) Based on (i) H2 outgassing rate of 10^{-7} Pam^3/m^2 from an estimated outgassing area of $25x10^3$ m^2 of warm vacuum facing stainless steel and an effective pumping speed of 61.4 m^3/s/cryostat cryopump (=77.2•$\{190/300\}^{0.5}$) for H_2 at 190 K and (ii) maximum sustainable helium partial pressure of $4x10^{-2}$ Pam^3/s with a helium pumping speed of 41 m^3/s/cryostat cryopump (=51.5•$\{190/300\}^{0.5}$). Water not included as mainly pumped by 80 K panels. Note (6) Tabulated heat loads include no margin.

Table 4.13-23Heat load at 80 K for 1 cryostat cryopump

80 K heat load component (4)	Value (kW)
Cooling of pumped gas throughput during pumping: Room Temperature pre-cooldown evacuation of H2O and air (1) Post-cooldown steady state pumping of helium and H_2(2)	 0.01 $<10^{-4}$
Radiation emitted from outer warm cryopump enclosure (3)	1.29
Radiation emitted from warm valve disc and shaft (inlet valve closed) (3)	0.2
Radiation emitted from warm valve disc and shaft (inlet valve open) (3)	≤0.2
Solid conduction (3)	0.13
Residual gas conduction (3)	0.08
Nuclear heating	0
Eddy current heating	0
Total for pre-cooldown transient pumpdown of N_2 & H_2O (0.01+1.29+0.2+0+0.13+0.08+0+0)	1.7
Total for cold steady state pumping of H_2 & He at ~ base pressure (10^{-4}+1.29+0.2+0+0.13+0.08+0+0)	1.7
Regeneration heat load (chiefly gas conduction) at 100 K (3)	19
Note (1) Based on stage 1 cryo-pumpdown with air (N_2) & H_2O throughputs of 0.4 Pam^3/s and 0.5 Pam^3/s respectively, inlet temperature/pressure 300 K/10 Pa Note (2) Based on pump inlet molecular temperature of arithmetic mean of cryostat vessel and thermal shield temperatures, i.e. 190 (={300+80}/2) K Note (3) Torus cryopump value used. Note (4) Tabulated heat loads include no margin.	

Table 4.13-24 Cooling conditions for 1 cryostat cryopump(1)

Parameter(2)	Unit	Value
Sorbent cryopanel		
Sorbent panel SCHe inlet/outlet temperature for nominal pumping	K	4.5/4.7
Sorbent panel SCHe inlet/outlet temperature for leak test pumping	K	40/42.5
Sorbent panel SCHe inlet/outlet pressure for nominal pumping	MPa	0.4/0.365
Mass flow 4.5 K Helium during nominal pumping (3)	g/s	75
Leak detection		
Sorbent panel SCHe inlet/outlet temperature (4)	K	4.5/4.75
Sorbent panel SCHe inlet/outlet pressure for nominal pumping	MPa	0.4/0.365
40 K heat load during leak test pumping	W	39
80 K heat load during leak test pumping	kW	1.97

Inlet baffle and radiation shield		
80 K gaseous Helium inlet/outlet temperature during nominal pumping	K	80/90
80 K gaseous Helium inlet/outlet pressure for nominal pumping	MPa	1.8/1.6
80 K Helium flow during pumping during nominal pumping (5)	g/s	33
Sorbent panel regeneration		
Temperature increase of 80 K helium flow during regeneration	K	20
80 K Helium flow during regeneration	g/s	33+TBD (6)
80 K gaseous Helium inlet/outlet pressure for regeneration	MPa	1.8/1.6
Inlet valve heating (all modes) (7)		
300 K supply/return temperature	K	300/290
300 K supply/return pressure	MPa	1.8/1.6

Note (1) All heat loads are pre-design estimations scaled from corresponding torus cryopump values.
Note (2) Tabulated heat loads include no margin.
Note (3) Based on heat load of 60 W and ΔT of 0.2 K.
Note (4) The possibility of increasing the temperature increment from 2.5 K to 5 K is being studied.
Note (5) Based on heat load of 1.7 kW for ΔT of 10 K
Note (6) 33 W relates to 1 pump under nominal pumping. TBD relates to the 1.8 MPa warm gas used to heat the sorbent to 100 K for release of the sorbed gases; thermo-hydraulic code studies are in progress to optimise gas release warming conditions.
Note (7) Inlet valve is radiatively cooled by the tokamak thermal shield.

Heat loads at 4.5 K for 4 neutral beam cryopumps :

The NB cryopumps are under re-design (DCR-49 Late 2006) to comply with top-opening maintenance access requirements

4.13.3.7 Type 1 Diagnostic Vacuum Pumping System

Type 1 Diagnostics are those which are separated from torus vacuum by a material vacuum barrier (other than a vacuum isolation valve) and therefore from tritium under normal operation conditions. The interspaces behind Type 1 Diagnostic material vacuum barriers are connected to the Service Vacuum System in order to establish and maintain the required vacuum conditions for normal diagnostic operation and to enable continued operation until the next planned shutdown by differential pumping in the event of vacuum leaks occurring in the diagnostic which if not pumped would be inoperable.

4.13.3.8 Type 2 Diagnostic Vacuum Pumping System

Type 2 Diagnostics are those in direct contact to the torus vacuum and therefore with tritium. Some of the Type 2 Diagnostics have elements that need a lower pressure than that prevailing in the torus during the required diagnostic operating window and therefore need to be differentially pumped. Figure 4.13-5 shows a conceptual arrangement for a Type 2 Diagnostic Vacuum Pumping System.

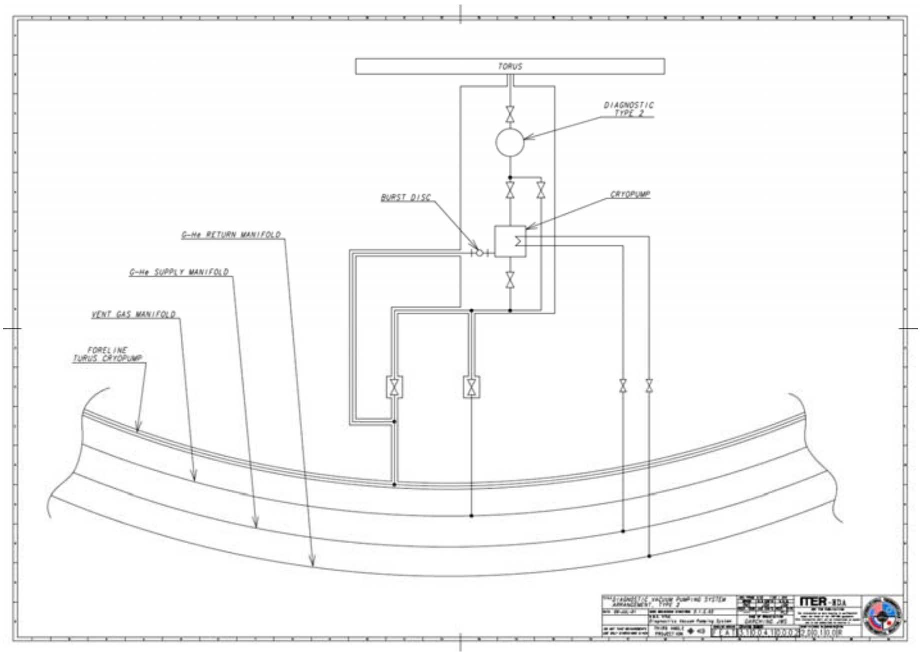


Figure 4.13-5 Conceptual arrangement for a type 2 diagnostic vacuum system

All Type 2 Diagnostics are connected to torus vacuum via a torus isolation valves. Pumping of tritiated gas from the torus is performed by cryo-cooler pumps of adequate crossover quantity, capacity and pumping speed to meet the requirement of each type 2 diagnostic. The cryo-cooler pumps are of the sorption type since helium has to be pumped and are tritium compatible. The cryo-cooler pumps are supplied with helium refrigerant gas from the Working Helium Supply System, are vented from the Venting System and the cryo-cooler pump relief devices discharge into the Relief Valve Capture System. Regeneration of the cryo-cooler pumps is via the Torus Cryopump Regeneration Foreline to the tritium plant. The cryo-cooler pumps are foreseen to accumulate during a plasma operating day and to regenerate in silent hours. The cryo-cooler pumps and associated valves are doubly contained.

4.13.3.9 Service Vacuum System (SVS)

The role of the SVS is to provide vacuum functions to all clients (mostly located external to the cryostat) that need vacuum and which are not tritium bearing under normal operation conditions. SVS is designed for compatibility with tritium at low levels (TBD), i.e. material of wetted parts and quality compatible with tritium. The SVS is located at the equatorial port level (except for the NB cell region where it routes above and below the NB cell). The SVS consists of 5 sub-systems, namely the Service roughing System (SRS), Service Guard Vacuum System (SGVS), Service Leak Detection System (SLDS), Venting System, Relief Valve Capture System and the Working Helium Supply System. An overview of the functions and provision of the SVS sub-systems are given in the following table.

Table 4.13-25 Overview of the functions and provisions of SVS sub-systems

SVS sub-system	Function	Provision	Remarks
SRS	Rough all non-tritium vacuum clients from atmospheric pressure to ~ 10Pa	A ring manifold at equatorial level (under/over NB cell) connectable to all clients and to its forepump in the VPR	Includes roughing of SGVS and SLDS systems at the same level
SGVS	Maintain a steady guard vacuum on all non-tritium clients that need it (e.g. cryo thermal insulation vacuum)	A ring manifold at equatorial level (under/over NB cell) with 5 integrated cryo-refrigerator pumps connectable to all clients needing guard vacuum	The cryo-refrigerator pumps are sorption-type to pump helium. Typical guard vacuum pressure 0.1 mPa
SLDS	Leak test all non-tritium client systems	A ring manifold at equatorial level (under/over NB cell) with 5 integrated cryo-refrigerator pumps connectable to all clients needing guard vacuum and a branch pipe to the leak detection unit in the VPR	The cryo-refrigerator pumps are condensation-type to pump impurity gases but not helium.
Venting System	Controlled venting of individual clients and SVS	A ring manifold at equatorial level (under/over NB cell) connectable through isolation valves to individual clients and SVS elements	Venting gas supplied from Tritium Plant
Relief Valve Capture System	Captures relief device discharges and routes them to Tritium Plant	A ring manifold at equatorial level (under/over NB cell) connected to pressure relief devices individual clients and SVS elements	
Working Helium Supply System	Supply and return working helium for SVS cryo-refrigerator pumps	Two ring manifolds (supply and return) at equatorial level (under/over NB cell) connected to each cryo-refrigerator pump	Working helium gas compressed in the Cryoplant Compressor Building and piped to Tokamak Building

4.13.3.10 Cryogenic Service Vacuum System (CSVS)

The Cryogenic Service Vacuum System (CSVS), shown in overview in Figure 4.13-1, provides permanent vacuum pumping of the Cryoplant and Cryodistribution (WBS 3.4). It is physically decoupled from the main Service Vacuum System (SVS) to preclude cross contamination from the SVS which will contain low levels of tritium during DT operation. The CSVS is an integral part of the ITER cryogenic system and provides the vacuum pumping of all parts of the cryoplant and the cryogenic distribution including all cryolines. The CSVS enables automatic pumping of any part of the cryosystem in case of leaks or outgassing, without stopping operation and without human intervention between two planned shutdowns. The CSVS maintains a pressure of less than 100 mPa in a leaking vacuum space for leaks up to 0.05 Pa.m^3/s. Leaks above this level may be pumped by portable vacuum pumping sets and provision is made to connect such portable pumps. The CSVS also includes the overall vacuum pressure control necessary for automatic leak detection and emergency pumping. The CSVS comprises 46 vacuum pumping sets for the cryodistribution system and approximately 10 for the cryoplant.

4.13.3.11 Leak Detection System

4.13.3.11.1 Leak Detection Approaches

Table 4.13-26 Leak Detection Approach

Hood testing	The component to be tested is internally pressurized with tracer gas while in a vacuum, with the vacuum monitored by a MSLD
Evacuation testing	The component to be tested is evacuated and the exterior probed with a tracer gas
Probe testing	The component to be tested is pressurized and the exterior probed with a high sensitivity sniffer probe
Spiking	An extension of the hood testing method in which trace amounts of additives are combined with the water in each cooling loop, replacing the tracer gas, to produce a unique cracking pattern when measured by the MSLD
Accumulation testing	An extension of the hood technique in which the change in global mass spectrum of a sample from the monitored vacuum space is compared over time; the sample is collected by an accumulation pump, e.g., cryopump, and the mass spectrum analyzed on regeneration
Pressure change	A change in pressure within a space is used for leak indication
Torus global leak detection system	Used for monitoring and leak location of all components within torus
Cryostat global leak detection system	Used for monitoring and leak location of all components within cryostat. Used in conjunction with cryostat helium pumping system for accumulation leak testing
Cryostat air leak detection system (ozone hazard)	Used to measure and limit air leak access to 4.5 K surfaces inside the thermal shield in conjunction with the cryostat cryosorption pumps and thermal shield labyrinths
Service leak detection systems	Used for testing, port interspaces, double seals and components external to torus and cryostat boundaries.

4.13.3.12 Vacuum Vessel Leak Detection

To implement a cost effective manufacture leak test of the vacuum vessel sectors, ports and port/stub extensions and to avoid duplication of expensive test facilities at multiple manufacture sites, these components should be tested to leak levels not requiring overly expensive test facilities, but which limit impurity inleakage to primary vacuum to levels compatible with plasma purity requirements.

Table 4.13-27 Vacuum Vessel Manufacture/Assembly Leak Detection Levels

Vacuum Vessel Component	Helium leak test level (manufacture) Pam^3/s (1), (2)	Rationale
Complete vacuum vessel & all appendages (global leak test)	**10^{-6}**	**For first plasma after baking/discharge cleaning, impurities limited to <0.1% plasma density (assumed~10^{20} ions/m^3). Plasma volume~800m^3, total ions in plasma~8•10^{22}, max. O ions~ 8•10^{19}~ 0.297 Pam^3. During 10^6 s (11.6 d, time to transition from vessel conditioning to plasma operation), air leak rate ~2.97x10^{-7} {=0.297/10^6} Pam^3/s (corresponding air leak at 79%N2, 21%O_2 =1.14x10^{-6} Pam^3/s). Plasma facing surfaces ~30% of impurity accumulation surface so allowable air leak=3.42x10^{-6} (=1.14x10^{-6} /0.3) Pam^3/s. Total vacuum vessel (9 sectors) leak rate ~ 10^{-6} Pam^3/s compatible with impurity limit of 0.1% and also with requirements of PDD2.7.4.1 ("A leak in the 10^{-4} to 10^{-6} Pam^3/s range would not result in a deterioration of the plasma performance in ITER").**
VV sectors	**10^{-7}**	**10^{-6}/9 (field joint welds leak tested to <10^{-7} Pam^3/s)**
Port/stub extensions	**10^{-8}**	**~ 10^2 parts so individual level ~ 10^{-8} to attain total of 10^{-6} Pam^3/s (field joints leak tested to <10^{-8} Pam^3/s)**
Blanket modules cassettes, small components	**10^{-9}**	**~ 10^3 parts so individual level ~ 10^{-9} to attain total of ~10^{-6} Pam^3/s (field joints leak tested to <10^{-9} Pam^3/s)**

Note (1) The tabulated leak levels are the maximum acceptable manufacture levels for the components (level at which repair is not required to reduce level).
Note (2) Baking to 200 C minimum needed to attain this leak test level

4.13.3.13 Neutral Beam Leak Detection Approach

Potential leak fluid candidates into the NB vacuum enclosure are external air, internal leaks of water from the in-vacuum water cooled components, internal helium leaks from the cryopump 5 K and 80 K cooling loops and internal leaks from the beamline absolute isolation valve actuators and fuel gas injection system.
In general, the cryopumps have to be warmed up to allow leak detection since helium tracer gas is sorbed at 5 K. Leaks of helium coolant are detected by mass analysis during periodic regeneration at 80 K. Air leaks through relatively vulnerable feedthroughs are mitigated by providing such feedthroughs with a second vacuum barrier and interspace monitoring and leak suppression by differential pumping using the service vacuum system.
Water leak detection is by mass analysis during regeneration at 475 K. The feasibility of water leak detection by measuring the quantity of water lost from the cooling circuit in a given period is being studied.
In-situ leak localisation for the internal components is performed at atmospheric pressure by filling components with tracer helium and traversing a sniffer or pumping probe over the component external surface (probes transported by manipulator accessing inside NB vessel through removable lid).
In-situ leak detection at the outer surface of the NB vacuum enclosure is performed by pumping internally through NB leak detector system and locally spraying tracer helium over candidate external leak sites. Leak potential is minimised by the use of secondary vacuum barriers at all likely leak sites and monitoring (and if necessary differentially pumping by SVS) the interspaces.

4.14 Tritium Plant , Ventilation and Detritiation (WBS 3.2)

Partial Update M.Glugla Oct 2006.
A more complete update will be carried out in 2007, following the ongoing review (M.Glugla).
Ventilation (HVAC) covers all site ventilation systems will be moved out of this section, and updated in 2007.

4.14.1 Functional Requirements

The tritium plant including ventilation and detritiation systems has the following following major functions.

1. Handling of incoming and outgoing tritium shipments and transfers to/from the fuel cycle. *(July 2005; Currently, the design of subsystem is not yet available for outgoing tritium shipments from the site.)*
2. Storage of tritium and deuterium.
3. Measurement and determination of tritium inventories.
4. Preparation and delivery of tritium, deuterium and their mixtures for fuelling..
5. Supply of divertor impurity seeding gases.
6. Processing of tokamak exhaust and neutral beam exhaust for recycling of tritium and deuterium, including:
   - Extraction/purification of fuel gases, including conditioning gases;
   - Protium rejection from the system to the environment;
   - detritiation of the exhaust into environment.
7. Extraction of tritium from breeding blanket modules and processing of tritium from test blanket modules (if required).
8. Radiological control of atmospheres for normal operation including maintenance and mitigation of tritium release at accident.
9. Detritiation of tritium contaminated water recovered from the detritiation systems.
10. Room air ventilation of the controlled areas for human access during maintenance operation *(to be moved to a separate section, and updated in 2007)*
11. Negative room pressure control (air depression) during normal operation and for establishing a release path at accident *(to be moved to a separate section, and updated in 2007).*

4.14.2 Configuration

4.14.2.1 Overall

4.14.2.1.1 Tritium Plant Overall Arrangement

The tritium plant consists of storage and delivery, tokamak exhaust processing, air, vent and water detritiation systems, hydrogen isotope separation, and tritium plant analytical systems.

The tritium plant shall be designed and arranged to meet the fuelling and exhausting requirements for all operational states and transients, including interfaces with other systems connected.

The components that form the primary boundary shall be designed in accordance with appropriate codes and standards to assure their structural integrity for the design loads. The components that mitigate the consequence of accident shall be designed to maintain their safety functions.

The tritium inventory in the tritium plant can be measured during and after plasma operation. The total tritium inventory in ITER is estimated to be about 3 kg based on the current design but this does not correspond to the maximum allowable inventory for ITER operation which should be determined in accordance with safety assessment.

4.14.2.2 Storage and Delivery System

The Storage and Delivery System (SDS) includes the long-term storage (LTS), and the fuel storage and delivery (SDS) systems. The main design features are given in Table 4.14-1.

4.14.2.2.1 Tritium Storage Arrangement

The long-term storage system (LTS) serves for receiving tritium from external facilities, and supplying tritium to the storage and delivery system, which is part of fuel cycle subsystems, required for machine operation. Prior to unloading, calorimetric determinations of the amount of tritium contained in the transport capsules are carried out. Also determinations of the tritium amount remained in the capsules after tritium recovery process and that transferred to the storage and delivery system. The long-term storage system also provides storage of tritium recovered from the fuel cycle subsystems during tokamak shutdown times as well as for tritium recovered from the plasma-facing components and dust in the hot cell building. This system is composed of 10 metal hydride beds (~100 g-T / bed) to ensure long-term storage of the whole tritium inventory in the vacuum vessel, fuel cycle subsystems, hot cell and detritation systems. The maximum tritium enrichment for storage is estimated to be 90%T-10%D due to tritium inventory limitation in the hydrogen isotope separation system.

4.14.2.2.2 Fuel Delivery Arrangement

The storage and delivery system (SDS) supplies specific fuel gases (>90%T - <10%D, 50%T-50%D, D_2, H_2, He-3) and impurity gases (N_2, Ar, He-4) through independent gas supply lines, and receives tritium containing hydrogen gases from the tokamak vacuum vessel, NBI systems and Water Detritiation System via the isotope separation system and the tokamak exhaust processing systems during plasma operation. Typical design parameters of the gas fuelling lines to the torus, pellets and neutral beams are given in Table 4.5-1.

During the time between plasma campaigns, tritium containing hydrogen gases can be stored, if required, in the metal hydride beds and D2 gas holding tanks with a specific isotopic compositions; i.e., 90%T-10%D, 50%T-50%D and tritium bearing D2. The metal hydride-beds in the storage and delivery system are designed with specific functions for rapid tritium inventory measurement (in-bed calorimeter), rapid delivery and and rapid recovery of fuel gases from the fuel cycle subsystems. The isotopic composition of is the gases in the SDS can be measured by the analytical system (ANS)

Table 4.14-1 Fuel Storage and Delivery System Configuration

Long-Term Storage System (LTS)	shipping container unpacking chamber	Receiving and unpacking of the IAEA B(U) type shipping containers
	tritium containing capsule temporary storage glovebox	After unpacking, the capsule containing tritium is temporary stored where necessary.
	calorimeter room	Accurate measurement of received tritium, and residual tritium in the shipping capsule after tritium unloading.
	tritium unloading furnace and high vacuum pumping glovebox	Transfer T2 gas to PVT-c measurement tank in SDS for tritium inventory measurement.
	waste packing and temporary storage box	Transfer T2 gas to PVT-c measurement tank in SDS for tritium inventory measurement.
	long-term tritium storage system	About 10 metal hydride beds (~ 100 g-T /bed)
Storage and Delivery System (SDS)	90%T-10%D storage bed train	About 9 metal hydride beds:
	50%T-50%D storage bed train	About 8 metal hydride beds:
	D2 storage bed train	About 3 metal hydride beds
	Fuel gas supply pumps	Supply pressure: 120kPa to torus GIS, 600 kPa to NBI-GIS
	In-bed calorimeter loop	Tritium inventory measurement accuracy (expected): 3% within 8 hr, 1% within 24 h after rapid supply or rapid recovery
	He-3 recovery loop	
	Over pressure protection systems	Pressure relief tank and header
	Gas analysis manifold	Connected to ANS

4.14.2.3 Tokamak Exhaust Processing System

4.14.2.3.1 TEPS Arrangement/Performance

This system processes torus exhaust gases during all modes of tokamak operation, including plasma operation, pump-down, baking and wall conditioning.

In addition, tritiated gaseous process wastes from other sources (NB injector cryopump, fuelling system purge, tritium plant analytical system, helium glow discharge cleaning molecular sieve beds, tritiated gases recovered in the hot cell) are also accepted by the tokamak exhaust processing system. Depending on the design of the test blanket module, the purge gases would also be accepted as required (TBD).

The waste gas stream from the tokamak exhaust processing system is discharged into the environment via normal vent detritiation system (VDS)

The tokamak exhaust processing system is composed of a front-end permeator (once-through process), impurity processing (recycling process), and a final clean-up system (once-through process).

4.14.2.3.2 Front-End Permeator

The front-end permeator comprises a palladium/silver alloy membrane for separation of elemental hydrogen isotopes from impurities included in the tokamak exhaust, and a nickel catalyst reactor for direct conversion of tritiated impurities into elemental tritium for full regeneration of cryo pump panels.

A separate permeator receives the NB injector cryopump regeneration gases and separates this stream into a pure molecular hydrogen stream and impurity stream. The hydrogen stream is mixed with the tritiated protium stream from the water detritiation system and sent to the protium column of the isotope separation system.

The impurities are sent to the impurity processing subsystem of the tokamak exhaust processing system.

4.14.2.3.3 Impurity Processing

During tritium recovery from the plasma-facing components, the tritium recovery gas may include high concentrations of tritiated hydrocarbons, water vapour (DTO), CO, CO_2, O_2 and carrier gas (N_2 or Ar).

The tritiated impurities are directly converted into elemental hydrogen (DT) in the nickel catalyst reactor, and the product gas sent to one of the front-end permeators for separation of DT from un-converted impurities and carrier gas. The DT stream is transferred to the isotope separation system. The stream of un-converted impurities and carrier gas is sent to the final clean-up system. Exhaust gas from the final clean-up system is sent to the normal VDS.

The impurity processing (re-circulation loop process) comprises a nickel catalyst reactor and a membrane permeator for continuous recovery of elemental tritium from tritiated impurities by heterogeneously catalysed cracking or conversion (chemical) reactions.

This subsystem also treats impurity streams from other sources including during glow discharge cleaning, cryogenic molecular sieve bed regeneration, and the analytical system.

4.14.2.3.4 Final Cleanup Process

The final cleanup process comprises a counter-current membrane reactor for final detritiation of waste gas by means of isotopic exchange with protium or deuterium. The detritiated impurity gas is then routed to the gamma-decay system. These three subsystems and gamma-decay system are placed in a shielded room.
The gamma decay system, which is composed of tanks equipped with γ activity monitors, compressors and transfer pumps has the function of temporary retention of activated (short half-life) inert gas species to allow for their decay to an acceptable level, at which point they are discharged through the normal VDS.

4.14.2.3.5 GDS Purge Cleanup

During glow discharge cleaning, deuterium and helium will be purged over prolonged periods (up to 100 h) through the plasma vacuum vessel, at a flow rate of ~ 50% of the nominal fuelling rate for plasma operations.

The gas stream from the torus roughing pumps during the helium glow will be processed through one of a pair of 80K liquid nitrogen-cooled molecular sieve beds to trap the impurities, and the helium is recycled directly back to the tokamak.

The molecular sieve beds will be regenerated every 24 h, by warming to room temperature for elemental hydrogen desorption, and (at a lower frequency) to higher temperatures for tritiated moisture desorption.

The desorbed gas is directed to the impurity processing loop of the tokamak exhaust processing system. This processing system will also be used for helium plasma operation.

4.14.2.3.6 Tritium Plant Auxiliary Cleanup

An auxiliary system, composed of a high vacuum manifold and a vacuum manifold, receives high vacuum pump exhaust from the tritium process component vacuum jacket and from various other sources, respectively.

4.14.2.4 Hydrogen Isotope Separation System

The isotope separation system utilizes cryogenic distillation and catalytic reaction for isotope exchange to equilibrate separate elemental hydrogen isotope gas mixtures.

The various feed streams are introduced at two feed locations in the column cascade to produce different products. These (feed and product streams) are

- a detritiated protium stream to recycle to the water detritiation system,
- a tritium enriched protium stream from the water detritiation system,
- two distinct deuterium streams for gas fuelling and NB injector source gas,
- two tritium fuelling streams, one with 50% T-50%D and one with 90% T-10%D isotopic composition.

The column cascade consists of four distillation columns (CD1, CD2, CD3 and CD4). The first column (CD1) removes tritium from predominantly protium-rich streams and produces a detritiated protium stream as the overhead product. The second column CD2 separates H and D from the CD4 overhead stream, and produces a partially-purified D_2 stream which is used for gas fuelling and as the feed to the third column CD3. CD3 removes tritium and protium from the deuterium for NB injector source gas. The bottom product of CD2, a D_2-DT mixture stream, is fed into the high-tritium-column CD4, which also receives an elemental hydrogen stream separated in the tokamak exhaust processing system from the plasma exhaust stream.

Table 4.14-2 Isotope Separation

Design approach			**Cryogenic distillation**			
Distillation columns			4			
Equilibrators			7			
Parameter	**Location**	**Dimension**	CD10	CD20	CD30	CD40
# of stages			125	100	60	70
Pressure	top/bottom	(kPa)	110/115	110/115	110/115	140/150
Column dia.	top/bottom	(mm)	87	116	109	16
Liquid Holdup (Top/Bottom/Reboiler)		mol/stage	1.03/1.03/3.1	1.39/1.17/3.5	0.84/0.84/2.5	1.15/0.79/2.4
Refrigeration load		(W)	350	550	350	550

Table 4.14-3 Hydrogen Isotope Separation System (ISS)

Parameter	**Value**
T inventory	~ 220 g (for short pulse) ~ 450 g (for long pulse)
Q_2 (hydrogen isotopes) inventory	~ 732 mol
He inventory	~ 10 m^3
Cold box	
Volume	~ 6.6 m^3
Design temperature	400 K
Operation temperature Column surface Vessel surface	 25 K 300 K
Set point for pressure relief to expansion tank	0.28 MPa(TBC)
Cross section of pressure relief pipe	TBD
Length of pressure relief pipe	~ 5 m

4.14.2.5 Ventilation and Atmosphere Detritiation Systems

(Note: This section will be completely rewritten after the major revisions of the DDD's in 2007).

The ventilation and atmosphere detritiation systems consist of the Vent Detritiation System (VDS) and the Atmosphere Detritiation System (ADS) for the controlled areas that require air ventilation for hand-on maintenance, negative pressure for normal operation and mitigation at an accident. The controlled areas are located in the radiologically controlled buildings such as tokamak building, tritium building and hot cell building.

4.14.2.5.1 Detritiation systems for Tokamak building and Tritium plant building

The normal vent detritiation system No.1 (N-VDS-1) is the system to process normal vent/purge gases from all tritium processing systems in Tritium Plant, depression gas from Tokamak Vault, depression/purge gases coming from GBs. N-VDS-1 has a function to process vent gas through ST-VS, and depression gas through S-ADS during maintenance of VV, drying/cover gases from maintenance of Divertor/PHTS. The Maximum flow rate is 700 m^3/h. This system is not a safety system but an investment protection system, backed up by S-VDS and S-ADS.

The normal vent detritiation system No.2 (N-VDS-2) is the dedicated system for processing depression gases from all Prot cell lines. The Maximum flow rate is 700 m^3/h. This system is not a safety system.
The Glovebox Atmosphere Detritiation System (GDS) is a centralised detritiation system for gloveboxes located in different rooms on various floors of the tritium plant building. The GDS is designed to;

a) Reduce tritium permeation and leakage from gloveboxes into rooms
b) Reduce HTO produced by radiolytic reactions between T_2, H_2O moisture and O_2.
c) Maintain negative (sub-atmospheric) pressure inside gloveboxes
d) Maintain the tritium concentration below the target value inside glove boxes

The Maximum flow rate is 150 m^3/h. This system is not a safety system.

The Standby vent detritiation system (S-VDS) is the mitigation (safety) system for processing contaminated depression gas coming from Tokamak Building Gallery (except vault) and Tritium Plant before environmental exhaust. During normal operation, the room atmosphere pressure is maintained at sub-atmospheric pressure by the room depression system. When the room air tritium concentration exceeds a set point, the relevant room isolation valves in the HVAC ducts are closed, and simultaneously this triggers the isolation valves connected to the room depression system to switch from stack exhaust to standby vent detritation system (S-VDS) and to start the S-VDS. The Maximum flow rate is 3000 m^3/h.

The standby atmosphere detritation system (S-ADS) serves for air recirculation through the vacuum vessel during tokamak maintenance to remove tritium from the vacuum vessel, and for room air enhanced detritation for early entry where necessary, in case tritium is released in either the tokamak building or the tritium building. This system is located in the tritium building. The condensate of the S-VDS is sent to the tritiated water holding tank systems in the tritium building. The maximum capacity is 4500 m^3/h. This system is a *safety system* backed up S-VDS.

All the above detritiation systems are located in the tritium building. The condensate from the driers of each detritiation systems is sent to the tritiated water holding tank systems in the tritium building.

4.14.2.5.2 Detritiation systems for Hot Cell building

The hot cell atmosphere detritation system (HC-ADS) controls the HTO concentration at a controlled level (< 500 DAC-HTO* or 1.6 x10^8 Bq/m^3) by continuous removal of tritium from hot cell red zone rooms where in-vessel components are received, stored, repaired, and radioactive wastes are processed. The HTO level of 500DAC can be achieved for maximum tritium release rate of ~ 750 GBq/h (~20Ci/h) by the system capacity of 4500 m3/h. After removal of divertor cassettes or other tritium source terms from one of the red zone rooms, reduction of the HTO concentration from 500 DAC to 1DAC will be achieved within a week using the HC-ADS at full capacity. The system is a safety system that backs up HC-VDS.

*(*DAC:Derived Air Concentration, unprotected exposure to 1 DAC = 10 μSv/h, see zoning Table 3.1-12).*
Note : This section requires updating to French Regulations, using VDO as in Table 3.1-12).

The hot cell vent detritiation system (HC-VDS) serves for depression of Red & Amber zones of Hot Cell. During normal operation, the Green zone rooms' atmosphere pressure is maintained also at sub-atmospheric pressure by the hot cell depression system. When the tritium concentration in Green zones exceeds a set point, the depression system switches from stack exhaust to HC-VDS. The process capacity is required to be at least ~2000 m^3/h, though it could be changed after safety evaluation.

All the above systems are located in the hot cell building. The system condensate is sent to the tritiated water temporary holding tank in the hot cell building or directly transferred through a tritiated water transfer pipeline to the tritiated water holding tank systems in the tritium building.

The major part of the activated dust from in-vessel components introduced in the red zone is collected and removed at the receiving dust collection and cleaning filter unit and storage cell, and by an air-recirculation loop installed in a green zone in the hot cell. The five cells in the red zone are connected to five independent loops, for reduction of dust concentration levels in the red zone, and removal of heat dissipated from in-vessel components (removed from vessel) and hot cell equipment, and for removal of dust by the filter units. These loops comprise HEPA filter units, air cooler cooled by normal chiller and air re-circulation blowers.

4.14.2.6 Water Detritiation System

4.14.2.6.1 Tritiated Water Sources

During operation of ITER, tritiated water will be produced in various systems. The expected sources are:

Table 4.14-4 Tritiated Water Sources

(to be updated once revised ADS / VDS DDD'S will become available in 2007)

<table>
<tr><th>class</th><th>Source</th><th>Process and Storage</th></tr>
<tr><td>a)</td><td>condensate generated from the normal operation of various atmosphere and vent gas detritiation systems</td><td rowspan="4">processed by the water detritiation system to minimize tritium-bearing waste water to the environment,</td></tr>
<tr><td>b)</td><td>tritium process component maintenance,</td></tr>
<tr><td>c)</td><td>condensate from the air coolers in the containment volume (designed to limit overpressures from an ex-vessel coolant leak),</td></tr>
<tr><td>d)</td><td>air detritiation dryers in the TCWS vault annex</td></tr>
<tr><td>e)</td><td>the tokamak cooling water system maintenance drain and the tokamak cooling water system vent gas condensate,</td><td rowspan="2">recycled to the tokamak cooling water system.</td></tr>
<tr><td>f)</td><td>in-vessel component maintenance drain collected in the hot cell, and</td></tr>
<tr><td>g)</td><td>condensate from the standby vent detritiation system and the standby atmosphere detritiation system operated during tritium contamination accidents.</td><td>stored in the emergency holding sump tanks.</td></tr>
<tr><td>h)</td><td>Ingress of coolant into vacuum vessel</td><td></td></tr>
<tr><td>i)</td><td>Condensate generated from HVACs</td><td></td></tr>
</table>

4.14.2.6.2 Tritiated Water Source Classification

The source water will be stored in the relevant holding tank system with the following level classification:

Table 4.14-5 Tritiated Water Source Classification

	Tritiated Level	Source (see above)
H Level	> 3.7 GBq/g (> 100 Ci/kg)	a)
M Level	37 kBq/g – 3.7 GBq/g (10^{-3} Ci/kg – 100 Ci/kg)	a), b)
L Level	60 Bq/gm – 37 kBq/g (1.6 x 10^{-6} Ci/kg - 10^{-3} Ci/kg)	c),d),e),f)
LL Level	< 60 Bq/g (<1.6 x 10^{-6} Ci/kg) for direct release after assaying;	i)
Emergency holding sump tanks:	total capacity 400 m^3	g), h)
Note : 1g Tritium ~10,000 Ci, 1Ci = 3.7e10 Bq		

4.14.2.6.3 Water Detritiation Process

(the process description below will undergo a major revision in 2007)

Tritiated water sent from the holding tank system is purified to remove metal and other ions and organic species to the catalytic exchange process and electrolysis.

The purified water (tritium concentration < 370 MBq/g (< 10 Ci/kg)) is processed (~ 20 kg/h) by the catalytic exchange towers and the electrolysers (tritium feed concentration ~ 3.7 GBq/g – 20 GBq/g (~ 100 – 500 Ci/kg)).

Through counter-current catalytic exchange reactions, tritium (T) is enriched in the water, which is flowing to the bottom of the towers, and hydrogen gas, which contains virtually no tritium, is flowing to the top of the tower. The enriched water is then dissociated into hydrogen gas H_2 (T) and O_2 gas by the electrolysers. The H_2 (T) is returned to the bottom of the tower, and a part of the hydrogen gas stream (~280 mol/h) is sent to the hydrogen isotope separation system to recover tritium via a membrane permeator system.

The H_2 stream at the hydrogen separating column (CD1) in the ISS, which recovers tritium included in the H_2 (T), is then recycled to the bottom section of the water detritation tower for further reduction of tritium in the H_2 prior to discharge into the environment.

The hydrogen stream from the top of the tower is discharged to the environment through the flame arrester. The O_2 stream from the electrolysers is sent to the normal vent detritiation system (N-VDS-1) via the O_2 gas processing system composed of molecular sieve dryers.

4.14.2.7 Tritium Plant Analytical System

4.14.2.7.1 Tritium Plant Analysis System

In addition to local instrumentation, a central analytical system is an integral part of the tritium plant with the main functions:

i) verification or additional control of the correct functioning of various processes by determination of the composition of various gas mixtures,
ii) determination of tritium concentrations in various gas mixtures for inventory accounting,
iii) monitoring and calibration of local instrumentation, e.g. ionisation chambers.

Samples from the storage and delivery system, tokamak exhaust processing system, and isotope separation system, are sent to the analytical system via interconnecting lines. Four manifolds are installed in the analytical system to receive samples, and are organised to minimise cross contamination between samples of different compositions. The samples to be analysed are injected into analytical gas chromatographs for determination of their gas composition.

Five gas chromatographs (three micro-gas chromatographs and two cryogenic micro-gas chromatographs) are chosen to perform the analytical tasks required, i.e. determination of the tritium concentration, as well as of the six hydrogen molecular species and of the other impurities expected.

The exhaust gases of the five gas chromatographs containing carrier gas~~,~~ and injected samples are either sent for final detritiation to the normal vent detritation system (N-VDS 1) or to the tokamak exhaust processing system, depending on their tritium concentrations.

The gas chromatographs, manifolds and pipework connected to the tokamak exhaust processing system are placed in a shielded barrier.

4.15 Cryoplant and Cryodistribution (WBS 3.4)

(Update V.Kalinin Oct.2006)

4.15.1 Functional Requirements

The cryoplant provides cooling for the following components:

- the superconducting magnet system,
- the torus, NB, cryostat and pellet cryogenic vacuum pumps,
- the 80K shields of the magnet system, the cryopumps and cryogenic transfer lines,
- small users such as gyrotrons and diagnostics.

4.15.2 Configuration

The ITER cryogenic system is subdivided into three parts: the cryoplant, the cryodistribution system and the system of cryogenic lines and manifolds.
Main guidelines for the design of the cryogenic system are the follows:

- active smoothing of pulsed heat loads deposited in the magnet system;
- usage of cold helium pumps for forced flow cooling of the magnet system and cryogenic vacuum pumps;
- stable operation over wide range of plasma scenarios;
- cost minimization by using standardized components.

4.15.2.1 Cryoplant

4.15.2.1.1 Cryoplant Equipment

The cryoplant equipment includes the following components:

- four identical cold process boxes of the LHe plant and LHe storage;
- helium gas compression station of the LHe plant;
- external He gas purification unit;
- 1.8 MPa warm (300K) and 80K tanks for storing the gaseous He,
- two identical 80K He cold boxes linked to two identical liquid nitrogen (LN_2) auxiliary cold boxes for final cooling of the He flow to 80K;
- helium gas compression station of the 80K He loop;
- LN_2 subsystem with two identical cold process boxes and LN_2 storage;
- nitrogen gas compression station.

The LHe plant is presently under optimization for the updated design option to use the HTS current leads (DCR-44) cooled with 50K He instead of the conventional current leads cooled by liquid helium. The expectation is that the total number of process boxes of the LHe plant can be reduced from four to three.

4.15.2.1.2 Cryoplant Location

The cryoplant equipment is located in two buildings: the cryoplant compressor building and the cryoplant cold box building.
The compressor building houses all the warm compressors for the LHe plant, the liquid nitrogen (LN_2) subsystem and the 80K He loop as well as the He and N_2 gas dryers.
All cold boxes of the LHe plant, the LN_2 subsystem, the 80K He cooling loop, and He gas purification unit, are installed in the cryoplant cold box building and connected with the

cryodistribution boxes of the tokamak building via two long cryogenic lines. One LHe tank of 25 m^3 is used for storing LHe when the plasma pulsing allows accumulation of LHe.
The 1.8 MPa warm He tanks, 80K He quench tanks, a low pressure He balloon, and liquid nitrogen tanks are placed in an outside area, adjacent to the cryoplant cold box building.
The 1.8 MPa warm He tanks are used to store the He inventory of the cryogenic components of ITER after machine warm-up. The cold 1.8 MPa tanks are needed to facilitate storing of cold He expelled from the TF coils during fast energy discharge and quench.

4.15.2.2 LHe Plant

4.15.2.2.1 LHe Plant Arrangement

Large pulsed heat loads deposited in the magnet system due to magnetic field variation and DT neutron production, are removed by the LHe plant, which is designed to be stable in operation despite a certain level of fluctuation of the heat load. The specification on the heat loads from different components of the magnet system, namely the TF winding packs, the TF cases together with intercoil structures, the PF and correction (CC) coils and central solenoid (CS) is given in "Table 3.6-8 Heat Loads in the Magnets and Feeders" for the scenario with a plasma pulse repetition rate of 1800 s and plasma burn time of 400 s.

The coils use HTS current leads cooled with 50K He. The current and heat load generated in the current leads of the CS, PF and CC coils vary in accordance with the current scenario of each coil. The current leads of the TF coils operate in steady state cooling conditions. Cooling requirements for the leads are given in Table 3.6-9.

The minimal heat load fluctuations are expected for the short (400 s) plasma pulses for which a special active cooling control of a constant heat load on the LHe plant is utilized. The maximal heat load variations will appear for the long (3000 s) standalone plasma pulses.

Another large cryogenic user is the vacuum pumping systems that include eight torus cryopumps, four NB cryopumps and two cryostat cryopumps. The specification of the heat loads for these cryopumps is shown in Table 4.15-1. The key requirement that drives the design of the LHe plant is fast cool-down of the cryopumps that is required for their periodical regeneration at 100K.

Table 4.15-1 Requirements for cryopumps at 4.5K

		8 Torus cryopumps	**4 NB cryopumps**	**2 Cryostat cryopumps**	**Cryopumps of pellet units**
Refrigeration heat load from cryopumps	kW	1.8	0.6	0.6	0.6 (TBC)
Refrigeration heat load from cryolines	kW	0.2	0.08	0.02	0.07
Heat load of He circulating pumps	kW	0.6	0.25	0.2 (TBC)	0.3 (TBC)
He flow rates during nominal pumping	kg/s	1.2	0.5	0.4	0.6 (TBC)
Temperature increase for He flow	K	0.2 (TBC)	0.4 (TBC)	0.2 (TBC)	0.2 (TBC)
Total for LHe plant					
Refrigeration	kW	2.6	0.93	0.82	0.97 (TBC)
Mean liquefaction	kg/s	0.021	0.007	0.002	TBC
Maximum liquefaction during cool-down stage	kg/s	0.08	0.3(TBC)	0.08 (TBC)	TBC

The mean liquefaction capacity for cool-down all of the cryopumps is 0.03 kg/s. The cool-down of the eight torus cryopumps requires the steady state liquefaction. However the liquefaction for the 4 NB cryopumps varies. The four NB cryopumps are regenerated sequentially one after another. Every hour one of the NB cryopumps will be taken off-line for the regeneration during dwell time of 1400s between the plasma pulses. Every hour a spike of the liquefaction capacity of 0.3 kg/s appears during 130 s cool-down of one NB cryopump.

The heat load specification on the LHe plant from all the ITER cryogenic components is listed in the following table. The two-stage pressure (2.0/0.4 MPa) cooling cycle of the LHe plant modules includes a LN_2 pre-cooling stage. The gradual cool-down of the magnet system will be performed at high cooling efficiency due to the usage of the LN_2 pre-cooling stage. The LN_2 plant will operate in the most effective refrigeration mode during cool-down of the magnet system.

No basis exists yet to optimize the operating temperature of the LHe plant. Further cooling simulations for magnets are required in order to clarify necessity to reduce operating temperature from 4.2 K to 4.15 K.

Table 4.15-2 Specification for the LHe plant

ITER cryogenic components	Requirements	Temperature of LHe plant
Magnets		
Total static heat input including, distribution boxes and cryolines to the magnets	11.9 kW	4.2 K
Averaged variable heat load due to nuclear heating and electro-magnetic heat losses	10.7 kW	4.2 K
Static heat load from cold helium circulating pumps (maximum value)	11.4 kW	4.2 K
Helium flow for cooling HTS current leads	0.15 kg/s	50 K
Cryo-sorbent vacuum pumps		
Total heat deposition and static heat input	5.5 kW	4.5 K
Averaged liquefaction to support regeneration of cryopumps	0.03 kg/s	4.5 K
Small cryogenic users		
Total heat deposition and static heat input to gyrotrons, fuelling and diagnostic	1.0 kW	4.5 K
Contingency		
Contingency on complexity of cryoplant operation, including wide range of plasma scenarios	5 kW	4.2 K
Total: 39 kW at 4.2 K + (0.03 kg/s + 6.5 kW) at 4.5 K + 0.15 kg/s at 50 K		

The LHe plant will operate at three temperature levels, namely 4.2K, 4.5K and 50K to satisfy the nominal plasma operation. A backup mode to cool the magnets at 3.7 K with reduced (by ~ 25 %) cooling capacity of the LHe plant is also included in the specification.

The design strategy of the ITER cryoplant and cryodistribution (ACBs, CCB and cryogenic transfer lines) is to satisfy for the maximum cooling demands expected from the different ITER cryogenic users and the maximum expected value of the heat load from cold helium circulating pumps is shown in the above Table. However the circulating helium pumps are designed to have adjustable rotating speed and hence the reduction the helium mass flow rate through the magnets can be minimised when the plasma experiments allows the reduced flow for circulating cooling.

The most challenging requirement for the design of the LHe plant is the removal of large pulsed heat loads deposited in the magnet system. The LHe plant has to be stable in operation over a wide range of plasma scenarios. Each of these scenarios will produce different thermal loads due to neutronic and magnetic field variation.

The ITER experimental program includes also the following extended plasma scenarios:
(Section 3.2 Plasma Operation Scenarios)

- plasma pulses with the plasma burn phase of 500 s for a fusion power of 500 MW and repetition time of 1800 s
- plasma pulses with the enlarged plasma burn phase of 1000 s for a fusion power of 400 MW; repetition time > 4000s
- plasma pulses with the enlarged plasma burn phase of 3000 s for a fusion power of 360 MW; repetition time > 12000s
- plasma pulses with enlarged fusion power of 700 MW with plasma burn duration of at least 100 s; (undefined repetition time)

The special cooling procedure has been developed for allowing active smoothing of the pulsed heat load. Due to smoothing of the pulsed heat load the LHe plant operates close to the most stable steady–state cooling mode for plasma scenarios with short burn duration, including 1000s. By use of control procedures, the ITER LHe plant is designed to cope with the average heat load from the magnets that is much smaller than the maximum variable heat load without the cooling control.

The exception is the plasma pulses with the enlarged plasma burn phase of 3000s. The heat load on the LHe plant will vary for this scenario. The present understanding is that the contingency on the heat load of 5 kW given in above Table will cover possible variation of the heat load for 3000s plasma pulses.

The ITER plasma experimental program starts with protium plasmas (H-plasma phase) at minimum cooling demand for plasma pulsing (no nuclear heating).

Then the tritium plasmas will be generated at reduced parameters (i.e. plasma current magnitude and burn time) and these parameters result in a reduced cooling demand on the LHe plant compared with that for the reference plasma scenario. Hence the LHe plant must be designed to be adjustable to allow reduced plasma operation. The expectation is that the LHe plant can operate at 4.5K to satisfy cooling requirements of the protium plasmas.

The LHe plant can be also adjusted to allow other typical operating modes, e.g., a stand-by mode during "silent" (night) hours, long term maintenance, baking of vacuum vessel, re-cooling of magnet system after fast energy discharge or quench, cool-down from 300K to 4.5K and so on (see the following table). The stand-by mode during "silent" (night) hours includes the high temperature (300K & 470K) regeneration of the torus, NB and cryostat cryopumps.

The requirement on the He mass flow rate for the gradual cool-down of different ITER components from 300 K to 80 K is shown in Table 4.15-3, and the typical operating modes are shown in Table 4.15-4.

Table 4.15-3 He mass flow rate for the gradual cool-down of different Components

Component	Total He mass flow rate (kg/s)	Notes
All coils	2.0	Constant He flow
Magnet structure	2.0	Constant He flow
Torus and cryostat cryopumps	From 0.5 to 1.0 (TBC)	Variable He flow
NB cryopumps	TBD	TBD
80K shields (VVTS & CTS, supports)	From 1.5 to 3.0	Variable He flow

Table 4.15-4 Typical operating modes of the cryoplant

Operating modes	Comments
Nominal plasma operation	Operation with different plasma scenarios according to ITER experimental program
Ready for plasma pulsing	No plasma operation. Cold helium circulating pumps and compressors are working.
Stand-by mode during “silent” (night) hours	8 –10 night hours per each day
Long term maintenance (one week or longer)	Cold helium circulating pumps and cold compressors are stopped.
Baking of vacuum vessel	Baking is performed during 10 days of each month.
Release of helium from magnet system during fast energy discharge or quench	Released helium is collected inside 80K quench tanks at 1.8 MPa as far as possible.
Operation with stand alone plasma pulses	Uncertainties of calculated values of heat loads are large and the real heat loads must be measured by using operating patterns with long dwell time between the plasma pulses.
Re-cooling of magnet system after fast energy discharge or quench	Time for re-cooling of the magnet system must be minimized (about 4 days).
Re-cooling after plasma disruption. (No overheating of the coils is expected for plasma disruption conditions.)	An additional time interval of about 3,000 s is expected after plasma disruption, in order to restore the cooling of the magnet system, before continuing with nominal plasma pulsing.
Standby mode to maintain tokamak thermal shields at 80K if one of two identical main cryolines, CTCBs, LN2 units, 80K helium boxes are out of service (allows repair one set of above identical cryogenic components)	Cool-down duration of ITER machine from 300K to 4.5K is long (about one month). It is why the design strategy is to have operational redundancy for maintaining of the thermal shields at 80K as long as possible.
Cool-down and nominal cooling of 8 torus and 2 cryostat cryopumps before cool-down of magnet system	This mode is used for pump out of VV and cryostat, and high temperature outgassing of 80K thermal shields.
Gradual cool-down (warm-up) of magnet system from 300K to 80K	Average cool-down (warm-up) rate is 0.5K per hour
Gradual cool-down (warm-up) of tokamak thermal shields from 300K to 80K in parallel with cool-down of magnet system from 200K to 80K	Average cool-down (warm-up) rate of thermal shields is 1K per hour.
Cool-down (warm –up) of magnet system from 80K to 4.5K	Expected duration of this mode is about one week.
Filling of magnet system with large quantity of supercritical helium	Expected duration of this mode is about one day.
Cool-down of NB cryopumps and pellet units after cool-down of magnet system	Expected duration of this mode is about one day.

4.15.2.3 80K He Loop and LN2 Subsystem

4.15.2.3.1 He Loop at 80K

Compressed helium at 80K is used for the forced flow cooling of the 80K thermal shields of the ITER machine. This compressed helium is cooled finally down to 80K by liquid nitrogen from the ITER LN_2 subsystem.

The 80K He cooling loop includes two identical cold He boxes and two LN_2 auxiliary cold boxes. If one 80K He box or LN_2 auxiliary box is removed from service, another box will cool the thermal shields with a higher outlet temperature of 120K.

Each 80K He box contains a counter flow He heat exchanger for cooling the supply He flow from 310K to about 103K by a return He flow heated from 100K to 307K. Final cooling of the He flow from 103K to 80K performs in the LN2 auxiliary cold box. The requirements for the above heat exchangers are given in Table 0.1-4. The counter flow heat exchangers are of conventional design, such as for the LN2 precooling stage of the LHe plant modules.

Table 4.15-5 Requirements for two heat exchangers of the 80K He loop

Components	He flow (kg/s)	ΔT for component (K)
80 K thermal shields of magnet system	4.5 (overestimated)	20 (nominal) 40 (baking)
Thermal shields and baffles of cryopumps	3.5 (TBC)	< 10 (nominal) 10 (regeneration)
300-80Kcounter- flow heat exchangers of He gas process cold box	9.0	3 (recuperation between supply and return flows)

The main requirements for the warm compressors of the 80K are shown in Table 4.15-6**.** Due to usage a high inlet pressure of 1.6 MPa and relatively small pressure drop for the 80K helium loop of 0.2 MPa the electric power supply of the He warm compressors is modest.

Table 4.15-6 Requirements for the warm compressors of the 80K He loop

80K helium loop	Total He flow (kg/s)	Inlet pressure (MPa)	Outlet pressure (MPa)
Warm high-pressure compressors	8.0	1.6	1.8

4.15.2.3.2 LN2 Plant

The LN_2 production subsystem together with the 80K He loop provides cooling for the following ITER components:

- 80K thermal shields inside the tokamak cryostat;
- gravity supports of the magnet system and the vacuum vessel;
- 80K chevron baffles and thermal shields of the cryopumps;
- 80K thermal shields of all cryogenic transfer lines and cryo-manifolds;
- pre-cooling the He flow to 80K as required for the cryogenic cycle of the LHe plant;
- external He gas purification unit.

As shown in Table 4.15-7, the LN_2 subsystem will operate close to the refrigeration mode. The main cryogenic users are the 80K shields of the magnet system and vacuum vessel. The detailed heat load requirements for the cryopumps and pellet units are listed in Table 4.15-8.

Table 4.15-7 Specification for the LN2 plant

Components	Nominal plasma operation	VV baking
80K shields and cryogenic gravity support of magnet system (MS) and vacuum vessel (VV)	270-kW (TBC)	490 kW (TBC)
80K shields of all cryogenic lines	50 kW	50 kW
80K shields/baffles all of cryopumps	152 kW + 0.06 kg/s	70-kW (TBC)
Subtotal capacity of LN_2 plant for all above components, including heat losses in heat exchangers of the 80K He loop	565 kW + 0.1 kg/s	700 kW
LN_2 for supplying the LHe plant for the option with HTS current leads	480 kW (TBC)	280 kW (TBC)
LN_2 for supplying the external He gas purification unit	30 kW + 0.05 kg/s	30 kW + 0.05 kg/s
Total cooling capacity of the LN_2 plant	1075-kW + 0.2 kg/s	1010-kW + 0.05 kg/s
Equivalent total refrigeration capacity	1125-kW	1025-kW

Two LN_2 refrigeration units each of 600 kW are incorporated in the cryoplant. These units are of conventional design, such as used in air separation plants. The LN_2 subsystem includes also two LN_2 tanks, to improve operation flexibility of the cryoplant.

Table 4.15-8 Requirements for cryopumps and pellet units at 80K

		8 Torus cryopumps	**4 NB cryopumps**	**2 Cryostat cryopumps**	**Pellet units**
Refrigeration heat load for nominal pumping	kW	18	84	6 (TBC)	TBD
Refrigeration heat load for cryolines and CVBs	kW	1.5	0.4	0.2	TBD
Refrigeration load during regeneration	kW	40 +18	30 + 52	From 0 to 22 (TBC)	TBD
He flow rate for nominal pumping and regeneration	kg/s	1.2	1.9	0.4	TBD
Temperature increase of He flow at 80K	K	10 (nom.) 10 (reg.)	10 (nom.) 10 (reg.)	3 4 (nom.) 10 (reg.)	TBD
Total for LN_2 plant:					
Average refrigeration	kW	59.5	86	6.3	TBD
Average liquefaction	kg/s	0.0	0.06	0.02 (TBC)	TBD
Heat load during cool-down from 300K to 80K	kW	60	TBD	40 (TBC)	TBD
He flow during cool-down	kg/s	From 0.5 to 1.0	TBD	From 0.1 to 0.2 (TBC)	TBD

4.15.2.4 Cryodistribution System

4.15.2.4.1 Cryodistribution Arrangement

The main function of the cryodistribution system is the forced flow cooling of the coils and the cryopumps by using cold helium circulating pumps. As shown in Table4.15-9 the helium flows requested for the magnet system and the cryopumps are much larger than the helium flow of about 2 kg/s produced by the main compressors of the LHe plant itself.

Table4.15-9 Requirements for Cold Circulating Pumps

Component	He Flow (kg/s)	Pressure Drop (MPa)	Heat load for SHEXs of ACBs (kW)
Pumps of TF cases and structures	2.5	0.03	12.5
Pumps of TF winding packs	2.0	0.12	12
Pumps of PF and correction coils	1.8 to 2.0	0.065	4.3
Pumps of central solenoid	2.0	0.12	5.6
Pumps all of cryopumps	2.7	0.05	5.5

The helium circulating pumps are located in the following five separate auxiliary cold boxes (ACBs) of each individual cryogenic ITER user:

- ACB-1 for forced flow cooling of the central solenoid (CS);
- ACB-2 for forced flow cooling of the 18 TF winding packs;
- ACB-3 for forced flow cooling of the TF coil cases and intercoil structures;
- ACB-4 for forced flow cooling of the 6 poloidal field coils (PF) together with the correction coils (CC);
- ACB-5 for forced flow cooling all of the cryopumps.

Each ACB contains a supercritical helium circulating pump~~s,~~ a bath with boiling liquid helium, a supercritical helium flow heat exchanger (SHEX) and a set of control valves, including a helium flow bypass valve for smoothing the pulsed heat load to the LHe plant. The heat load requirements for the SHEXs of different ACBs are included in the table above.

Cold compressors are incorporated in the cryodistribution system for maintaining a temperature of 4.2 K in the boiling He baths all of the ACBs of the magnet system. The specification for the cold compressors is given in the following table.

Table4.15-10 Specification for cold compressors

	Total He flow (kg/s)	Head of compressor (MPa)
Helium vapour compressors	2.0	0.04

All cold boxes of the cryodistribution system are located inside the main tokamak building. Each ACB is connected through the cryogenic transfer lines and ring manifolds with the cold termination boxes (CTB) or cold valve boxes (CVBs) of each individual cryogenic user.

4.15.2.5 System of Cryogenic Lines and Manifolds

4.15.2.5.1 Cryoline Systems

The system of the cryogenic transfer lines and manifolds that is located inside the tokamak building includes the following components:

- a common manifold for connecting together the ACBs, and CCB of the cryodistribution system;
- two cryolines and two half ring manifolds for supplying the nine CTBs of the TF coils;
- two cryolines and two manifolds for supplying the 11 CTBs of the PF coils and correction coils;
- cryoline and two manifolds for supplying the 6 CTBs of the CS;
- two cryolines and a short manifold for supplying the 3 CVBs of the magnet structures;
- two parallel cryolines for supplying the 2 CVBs of the 80K thermal shields;
- cryoline and manifold for supplying the 4 CVBs of the NB cryopumps;
- two cryolines and two half ring manifolds for supplying the 8 CVBs of the torus cryopumps and 2 CVBs of the cryostat cryopumps,
- short cryolines to connect the ACB of the magnets LHe tank~~s~~.

There are also the two long cryolines connecting the cold process boxes of the cryoplant with the two CTCBs in the main tokamak building.

In order to minimize the total number of the cryogenic transfer lines each manifold (or cryoline) contains five (for the coils) or six (for the cryopumps) cold helium tubes. Each manifold includes also bellows for allowing free thermal contraction of the cold tubes, fixed and sliding supports, and a thermal shield including multilayer superinsulation.

The manifolds will be built from several types of typical sections, in particular straight pipe sections and curved sections installed in such a way as to form a ring-shaped manifold. Each manifold includes also several tee-sections for connecting branched cold tubes from the manifold to the CTBs or CVBs of each individual ITER component.

The cryolines and ring-shaped manifolds are located around the tokamak cryostat on three different levels of the tokamak building, in particular the TCWS vault level (upper CTB level), the basemat level (lower CTB level) and the lower pipe chase level. The design of the cryolines and manifolds will be based on the current manufacturing technology available on the market.

4.16 Pulsed and Steady State Power Supplies (WBS 4.1,4.2,4.3)

(Update Nov 2006, I.Benfatto)

4.16.1 Functional Requirements

4.16.1.1 Pulsed Power Supply System

The main operational functions of the Pulsed Power Electrical Network (PPEN) are to provide:

- pulsed power for energizing the TF, PF, CS and correction coils to generate, confine and control the plasma current;
- pulsed power to the additional Heating and Current Drives (H&CD) systems (IC, EC, LH and NB systems) for the plasma heating and current drive.

This system protects:

- the super-conducting coils against fast discharges in case of quench;
- the coils against over-voltages or/and over-currents due to abnormal or fault operation of power supplies or in case of plasma current disruption;
- the machine structures against high induced voltages due to electrical ground faults, by grounding the magnet system components and monitoring ground currents.

4.16.1.2 Steady State Power Supply System

The main operational functions of the Steady State Electrical Power Network (SSEPN) are to provide:

- steady-state power, in class IV, to all the electrical loads of the ITER plant auxiliary and utility systems during normal operation, from the power grid;
- steady-state power, in class I, II and III, to the Safety Related and Investment Protection loads of the ITER plant auxiliary systems in case of unavailability of the class IV network; from diesel motor generators.

The SSEPN supplies the class I and II AC power to plant loads, from class III or IV power through AC/DC/AC inverters associated to batteries.

4.16.1.2.1 Power Supply Classification

The SSEPN is made of several electrical power circuits according to voltage levels and voltage classes. The voltage class defines the degree of availability of the power delivery. Four voltage classes are defined as follows:

Table 4.16-1 Power Supply Classification

Class	Required Reliability
Class I	Uninterruptible DC
Class II	Uninterruptible AC
Class III	Temporarily interruptible AC
Class IV	Indefinitely Interruptible

4.16.2 Configuration

The pulsed and steady-state power supplies consist of the following four major systems:

- pulsed power distribution system;
- coil power supplies;
- heating and current drive (H&CD) power supplies (PS);
- steady state electric power network (SSEPN).

4.16.2.1 Pulsed Power Distribution System

4.16.2.1.1 Pulsed Power Demand

The active and reactive power limits reported in the following section are based on the support studies performed in 2002-03 for the preparation of the Cadrache site proposal. The limits will be reviewed during the negotiation of the contract for the electricy supply of ITER.

The total peak pulsed power demand will be limited to 500 MW active power and 180 Mvar reactive power. This includes power required for the pre-programmed PF scenarios, power needed for the plasma current, position and shape control, including the vertical stabilisation, and power to supply the H&CD systems.

The active power demand will be limited in derivative to 200 MW/s, with maximum step changes (i.e when power derivative is higher than 250 MW/s) of 60 MW in normal operation.

4.16.2.1.2 Pulsed Power Network Connection

AC power is assumed to be received from the HV grid at 400 kV voltage and is transformed to an intermediate level (69 kV) via 3 step-down transformers, each rated at 300 MVA continuous power.

All loads (ac/dc converters for the TF coils, CS, PF coils and CCs, and H&CD systems) are shared among the three 69 kV busbars as equally as possible. Most of these loads are directly supplied from the 69 kV busbars.

The loads with relatively lower power per unit (normally less than 20 MVA) are connected to the medium, 22 kV, busbars, which are powered from the 69 kV busbars through 3 step-down transformers, 50 MVA power each.

A reactive power compensation unit, including harmonic reactive power and high frequency filters, is connected to each 69 kV busbar to compensate for the reactive power produced by the ac/dc converters of the Coil and the Heating and Current Drive powers ssupplies. The rated power of each unit is expected in the range between 200 to 300 Mvar, depending on the final configuration of the pulsed power supplies and the value of reactive power acceptable by the 400 kV grid.

The following two tables refer to the 2001 baseline design and will be revised during the site adaptation design that is planned during 2007-2008. Substantial changes are expected mainly in the design ratngs and characteristics of the Filters described in Table 4.16-3

Table 4.16-2 Power Loads (DDD 4.1)

	Max. power at ramp-up		**Power consumption during burn**		**Power consumption during burn**	
	73 MW heating		73 MW heating		110 MW heating	
	P (MW)	Q (Mvar)	P (MW)	Q (Mvar)	P (MW)	Q (Mvar)
1. PF Scenario	80	600	30	640	30	640
2. Heating	205	170	205	170	335	290
NB H&CD	114	100	114	100		
EC H&CD	57	50	57	50		
IC H&CD	34	20	34	20		
3. Losses	25	20	25	20	25	20
4. Correction	10	10	10	10	10	10
5. PF control	100	0	100	0	100	0
TOTAL	**420**	**800** (*)	**370**	**840** (*)	**500**	**960** (*)

(*) The value is without the rective power compensation system.

Table 4.16-3 Main Parameters of the Filters (DDD 4.1)

Parameter	**Unit**	**Filter circuit**					
Order of harmonic		3	5	7	11	13	22
Reactive power (at 50 Hz)	Mvar	20	32	32	40	40	16
Inductance	mH	94.7	19.7	9.9	3.1	2.2	1.9
Capacitance	μF	11.9	20.5	20.9	26.5	26.6	10.7
Resistance	Ω	268	-	-	-	-	40.0
Quality factor		2	50	50	50	50	2

4.16.2.2 Coil Power Supplies and Fast Discharge

4.16.2.2.1 Coil Power Supplies

These consist of:

- One common power supply system for the 18 TF coils;
- Five PS systems for the CS modules: 4 individual PS systems for CS2 and CS3 upper and lower modules, and one common PS system for the CS1 upper and lower modules connected in series;
- Two PS systems for individual supply of the PF1 and PF6 coils;
- Two systems for plasma vertical stabilisation: VS1 for the 4 outer PF coils, PF2 - PF5, and VS2 for the CS2U, CS2L coils
- Six relatively small PS systems with identical configuration to supply the 6 upper and lower correction coils (CCs)
- Three PS systems for resistive wall mode control and error field correction with the side correction coils

The VS2 and RWM systems are not required simultaneously. Therfore, they use common power converter units.

4.16.2.2.2 TF Fast Discharge

The TF coils are normally connected in series in 9 series pairings of adjacent coils. The minimum normal discharge time is ~ 30 minutes.

See Table 4.1-14 for TF fast discharge parameters.

For each fast discharge unit there are two breakers in series: a multi-use current commutation unit and (in case of failure) a single-use highly reliable pyrobreaker. These fast discharge units plus a fast make switch are located in the tokamak building gallery near the coil terminal boxes. The discharge resistors,and the counterpulse capacitor banks to zero the current commutation units, are located in the diagnostics building.

4.16.2.2.3 Plasma Breakdown /Initiation

The loop voltage required for breakdown and plasma initiation is obtained by connecting resistors in series with the CS modules, PF1 and PF6 coils, causing a very large amount of power (about 2 GW) to be extracted. These circuits, called switching networks, are made up of circuit breakers, make switches and resistors.

After plasma initiation, the power required for plasma current, shape, and position control is provided by one 12-pulse, 4-quadrant thyristor converter, rated for 45 kA continuous current, 2 kV no-load voltage.

4.16.2.2.4 CS/PF Fast Discharge

The fast discharge units for CS and PF coil energy discharge are similar to those used for TF coil protection, with resistor values adapted to the discharge parameters.

See Table 4.1-14 for CS/PF fast discharge parameters.

4.16.2.2.5 Correction Coil Power Supplies

The 6 upper and lower CC power supplies have the same, simple, configuration: an ac/dc converter supplies dc power to each pair of coils. Acting in inversion mode they also provide fast discharge of the coil energy in case of quench.

If the converter fails to perform this action, it will be bridged by a parallel-connected make switch, and the coil energy will be dissipated in the busbars with a time constant less than 20 s (backup protection).

The CC coil capability is shown in Table 4.1-12 Parameters for CC coils.

4.16.2.2.6 Grounding for Coils

See also Section 3.9

A soft grounding via high impedance (1-2 kΩ) resistors is provided for all the coils at the power supply. The leakage current to ground will be measured and used for ground fault detection.

Table 4.16-4 Magnet Electrical Interfaces

Parameters	Unit	Value
Maximum TF magnet conductor current	kA	68
TF voltage to ground for slow (30min) charge and discharge	KV	± 0.5
Maximum TF voltage during normal operation (across current leads of two coils connected in series, but not including transient spike)	kV	7.1
Maximum CS current	kA	45
Maximum CS voltage (between current leads) for normal operation	kV	12
Maximum PF conductor current (normal/backup)(2)	kA	45/52
Maximum PF terminal voltage for normal operation (PF1, PF6) including fast discharge	kV	12
Maximum PF terminal voltage for normal operation (PF2 to PF5) including fast discharge	kV	16
Maximum CC conductor current	kA	10
Voltage (per coil) available from SNUs for for breakdown CS modules, PF1, PF6	kV	10
Voltage (per coil) available for PF scenario & slow control CS modules, PF1, PF6	kV	1.5-2
Voltage (per coil) available for breakdown PF2 to PF5	kV	5.6
Voltage (per coil) available for PF scenario & slow control PF2, PF5	kV	1.5-2
Voltage (per coil) available for fast control (for vertical stabilisation), PF2 to PF5	kV	6-8
Note: (1) Voltages indicated are not the normal operation voltages but maximum values to be used for interface purposes between the power supplies and the magnets. (2) PF "backup" is for the case of failure of one pancake. (see also 4.1.2.1.7)		

The following tables report the main interface parameters for the fast discharge units operated in case of coil quench. From the hot spot point the equivalent exponential constants include the assumption of a 2s delay from the start of the quench until the electrical discharge starts (which is equivalent thermally to adding 4s to the electrical constant). It is assumed (thermally) that the form of the current decay is exponential (i.e. constant resistors). However, the fast discharge units are equipped with non-linear resistors so that the electrical discharge is not exponential. The quoted coil voltages (and used as insulation specification) refer to the actual resistors used.

Table 4.16-5 TF Coil Discharge Design Data

Parameter	Unit	Value
Coil rated current	kA	See Table 4.1-9
Total stored energy	GJ	See Table 4.1-9
Equivalent time constant of fast discharge	s	See Table 4.1-14
Energy dissipated in discharge resistors	GJ	34.4
Number of fast discharge units		9
Max. voltage at fast discharge (across current leads)	kV	7.1(1)
Max. voltage-to-ground in normal operation	kV	3.55(2)
Slow discharge duration	min	25
Voltage of ac/dc converter at slow discharge	kV	0.7 to 0.8

(1) Voltage surge (of a few ms) caused by the discharge of the counterpulse capacitors may reach about 8 kV;

(2) The voltage surge up to 5 kV/5 ms is possible due to non-simultaneous operation of the circuit breakers.

Table 4.16-6 CS, PF1 and PF6 Coil Discharge Data

Parameter	Unit	CS1 U&L	CS2 U/L	CS3 U/L	PF1	PF6
Coil rated current	kA	see 4.1.2.6				
Max. energy to be dissipated	GJ	1.14	1.15	1.0	1.0	2.1
Equivalent time constant (from Table 4.1-14) (1)	s	7.5	7.5	7.5	14	14
Number of fast discharge units		2	1	1	1	1
Max. voltage at fast discharge (across current leads)(2)	kV	4.6	4.6	4.1	2.5	5.0
Max. voltage at fast discharge to ground in normal operation	kV	2.3	2.3	2.1	1.25	2.5

(1)$\int i^2 dt$ is the same as for an exponential current decay with the shown time constant;
(2) Voltage surge (of a few ms) caused by the discharge of the counterpulse capacitors may reach about 7 kV.

Table 4.16-7 PF2 through PF5 Coil Discharge Data

Parameter	Unit	PF2	PF3	PF4	PF5
Coil rated current	kA	45	45	45	45
Max. energy to be dissipated	GJ	0.42	1.8	1.5	1.6
Equivalent time constant (from Table 4.1-14) (1)	s	14	14	14	14
Max. voltage at fast discharge (across current leads)(2)	kV	0.9	3.6	3.0	3.2
Max. voltage at fast discharge to ground	kV	0.45	1.8	1.5	1.6

(1)$\int i^2 dt$ is the same as for an exponential current decay with the shown time constant;
(2) Voltage surge (of a few ms) caused by the discharge of the counterpulse capacitors may reach about 7 kV.

4.16.2.3 Heating &Current Drive Power Supplies

(This section refers to the 2001 baseline design and it is expected to be updated in 2007.)

The H&CD power supply (PS) systems supply dc or ac controlled voltages and/or currents to:

- the anodes and driver stages of the IC H&CD tetrodes;
- the cathodes, anodes and bodies of the EC H&CD gyrotrons;
- the collectors of the LH H&CD klystrons;
- the acceleration grids and auxiliaries of the NB H&CD and DNB injectors.

The ac power will be supplied from the pulsed power distribution system at the 69 kV or 22 kV busbars, according to the level of power required for their operation.

The power delivered to the plasma can be upgraded from the initial 73 MW to 110 MW, by increasing the number of IC, EC or NB units and/or by adding the LH system (see Table 2.2-1). The total peak power demand on the pulsed power system is 210 MW (330 MW with upgrade).

4.16.2.3.1 IC H&CD Power Supplies

The IC system uses individual power supply (PS) units for the tetrodes of the IC H&CD, based on "pulse step modulator" (PSM) technology. This has been adopted to control tetrode voltage with high accuracy. 36 separate voltage steps can be electronically switched in and out of the circuit.

Moreover, pulse width modulation, with an overall effective frequency per PS unit of 90 kHz, is employed to regulate the voltage with more accuracy, to smooth the 12-pulse rectification ripple, and to provide required 200 Hz modulation of the anode voltage.

Load protection is accomplished by an insulated gate bipolar transistor, which is part of the switched PS module; all modules are switched off in less than 10 μs, limiting the energy dissipated in a fault to 10 J.

Table 4.16-8 HVDC Requirements for IC H&CD System

Parameters	Unit	Value
Total power to the plasma (for PS interface definition)	MW	20
Number of power supply units	–	8
Number of generators/sources supplied by each power supply unit	–	1
Anode voltage range	kV	5 – 26
Anode modulation bandwidth	Hz	200
Accuracy of the anode voltage control (% of max. voltage)	%	± 1
Ripple of anode voltage (% of max. voltage)	%	± 1
Anode maximum current	A	150
Driver stage voltage range	kV	3 – 18
Accuracy of the Driver stage voltage control (% of max. voltage)	%	± 1
Ripple of driver stage voltage (% of max. voltage)	%	± 1
Driver stage maximum current	A	25
IC load protection system		
Fault energy (short circuit energy in case of load fault)	J	≤ 10
Response time of the load protection system	μs	≤ 10
Time to be ready for restart	ms	≤ 200
Rise time of the output voltage	ms	≤ 50

4.16.2.3.2 EC H&CD Power Supplies

EC system gyrotrons use two types of HV dc power sources to supply their cathode, anode and body.

The main, cathode, PS consists of 2 units each feeding six pairs of gyrotrons with a controlled dc voltage between cathode and collector. Another PS (one for each gyrotron) supplies a controlled and modulated (1 kHz) voltage between body and cathode. Part of this voltage is derived between anode and cathode, through a resistive voltage divider.

Load protection is accomplished by the insulated gate bipolar transistor switch, which is in series with the faulty gyrotron. This acts in less than 10 μs, limiting the energy dissipated in a fault to 10 J.

Table 4.16-9 HVDC Requirements for EC H&CD System

Parameters	Unit	Value
Total power to the plasma (for PS interface definition)	MW	20
Number of power supply units	–	2
Number of generators/sources supplied by each power supply unit	–	12
Nominal cathode voltage	kV	-50
Nominal cathode current (at –50 kV)	A	45
Cathode voltage range	kV	-45 to -55
Accuracy of Cathode voltage regulation	%	± 1
Cathode voltage ripple, overshoot and undershoot (% of max. voltage)	%	2
Acceleration (body) voltage range	kV	0 to +45
Maximum body-to-cathode voltage	kV	90
Accuracy of acceleration voltage control (% of max. voltage)	%	± 0.5
Acceleration voltage ripple (% of max. voltage)	%	± 0.5
Maximum acceleration current (per tube)	A	0.1
Acceleration voltage Modulation range	%	20 to 100
Maximum Acceleration Voltage modulation frequency	kHz	1
Anode voltage range	kV	0 to –50
Anode voltage control: by resistive voltage division from cathode voltage		
Fault energy (short circuit energy in case of load fault)	J	≤ 10
Response time of the load protection system	μs	≤ 10
Time to be ready for restart	ms	~ 200
Rise time of the cathode voltage	ms	1

4.16.2.3.3 LH H&CD Power Supply

For LH, one PS unit feeds six pairs of klystrons with a controlled dc voltage between collector and anode, with a scheme similar to that for EC H&CD.

Load protection is accomplished by an insulated gate bipolar transistor switch, which is in series to the faulty klystron. This acts in less than 10 μs, limiting the energy dissipated in the fault to 10 J.

Table 4.16-10 HVDC Requirements for LH H&CD System

Parameters	Unit	Value
Total power to the plasma (for PS interface definition)	MW	20
Number of power supply units	–	2
Number of generators/sources supplied by each power supply unit	–	12
Collector voltage of the main supply	kV	80
Collector maximum current	A	25
Accuracy of the voltage control (% of max. voltage)	%	± 1
Ripple of voltage, overshoot and undershoot (% of max. voltage)	%	± 2
Fault energy (short circuit energy in case of load fault)	J	≤ 10
Response time of the load protection system	μs	≤ 10
Time to be ready for restart	ms	~ 200
Rise time of the output voltage	ms	1

4.16.2.3.4 NB H&CD Power Supply

One NB PS system feeds one NB injector. The PS system is split into three parts. An acceleration PS provides a negative voltage (up to 1 MV to ground) to the beam source, together with intermediate voltages to the acceleration grids. An ion source PS provides different voltages and current to several items of equipment in the HV deck (SF_6-insulated). A third group of power supplies provides individual dc voltages to seven active correction/compensation coils and to the residual ion dump.

Frequency and voltage conversion is provided by gate-turn-off thyristor inverters and voltage transformation is provided by five transformers, connected in series, with 1 MV insulation between primary and secondary windings.

The connection between power supply and beam source is done by a multi-axial, SF_6-insulated, HV transmission line. The line is split in two parts by the HV deck, where the ion source power supply is located.

Load protection is obtained by fast switch-off of the gate-turn-off thyristor inverters. These act in less than 200 μs, limiting the energy dissipated in a fault to 50 J.

The DNB PS system is split into three parts, with a structure very similar to the NB PS. The main differences are that only one voltage (- 100 kV) is provided to the beam source and only five active correction/compensation coils are used.

Table 4.16-11 HVDC Requirements for NB H&CD System

Parameters	Unit	Value
Total power to the plasma (for PS interface definition)	MW	33
Total power from the AC supply	MW	115
Number of power supply units	–	2
Number of generators/sources supplied by each power supply unit	–	1
Output voltage of the main supply	kV	400-1000
Output current of one power supply unit	A	59
Accuracy of the voltage control (% of max. voltage)	%	± 2
Ripple of voltage (% of max. voltage)	%	± 5
Total voltage/current	kV/A	1000/59
Grid 1 voltage/current	kV/A	800/7
Grid 2 voltage/current	kV/A	600/6
Grid 3 voltage/current	kV/A	400/3
Grid 4 voltage/current	kV/A	200/3
Maximum continuous power per unit	MW	48
Auxiliary power referenced to 1MV per unit	MW	3
Auxiliary power referenced to ground per unit	MW	3
DNB acceleration voltage	kV	100
DNB acceleration current	A	71
Fault energy (short circuit energy in case of load fault)	J	≤ 50
Response time of the load protection system	μs	≤ 200

4.16.2.4 Steady State Power Supply

4.16.2.4.1 Steady State Power Demand

The SSEPN will provide ac power to all plant electric loads, except those supplied by the pulsed power distribution system. In case of a loss of off-site power, backup, autonomous power generators will be used. Loads that would not tolerate the 90 s interruption needed to start up the backup generators and to reconnect the loads, will get power from ac class II uninterruptible power supplies for loads of unit power lower than 20 kW, or from dc class I uninterruptible power supplies. The ac class II power delivery service will be centralized in a few load centres. The ac class II and the dc class I power delivery service will be centralized in several load centres, or localized as part of the individual systems requiring them.

4.16.2.4.2 SSEPN Network Connection

The SSEPN will receive up to 120 MW continuous power from an HV grid through two independent 400 kV transmission lines, each capable of supplying the entire plant maximum load. The grid power is then delivered to the loads at two voltage levels: 6.6 kV Medium Voltage (MV) and in 400/230 V Low Voltage (LV), through step down transformers. The most powerful loads over 200 kW unit power, except the powerful water heaters, are fed in MV 6.6 kV from the 6.6 kV switchgears in class III and IV. Other loads of power lower than 200 kW, including the powerful water heaters, are fed in LV 400/230 V through 6.6/0.4 kV transformers located in several load centres, spread over the ITER site.

4.16.2.4.3 Emergency Power Supplies

Emergency backup power will be generated by two diesel generators, each rated for 6.3 MW with a minimum power factor of 0.85. These generators are connected to 2 separate 6.6 kV busbars for supplying Class III (i.e. temporarily interruptible AC) power to the loads, most of which are safety classified. The two redundant circuits or trains for the supply of the safety-related loads, starting from the diesel generators and including switchgear components, transformers and cables, are entirely segregated from each other.

Table 4.16-12 Installed Power P0 (kW) per WBS system, voltage and class

(This table will be revised in 2007.)

WBS systems		0.4 kV			6.6 kV		Total P0
Names	**Codes**	**Class II**	**Class III**	**Class IV**	**Class III**	**Class IV**	
Magnet	1.1	164	0	0	0	0	164
Vacuum Vessel	1.5	5	0	0	0	0	5
Blanket	1.6	5	0	0	0	0	5
Divertor	1.7	5	0	0	0	0	5
Assembly tooling	2.2	0	0	0	0	0	0
Remote Handling Equipment	2.3	90	0	726	0	0	816
Cryostat	2.4	5	0	0	0	0	5
Cooling Water System	2.6	29	524	15671	4400	48160	68784
Thermal shields	2.7	1	0	1616	0	0	1617
Vacuum Pumping & Fuelling System	3.1	44	520	1055	0	0	1619
Tritium Plant & Detritation System	3.2	49	4089	2271	0	2800	9210
Cryo-plant and cryo-distribution	3.4	61	0	1998	0	30860	32919
Pulsed Power Supplies	4.1	251	32	1823	0	0	2105
RF Heating & Current drive	4.2	38	50	130	0	0	218
Steady State Electrical Power Network	4.3	10	288	305	0	0	602
General control & Instrumentation	4.5	246	0	0	0	0	246
Ion Cyclotron Heating & Current Drive System	5.1	6	0	2200	0	0	2206
Electron Cyclotron Heating & Current Drive System	5.2	8	0	510	0	0	518
Neutral Beam Heating & Current Drive System	5.3	5	0	0	0	0	5
Lower Hybrid Heating & Current Drive System	5.4	0	0	0	0	0	0
Diagnostic systems	5.5	301	0	423	0	0	724
Test blanket module (included in WBS 2.6)	5.6	0	0	0	0	0	0
Building HVAC & Fire detection	6.2	23	992	16582	0	0	17597
Hot Cell Processing & Waste Treatment System	6.3	36	0	1034	0	0	1070
Radiological & Environmental Monitoring	6.4	199	0	138	0	0	337
Liquid Distribution System	6.5	2	0	261	1000	0	1263
Gas Distribution System	6.6	2	0	139	0	1450	1591
Total:……		1582	6495	46881	5400	83270	**143628**
Total class III in 6.6kV (trains A & B)		**13477**					

4.17 CODAC, Interlock and Safety Systems (WBS 4.5, 4.6)

CODAC = COntrol, Data Access and Communication

Editors Note: This section has been revised (J.Lister) in 2006 and currently represents an intermediate status from June 2006. A final revised version is planned for December 2006.

4.17.1 Scope

ITER operation requires coordinating the activity of 80-120 Plant Systems, procured "in kind" from the Participant Teams, including the technical systems (such as cryoplant, vacuum, power supplies, tritium plant, HVAC, personnel access, radiation monitoring) as well as the plasma diagnostic systems. This coordination is guaranteed by three clearly separated tiers. These three tiers are: CODAC, Interlock Systems and Safety Systems. This is illustrated in Figure 4.17-1.

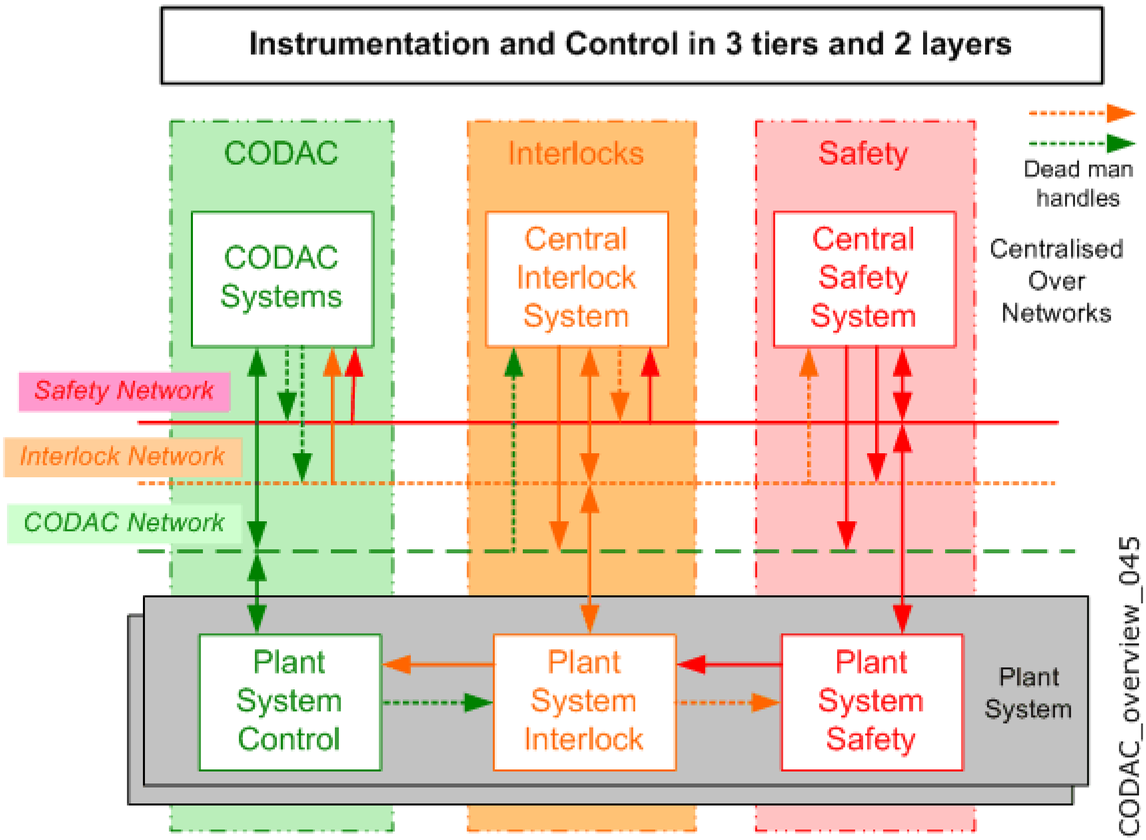


Figure 4.17-1 Three tiers provided by CODAC, the Interlock System and the Safety System.

CODAC provides the COntrol, Data Access and Communication functions for ITER, allowing the integrated operation of all the ITER project equipment. This includes: continuously monitoring the Plant Systems; displaying their status to operators; preparing and automating scheduled operations (including plasma pulses); recovering data from Plant Systems; storing and making all the experimental data available.

The Interlock Systems provide protection of investment for ITER. Each Plant System may have a Plant Interlock System to constrain its own operation, provided as part of its procurement. A Central Interlock System provides interlocking between multiple Plant Systems, where the uncoordinated operation of several Plant Systems can cause potentially damaging interactions. The signal sources, networks and logic will have a higher degree of reliability and availability than CODAC.

The Safety Systems provide protection of personnel and the environment during the operation of the ITER project. The Safety Systems can shut down plasma operation and can inhibit access to potentially dangerous areas. The Safety Systems have the highest level of reliability and availability, provided by redundancy and proof of functionality, appropriate to the ITER safety case. The Safety Systems use a very small number of signals. These are not derived from CODAC.

Separating the surveillance and operation of ITER into these three tiers is a major design feature, allowing CODAC to be inevitably formally unprovable due to its complexity. The Safety Systems are the only systems which are subject to licensing restrictions and must be demonstrably safe. All instrumentation and control must clearly fall into one or other of these categories and comply with the corresponding standards.

The three tiers are inter-related by a set of **Operating Limits and Conditions** (OLC). A small set of these OLC corresponds to the Safety requirements. Some are designed to protect investment. Some are used to restrict normal operation within these protection limits. One aim of CODAC is to use all the OLC to avoid triggering the Interlock Systems and in turn, it is the aim of the Interlock Systems to avoid triggering the Safety Systems, providing some defence in depth.

Plant Systems will be maintained by the ITER project after final acceptance. To reduce the number of technologies and methodologies to be guaranteed by the ITER team, the Plant Systems are subject to a set of standards and methods for each of these three tiers. Each Plant System Instrumentation and Control implementation must conform to these standards. Any non-compliance with these standards has to be formally agreed with CODAC and the deviations must be fully documented.

Each of the three tiers is distributed over the ITER site and relies on a multiplicity of networks; Figure 4.17-2.

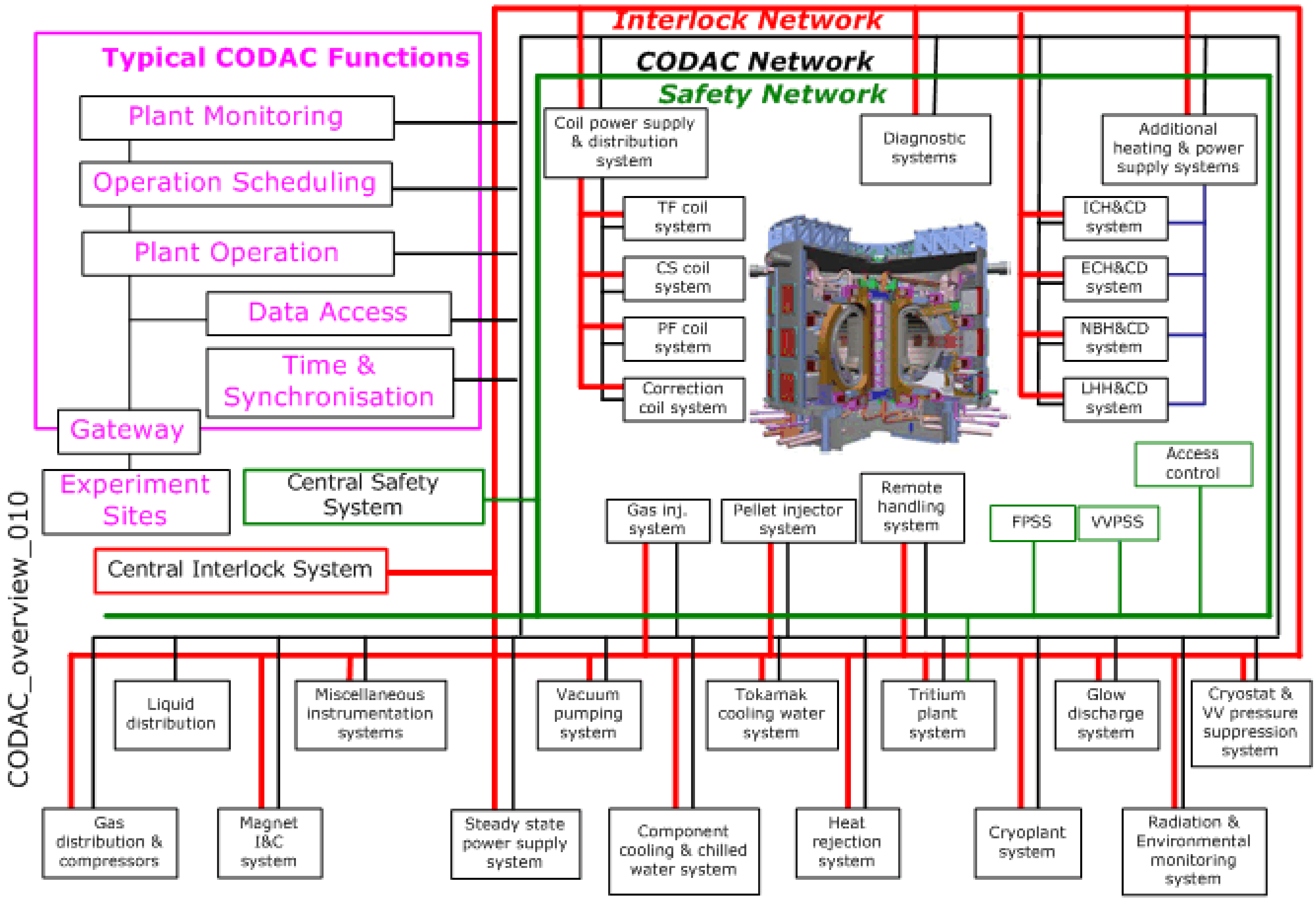


Figure 4.17-2 Plant Systems connected to CODAC, CIS and CSS

with separated networks (CIS = Central Interlock System, CSS=Central Safety System)

4.17.2 CODAC Functional Requirements

The clearly identified functional requirements on the CODAC tier are outlined in Table 4.17-1, which indicates the origins of these requirements and also indicates the implementation within the CODAC Systems. Functional requirements express the functions required of CODAC without reference to methods or performance properties.

Table 4.17-1 CODAC Functional Requirements

Requirement – CODAC shall…	Origin and implementation
... provide networks to communicate information between all Plant Systems and CODAC Systems. (Wide Area Networking is outside CODAC).	Distributed nature of ITER. CODAC Networks.
... monitor the ITER plant and display the status to operator stations.	Support for operators and for automation. Plant Monitoring System; Data Visualisation System.
... provide a method for automating the ITER plant.	Complexity of ITER, ease of scheduling, reduction of operator errors. Plant Automation System.
... acquire and archive all engineering and scientific data concerning the operation of ITER.	Support for operation and analysis. Data Handling System.
... provide access to all archived data both on-site and remotely.	Support for international operation and analysis. Data Handling System.
... provide methods for specifying and verifying the parameters used during ITER operation, including during plasma pulses.	Support for operation. Schedule Preparation System; Schedule Validation System.
... provide and monitor a project-wide time reference.	Synchronisation of operation. Time Communication Network.
... provide and record audio and video information inside the Plant.	Archival requirements. Data Handling System streams.
... operate ITER in terms of Global Operating States, which are linked to the Operation Limits and Conditions.	OLC methodology. Plant Automation System.
... validate all data and instructions originating outside a protected "Plant Operation Zone".	Support for remote experimentation. Operation Request Gatekeeper.
... provide the infrastructure for controlling plasmas.	Plasma Control requirement. Plasma Control System.
... provide all calculations needed for ITER operation, including verification before and during plasma pulses and diagnostic data evaluation during plasma pulses.	Boundary of CODAC.
... provide features which allow ITER to be efficiently exploited from remote Experiment Sites as well as from the Cadarache Experiment Site.	ITER mandate.
… allow equipment to be operated independently, for commissioning and testing.	Assumed requirement.
… provide features to allow evolution of all systems over the 35 year lifetime of procurement and operation.	Assumed requirement.
… provide features which allow the collaborative research activity to be efficiently executed given the distributed nature of the research teams.	Assumed requirement.
… provide operator consoles which allow the operation of the ITER plant at the engineering level of each plant, and at the operation level of the integrated plant.	Assumed requirement.

4.17.3 CODAC Non-Functional Requirements

The clearly identified non-functional requirements for the CODAC tier are outlined in the Table 4.17-2, together with the origins and implementation of these requirements. Non-functional requirements define the qualities of the system to be produced.

Table 4.17-2 CODAC Non-Functional Requirements

Requirement – CODAC shall…	Origin and implementation
... provide clocked transfer of at least 3000 data values between Plant Systems and CODAC Systems with a latency of 1msec	Best estimate to allow stabilisation of the vertical positional instability; best estimate of required volume of data. Synchronous DataBus Network.
... provide low latency point-to-point communication of events between Plant Systems and CODAC systems, with a maximum latency of 10 microseconds	Best estimate to allow synchronisation of plasma or engineering events. Event Distribution Network.
... provide high reliability of all network communication, resulting in unscheduled down time of less than 1 hour per month.	Proposed target. Communication Network System.
... monitor 1000 operational video channels (not plasma diagnostic channels) and 1000 audio channels for communication within the Plant Operation Zone.	Best estimate. Audio-Video Network
... have a maximum jitter in the time reference of 10 nsec.	Proposed target. Time Communication System plus local reconstruction of high frequency references.
... maintain a guaranteed backup of all archived data in a remote location, protected against evolution of technologies.	Analysis support. Data Handling System.
... provide features of remote experimentation which allow researchers world-wide to participate with equivalent efficiency to researchers on site.	International participation. Operation Request Gatekeeper, remote data access, remote monitoring.
... provide access to all ITER data in a user-friendly way to provide for efficient analysis on-site or off-site.	International participation. Data Access System; Data Visualisation System.
... handle up to 1GByte/sec sustained data flow from Plant Systems during normal operation without interruption.	Technology demonstration. Data Handling System.
... not interrupt the data archiving for more than 10 seconds during scheduled outages.	Proposed allowable. Data Handling System.
... be able to continuously monitor up to 1 million input/output data points and status values during normal operation at a frequency of at least 3 Hz.	Best estimate of volume with a margin; human reaction timescale and proposed target for automation. Plant Monitoring System.
... not interrupt plant monitoring or display for more than 10 seconds during scheduled outages.	Proposed allowable. Plant Monitoring System.
... handle a stream of alarms, providing appropriate filtering, archiving and interpretation.	Operation support.
... provide computing power to perform feedback control during the plasma pulse on a distributed set of processors connected to the Synchronous DataBus.	Plasma control support. Computing Support System.
... provide a user-friendly graphical interface for generating Pulse Schedules and for verifying them, both inside and outside the Plant Operation Zone.	Plasma operation support. Operational Schedule System.
… comprise CODAC Systems which individually respect a set of General CODAC Requirements.	ITER standards on power, seismic, fire, electromagnetic, grounding, insulation, HVAC etc.

4.17.4 Major CODAC Design Features

The major design features easing the implementation of CODAC and the integration of the Plant Systems are outlined in Table 4.17-3.

Table 4.17-3 Major CODAC design features

Design decision	Driving reason
Provide a methodology for accumulating a structured data description of all Plant Systems, including their construction, input/output list and dynamical behaviour. This "self-description" shall be part of each procurement package, using tools provided by CODAC.	Complexity and uniformity to allow data-driven solutions and data-driven integration.
Define a minimum set of acceptable hardware and software standards.	Complexity and "in-kind" procurement, followed by ability to guarantee long-term maintenance.
Impose a restrictive set of ITER-wide message protocols for communicating with Plant Systems.	Uniformity.
Provide a method and support for fully validating each Plant System at the factory.	Procurement tracking and integration.
Maximise the use of structured data in all aspects of its implementation.	Long-term nature of ITER, allowing evolution and maintenance and reducing the generation of code.
Inhibit initiation of communication by Plant Systems with other Plant Systems.	Simplicity and also control of the communication bandwidth.
Protect the Plant Operation Zone against all inappropriate commands via an Operation Request Gatekeeper which shall decide whether any incoming request is to be passed on to the targeted Plant System or CODAC System.	A model for allowing remote experimentation while maintaining access integrity.
Automate Plant Systems in terms of the standard Sequential Function Chart formalism, IEC 61131-3, using the SCXML representation.	Uniformity, clarity and data-driven use.

4.17.5 Principal CODAC features in procured Plant Systems

Plant Systems are procured with specific technical specifications to meet their design purpose and with generic CODAC specifications to allow their integration into CODAC. Plant Systems will have differing degrees of complexity and examples are shown in Figure 4.17-3.

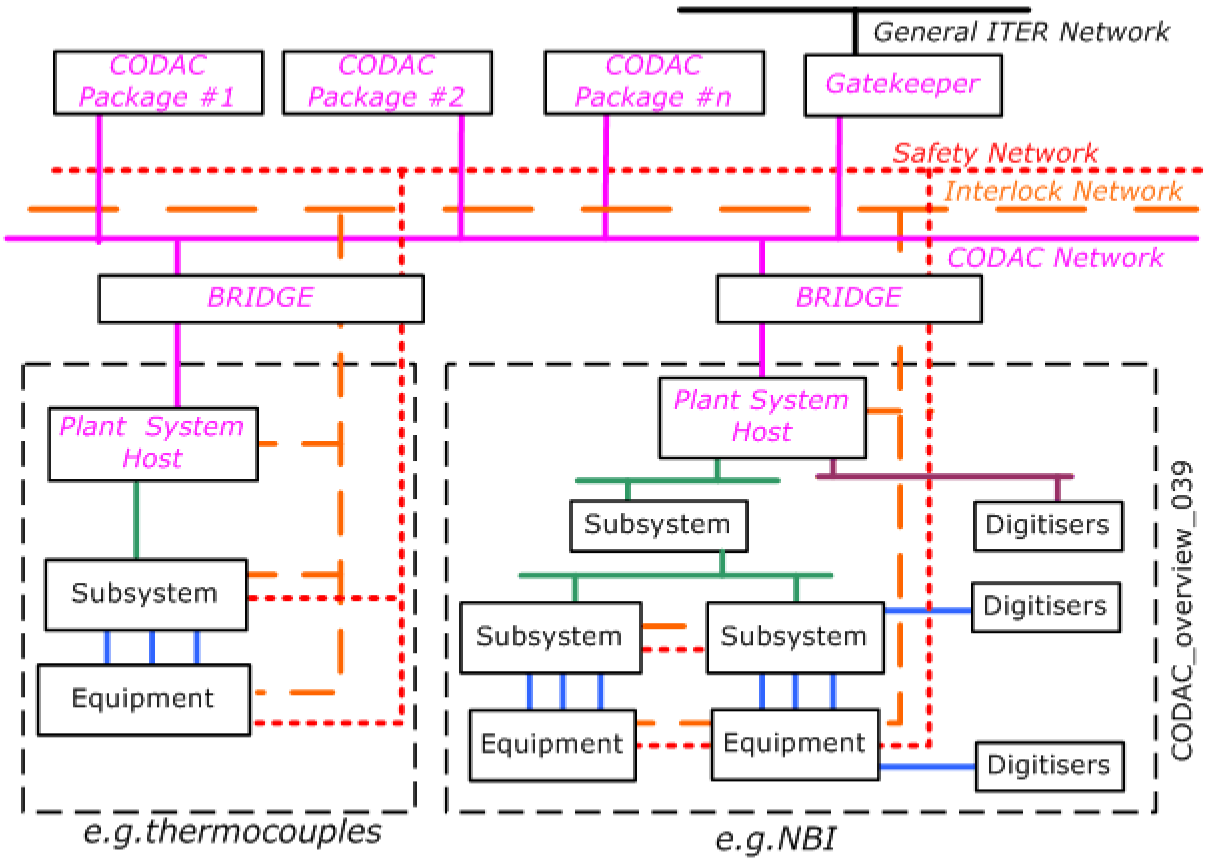


Figure 4.17-3 Interfacing Plant Systems to CODAC

(CODAC components are marked magenta/italics)

Each Plant System has a single Plant System Host which is responsible for the activities in Table 4.17-4.

Table 4.17-4 Plant System Host functions

Marshalling the data flow from the Plant System
Containing the self description of the Plant System
Containing the documentation and drawings of the Plant System
Handling configuration data
Handling transition requests to the Plant System

The Plant System Host is not responsible for the integrity of the operation of the Plant System, since the reliability of a single computer might not be adequate. This responsibility lies with the Subsystem Controller(s) which could be one, more or hierarchical, depending on the choice of the Plant System designer. Subsystem controllers are normally PLCs to be prescribed by CODAC.

Beneath the Subsystem level is the equipment level, interfaced by a prescribed set of fieldbuses. Equipment such as digitisers can communicate with the Plant System Host, or with the controllers, depending on choices made by the Plant System designer.

This architecture leaves considerable freedom in the design of each Plant System, while presenting a generic image via the Plant System Host and restricting the implementation to a set of ITER CODAC standards.

Most ITER Plant Systems will be delivered to operate independently. However, some will be procured as multiple packages due to procurement choices. Examples are the ECRH/ICRH Heating and Current Drive systems in which antennae, transmission lines and power sources are delivered separately. Integration of multiple Plant Systems within CODAC can follow either of two models, shown in the following figure.

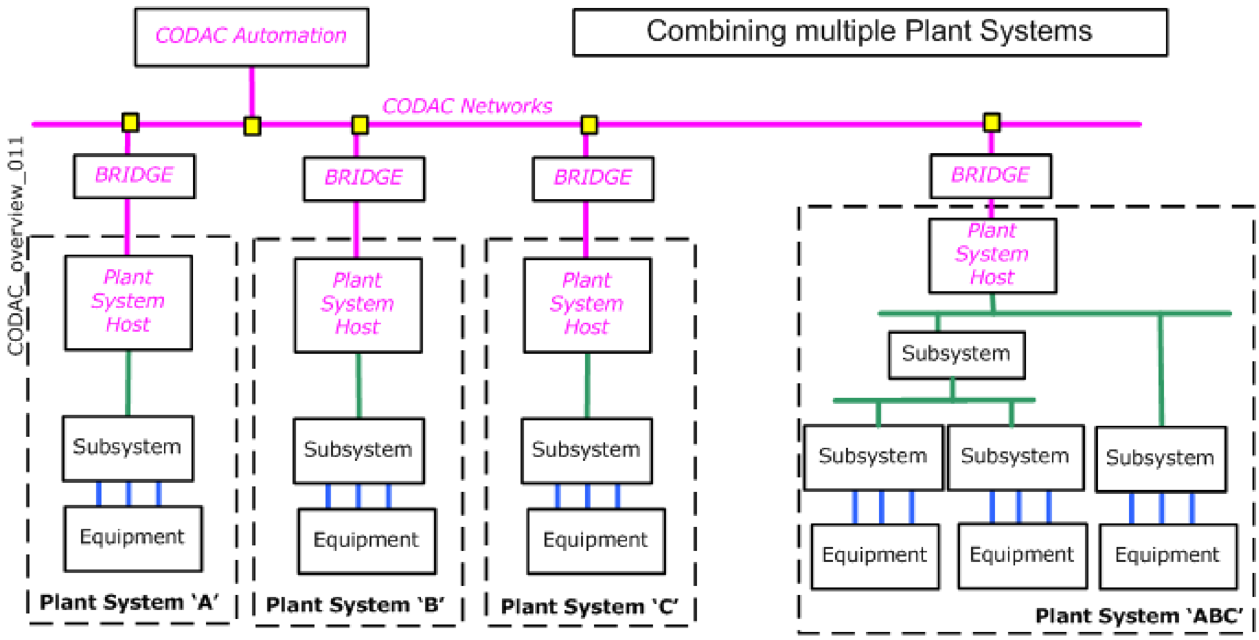


Figure 4.17-4 Integration of multiple procurement packages into CODAC

On the left of Figure 4.17-4, three Plant Systems are separately integrated into CODAC. Coordination of their activity during operation is guaranteed by standard CODAC automation methods.

On the right of the same figure, the systems are integrated into a single Plant System. This method is suitable when the guaranteed operation of the full system has to be more reliable than combining several Plant Systems under CODAC. An example is the ITER tritium handling system, in which one procured system has overall responsibility for the other three.

Integrating complex Plant Systems on site is excluded. This integration must be part of a System Integration package during procurement of the set of systems, to generate a single Plant System. Most CODAC functionality can be factory tested even if the functionality of the Plant System itself cannot be factory tested.

Selecting the appropriate method is a choice to be made by the Plant System designer. Criteria for making this choice include: reliability of the communication; requirement to communicate data between systems at a faster rate than CODAC can provide; requirement to operate independently of CODAC.

The Plant System Host or the controllers can signal errors on the Central Safety Network and on the Central Interlock Network.

The Plant System Host is responsible for exporting all the information concerning the Plant System, summarised in Table 4.17-5. This information is exchanged between CODAC and the Plant System at the moment of final acceptance and is then stored in a central CODAC database. All the structured information from the Plant System Host is exported as data validated against explicit schemas and using tools provided by CODAC. Exporting this Plant Self-Description is tested during construction and commissioning at the factory using a "mini-CODAC" emulator, ensuring problem-free integration at the CODAC level during on-site acceptance.

Table 4.17-5 Summary of major elements inside the Plant System self-description

Schema	Information
Signal list	Signals on wires, origins, drawing references…
Input-output list	Signals exported by the Plant System Host, names, meanings, units, properties…
Module list	Hardware modules, origins, types, conversions, limits, configuration data, jumpers…
Software	Firmware in modules, software in the Subsystem controllers, software in the Plant System Host…
Documentation	List and source of all documents, drawings, concerning the Plant System
Alarms and warnings	List of alarm conditions and their meanings
Common Operating States (COS)	List of COS and their transition commands and transition conditions, standard delays, alarm delays, using SCXML…
Plant Operating States (POS)	List of POS and their transition commands and transition conditions, standard delays, alarm delays, using SCXML…
Mimics	Structured data description of engineering mimics using a CODAC symbol library, mimics delivered by the supplier…
Data handling	Data sampling frequencies, dependence on COS, upper and lower limits…
Operating Limits and Conditions	Restrictions on operating this Plant System, independently of other Plant Systems…
Contacts	Information for contacting sub-contractors for this equipment…
Network use	Requirements for use of the Synchronous DataBus and the Event Distribution Network, connection to the Time Communication Network and to the General ITER Network…
Data Streams	Description of the data streams produced by the Plant System Host…
Operation Request Gatekeeper	Operations to be allowed on the Operation Request Gatekeeper, including access rights…

4.17.6 The Plant Operation Zone

The operation of ITER using CODAC takes place within a Plant Operation Zone, which is a logical and physical separation to guarantee operational integrity. The Plant Operation Zone is shown as a light background in Figure 4.17-5 and includes most of CODAC, the Central Interlock Networks and the Central Safety Network.

The major data-flow is outgoing signals and status values, presenting no security risk. The data are subsequently stored outside the Plant Operation Zone, again to avoid any operational security requirement when accessing the data from outside the Plant Operation Zone.

Actions and data entering from outside the Plant Operation Zone are examined by the Operation Request Gatekeeper. The Operation Request Gatekeeper interprets all incoming commands and data and decides whether they should be transmitted inside the Plant Operation Zone. This decision is made on the basis of: the authenticated originator; the current role of the originator; the current operation mode of the equipment; the operation mode of ITER. The Plant Operator also intervenes if the Request is neither accepted nor rejected. This model allows a degree of automation for some requests and interception by the Plant Operators for other requests, and will evolve during ITER operation.

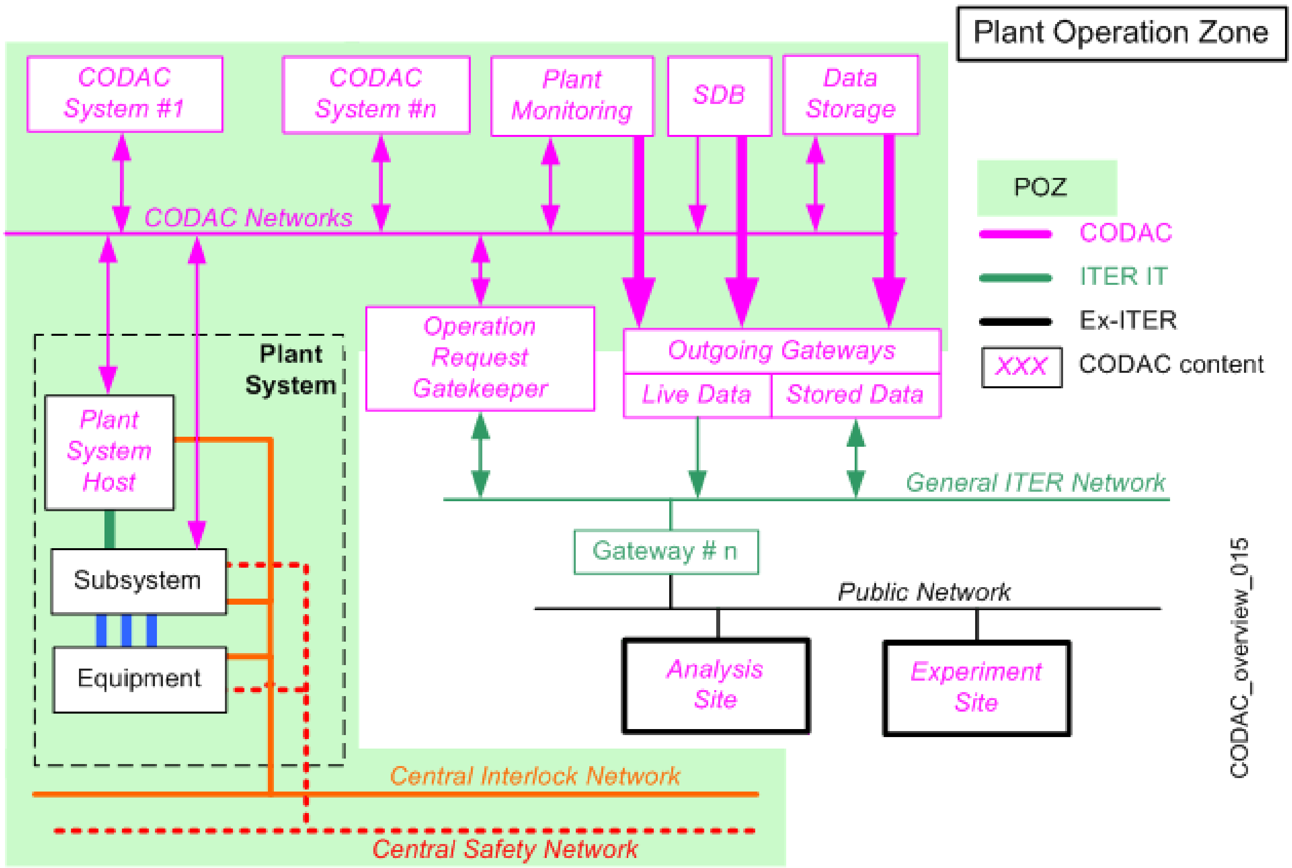


Figure 4.17-5 The Plant Operation Zone.

(CODAC components are shown in magenta)

The exploitation of ITER will take place in Experiment Sites, one of which is in Cadarache and the others are Remote. The functionality of these sites is common, ensuring that Remote Experiment Sites can exploit ITER with the same efficiency, using the same interfaces and tools, as the Cadarache Experiment Site. All Experiment Sites therefore interact with the Plant Operation Zone via the Operation Request Gatekeeper, shown in Figure 4.17-6, allowing the ITER plant to be continuously operated under “CODAC control”.

Plant Systems can be operated under “local control”, if authorized by the Plant Operators. The equipment then functions under front panel control, if provided by the Plant System supplier, for commissioning and testing. Note that Local Control is not an obligatory feature of a Plant System. Cost can be significantly reduced by not providing Local Control, but using CODAC control for commissioning and testing. When the equipment is transferred back under “CODAC control”, the local control room role is outside the Plant Operation Zone and all commands and data once again pass through the Operation Request Gatekeeper.

Plant Systems integrated into CODAC can never be operated under “direct communication” between any computer, inside or outside the Plant Operation Zone and the Plant System. This would present too high a security risk to CODAC. The implication on Plant System design is strong. All Plant Systems must be designed so that all communication between a user and the equipment is established as Operation Requests during the construction of the Plant System. There will be no means of establishing “on the fly” communication inside the Plant Operation Zone once the equipment is operational. Should this become necessary for any reason, the Plant System must be taken out of CODAC control and connected to the assumedly hostile ITER General Network. The Plant System must be totally re-commissioned before re-integration into CODAC.

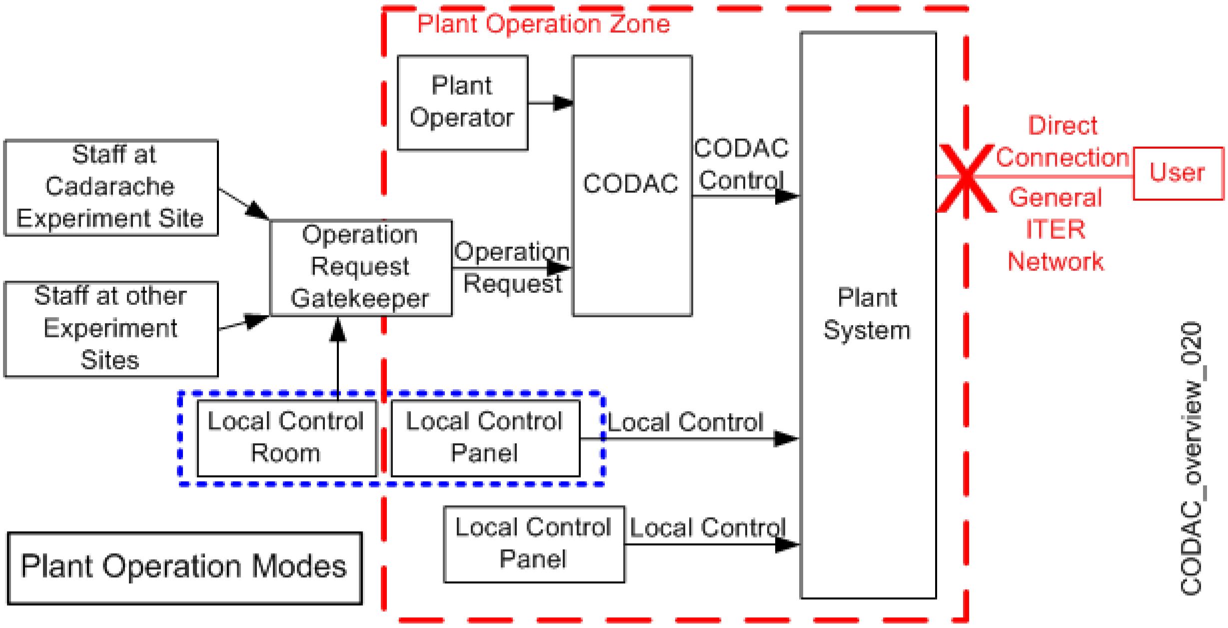


Figure 4.17-6 Plant Operation Modes
(Showing the interaction with the Experiment Sites and with the Local Control Rooms)

4.17.7 CODAC Systems

The internal implementation of the CODAC Systems does not generate an interface to the Plant Systems. The structuring of the CODAC Systems and their component systems is is indicated in Table 4.17-6. Each of these CODAC systems is has its own functional and non-functional requirements in the Detailed Design Document, together with a discussion of its design methodology and a strawman design. The engineering design of the CODAC Systems has not started (June 2006).

This list is followed by a brief comment on each component.

Table 4.17-6 CODAC Systems structuring

CODAC System	Systems
Supervisory Control System	Supervisory Control System
	CODAC Monitoring System
Plant Monitoring System	Main Control Room
	Backup Control Room
Plant Automation System	
	Plant Monitoring System
	Data Logging System
	Error Avoidance System
	Schedule Preparation System
Data Handling System	Schedule Validation System
	Pulse Control System
	Schedule Execution System
Network System	Five CODAC Communication Networks
	Synchronous DataBus Network
	Time Communication Network
	Event Distribution Network
	Audio-Video Network
	Access Gateway System
	Central Interlock and Safety Networks (WBS 4.6)
	Data Exchange System
Services	
	Data Visualisation System
	General Reporting System
	Computing Support System
	Event Notification System
	Plant Simulator System
	Automated Maintenance System
Management System	Operation Training System
	Database Tools System
	CODAC Development System
	Plant Simulator System
	Performance Testing System
	Generic Plant System
	Five CODAC Communication Networks

4.17.7.1 Supervisory Control, Plant Monitoring and Plant Automation Systems

The principal CODAC System is the Supervisory Control System (SCS). SCS allocates all resources to ITER operation. SCS manages a dynamically evolving set of concurrent activities, each of which is driven by an Operation Schedule. SCS initialises and monitors all CODAC functions.

The present status of the ITER plant is monitored by the Plant Monitoring System, which also generates a data stream for the Data Logging System. The maximum refresh frequency is 3 Hz, corresponding to a human reaction timescale. The minimum refresh rate is 0.1 Hz. The monitoring data are made available to the Remote Experiment Sites to enhance contact with the experiment.

The functionality provided is typical of a SCADA (Supervisory Control And Data Acquisition) system in industrial plants, namely display on mimic diagrams, trending, warning and alarm handling, manual triggering of commands or changes to set-points. Operator stations are provided in the Main Control Room.

The Supervisory Control System dynamically allocates ITER resources, normally individual Plant Systems or groups of Plant Systems, to an Operation Schedule. The Operation Schedule is previously prepared by the Schedule Preparation System. Each Operation Schedule is then validated by the Schedule Validation System before becoming executable. An Operation Schedule can be executed by the Schedule Execution System at any time it is requested if the resources can be made available by the Supervisory Control System. There is therefore a strong interface between the scheduling and the planning of the operation of ITER.

The Plasma Control System is implemented as a specific Operation Schedule to maximise reuse of both automation code and plasma control methods.

The Supervisory Control System has overall responsibility for respecting the Operation Limits and Conditions. In view of the importance of protecting the investment of the ITER device, procedures whose functions are closely linked to protection of investment are isolated in a Fault Protection System. This enhances their degree of verification and avoids pollution from other evolving Systems.

4.17.7.2 Data Handling System

The Plant System Host is responsible for marshalling all experimental data streams (signal data, undersampled signal data, plant monitoring data, configuration data, source code, some simple analysis, etc.), converting it to engineering units and delivering it to the Data Logging System. This in turn marshals the data from all the Plant System Hosts.

The Data Logging System provides this re-marshalled data to the Data Storage System, which is responsible for physically storing the data outside the Plant Operation Zone, archiving it and backing it up. An undersampled data stream is set up, continuously available to all Experiment Sites and archived separately.

The Data Access System provides access all ITER data to users inside and outside the Plant Operation Zone. It provides equivalent facilities to users on-site or at Remote Experiment Sites. Uniform access is provided to all data streams. The features provided by the Data Access System include management of the signal names, server-side evaluation of data, server-side undersampling of data, and other features used on existing experiments.

4.17.7.3 Network System

The networks listed in Table 4.17-6 are logical networks. The implementation of each network, and the combination of logical networks into physical networks will be decided as late as possible, to use the most appropriate technologies and methods.

4.17.7.4 CODAC Services

Services are provided when they can be used by multiple CODAC Systems.

The Data Visualisation System groups the visualisation tools for plant monitoring, undersampled data monitoring, trending and scientific visualisation needed during operation.

The General Reporting System allows all CODAC Systems to report correct or incorrect functionality using a standard interface for recording, archiving and tracing reports, including error reports and performance reports.

The Computing Support System provides guaranteed access to distributed computing power to allow operation of ITER as well as file systems to store CODAC Systems data and the experimental data.

The Message Service System provides an ITER-wide definition of inter-process and inter-processor communication using standard ITER protocols.

The Database Tool System provides database support to CODAC.

The Event Notification System allows signalling between processes in the distributed CODAC Network.

4.17.7.5 Management System

CODAC requires automated maintenance and performance testing procedures as part of the CODAC QA. These will be interfaced to an ITER-wide maintenance System.

The Performance Testing System emulates CODAC with a “mini-CODAC” which can be used by ITER staff, Domestic Agency staff or supplier staff to verify the correct functionality and correct performance of their Plant System during factory testing and acceptance, and again during on-site acceptance.

The Plant Simulator System uses the description of the Plant Systems to build a simulator of the ITER plant. Combined with CODAC, this allows early testing of the integration of the Plant Systems into CODAC, and identifies any potential problems well before on-site commissioning.

The Operator Training System provides a fully realistic simulator of ITER using the Plant Simulator System and a copy of all CODAC Systems. Replaying incidents allows operators to test new responses to problems. The Operator Training System also doubles up as a backup to the Main Control Room in case the latter becomes non-operational. Full functionality of ITER is not maintained in the Backup Control Room, but vital functions are guaranteed at the same level as the Main Control Room.

The CODAC Development System provides a full replica of CODAC without the required network reliability, to develop and test CODAC systems. All networks are replicated.

4.17.8 Central Interlock System

The Central Interlock System provides protection of investment for ITER by inhibiting combinations of actions among all Plant Systems which might endanger the integrity of any ITER component. It respects a wide set of Operation Limits and Conditions. It receives digital inputs from any Plant System and delivers outputs on the basis of Boolean logic on the full set of inputs and on the latched outputs. The Operation Limits and Conditions are expressed as structured data. The Central Interlock Network and the logic are doubly redundant. The Central Interlock System is totally independent of CODAC operation, but is aware of the operational status of each CODAC component. All signals used by the Interlock Systems are clearly tracked from signal source through to the actuators with the same QA to ensure consistency of reliability.

The Central Interlock System signals its internal and external status to CODAC for monitoring, display and archiving, Figure 4.17-1.

4.17.9 Central Safety System

The Safety Systems are part of the ITER safety case. They respect international standards which are acceptable to licensing in France, notably IEC 60880-2 for any software implementation. The Safety Systems shall be as small as compatible with the licensing, to enhance the provability of the implementation. Plant Systems which provide inputs to the Safety System or use outputs from the Safety System are classed as SIC (Safety Important Component) and are to be kept as small as possible. Safety functions which transmit information outside a Plant System use the Safety Network as part of the Central Safety System, which is probably hardwired. The Safety Systems are totally independent of the Central Interlock System and of CODAC operation. All signals used by the Safety Systems are clearly identified so that the SIC QA is guaranteed from signal source to the Safety Systems and back to the actuators.

An example of a Safety System is the fusion power shutdown system (FPSS) which guarantees the safe shutdown of the fusion power in case of an ex-vessel coolant leakage.

Part of the Main Control Room is attributed to the Safety Systems during operation. The Safety Operator’s desk has all the Safety System data available with SIC reliability, and provides access to all Central Interlock System data and CODAC data if these systems are operational.

The Safety Systems signal their status to CODAC and to the Central Interlock System, Figure 4.17-1.

4.17.10 Plasma Control and Diagnostic Data Evaluation

The ITER plasma control system comprises four major elements: control of scenario sequencing, plasma magnetic control, kinetic and divertor control, and fast plasma termination by impurity injection. Plasma Control methodology is under WBS 4.7 and CODAC provides the software and hardware infrastructure for implementing all features of Plasma Control.

Plasma Control uses the Synchronous DataBus to communicate data in physics units between Plant Systems and CODAC Systems, including an estimate of the error on each signal. Evaluation of plasma diagnostic information is carried out locally in the diagnostic Plant Systems if this is straightforward. Information is then collected over the Synchronous DataBus for analysing data from multiple Plant Systems and finally transmitted over the Synchronous DataBus to the actuators. Two “hops” on the Synchronous DataBus in addition to computational time, satisfy the requirements for vertical plasma position stabilisation and RWM stabilisation, namely a minimum delay of 5 msec between sensor and actuator response.

Plasma Control is formulated within the frame of general operational scheduling, allowing reuse of complex code and taking advantage of the relative slowness of ITER plasma control compared with existing tokamaks.

Figure 4.17-7 illustrates different models for evaluating the diagnostic data used for feedback control. These models are summarised as follows:

- Diagnostic #1 is evaluated inside its Plant System Host and no external evaluation is required. It writes the evaluated data to the Synchronous DataBus Network (SDN).
- Diagnostic #2 has a more complex scientific evaluation which is not implemented inside the Plant System Host. This allows evolution of the method without having to modify the diagnostic itself. It writes its data to the SDN and the evaluation process reads the data from the SDN, evaluates them and writes the results to the SDN.
- Diagnostics #3 and #4 must be evaluated together for another physics analysis. Both diagnostics write their data to the SDN and they are evaluated externally and the result is rewritten back to the SDN.
- Diagnostic #5 requires the analysis of Diagnostics #3 and #4 for its own evaluation, which is again performed externally.
- Feedback controller#1 reads the diagnostic data originating from all these diagnostics and from the evaluators, and then writes back an actuator demand signal to the SDN. The actuator Plant System reads the actuator demand signal, acts accordingly and supplies its own diagnostic data to the SDN.

The SDN will use conventional network technologies to meet the latency requirements.

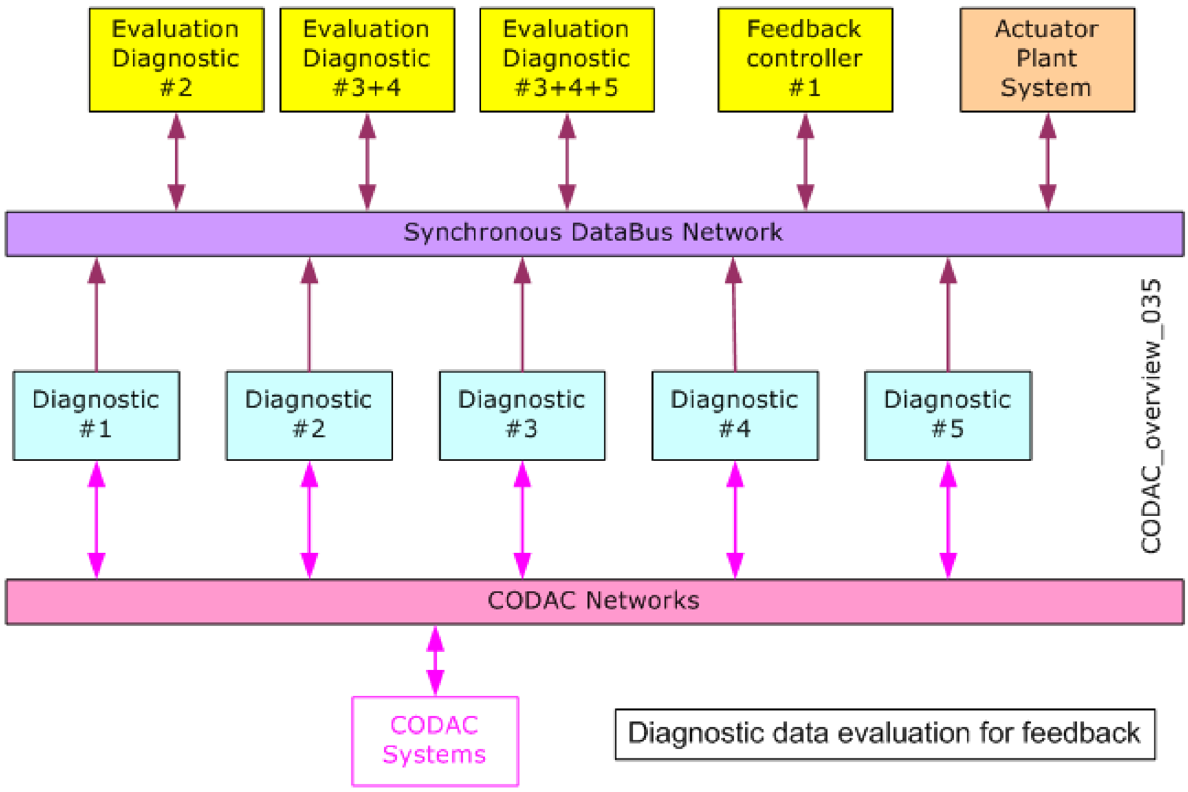


Figure 4.17-7 Data evaluation and analysis combining different diagnostic data sources

4.18 Ion Cyclotron H&CD (WBS 5.1)

This section was extensively updated in 2005.
A further update will be done in 2007 after selection of the antenna concept.

4.18.1 Functional requirements

The IC H&CD system is designed to:

- access H mode and heat plasma at Q>10 (with preference to bulk ion heating),
- provide steady state current drive capability for DT, D, H and He plasmas, in particular to provide central current drive in high bootstrap fraction scenarios,
- accomplish several functions of plasma control, including burn and plasma transport

4.18.2 Desirable Capabilities

- Achieve plasma break-down, burn-through and assisted current rise at low start-up electric fields.
- Conduct IC resonance discharge cleaning (ICR-DC) at full toroidal field. The control system must provide adequate protection to allow for this mode of operation without increased risk of damage to the IC system.

(Editors Note: There is an ongoing review of the ITER Wall Cleaning Strategy (See TCM-21))

4.18.3 IC H&CD Configuration

The main heating scheme of the IC system is at the tritium second harmonic, in a 50-50% DT mixture at f = 53 MHz and B_T = 5.3 T with typical 50-50% power partition among the bulk ions and electrons. Addition of ^{3}He (< 3%) minority [DT-(^{3}He)] results in a significant increase of the fraction (up to 70%) deposited on bulk ions.
The frequency window for on-axis current drive is at the peak of the electron absorption (f = 55 MHz, with a central current drive efficiency of ~ 20 kA/MW). Ion minority current can be driven at the outboard q = 1 for the control of the sawtooth period, at a frequency of 40 MHz. The operating range Δf = 40 – 55 MHz encompasses all the IC physics scenarios of interest.
The main system components are the launcher and its support structure, the power transmission line, and power source.

Table 4.18-1 Ion Cyclotron Resonances

Resonance	MHz	Comments
$2\Omega_T=\Omega_{3He}$	53	Second harmonic + minority heating
Ω_D	40	Minority heating. strong competition of Be and α-particles
FWCD	55	On axis current drive
Ω_{3He}	45	Minority ion current drive at sawtooth inversion radius (outboard)

Table 4.18-2 IC Heating and Current Drive Parameters

	Unit	H	DT
IC H&CD power delivered to the plasma(1,2)	MW	→	20 – 40
Resonance frequency for second tritium harmonic and ^{3}He minority heating	MHz	→	53
IC resonance frequency for D-minority heating	MHz	→	40
On-axis CD	MHz	→	55
IC number of allocated equatorial ports(1)		→	1 – 2

Notes:

(1) The lower value of the range represents the foreseen design requirement. The upper value of the range represents a design requirement in the event of an upgrade, in other words the design will be upgradable (with additional investments) to the upper value of the range. The upgrade of the LH, IC, and NB cannot be all carried out at the same time, as only 2 upgrades are required and compatible with the port allocation. In all cases no more than ~130 MW of installed power will be present and no more than 110 MW will be available at the same time to the scenario.

(2) Assumes R′ = 4 Ω

Table 4.18-3 Maximum Antenna Electric Field Requirements

Parameter	Value
Max E field || to B	1.5 kV / mm
Max E field ⊥ to B	3 kV / mm

4.18.4 Launcher, Tuning and Port-Plug

The IC launcher is an array of 4x6 elements fed by eight or twelve coaxial transmission lines carrying a total power of 20 MW. The straps are short, with one end connected to ground. This arrangement makes them self-supporting and able to operate at low voltage.

The tuners consist of either capacitive tuning elements connected in series to the ungrounded ends of the current straps through connecting leads, or some combination of external matching components (trombones, shorted stubs, hybrid power splitters) connected to the antenna feed lines on the pressure side of the vacuum windows.

The IC array has a modular construction. A box-like, Be-plated, Faraday shield in Cu-Cr-Zr alloy, is the plasma-facing component.

The components are mechanically assembled with and supported by the outer mechanical structure of the launcher support structure, also enclosing coolant manifolds for the front-end components. The closure plate supports the IC vacuum transmission line ceramic feed-throughs, contributing to the vacuum and tritium containment. The windows are also accessible from behind and can be easily maintained.

The support structure of the launcher is common to all RF systems. It includes the vacuum vessel port closure plate, which transfers to the port the gravity load and the bending moments induced by plasma disruptions. The structure is also used to distribute coolant to the different RF components from blanket and VV cooling loops. The support structure is not in contact with other in-vessel components.

A nominal gap of 20 mm is allowed from the port walls all around the plug perimeter and an overlap of 30 mm is provided by the adjacent blanket modules, to shield direct neutron streaming in the gap.

The Faraday shield is the primary plasma-facing component of the RF launcher assembly. It is designed to accept the same thermal loads from plasma radiation and neutrons as the adjacent shield blanket modules. However, it is not designed to be exposed to conduction heat loads from the plasma (e.g. during the start-up phase) from which they are protected either by the blanket or by shielding structures built in the blanket.

The RF launchers include a neutron shield, about 1 m thick, which reduces the high energy (E > 1 MeV) neutron flux outside the VV closure plate to 1.18×10^7 n/cm^{2}s, and limits the integrated activation level in the area to produce < 100 μSv/h, 15 days after shut-down. Hands-on maintenance of the components in the port inter-space would be therefore permitted.

The neutron shield is penetrated by transmission lines carrying power to the antenna current straps. Artificial bends (doglegs) can be included in their path if needed, to reduce the radial neutron streaming to acceptable levels.

Plasma-facing components and neutron shield are cooled in parallel by pressurised water distributed by the blanket cooling loop. The coolant manifolds are located within the support structure. The closure plate and the components integrated in it (e.g. windows) are instead connected to the VV cooling loop.

The boundary of the primary vacuum confinement is located at the closure plate of the vacuum vessel. A double boundary window system will be included in the lines. Water-cooled ceramic windows are used in each coaxial line.

4.18.5 Power Transmission Design

Sections of evacuated coaxial transmission line (VTL) penetrate the bioshield. The vacuum transmission lines are manually disassembled, to allow a remote-handling cask to dock to the VV port.

Monitoring and safety-related equipment is installed at the end of the VTL, in locations easily accessible for maintenance. Among others, DC breaks (to prevent ground loops being closed outside the pit) and rupture disks, exhausting in the volume served by the detritiation system, are installed at this location.

Outside the bioshield, the VTLs are connected to the main transmission lines (MTLs) and to some matching or decoupling components that are located there. The MTLs run on the ceilings of the appropriate port cells and gallery, and are then routed to the RF heating area of the laydown, assembly and RF heating building.

For continuous operation, VTLs and MTLs are water-cooled by loops of similar specifications, connected to the cooling water system. The MTL use a pressurised gas as the dielectric.

The coaxial transmission lines used by the IC system are commercial items. The MTL is a rigid coaxial line, having a characteristic impedance of 30 Ω. The inner conductor is water or air-cooled and operates at $T_{in} < 110$°C. The outer conductor is water-cooled and operates at $T_{out} \sim 45$°C.

The voltage stand-off of the main transmission line is TBD kV, well in excess of the expected maximum RF voltage (< 21 kV). This large margin should provide low maintenance and a high reliability to this component.

4.18.6 RF Power Sources

The RF power sources (gridded–tube amplifiers), are connected to the MTLs by RF conditioning and matching components.

The RF power sources are connected to their main DC supply, located at ground level in the RF heating area of the laydown, assembly and RF heating building. When a voltage fault is detected, a fast protection circuit rapidly (10 μs) removes the DC supplies to prevent arc damage in the tubes. In view of the long pulse operation, the protection is designed for multiple response and power re-application, after a time suitable for the arc to clear.

The IC power sources are commercial multi-stage amplifiers equipped with tetrode tubes. The sources must have the capability to deliver 20 MW total power *to the plasma* (i.e., enough power to compensate for rf losses in the transmission lines and antenna). If eight sources are used to drive eight antenna elements, then each source should be able to deliver ≈ 3 MW; if 12 sources are used, each should be able to deliver ≈ 2 MW. Each power source must be able to operate steady-state into a load mismatch with VSWR < 1.5 CW, and transiently into a mismatch with a VSWR up to 2.

4.18.7 Antenna Alignment to Plasma

Table 4.18-4 IC Antenna FW Position

Parameter	Unit	Value
Largest antenna-separatrix gap(1)	mm	120
Antenna location relative to FW	mm	10 behind FW
Radial tolerance	mm	± 5
Vertical/Toroidal tolerance	mm	± 10

(1) This gap is measured at the poloidal location where the distance between the antenna and the separatrix is a minimum. This is the largest value that will allow the rf system to deliver full power to the plasma; a smaller distance is desirable.

4.18.8 ICRF Cooling Conditions

See Section 4.11 Cooling Water

4.18.9 ICRF-PS Interfaces

See 4.16.2.3.1

4.19 Electron Cyclotron H&CD (WBS 5.2)

(Updated Dec06/Jan07 A.Tanga, N.Kobayashi)

4.19.1 Functional Requirements

The EC H&CD system is designed to:

- Access H mode and heat plasma to Q>10. (with electron heating),
- Provide steady state current on-axis and off-axis drive,
- Control MHD instabilities by localised current drive, such as stabilisation of neo-classical tearing modes (NTMs) and sawtooth instability.
- Conduct wall conditioning during the inter-pulse and machine conditioning phase
- Assist the poloidal field system in establishing breakdown and current initiation

(Editors Note: There is an ongoing review of the ITER Wall Cleaning Strategy. See TCM-21, and ITER Physics Basis Chapter 6 Section 2.4 EC Discharge Cleaning.)

4.19.1 Possible additional Capabilities

- Provide counter-ECCD in the range of $0<\rho_p<0.3$. (TBD)

4.19.2 Configuration

The EC H&CD system consists of:

- Gyrotrons for H&CD (24MW installed power at 170 GHz with minimum pulse length of 500 sec),
- Gyrotrons for start-up (3 MW installed power at ~127 GHz with minimum pulse length of 5 sec),
- Power supplies for the above sources,
- 24 evacuated low-loss transmission lines,
- One equatorial launcher (3 waveguide lines shared by the 170 and ~127 GHz sources),
- Four upper launchers,
- In-line, automatic and remotely controlled switching system to share the RF power between the upper and equatorial launchers.

Table 4.19-1 EC H & CD Parameters

Frequencies - H & CD - Start-up	 170 GHz ~127 GHz (presently 127.5 GHz TBD)
Design parameters for launcher	Equatorial Launcher; Toroidal steering range: 20º ~ 40º in a horizontal plane with access of $0<\rho_p\leq\sim0.5$. Upper Launchers; Poloidal steering range(1) : 53º~69º with 18º inclination between beam and poloidal cross section for upper steering mirror (USM) and 39º~61º with the inclination of 20º for lower steering mirror (LSM). Total access of $0.4<\rho_p<\sim0.9$. Max. steering speed: 39º ~ 61º for LSM in ≤3s. All components shall be designed for the whole ITER lifetime of 3×10^4 pulses. Neutron shield capability; For a torus window: fluence≤10^{20} n/m^2. For hands on maintenance: dose rate≤100µSv/hour at 10^6 sec after shutdown.
RF Power source - Efficiency - Unit power	Gyrotron 50% ≥1 MW (at an output of a MOU) (TBD)
Injection power to plasma - H & CD - Start-up	 20 MW with pulse length ≥ 500s (TBD) ≥2 MW with pulse length ≤10s (TBD)
Power modulation (H&CD) -Modulation Frequency - Duty cycle - Range	 1~5 kHz (This should be TBD. Need more assessment and experiment.) 50% (50% on, 50% off) 20 to 100%

1. The exact value of the steering angles depends on the specific launching geometry.

Table 4.19-2 EC Heating and Current Drive Parameters

	Unit	H	DT
EC power (1) for initial breakdown assist	MW	→	≥2 (TBD)
EC frequency for initial breakdown assist	GHz	→	~127(TBD)
EC power (2) for H&CD	MW	→	20 – 40
EC frequency for H&CD	GHz	→	170
EC number of allocated upper ports		→	4
EC number of allocated equatorial port except Scinario4, where no equatorial port allocation.		→	1

Notes:
1. Pulse duration of 5 s required. Longer pulse length will be assessed, if it is necessary during the current ramp-up.
2. The lower value of the range represents the foreseen design requirement. The upper value of the range represents a design requirement in the event of an upgrade (with additional investments).

4.19.3.1 Launcher

A launcher is composed of a front shield, steering mirrors with a drive mechanism and a drive force generator, internal shields, transmission lines with torus windows (and isolation valves for the upper launcher), casing and water pipes.

The structure is summarized in Table 4.19-3.
The upper launcher is optimised for MHD control (NTM & sawteeth). As the steering range is divided into two regions accessible with either one of the two steering mirrors (USM and LSM), the upper launcher has 8 entry lines per port. Switching between the two steering mirrors is performed with an in-line line remote controllable switch (~1 s) located near the cryostat entrance.

Internal circular corrugated waveguides has a dogleg structure to reduce neutron fluence at the back of the launcher in order to satisfy requirements on maximum nuclear damage at the diamond window and maintenance dose rate requirement at the primary closure plate.
All of in-vessel components and casing of launchers are cooled by water of blanket PHTS.

The first tritium confinement boundary is composed of combination of the torus window and an isolation valve in the port interspace. although order of insertion is different between the upper launcher and the equatorial one.

Common components design strategy is included in the design of launchers and transmission lines, for example for chemical vapour deposition (CVD) diamond windows, waveguides, isolation valves, etc.

Table 4.19-3 Launcher Structure

	Equatorial launcher	**Upper launcher**
Number of launcher	1	4
RF beams/launcher	24	6
Slot of Front shield	3 (horizontal)	1(vertical)
Focusing mirror	0	1
Steering mirror	3	2
Mirror material	dispersion strengthened Cu (or CuCrZr)	
Cooling of mirror	Water of Blanket PHTS	
Water supply	Spiral tube	
Transmission lines	24 3 waveguides are shared with the start-up gyrotrons (RF power source switched by waveguide switches near the gyrotrons).	8
Torus window/laucher	24	8
Max. pressure to disk	0.2 MPa	
Cooling of disk	Water of CCWS	

4.19.3.2 Transmission Line
Transmission lines are commercial items and consist of:
1. straight sections of circular corrugated waveguide;
2. an in-line switch connected to a short pulse load at a MOU.
3. a number of mitre-bends, two of which are polarisers and one a power monitor (arc detection systems are to be incorporated in a majority of the mitre bends);
4. auxiliary equipment such as DC breaks, expansion segments, vacuum pumping sections (including vacuum monitoring systems), isolation valves and miscellaneous systems (water cooling, support structures, etc).
5. high power, remotely operated switches (~1 s) to direct the power between the upper and equatorial launchers.

6. a dummy load used for calorimetric measurement of RF power.

Flexible support structure of waveguides is necessary at gap between the seismically supported Tokamak building and the ground fixed RF building in order to accommodate relative displacement between two buildings.

Table 4.19-4 Basic parameter for transmission line

Vacuum level	$<10^{-2}$ Pa
Cooling	Component cooling water Long straight sections of waveguide will be cooled by air.
Transmission efficiency	≥83 %.
Max. line length	~100 m

4.19.3.3 RF Power Source

RF power source is mainly composed of a gyrotron unit and a mating optical unit (MOU) (TBD). High power mm waves are generated by a gyrotron. The gyrotron uses a depressed-collector geometry for an efficiency of up to ~50%. The generated RF power is output from a CVD diamond window. The RF output power is ~1 MW (TBD) at an output of the MOU.

The MOU converts the mm-wave power beam at the exit of the gyrotron to the waveguide (HE_{11} mode) for long distance transmission. It consists of a small vacuum chamber, focusing mirror(s), stray radiation absorbing plates, matching optics and a pumping port. The MOU should provide optimal coupling of the RF power (~95% of measured power at the CVD window or design Gaussian content of 99.5 % (TBD)) into the transmission line with the HE_{11} content ≥96%.

The gyrotron tubes and associated power circuitry are protected by fast protection circuits, which automatically (< 10 μs, < 10 J in a nominal 100 V arc) remove the DC body and cathode voltage in case of tube, line or load breakdown. The DC high voltages can be re-applied when suitable vacuum conditions are recovered after the arc (gyrotron conditioning is automatically integrated into the re-start procedure). (TBD)

The gyrotron and the MOU are cooled by water of the CCWS.

The gyrotron units are located at the mezzanine level in the RF building and are arrayed in 7 groups (pods). A pod contains 4 tubes for the 170 GHz frequency and 3 tubes for ~127 GHz. All gyrotrons within a given pod are to be from the same supplier.

Each pod has common auxiliary supplies such as cooling water. Power supplies, series switches, and other auxiliary support equipment are located immediately below the gyrotrons on the lower level of the RF building.
Each pod has separate control unit, interfacing to the power supply and capable of an independent operation.

4.19.3.4 DCHV Power Supply

The power supply scheme is based on a main high voltage power supply, typically up to 60 kV, connected between a cathode and a grounded collector and providing the beam current and a body power supply, connected in the baseline between the body and the collector, establishing and controlling the required beam voltage (typically - 85 -90 kV). It is necessary that the body voltage (and an anode voltage for a triode gun) can be independently controlled because characteristics of gyrotrons are different with each other.

The body power supply can also modulate the applied beam voltage in order to control the output RF power, in particular during NTM stabilisation.
Semiconductor elements are cooled by water of CCWS.

4.19.3.5 Control of EC H&CD system
Monitoring and control of the EC H&CD system include:

1. Monitoring & data acquisition of plant parameters (e.g. vacuum, electrical, hydraulic, temperatures, RF power, etc.).
2. Plant control: dedicated diagnostics (including arc detection) and control/local safety/protection systems integrated into the PS, gyrotron, transmission lines and launchers.
3. Safety systems: vacuum and T contamination monitoring, for the launchers/waveguide systems (including CVD diamond window monitoring).
4. Launcher deposition control, including feedback control systems: these require local controllers and actuators as well I/O with CODAC (synchronous data bus).
5. Feedback control of the PS voltage (amplitude and modulation frequency) for NTM control, deposition localisation and control of plasma performance.

4.20 Lower Hybrid H&CD (WBS 5.4)

(There will be a further update in 2007)

4.20.1 Functional requirements

The Lower Hybrid Heating and Current drive (LH H & CD) system is designed to:

1. provide steady state off-axis current drive capability for DT, D, H and He plasmas,
2. modify current density and q profile.

4.20.2 Configuration

4.20.2.1 Overall

4.20.2.1.1 LH H&CD System Usage

The lower hybrid H&CD system is specialised for off-axis current drive and current profile control. It is designed to deliver a total power of 20 MW at 5 GHz to the ITER plasma.

An efficient coupling of LH to the plasma requires the presence of a sufficient plasma density between the launcher and the separatrix ($4 \times 10^{17}\ m^{-3} \leq n_e \leq 10 \times 10^{17}\ m^{-3}$).

The system is designed to operate at:

- a waveguide maximum electric field of 3.2 kV/cm (or a power density of 33 MW/m^2) at full power;
- a wave number $1.9 < N_{//} < 2.1$ modified by electronically varying the array phasing;
- a maximum load reflection coefficient of 5%, measured at the antenna input.

Table 4.20-1 Heating and Current Drive Parameters

	Unit	H	DT
LH H&CD power(1)	MW	0	0 – 40
LH H&CD frequency	GHz	→	5
LH H&CD # of allocated equatorial ports(1)		→	0 – 2

(1) The lower value of the range represents power requested for the start-up configuration. The design value corresponds to 20 MW. The upper value of the range represents a design requirement in the event of a full implementation, depending on the choice of future scenarios.

4.20.2.2 Launcher Design

4.20.2.2.1 LH Launcher Arrangement

The plasma-facing component is a phased array, made of four modules (2x2) of passive-active multi-junction (PAM) stacks. Each PAM module is composed of 24 active and 25 passive rectangular cross section waveguides, progressively phased in quadrature, so as to synthesize a travelling slow wave propagating in the toroidal direction. The phase shift is obtained by inserting phase shifters of multiples of $\pi/2$ in adjacent groups of waveguides.

The PAM assembly is constructed in precipitation-hardened copper alloy (CuCrZr) or in dispersion-strengthened copper alloy (Glidcop Al25) and covered by a Be protection plate 18 mm thick. Cooling channels are provided in the thickness of the passive waveguide wall, by twin cooling channels lined with thin 316LN SS tubing, and supplied by a manifold housed in the supporting frame.

Forty-eight TE_{01-03} mode converters form a 2x2 matrix, held in position by a 316LN SS frame, which also provides the attachment of the launcher to the support structure. The mode converters are cooled by means of channels bored in their top and bottom SS walls. The same cooling circuit feeds all cooling channels in the module. The mode converters are fabricated of 316 LN SS, Cu-plated on the inside surfaces.

Forty-eight double-disk ceramic windows (12 per module) are located at the end of each converter, outside the primary closure plate. BeO (99% purity) is used as dielectric material. The windows have an insertion loss of 0.1%, a voltage standing wave ratio VSWR ~ 1.5, and a working frequency of 5 GHz. The ceramic disks can withstand 0.2 MPa dry air pressure on one side. The coolant is routed to the components of the four PAM modules from a common manifold located in the support structure, and connected to the primary heat transfer system.

Table 4.20-2 Mechanical Dimension of PAM Module

Parameter	Value
Number of active waveguides	24 x 12
Number of passive waveguides	25 x 12
Cross section of active waveguide (mm2)	9.25 x 56
Mechanical length (mm)	925 to 1050
Fundamental transmission mode	TE_{01}
Cross section of passive waveguide (mm2)	7.25 x 56
Mechanical length (mm)	925 to 1050
Wall thickness (mm)	3
Phasing among active waveguides	3 $\pi/2$
Phase shifter dimensions (mm)	14 x 750
Typical $N_{//}$ value	1.9 – 2.1
Max electric field in nominal power (for high power reflection from plasma of 22%) (kV/cm)	5

4.20.2.3 Power Transmission Design

4.20.2.3.1 LH Transmission Line

The main transmission line (MTL) is in four sections. Sections, with 6 main transmission lines.

i) A vacuum section which runs between vacuum vessel and first window. The VTLs are commercial rigid rectangular waveguide sections (type WR229), connected in pairs to 24 3-dB hybrid junctions. At the cryostat exit, twenty-four standard vacuum windows or gas barriers are used for secondary vacuum containment.

ii) A pressurised splitting network section (SN), running from the cryostat wall and the circular transmission line. Each circular wave-guide feeds 4 standard vacuum transmission lines through its SN.

iii) A pressurised circular transmission line section (CTL), running from the splitting network to the klystron cabinet. The total CTL length is approx. 80 m with six 90° bends. The circular transmission line is made of sections of commercial, rigid circular cross-section waveguide and elbows, filled with pressurised air. The main propagating mode is TE_{01}. Mode filters are inserted along the line, to avoid propagation of high order modes.

iv) A pressurised recombining network (RN) section (symmetric with the SN), located in the klystron cabinet, running from the CTL to each individual power klystron. Each MTL is fed by 4 klystrons through the RN.

Four klystron generators are connected to the same CTL through the RN, which includes an RF switch and a 1 MW test load.

Table 4.20-3 Circular Transmission Line Parameters

Design	Feature
Wave guide	standard C10: 107.57 mm radius
Flange	remote handling and standard C 10 types
Material	OFHC, 3 mm thick
Attenuation TE01 @ 5 GHz (dB/m)	8×10^{-4}
RF power rating	4 MW
Electric field for nominal operation at VSWR = 1	5 kV/cm
VSWR	< 1.4
Admissible operating pressure	4 bar absolute
Sealing material	Silicon rubber
Insertion loss for a 90° bend	< 0.3 %

4.20.2.4 RF Power Sources

4.20.2.4.1 LH Klystrons

The LH RF power system consists of 24 RF power sources for 20 MW of total injection power, each consisting of one power klystron amplifier and of auxiliaries (focusing block with superconducting magnet and regulated supply, shielding, filament power supplies, and oil tank). The amplifier is driven by a solid state driver amplifier (20 W).

Phase and amplitude of the output wave of all klystrons are electronically controlled using a single reference.

The klystrons are connected in groups of four to the HVDC power supply unit and common protection circuits are installed close by so as to minimize stray capacitance in the connections.

Table 4.20-4 Main Features of the Klystron Amplifier

Nominal power	1 MW (VSWR < 1.4:1 any phase), CW
Duty cycle	CW or 1000s every 20 min
Frequency	5 GHz
Bandwidth	+/- 5 MHz
Efficiency	> 60 %
Gain	53 dB
Amplitude accuracy	1%
Phase accuracy	± 2°

4.20.2.5 Cooling Conditions

TBD

4.20.2.6 PS Interface

See Table4.15-10 Specification for cold compressors

4.21 Neutral Beam H&CD (WBS 5.3)

(An extensive review is underway (Oct 2006) of the NB Cell. This section will be updated after the review.)

4.21.1 Functional requirements

The NB H&CD system is designed to:

- help in accessing the H-mode and heating plasma at Q>10,
- provide steady state current drive capability (on-axis, off-axis) for DT, D, H and He plasmas,
- modify current density and q profile,
- provide plasma rotation.
- provide power to sustain the density during shutdown and allow for controlled transition from H to L-mode at the end of burn,

4.21.2 Configuration

4.21.2.1 Overall

4.21.2.1.1 NB H&CD System Usage

The neutral beam (NB) system design consists of two heating and current drive (H&CD) injectors and one diagnostic neutral beam (DNB) injector (see 4.21.2.3). Space is available in the building and on the tokamak for a third NB system. Each H&CD injector will deliver a deuterium beam of 16.5 MW, with energy of 1 MeV, and will be able to operate for long pulses (up to 3,600 s for steady state operation). A system based on negative (D^-) ions is used.

In addition to H&CD, a small amount of plasma rotation is also provided by the NB H&CD injectors.

For the H operation phase, the H&CD injectors can be operated in hydrogen, with beam energy ≤ 0.8 MeV and beam power ≤ 13 MW.

Table 4.21-1 NB Heating and Current Drive Parameters

NB H&CD injection power(3)	MW	27	33 – 50
NB H&CD beam energy	MeV	0.8	1
NB H&CD # of allocated equatorial ports(3)		→	2 – 3
NB tangency radius(1)	m	→	5.276
NB lowest beam axis level at the tangency point(1)	mm	→	–420
NB highest beam axis level at the tangency point(1)	mm	→	+154
NB e-folding length of beam profile at the tangency point in vertical direction, B (2)	m	→	0.32
NB e-folding length of beam profile at the tangency point in horizontal direction, A (2)	m	→	0.22

(1)The parameters refer to beam aiming at the first wall between ports (no beam strike area in the ports).

(2)Beam profile at tangency point described as $\mathbf{P(x,y) = Ce}^{-\left[\left(\frac{x}{A}\right)^2+\left(\frac{y}{B}\right)^2\right]}$

(3)The lower value of the range represents the foreseen design requirement. The upper value of the range represents a design requirement in the event of an upgrade, in other words the design will be upgradable (with additional investments) to the upper value of the range. The upgrade of the LH, IC, and NB cannot be all carried out at the same time, as only 2 upgrades are required and compatible with the port allocation. In all cases no more than ~130 MW of installed power will be present and no more than 110 MW will be available at the same time to the scenario.

4.21.2.2 NB System

4.21.2.2.1 NB Injection Angle

The horizontal angle of injection is defined by the NB duct size (including beam envelope, vacuum confinement, neutron shielding, tolerances and clearances) and the space available between the toroidal field coils.

The strike area on the far wall of the vacuum vessel is constrained not to include a port so as not to damage items in the port plug.

Beam injection is possible when the plasma density is $\geq 0.35 \times 10^{20}$ m^{-3}, assuming that 1 MW/m^2 is the acceptable power density on the first wall.

Within the NB duct height, the beam can be aimed at two extreme (on-axis and off-axis) positions by tilting the beam source around a horizontal axis on its support flange.

In order to cover a range of vertical positions from the machine equatorial plane (at the tangency point) suitable for both on- and off-axis CD, the beam axis is tilted vertically in the range 40 - 60 mrad. This tilted beam geometry enables the NB duct to be compatible with a port of the same height as the regular equatorial port, the other relevant structures of the machine (toroidal field coils, intercoil structures, poloidal field coils and thermal shields) and the building design.

4.21.2.2.2 NB Port Location

The NB injectors are located on the north side, at the equatorial level of the tokamak building. Port 4 is shared between an H&CD injector and the diagnostic injector, and port 5 is allocated to the second H&CD injector. A third injector could be mounted on port 6.

4.21.2.2.3 NB Injector Vessel

The injector's vessel is an extension of the primary vacuum boundary and is part of the primary barrier for contamination confinement. The common enclosure for all the injectors, the NB cell, performs the function of a secondary confinement barrier.

4.21.2.2.4 NB Ion Source

The ion source design assumes 200 A/m^2 as the D^- accelerated current density, at the grounded grid, a total accelerated current of 40 A.

In the ion source D^- ions are extracted from an arc discharge produced between tungsten filaments (cathode) and the source body (anode). Small quantities (< 1 g) of caesium are introduced into the discharge. The plasma grid (PG) operates at high temperature, between 250 – 300°C. A current is passed through the plasma grid (2 – 4 kA), which generates a magnetic field (PG filter) that reduces the loss of negative ions by collision with fast electrons, and the number of extracted electrons. The plasma and extraction grids form the extractor (height ~ 1,540 mm, width ~ 580 mm). Each grid is made up of 4 horizontal segments, divided into 4 groups. Each group has 5 x 16 apertures, so that the total number of apertures is 1,280.

4.21.2.2.5 NB Accelerator

The MAMuG (Multi Aperture Multi Grid) concept is considered as the basis for the ITER design. In this process, the accelerator consists of 5 stages, each stage consisting of 4 grid segments for beam aiming. Post insulators support the intermediate acceleration grids and the ion source. The design current of the negative ion beam, at the exit of the grounded grid, is 40 A.

The beam source is in the primary vacuum. The primary vacuum is sealed by a 1 MV, five stage bushing made up of two coaxial insulators: the inner, facing the vacuum, in ceramic (1.56 m outer diameter, OD), and the outer, facing SF_6 gas, in fibre-reinforced plastic (1.76 m outer diameter).

The space between the two insulators is filled with "guard gas" to avoid contamination of the primary vacuum with SF_6. The guard gas could be nitrogen or dry air.

4.21.2.2.6 NB Neutraliser

The gas neutraliser provides ~ 60% neutralisation efficiency for the 1 MeV D^- ions. The beam, at the exit from the neutraliser, consists of neutrals and residual ions (D^- and D^+, each ~ 20%).

In the worst case (7 mrad core divergence plus 2 mrad horizontal misalignment) the maximum power deposited on the walls of the neutraliser channels is 4.2 MW and on the neutraliser leading edges is 370 kW; there the maximum power density is 2.1 MW/m^2.

4.21.2.2.7 NB Residual Ion Dump

The residual ion dump (RID) uses an electric field to deflect the ions that are dumped on the five RID panels. The field is produced by applying zero potential to the odd panels (the two external and the central panels) and - 20 kV to the two even panels.

In the worst case (7 mrad core divergence plus 2 mrad horizontal misalignment) the maximum power deposited on the RID panels is 19.6 MW**.** The maximum power density that could be encountered, 6.0 MW/m^2, occurs if the beam core divergence is 3 mrad.

4.21.2.2.8 NB Calorimiter

A movable calorimeter intercepts the neutral beam during the commissioning and conditioning phases. A single V-shaped calorimeter is selected. The maximum power deposited in the panels of the calorimeter, 21.6 MW, and the maximum power density, 22 MW/m^2, arise if the beam core divergence is 3 mrad.

The calorimeter panels consist of an array of swirl tube elements, parallel to the beam direction. A limited deflection of the swirl tube array (about 40 mm) can be allowed, and the secondary stresses can be limited to < 200 MPa.

4.21.2.2.9 NB Cooling

The neutraliser, the RID and the calorimeter are water-cooled. Swirl tube elements made of CuCrZr-alloy are foreseen in the RID, the calorimeter and in the neutraliser leading edge.

4.21.2.2.10 NB Vacuum Pumping

Large cryopumps maintain low pressure in the injector outside the ion source and the neutraliser. They are located close to the wall of the beam line vessel from the entrance of the neutraliser to the exit of the calorimeter. They cover all but the lower section of the beam line vessel where the support structure of the beam line components is located together with the coolant and the gas supply lines. A gas baffle is mounted at the RID entrance to subdivide the injector volume and therefore to achieve differential pumping.

4.21.2.2.11 NB Pressure Vessels

The injector is contained inside two large pressure vessels: the beam source vessel and the beam line vessel. Both are designed to guarantee the confinement of radioactive materials in case of an accidental overpressure: they are designed for a maximum internal pressure of 0.24 MPa (absolute pressure).

4.21.2.2.12 NB Shutter/Valve

A shutter is located at the exit of the injector vacuum vessel to prevent tritium flowing from the torus to the injectors and to allow regeneration of the injector cryopumps without significantly increasing the torus pressure. This is essentially a “door” which can be moved across the beam path to seal against a frame built into the exit of the beam line vessel.

A flexible metal seal on the frame ensures a low conductance between the injector and the torus of $\approx 10^{-4}$ m^3/s for D_2.

The shutter has an additional function to provide a vacuum barrier such as an isolation valve in case of failure in the flexible metal seal and/or the shutter action. A design change (DCR-33) is currently under investigation (2005) to establish whether this shutter can be made into a full absolute valve, allowing repairs to the NB lines without the necessity of a torus vent.

4.21.2.2.13 NB Drift Duct

The drift duct allows the flexible connection between the fast shutter casing (connected rigidly with the beam line vessel) and the NB duct, connected rigidly with the vacuum vessel.

Two bellows with equalizing rings connected with an intermediate cylinder are foreseen between the two extremity flanges. The drift duct liner faces the beam and protects the two bellows.

The duct box encloses the fast shutter and the drift duct and provides a double barrier, with guard vacuum, between the NB cell and the primary vacuum. It provides also the nuclear shielding in this region.

4.21.2.2.14 NB Magnetic Field Reduction System

The magnetic field reduction system consists of a passive magnetic shield and of active compensation and correction coils.

The passive magnetic shield (PMS) encloses the beam line and the beam source vessels, the HV bushing, the transmission line and the HV deck. The PMS is made of ferromagnetic steel with

150 mm thickness. It is a simple structure made of two construction steel plates each 75 mm thick, bolted together.

Six active compensation coils (three above and three below the PMS) limit the flux in the space inside the PMS and reduce the field in the PMS below its saturation value. Finally, an active correction coil, located between the beam line vessel and the fast shutter, could limit, if required, the error field produced by the NB system on the plasma. All coils are made of copper and are water-cooled.

4.21.2.3 Diagnostic Neutral Beam

The overall design of the DNB injector is identical to the one used for the H&CD injectors. This allows to utilise most of their R&D and design and to standardise the components, maintenance equipment and procedures.

4.21.2.3.1 DNB Source

In the beam source, one single stage of acceleration is used. The design assumes 300 A/m^2 as the accelerated current density of H^- corresponding to a total accelerated current of 60 A.

4.21.2.3.2 DNB Magnetic Field Reduction System

The requirements for the magnetic field reduction system are about a factor 4 more stringent than in the H&CD injectors, due to the lower beam energy. The line integral of the residual magnetic field in the space between the accelerator grounded grid and the neutraliser exit (about 4 m) has to stay at the level of $1x10^{-4}$ Tm.

4.21.2.3.3 DNB Residual Ion Dump

A RID is used for the deflection of the ions (H^- or H^+) exiting the neutraliser. The RID length is 1 m and the required deflecting voltage is limited to 4 kV. During "beam on" time the peak power density on the RID panel is about 0.75 MW/m^2 and total power to the RID is about 2.5 MW.

4.21.2.3.4 DNB Calorimiter

During "beam on" time the peak power density at the calorimeter position is 12 MW/m^2 (normal incidence) and total power is about 3.5 MW. The power is accommodated by a movable calorimeter with a heat receiving panel inclined 45° with respect to the beam line axis. Peak power density on the panel surface is about 8 MW/m^2.

Table 4.21-2 Diagnostic Neutral Beam, Main Parameters

Parameter	Unit	Value
Accelerated ion		H^-
Injection axis vertical position	mm	+1070
Beam energy	keV	100
Injected neutral (H^o) equivalent current	A	≥20
DNB pulse train duration	s	up to 3
DNB pulse train modulation frequency	Hz	5
Modulation wave form		Square (100 ms on/100 ms off)
Repetition time	s	20
Integrated number of pulse trains		1.0×10^6
Integrated number of modulations		1.5×10^7

Table 4.21-3 Neutral Beam Volumes

Neutral beam volume	Number	Volume (m^3)	Area (m^2)
Heating NB injector	2	150	600
Diagnostic NB injector	1	110	400

4.21.2.4 Cooling Conditions

See Table 4.11-4

4.21.2.5 PS Interfaces

See 4.16.2.3.4

4.22 Plasma Diagnostics (WBS 5.5)

(Updated A.Costley Oct 2006)

4.22.1 Functional Requirements

The function of the ITER plasma diagnostic system is to provide accurate measurements of plasma behaviour and performance. There are three categories of parameters to be measured:

group 1a	those needed for machine protection and basic machine control;
group 1b	those required for advanced plasma control;
group 2	those required for evaluation and physics studies.

The machine is unable to operate without a working diagnostic providing every group 1a parameter (1b for advanced operation). The machine may operate without a group 2 parameter diagnostic in operation. Measurements in each group are shown in Table 4.22-1. The requirements for each measurement are shown in Table 4.22-2.

Table 4.22-1 List of Required Plasma Measurements classified by their Operational Role

GROUP 1a Measurements For Machine Protection and Basic Control	GROUP 1b Measurements for Advanced Control	GROUP 2 Additional Measurements for Performance Evaluation 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. (divertor & upper plates) • Surface temperature (first wall) • Runaway electrons • 'Halo' currents • Radiated power (main plasma, X-point & divertor). • 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) • Resistive Wall Modes • 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 • ne of divertor plasma • Te of divertor plasma • Alpha particle loss • Low m/n MHD activity • Sawteeth • Net erosion (divertor plate) • Neutron fluence • Neoclassical Tearing Modes	• Confined α-particles • Fast ions (esp. H, D, T and He^3) • Profile of He^3 concentration • Total perp. fast ion energy content • TAE Modes, fishbones • Te and Ti profile (edge) • ne, Te profiles (X-point) • Ti in divertor • Plasma flow (divertor) • nT/nD/nH (edge) • nT/nD/nH (divertor) • Te fluctuations • ne fluctuations • Radial electric field and field fluctuations • Core and Edge turbulence • MHD activity in plasma core

Table 4.22-2 Requirements for Plasma and First Wall Measurements

Parameter Ranges, Target Measurement Resolutions and Accuracy

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	25 – 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	–	10 mm
		I_p Quench	–	10 ms	–	20 mm
	Divertor channel location (r dir.)	Default	–	10 ms	–	10 mm
		I_p Quench	–	10 ms	–	20 mm
	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 – 5	1 ms	Integral	5 % at β_p=1
		I_p Quench	0.01 – 5	1 ms	Integral	~ 30%
5. Radiated Power	Main Plasma P_{rad}	Default	TBD – 1 GW	10 ms	Integral	10 %
	X-point / MARFE region P_{rad}	Default	TBD – 0.3 GW	10 ms	Integral	10 %
	Divertor P_{rad}	Default	TBD – 0.3 GW	10 ms	Integral	10 %
	Total P_{rad}	Disruption	TBD – 50 GW	3 ms	Integral	20 %
6. Line-Averaged Electron Density	$\int n_e\, dl / \int dl$	Default	$1\bullet10^{18} – 4\bullet10^{20}$ /m^3	1 ms	Integral	1 %
		After killer pellet	$8\bullet10^{20} – 2\bullet10^{22}$ /m^3	1 ms	Integral	100 %
7. Neutron Flux and Emissivity	Total neutron flux		$1\bullet10^{14} – 5\bullet10^{20}$ n/s	1 ms	Integral	10 %
	Neutron / α source		$1\bullet10^{14} – 4\bullet10^{18}$ n/m^3/s	1 ms	a/10	10 %
	Fusion power		TBD – 1 GW	1 ms	Integral	10 %
	Fusion power density		TBD – 10 MW/m^3	1 ms	a/10	10 %
8. Error Field, Locked Mode and RWM	Br(mode)/Bp		$10^{-4} – 10^{-2}$	1 ms	(m,n) = (2,1) RWM, n=2,3	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					
10. Plasma Rotation	VTOR		1 – 200 km/s	10 ms	a/30	30 %
	VPOL		1 – 50 km/s	10 ms	a/30	30 %
11. Fuel Ratio in Plasma Core	nT/nD	r/a < 0.9	0.1 – 10	100 ms	a /10	20 %
12. Impurity Species Monitoring	Be, C rel. conc.		$1\bullet10^{-4} – 5\bullet10^{-2}$	10 ms	Integral	10 % (rel.)
	Be, C influx		$4\bullet10^{16} – 2\bullet10^{19}$ /s	10 ms	Integral	10 % (rel.)
	Cu rel. conc.		$1\bullet10^{-5} – 5\bullet10^{-3}$	10 ms	Integral	10 % (rel.)
	Cu influx		$4\bullet10^{15} – 2\bullet10^{18}$ /s	10 ms	Integral	10 % (rel.)
	W rel. conc.		$1\bullet10^{-6} – 5\bullet10^{-4}$	10 ms	Integral	10 % (rel.)
	W influx		$4\bullet10^{14} – 2\bullet10^{17}$ /s	10 ms	Integral	10 % (rel.)
	Extrinsic(Ne,Ar, Kr) rel. conc.		$1\bullet10^{-4} – 2\bullet10^{-2}$	10 ms	Integral	10 % (rel.)
	Extrinsic (Ne, Ar, Kr) influx		$4\bullet10^{16} – 8\bullet10^{18}$ /s	10 ms	Integral	10 % (rel.)

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION Time or Freq.	RESOLUTION Spatial or Wave No.	ACCURACY
13. Zeff(Line-averaged)	Zeff		1 – 5	10 ms	Integral	20 %
14. H-mode: ELMs and L-H Transition Indicator	ELM D_α bursts	Main Plasma	–	0.1 ms	One site	–
	ELM density transient	r/a > 0.9	TBD	TBD	TBD	TBD
	ELM temperature transient	r/a > 0.9	TBD	TBD	TBD	TBD
	L-H D_α step	Main Plasma		0.1 ms	One site	–
	L-H Pedestal formation (ne, Te)	r/a > 0.9	–	0.1 ms	–	TBD
15. Runaway Electrons	E_{max}		1 – 100 MeV	10 ms	–	20 %
	$I_{runaway}$	After Thermal quench	(0.05 – 0.7) $\bullet I_p$	10 ms		30 % rel
16. Divertor Operational Parameters	Max. surface temperature		200 – 2500°C	2 ms	–	10 %
	Erosion rate		1 – 10 x 10^{-6}m/s	2 s	10 mm	30 %
	Net erosion		0 – 3 mm	Per pulse	10 mm	12 x 10^{-6}m
	Gas pressure		$1\bullet10^{-4}$ – 20 Pa	50 ms	Several points	20 % during pulse
	Gas composition	A = 1–100 ΔA = 0.5	TBD	1 s	Several points	20 % during pulse
	Position of the ionisation front		0 – TBD m	1 ms	100 mm	–
17. First Wall (FW) Visible Image & Wall Temperature	FW image		TBD	100 ms	TBD	–
	FW surface temperature		200 – 1500°C	10 ms	TBD	20°C
18. Gas Pressure and Composition in Main Chamber	Gas pressure		$1\bullet10^{-4}$ – 20 Pa	1 s	Several points	20 % during pulse
	Gas composition	A = 1–100 ΔA = 0.5	TBD	10 s	Several points	50 % during pulse
19. Gas Pressure and Gas Composition in Ducts	Gas pressure		< 7 kPa	100 ms	Several points	20 % during pulse
	Gas composition	A = 1–100 ΔA = 0.5	TBD	1 s	Several points	20 % during pulse
20. In-Vessel Inspection	Wall image		100 % coverage of FW and divertor	–	1 mm	
21. Halo Currents	Poloidal current	In disruption	0 – 0.2 I_p	1 ms	9 sectors	20 %
22. Toroidal Magnetic Field	B_T		2 – 5.5 T	1 s	2 locations x 2 methods	0.1 %
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	5 mm	10 %
24. Electron Density Profile	Core N_e	r/a < 0.9	$3\bullet10^{19}$ – $3\bullet10^{20}$ /m^3	10 ms	a/30	5 %
	Edge N_e	r/a > 0.9	$5\bullet10^{18}$ – $3\bullet10^{20}$ /m^3	10 ms	5 mm	5 %
25. Current Profile	q(r)	Physics study	0.5 – 5	10 ms	a/20	10 %
			5 – TBD	10 ms	a/20	0.5
	r(q=1.5,2)/a	NTM feedback	0.3 – 0.9	10 ms	–	50 mm/a
	$r(q_{min})/a$	Reverse shear control	0.3 – 0.7	1 s	–	50 mm/a
26. Z_{eff} Profile	Z_{eff}	Default	1-5	100 ms	a/10	10 %
		Transients	1-5	10 ms	a/10	20 %
27. High Frequency Instabilities (MHD, NTMs, AEs, turbulence	Fishbone–induced perturbations in B,T,n		TBD	0.1 – 10 kHz	(m,n) =(1,1)	–
	TAE mode – induced perturbations in B,T,n		TBD	30 – 300 kHz	n = 10 – 50	–
	NTM			10 – 100 KHz	10 mm	

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION Time or Freq.	RESOLUTION Spatial or Wave No.	ACCURACY
28. Ion Temperature Profile	Core T_i	r/a < 0.9	0.5 – 40 keV	100 ms	a/30	10 %
	Edge T_i	r/a > 0.9	0.05 – 10 keV	100 ms	TBD	10 %
29. Core He Density	n_{He}/n_e	r/a < 0.9	1 – 20 %	100 ms	a/10	10 %
30. Confined Alphas and Fast Ions	Alpha Energy spectrum	Energy resolution TBD	(0.1 – 3.5) MeV	100 ms	a/10	20 %
	Alpha Density Profile		$(0.1 – 2)\bullet10^{18}/m^3$	100 ms	a/10	20 %
	p,D,T,He^3	tbd	tbd	tbd	tbd	tbd
31. Escaping Alphas and Fast Ions	First wall flux	Default	TBD – 2 MW/m^3	100 ms	a/10 (along poloidal direction)	10 %
		Transients	TBD – 20 MW/m^3	10 ms	TBD	30 %
32. Impurity Density Profile	Fractional content, Z<=10	r/a < 0.9	0.5 – 20 %	100 ms	a/10	20 %
		r/a > 0.9	0.5 – 20 %	100 ms	50 mm	20 %
	Fractional content, Z>10	r/a < 0.9	0.01 – 0.3 %	100 ms	a/10	20 %
		r/a > 0.9	0.01 – 0.3 %	100 ms	50 mm	20 %
33. Fuel Ratio in the Edge	n_T/n_D	r/a > 0.9	0.1 – 10	100 ms	Radial integral	20 %
	n_H/n_D	r/a > 0.9	0.01 – 0.1	100 ms	Radial integral	20 %
34. Neutron Fluence	First wall fluence		0.1 – 1 MWy / m^2	10 s	TBD	10 %
35. Impurity and D,T Influx in Divertor	$\Gamma_{Be}, \Gamma_C, \Gamma_W$		$10^{17} – 10^{22}$ at/s	1 ms	50 mm	30 %
	Γ_D, Γ_T		$10^{19} – 10^{25}$ at/s	1 ms	50 mm	30 %
36. Plasma Parameters at the Divertor Targets	Ion flux		$10^{19} – 10^{25}$ ions/s	1 ms	3 mm	30 %
	n_e		$10^{18} – 10^{22}\ /m^3$	1 ms	3 mm	30 %
	T_e		1 eV – 1 keV	1 ms	3 mm	30 %
37. Radiation Profile	Main plasma P_{rad}		0.01 – 1 MW/m^3	10 ms	a/30	20 %
	X-point/MARFE region P_{rad}		TBD – 300 MW/m^3	10 ms	a/30	20 %
	Divertor P_{rad}		TBD – 100 MW/m^3	10 ms	50 mm	30 %
38. Heat Loading Profile in Divertor	Surface temperature		200 – 1000°C	2 ms	3 mm	10 %
			1000 – 2500°C	$20\ x10^{-6}$ s	3 mm	10 %
	Power load	Default	TBD – 25 MW/m^2	2 ms	3 mm	10 %
		Disruption	TBD – 5 GW/m^2	0.1 ms	TBD	20 %
39. Divertor Helium Density	n_{He}		$10^{17} – 10^{21}\ /m^3$	1 ms	–	20 %
40. Fuel Ratio in the Divertor	n_T/n_D		0.1 – 10	100 ms	integral	20 %
	n_H/n_D		0.01 – 0.1	100 ms	integral	20 %
41. Divertor Electron Parameters	n_e		$10^{19} – 10^{22}\ /m^3$	1 ms	50 mm along leg, 3 mm across leg	20 %
	T_e		0.3 – 200 eV	1 ms	50 mm along leg, 3 mm across leg	20 %
42. Ion Temperature in Divertor	T_i		0.3 – 200 eV	1 ms	50 mm along leg, 3 mm across leg	20 %
43. Divertor Plasma Flow	V_p		TBD – 10^5 m/s	1 ms	100 mm along leg, 3 mm across leg	20 %
44. n_H/n_D Ratio in Plasma Core	n_H/n_D		0.01 – 0.1	100 ms	a/10	20 %
45. Neutral Density between Plasma and First Wall	D/T influx in main chamber		$10^{18} – 10^{20}$ $at/m^2/s$	100 ms	Several poloidal and toroidal locations	30 %

4.22.2 Configuration

4.22.2.1 Overall

Meeting the functional specification requires approximately 50 diagnostic systems. These are shown in Table 4.22-3. For convenience, they are divided into seven groups, mainly by the principal technology employed in the instrumentation (WBS 5.5.A-G). Most systems have been included in the cost estimation and budget provision. However, since the costing was performed, the design of some systems necessary to meet the measurement requirements have matured. These 'uncredited systems' are included in the technical provisions and particularly in the system integration. They are shown in italics in in Table 4.22-3.

The common engineering needs of the diagnostic systems, for example support and shielding structures, are met by a separate group of systems (WBS 5.5.N). Plasma diagnostic components are located in and on the main vessel, and in all three levels of ports. In addition, diagnostic equipment exists in the cells, pit gallery, the three levels of the diagnostic building and certain other locations specific to particular diagnostics.

The distribution of diagnostics over the ports is shown in (click for link)
Figure 4.22-1 Distribution of Diagnostics Mounted on the Vacuum Vessel

and Table 4.22-3 List of diagnostic systems, showing their location

The Diagnostic Engineering Systems provide common interfaces between individual diagnostics and other tokamak systems and so are discussed first below. A brief description of the diagnostics of the seven other groups follows, emphasizing configuration and interface points. More details of the systems are in the Overview DDD (WBS 5.5).

4.22.2.2 Diagnostic Engineering (WBS 5.5.N)

4.22.2.2.1 In-Vessel Services

(WBS 5.5.N.01)

In-vessel services comprise the set of common electrical, gas distribution and similar hardware located within the primary vacuum up to and including the first feedthrough and installed to support the operation of in-vessel sensors distributed through the vessel as well as those inside the diagnostic port plugs and in the cryostat.

The majority of the hardware in this category is electrical wiring. The reference cable used in-vessel is Mineral Insulated Cable (MIC) sized to meet the electrical insulation and current carrying requirements for each application.

The reference insulation is Alumina, chosen for its performance under irradiation. The conductor depends on the application but in the majority of cases is copper. Noise reduction techniques (twisted pair, triaxial construction, etc.) will be used where appropriate. However, since the costing was performed, the design of some systems necessary to meet the measurement requirements have matured. Departures from this reference may be needed for specific diagnostics.

The cabling is marshaled into looms, which incorporate suitable mechanical and thermal attachment features depending on the application. The looms on the vacuum vessel inner surface run along notional conduit paths cut out of the back of blanket modules to connectors on the inner sides of the upper ports, themselves connected via secondary looms to feedthroughs mounted above the port flange. Wiring within ports is joined using automatic or remote handling compatible connectors with looms terminating at feedthroughs on the port closure. Wiring from divertor sensors terminates at a remote handling connector, connected in turn to a secondary loom bearing the signals to a feedthrough mounted on the side of the divertor ports.

Services other than wiring, for example special cooling loops, gas actuators, mechanical feedthroughs etc. may be required in specific diagnostic ports depending on the detail design of individual systems.

4.22.2.2.2 Port Plugs And First Closures

(WBS 5.5.N.03)

Diagnostic equipment in ports is mounted in a port plug assembly. This provides the primary vacuum boundary at that port, and bears the feed-outs for diagnostic signals (windows and feedthroughs) and feed-ins for control signals. Within the primary vacuum it provides the support for the diagnostic equipment, the shielding, and the support for the blanket shield module with its first wall protection and shielding and suitable diagnostic apertures.

Labyrinthine diagnostic channels are used in the port plug for shielding and this brings about the most significant limitation to the number of diagnostics that can be installed at any port. The primary concern is to provide sufficient shielding, equivalent to that lost accommodating the diagnostic channels, to limit the nuclear heating of the cryogenic coils of the magnet system as well as to prevent high activation of diagnostic system and the port cell.

Modular subassemblies are used within the port plugs. The port plug modularity allows a single concept for remote maintenance of all the port plugs and a standardised approach for hot cell maintenance. In some ports there is a relatively simple connection between the port plugs and the port cell equipment while in others there are substantial connections of transmission lines or vacuum extensions.

4.22.2.2.3 Interspace Blocks

(WBS 5.5.N.04)

Outside the closure flange of the port plug, a structure is located in the connecting duct between the vessel flange and the bioshield. This structure, termed an “interspace block” supports connecting waveguides, mirror assemblies and vacuum extensions as appropriate. It is removed as a unit to access the seal plate for port maintenance operations.

4.22.2.2.4 Divertor Components

(WBS 5.5.N.05)

Divertor diagnostic components includes:

- Special modular assemblies incorporating, supporting and /or cooling sensors and mounted in turn on the diagnostic and instrumented cassette sides.
- Diagnostic racks, located within the divertor level ports at diagnostic cassette locations, supporting diagnostic components in this region.

4.22.2.2.5 Ex-Bioshield Electrical Equipment

(WBS 5.5.N.06)

This system includes all wiring connecting all diagnostic sensors and actuators with their control and data acquisition equipment, as well as related hardware (cubicles, conduits etc.). It is an extensively distributed system, with a presence in every port as well as the gallery and the diagnostic hall.

In the port cell, wiring is collected from the port plugs and brought on cable trays into marshalling boxes on the side of the port cell. From there, it is taken through the port cell boundary by appropriate feedthroughs and laid on dedicated cable trays near the gallery ceiling across to the set of cable trays running along the gallery walls. From there it is brought to one of the penetrations into the diagnostic hall that are present at all three levels, through a second, low pressure, feedthrough and taken to the appropriate cubicle bay or diagnostic specific area.

4.22.2.2.6 Window Assemblies

(WBS 5.5.N.07)

For complete installation of the plasma and first wall diagnostic system in ITER there are planned to be more than 200 diagnostic windows on the ITER primary vacuum boundary. The windows, diffusion boned to metal ferrules, are incorporated in window assemblies that are welded into port plugs and seal plates. Their function is to:

1. Maintain vacuum boundary
2. Withstand failure pressures
3. Transmit diagnostic signals

There are approximately 30 types of window assembly, distinguished by size, material, angle and bond technique and RH port size. All types will be proved by destructive type testing and all windows in use will be provided with a replacement spare.

In addition there will be a somewhat smaller number of secondary boundary windows (some diagnostics use a valve, or the instrument itself as the second boundary) and a number yet to be determined of tertiary windows on crossing from the gallery to the diagnostic hall.

Table 4.22-3 List of diagnostic systems, showing their location

4.22.2.2.6.1.1 GROUP	WBS 5.5…	Diagnostic	Innermost Component Location
Magnetics	A.01	Outer Vessel Sensors	Vacuum Vessel (VV) Outer Surface
	A.02	Inner Vessel Sensors	VV Inner Surface
	A.03	Divertor Magnetics	Divertor Cassette (DC)
	A.04	External Rogowski	Within TF coil case
	A.05	Diamagnetic Loop	VV Outer surface; VV Inner surface; around TF case
	A.06	Halo Current Sensors	VV Inner Surface; DC
Neutron	B.01	Radial Neutron Camera	Equatorial Port (EP)
	B.02	Vertical Neutron Camera	Upper Port (UP); Cryostat
	B.03	Microfission Chambers	VV Inner Surface
	B.04	Neutron Flux Monitors	EP; Divertor Port (DP); UP;
	B.07	**Gamma Ray Spectrometers*	EP
	B.08	Activation System	VV Inner Surface
	B.09	* *Lost Alpha Detectors*	Side of Blanket module
	B.10	* *Knock-on-Tail Spectrometer*	EP/UP
	B.11	* *High Resolution Neutron Spectrometer*	EP
Optical	C.01	Thomson Scattering (Core)	EP
	C.02	Thomson Scattering (Edge)	UP
	C.03	Thomson Scattering (X Point)	DP
	C.04	* *Thomson Scattering (Divertor, Outer)*	DP
	C.05	Toroidal Interferometer/Polarimeter	EP; VV inner surface; Other port BSM
	C.06	Polarimeter	EP;UP; Blanket slots
	C.07	* *Collective Scattering*	EP; VV Inner surface
	C.08	* *Thomson Scattering (Divertor, Inner)*	EP
Bolometric	D.01	Bolometers (All)	EP, UP, DC,VV Inner surface
Spectroscopic and NPA	E.01	CXRS Based On DNB (Core)	UP
	E.02	H-Alpha	EP, UP, DP
	E.03	VUV (Main Plasma)	EP; UP
	E.04	Impurity / Influx Mon (Divertor Visible / UV)	EP, UP, DP
	E.05	X-Ray Crystal Spectrometer	EP, UP
	E.06	Visible Continuum Array	EP
	E.07	* *Soft X-Ray Array*	EP, UP, VV Inner surface
	E.08	Neutral Particle Analyser	EP
	E.10	* *Laser Induced Fluorescence*	DP
	E.11	MSE Based On Heating Beam	EP
	E.12	CXRS Based On DNB (Edge)	EP
Microwave	F.01	ECE(Main Plasma)	EP
	F.02	Reflectometer (Main Plasma, LFS)	EP
	F.03	Reflectometer (Plasma Posn)	VV Inner surface
	F.04	Reflectometer (Divertor)	DC
	F.07	* *Fast Wave Reflectometry*	EP; VV Inner surface
	F.09	Reflectometer (Main Plasma, HFS)	VV Inner surface
	F.10	Interferometer (Divertor)	DC
Plasma Facing and Operational	G.01	IR Cameras, Vis/IR TV (Midplane)	EP
	G.02	Thermocouples	DP
	G.03	Pressure Gauges	DC
	G.04	Residual Gas Analyzers	DP; NBI pump
	G.06	*IR Thermography (Div)*	DP
	G.07	Langmuir Probes	DC
	G.08	* *Erosion Monitor*	DP
	G.09	* *Dust Monitor*	DC
	G.10	IR Cameras, Vis/IR TV (Upper)	UP
Diagnostic Engineering	N.01	In-Vessel Services	VV Inner surface; DC
	N.03	Port Plugs And First Closures	In port BSM hole
	N.04	Interspace Blocks And Second Closures	On VV flange (air side)
	N.05	Divertor Components	DC
	N.06	Ex-Bioshield Electrical Equipment	Port Cell
	N.07	Window Assemblies	VV flange

Figure 4.22-1 Distribution of Diagnostics Mounted on the Vacuum Vessel

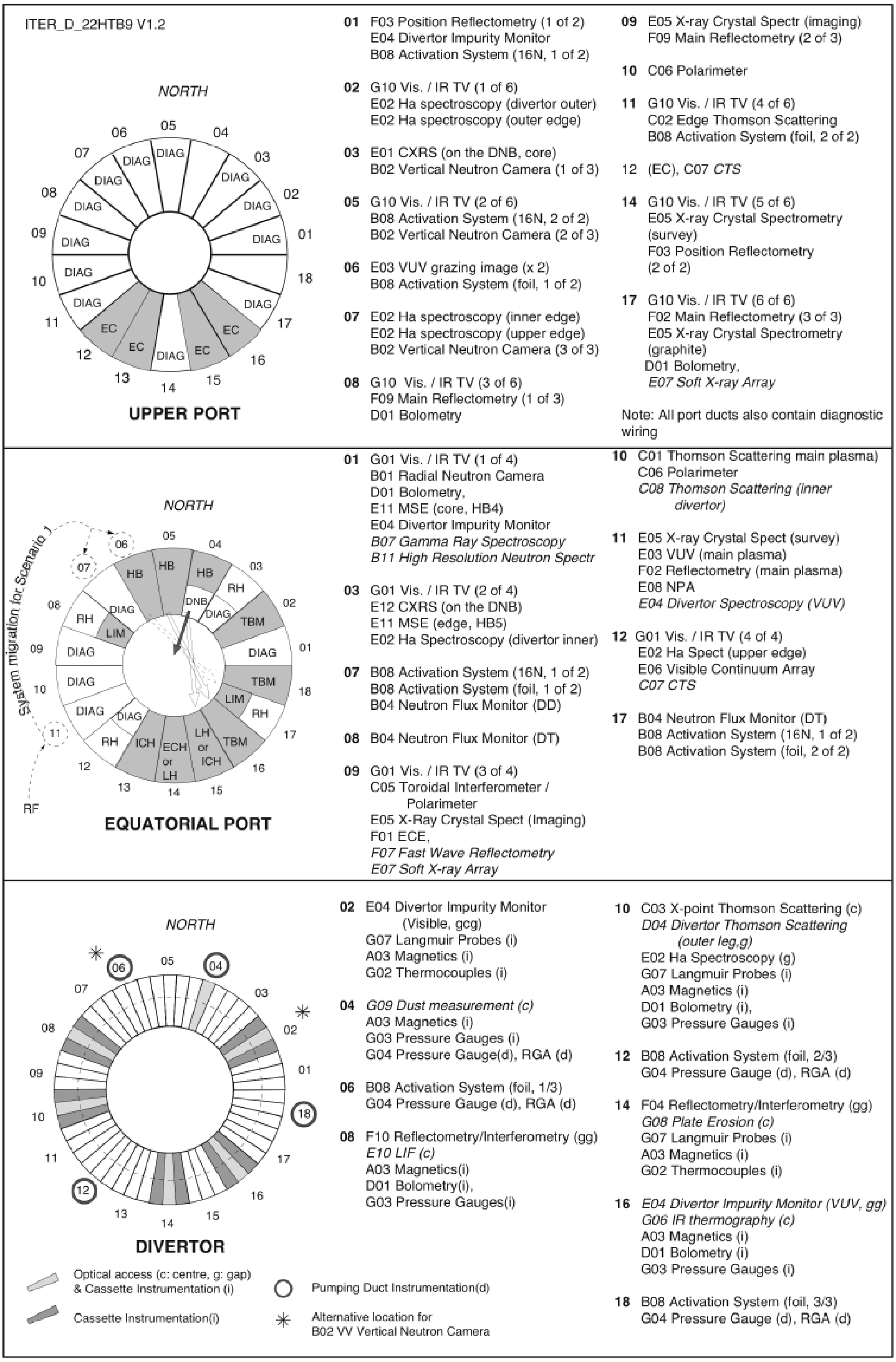


Table 4.22-4 Allocation for the port-based diagnostic systems

WBS	DIAGNOSTIC	Unit	SECTOR	1		2		3		4		5		6		7		8		9	
			PORT	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
-	Diagnostic Neutral Beam	Beam Duct					1														
A.01	Outer Flux Loops	Loop Feed Point					3												3		
A.01	Outer Vessel Sensors	2-axis Coil Pair					40		40		40					40		40		40	
A.01	Outer Vessel Sensors (steady state)	2-axis Hall Probe				40		40		40					40		40		40		
A.02	Continuous Flux Loop	Loop Feed Point					2						2						2		
A.02	HF Coil Poloidal Array	Coil				20			20			20			20			20			20
A.02	MHD – Dedicated Saddle Loop	Loop		8		8		8		8		8		8		8		8		8	
A.02	Partial Loops	Loop				18			18			18			18			18			18
A.02	Resistive Wall Mode sensors	Coil			1			1			1			1			1			1	
A.02	Tangential, Normal & Toroidal Coils, Hardwired	Coil			34						34						34				
A.02	Tangential, Normal & Toroidal Coils, Replaceable	Coil					34						34						34		
A.05	Diamagnetic Loop	Loop				1						1						1			
A.05	Diamagnetic Loop Compensation Coil	Coil				1						1						1			
A.05	Diamagnetic Loop Compensation Saddle	Loop				2						2						2			
A.06	Halo Rogowski Poloidal Array	Coil		14			14			14			14			14			14		
A.06	Halo Rogowski Toroidal Array	Coil		4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4
B.02	Vertical Neutron Camera (VV divertor level)	Collimator							2												
B.03	Microfission Chambers	Detector				6									6						
B.08	Neutron Activation (^{16}N Fluid)	Exposure Cell								1											
B.08	Neutron Activation (Foil Capsule)	Exposure Cell							4	1				3	1						1
B.09	*Lost Alpha Detector*	Alpha Spectrometer		2																	
C.07	*Collective Thomson Scattering*	Waveguide													8						
D.01	Bolometer Array	Miniature Camera		4					4	5	1							2	4	3	
E.07	*Soft X-ray Array*	Line of Sight	[1]	13	6	6	4	6						14		16	4				
F.03	Position Reflectometry	Waveguide & Antenna			2												2				
F.07	*Fast Wave Reflectometry*	Coil	[2]																		
F.09	Main Reflectometry (High Field Side)	Waveguide & Antenna									4									4	
F.11	*Refractometry*	Waveguide & Antenna										4									
G.02	Thermocouples, inboard	Thermocouple Element		4	4	4		4			4	4		4			4			4	
G.02	Thermocouples, outboard	Thermocouple Element		3			6						6						6		
G.09	*Dust Monitor*	Location	[3]																		
N.01	Cryostat Cabling	100-Core Loom				5		5		5					5		5		5		
N.01	Divertor Port Cabling	80-Way Feedthrough			8		3		1		8		4		1		6		7		1
N.01	Upper Port Cabling	48-Way Feedthrough		4	3	5	3	3	5	5	3	3	5	4	2	5	3	4	6	2	4
N.01	VV Wall Poloidal Cable Conduits	36-Core Loom		3	2	4	2	2	4	3	2	2	3	2	2	3	2	2	4	2	2

[1] Two to three lines of sight per location on vessel [2] The number and locations of receiving coils is TBD. [3] The type and location of dust monitoring instrumentation is TBD.

4.22.2.3 Magnetics (WBS 5.5.A)

The magnetic diagnostics comprise several subsystems:

- sets of pick-up coils, saddle loops and voltage loops mounted on the inner wall of the vacuum vessel for equilibrium and high frequency measurements (inner vessel sensors, WBS 5.5.A.02);
- sets of coils mounted in the divertor diagnostic cassettes (WBS 5.5.A.03);
- sets of pick-up coils and steady state sensors for back-up measurements mounted on the outer surface of the vacuum vessel (outer vessel sensors, WBS 5.5.A.01);
- continuous poloidal (Rogowski) loops mounted within the TF coil case (WBS 5.5.A.04);
- a diamagnetic system comprising poloidal loops on the inner wall of the VV and compensation circuits inside and outside the vessel (WBS 5.5.A.05);
- Rogowski coils mounted around earth straps of the blanket/shield modules and divertor structures for measuring the 'halo' currents (WBS 5.5.A.06).

The inner vessel system in turn comprises:

- tangential, normal and toroidal equilibrium coils mounted on the inner surface of the VV;
- tangential high frequency coils mounted on the inner surface of the VV;
- complete and partial flux loops mounted on the inner surface of the VV;
- saddle loops mounted on the inner surface of the VV dedicated to MHD measurements (dedicated saddle loops);
- a (temporary) PF and TF error field measurement assembly used during machine construction;
- Additional sensors dedicated to resistive wall mode stabilization (two sensors per error field correction coil located near the coil centre).

4.22.2.3.1 Inner Vessel Sensors and Divertor sensors

(WBS 5.5.A.02 & 03)

The pick-up coils mounted (with good thermal and mechanical contact) on the inner surface of the VV and within the divertor are constructed to fit available cutouts with suitable margins to ensure they survive blanket replacement. They are cooled by conduction to the vessel in the case of the equilibrium coils and by radiation in the case of the high frequency coils.

There are two equilibrium coil sets, a primary and a secondary set. The first is permanent. The other is in-situ replaceable by remote handling with a suitable welded connector. Each set sums the signals from three sectors, 120° apart.

The equilibrium loops are similarly grouped in two sets of three. They are all hardwired and made with mineral insulated cable (MIC), and held with frequent spot welded stainless steel clips. Special connectors that can be replaced by remote handling are provided for a small number of continuous flux loops for each of the nine field joints.

The high frequency coils are optimized to extend frequency response at the expense of low frequency sensitivity, temperature control and mechanical robustness.

The dedicated saddle loops used for MHD are similar to, but generally larger than, the equilibrium loops. They are one every one of the nine manufacturing sectors in order to provide toroidal mode number discrimination.

The coils, the cable and other components with embedded insulation are sufficiently shielded by the presence of the blanket modules that their lifetime is comparable to, or longer, than the lifetime of

ITER. All coil insulation is rated for the worst expected voltage (typically during disruptions) with a margin of a factor of ten to account for the presence of ionizing radiation during the high voltage state. The exception to this is the temporary field mapping system. This is intended to map machine error fields once during the late stages of construction.

4.22.2.3.2 Diamagnetic Loop

(WBS 5.5.A.05)

The diamagnetic loop system measures the magnetic flux expelled by the plasma. From this measurement, the perpendicular energy content can be derived. This in turn gives the confinement time. There are three diamagnetic loop sets to provide redundancy.

Each set uses a pair of poloidal loops on the inner wall of the vessel, and compensation circuits. The poloidal loops run in one poloidal plane, splitting apart in a symmetric manner to bypass vacuum vessel features. They respond on the vessel poloidal decay time scale. The compensation circuits measure the vacuum flux inside and outside the vessel, and the local vertical field. These are: a multi-turn compensation coil of the same surface area as the diamagnetic loop inside the vacuum vessel below the triangular support, a similar coil just outside the vacuum vessel, a Rogowski wound around the TF coil casing on the low field side and a pair of narrow saddle loops at each diamagnetic loop toroidal location.

The exit wires are twisted and routed out of the vessel through the nearest divertor port, using the feed-throughs and connectors of the divertor coils (5.5.A.03).

4.22.2.3.3 Halo Current Sensors

(WBS 5.5.A.06)

Separate sensors are used to measure halo currents by measuring the current in the earth straps and key divertor module sub-circuits using Rogowski coils. A blanket module requires two Rogowskis (one per strap). These are mounted on the blanket electrical connector on the vacuum vessel side. Their wires are brought out of the vessel in the same way as the inner vessel sensors (5.5.A.02). Eight Rogowskis are used to map the currents flowing within the cassette.

Not all modules are instrumented. The present design attempts to obtain reasonable coverage of the important areas, providing full poloidal coverage (including the divertor) at six locations 60° apart toroidally and in addition full toroidal coverage for certain key poloidal locations.

4.22.2.3.4 Outer Vessel Sensors

(WBS 5.5.A.01)

There are three types of these sensors:

- Sets of pick up coils tangential and normal to the vessel
- Sets of steady-state sensors based on the Hall effect
- A small number of continuous flux loops.

In all cases these sensors have to exist in the gap available between the vacuum vessel thermal shield and the vessel (typically less than 10 mm).

The coils are designed to have about ten times higher effective area than their inner vessel counterparts with much lower frequency response. The Hall sensors are of the self-calibrating type and will allow detection of any residual drift on the coils. They may also allow monitoring of the state of the ferromagnetic inserts in the absence of plasma. The flux loops use connectors that can be actuated from the inside of the vessel by remote handling prior to the sector assembly weld.

4.22.2.3.5 External Rogowski

(WBS 5.5.A.04)

The external Rogowski is designed to measure the total (vessel + plasma) current on a slow timescale. It is embedded in several TF coils, using a conduit similar and alongside that of the He cooling pipes of the inner surface of the TF coil casing. Its response time is limited by the casing (about 70 ms).

4.22.2.4 Neutron Systems (WBS 5.5.B)

The principal neutron systems are a radial neutron camera, a vertical neutron camera, neutron spectrometers, neutron flux monitors, and a neutron activation system.

4.22.2.4.1 Radial Neutron Camera

(WBS 5.5.B.01)

The radial neutron camera consists of a fan-shaped array of flight tubes, viewing the plasma through a vertical slot in the blanket shield module of an equatorial port plug. The sight lines intersect at a common aperture defined by the port plug and penetrate the vacuum vessel, cryostat, and biological shield through stainless steel windows. Additional sight lines with collimators and detectors mounted in the port plug are included to give measurements in the outer regions of the plasma.

4.22.2.4.2 Vertical Neutron Camera

(WBS 5.5.B.02)

Because ITER does not have vertical ports it is difficulty to measure the neutron emission in a vertical direction (necessary for combination with the radial measurements for tomographic reconstructions of the neutron source profile). A version installed in a divertor port is currently under development (October 2006).

4.22.2.4.3 Neutron Flux Monitors and Microfission Chambers

(WBS 5.5.B. 04 & 03)

The neutron flux is measured by fission chambers containing ^{235}U or other isotopes, situated at different locations in the diagnostic ports and outside the VV. In addition, micro-fission chambers are deployed behind the blanket modules in poloidal arrays at two toroidal locations.

4.22.2.4.4 Neutron Activation System

(WBS 5.5.B.08)

Two activation systems are planned. One uses pneumatic transfer to place a sample of material close to the plasma for irradiation. The second system measures the gamma rays from the decay of 16N produced in a flowing fluid. The activated samples in both cases are brought out to counting stations located in the tokamak building.

4.22.2.5 Optical Systems (WBS 5.5.C)

The principal optical systems are two multi-pulse Thomson scattering (TS) systems (core and edge), an equatorial plane interferometer, and a poloidal interferometer / polarimeter.

Additional planned optical systems under investigation are Thomson scattering systems for the X-point and divertor regions, and a collective scattering system to provide measurements of the confined alpha particle population.

4.22.2.5.1 Core Thomson Scattering

(WBS 5.5.C.01)

The core TS system operates on the time-of-flight (LIDAR) principle. Light from a high power laser is transmitted to the plasma using a folded mirror arrangement inside a shielded labyrinth at an equatorial port. The plasma-facing mirror is metallic and actively cooled. Scattered radiation returns along the same labyrinth to remote spectrometers. An active alignment system is employed to compensate for movements of different parts of the system.

The high power laser light is sent to the port cell using a wide bore trasmission line. The collected light is similarly brought back to the diagnostic hall.

4.22.2.5.2 Edge Thomson Scattering

(WBS 5.5.C.02)

In order to meet the requirements for high-resolution measurements in the edge region, a conventional Thomson scattering system is employed. The large upper ports permit the installation of separate input and collection lines in the same port.

The high power laser light is sent to the port cell using a wide bore light tube. The collected light is brought back to the diagnostic hall by fibres.

4.22.2.5.3 X-point Thomson Scattering

(WBS 5.5.C.03)

A LIDAR system will measure across the lower X-point by means of a special aperture at the top of a divertor cassette, with the primary optics in the diagnostic rack. Other than the location and design details (power, rep. Rate, wavelength), it is very similar to 5.5.C.01.

4.22.2.5.4 Divertor Thomson Scattering – Outer (Provisional)

(WBS 5.5.C.04)

A conventional Thomson scattering system (as 5.5 C.02) will measure along the leg of the plasma in the divertor region. It will view the plasma through the inter-cassette gap, with its optics in the diagnostic rack.

4.22.2.5.5 Equatorial Plane Interferometer

(WBS 5.5.C.05)

A vibration-compensated interferometer employing Faraday rotation techniques will be used to measure line-integrated density for use in feedback control. The plasma is probed along five lines of sight in the equatorial plane. Each beam path has collinear 10.6 μm (CO_2) and 5.3 μm (CO) laser probe beams.

The retroreflectors for this system are located at suitable locations behind the blanket modules. The blanket-to-blanket toroidally running gap, suitably enlarged, is used to channel the radiation to the retroreflector. A total of ten transmission lines consisting of lens relays in a light guard / alignment tube are used to couple the lasers and detectors to the diagnostic block optics.

4.22.2.5.6 Poloidal Interferometer/Polarimeter

(WBS 5.5.C.06)

Multi-chord polarimetry in the poloidal plane will be used to provide measurements of the q profile and/or to anchor reconstructions of the magnetic equilibrium.

As for the interferometer, in-vessel retroreflectors have to be used, but in this case it is proposed to mount them in the remote handling vertical slots in the blanket modules.

Two views are included in the design, one in an upper and one in an equatorial port, with a total of > 10 sightlines. This combination allows the system to come close to the required performance. Special transmission lines channel the radiation (118 μm wavelength) to and from the diagnostic hall at two levels. A special penetration in the diagnostic hall allows the use of the lasers / detectors in the equatorial hall to be used with upper level transmission lines and vice versa.

4.22.2.5.7 Collective Scattering (Provisional)

(WBS 5.5.C.07)

This is a system designed to measure key properties of the confined alpha and other fast particle populations. It employs a pair of gyrotron sources through a main port, and detectors in the same port and within the vessel. The launcher technology is thus similar to that of the ECH system and it is proposed to locate the gyrotrons in the RF hall. The detection hardware and waveguide access are similar to those for the reflectometer for plasma position (5.5.F.03).

4.22.2.5.8 Divertor Thomson Scattering – Inner (Provisional)

(WBS 5.5.C.08)

This is a LIDAR system under investigation for the measurement of the density and temperature profile along the inner divertor leg, where the capability of the present diagnostic set is very weak. If installed, it would operate through an equatorial port.

4.22.2.6 Bolometry (WBS 5.5.D)

The bolometric systems aim to provide the spatial distribution of the radiated power in the main plasma and in the divertor region, using sparse-data tomography. About 340 lines-of-sight are required to meet the measurement requirements. The lines are collected in groups by bolometer arrays.

4.22.2.6.1 Bolometer Arrays

The bolometer arrays will be installed in the equatorial and upper ports, in the specially instrumented diagnostic divertor cassettes and at selected locations on the VV. From each of these locations several arrays of lines of sight observe the plasma in a fan-shaped geometry. From the equatorial port, the inner divertor leg and the main plasma are viewed. From the upper port, the main plasma, the area of the X-point, and the largest part of the divertor legs, can be seen.

Bolometers mounted on the VV view the plasma through the poloidal gaps between adjacent blanket modules. In the equatorial and upper ports, the bolometer arrays are integrated rigid units with all wiring attached. In the divertor, there are multiple small heads assembled in a rigid conduit with the wiring attached to the sidewall of the instrumented divertor cassette. The plasma is viewed through the gap between cassettes.

4.22.2.7 Spectroscopic and Neutral Particle Analyzer Systems (WBS 5.5.E)

An extensive array of spectroscopic instrumentation will be installed covering the visible to X-ray wavelength range. Both passive and active measurement techniques will be employed. Table 4.22-5 shows a summary.

Most systems use labyrinths and mirrors to decrease the radiation flux to the detectors and maintenance areas. The exceptions are the NPA and systems operating in the UV and VUV region, where mirrors, either can't be used or have to be used at grazing incidence and are ineffective as neutron labyrinths. Most systems use a window for the primary barrier, with the exception of the NPA and VUV systems, which are directly coupled and located in the port cell. Fibres are used to transmit visible and IR signals to the spectrometers as early in the signal chain as possible.

Table 4.22-5 Summary of Spectroscopy and NPA Diagnostics

WBS	Instrument	Wavelength Range	Regions Probed/ Viewing Directions	Function
5.5.E.01 and 5.5.E.12	CXRS	Visible region	Core and edge	T_i (r), He ash density, impurity density profile, plasma rotation, alphas.
5.5.E.02	Hα system	Visible region	Main plasma: inner, outer and upper regions Divertor: inboard and outboard regions	ELMs, L/H mode indicator, n_T/n_D and n_H/n_D at edge and in divertor.
5.5.E.03	VUV (main plasma)	2.3 – 160 nm	Upper and equatorial regions; divertor region	Impurity species identification.
5.5.E.04	Divertor impurity monitor	200 – 1000 nm (*10 – 200 nm under consideration*)	Divertor and X – point regions	Impurity species and influx, divertor He density, ionisation front position, T_i.
5.5.E.05	X-ray (survey and high resolution)	0.1 – 2.5 nm	Core and edge	Impurity species identification, plasma rotation, T_i.
5.5.E.06	Visible continuum array	λ = 523 nm	Core plasma	Z_{eff}(r), line averaged electron density indicator.
5.5.E.08	NPA	N/A (10 – 200 keV, 0.1 – 4 MeV)	Core and edge	n_T/n_D and n_H/n_D at edge and core. Fast alphas.
5.5.E.11	MSE	Visible region	Core and edge	q (r)

Abbreviation: LOS = Line(s) of sight

4.22.2.7.1 CXRS and Beam Emmission Spectroscopy

(WBS 5.5.E.01 & 12)

This system, employing Charge Exchange Recombination Spectroscopy (CXRS) makes use of the dedicated diagnostic neutral beam (DNB). The emission of the beam, the spatial profile of ion temperature, plasma toroidal and poloidal rotation velocities and the spatial profiles of impurity densities (including helium ash density) will be measured.

The beam is viewed through optical labyrinths in the upper port above the DNB and in an equatorial port chosen for adequate spatial resolution. One system is installed at the upper port for measurements in the plasma core and two systems are installed at the equatorial port for measurements from the edge to mid-radius regions.

4.22.2.7.2 H-alpha Spectroscopy

(WBS 5.5.E.02)

The H-alpha spectroscopy system measures the emission in the Balmer hydrogen lines. Several wide-angular optical systems located in the upper and equatorial ports view the upper, inner, and outer regions of the main plasma and the divertor region from above. Collected light is transmitted by mirror optics through the labyrinths in port plugs and port ducts and then focused onto the fibre light guides leading to the spectrometer installed behind the biological shield. The divertor region is also probed with the divertor impurity monitoring system (5.5.E.04) that views the plasma through the divertor ports.

4.22.2.7.3 VUV Main Plasma Impurity Monitor

(WBS 5.5.E.03)

The VUV main plasma impurity monitor has two subsystems: one consists of two spectrometers located at an upper port viewing from the flux expansion region up to 1 m into the plasma, and the other consists of two spectrometers located at an equatorial port viewing the core and divertor region. The spectrometers are direct coupled and extend the tokamak vacuum to the biological shield plug where the spectrometers are installed.

4.22.2.7.4 Divertor Impurity Monitoring System

(WBS 5.5.E.04)

The divertor impurity monitoring system measures the emission in a wide spectral range along multiple lines of sight in the divertor region. The plasma is viewed at the divertor level and also from the equatorial and upper ports.

At the divertor level, the light emitted from different chords is collected by mirror optics mounted beneath the dome and on the side, of the special (central) optical divertor cassettes. It is transmitted through an optical penetration in the divertor remote handling port, the VV shielding plug. A labyrinth in the biological shield provides the necessary shielding.

The plasma in the upper part of the divertor region to the X-point is observed through the gap between the divertor cassettes. Optical transmission lines transmit the emission to the spectrometers and detectors located in the port cells, or in the diagnostic hall, depending on the wavelength region. A VUV system is also being considered.

4.22.2.7.5 X-ray Diagnostic

(WBS 5.5.E.05)

For the X-ray region, two types of spectrometry are needed: wide range survey spectrometry for impurity lines, and high-resolution imaging spectrometry for determining ion temperatures from the Doppler broadening, and ion rotation velocities from the Doppler shift of extrinsic impurities.

Sufficient spatial coverage is obtained by installing one of each type of spectrometer at the equatorial level and one survey plus two imaging systems at the upper level (five locations total). The survey spectrometers are direct coupled and extend the tokamak vacuum to the port duct where the spectrometers are installed. The multi-channel/imaging spectrometers are isolated from the tokamak vacuum by beryllium windows.

4.22.2.7.6 Visible Continuum Array

(WBS 5.5.E.06)

The visible continuum array measures the emission at λ = 523 nm along multiple lines of sight in the equatorial plane. By using many sight lines (~ 35), an accurate unfolding of the Z_{eff} profile can be obtained. The emission is multiplexed into one transmission line and an optical labyrinth provides the required shielding.

4.22.2.7.7 Neutral Particle Analyser (NPA)

(WBS 5.5.E.08)

The NPA is designed to measure the tritium-to-deuterium ion density ratio n_T/n_D in the plasma and to provide information on the distribution function of alpha particles in the MeV energy range. The system is directly coupled and samples the plasma through an aperture (<~ 200 mm) in the blanket at an equatorial port. There is an ex-vessel flight tube, approximately 8m long, tapering from the port plug to the input to the instrument. The tube near the port plug is enveloped in a stainless steel shield, The incoming neutral particles are ionized in a stripping target foil and then pass the magnetic and electric fields of the analyzing magnet and electrostatic condenser before detection. The NPA primary vacuum chamber (located outside the equatorial port) is enveloped in a secondary vacuum chamber.

4.22.2.7.8 Motional Stark Effect (MSE)

(WBS 5.5.E.11)

The heating NBs are used for MSE measurements. Two optical viewing systems are needed to achieve the required measurements with the available ports. One system will be installed in equatorial port number 1 for viewing the core using the heating beam installed on port number 4. The second system will be installed in equatorial port 3 and make measurements in the edge region using the heating beam on port number 5. With this combination high resolution measurements will be achieved over most of the plasma cross section.

4.22.2.8 Microwave Systems (WBS 5.5.F)

The principal microwave diagnostics will be a system to measure the electron cyclotron emission (ECE) from the main plasma, a divertor interferometer and three reflectometry systems for probing the main plasma, the divertor plasma, and for measuring the plasma position. An additional systems under study is a fast wave reflectometry system for measuring the fuel mix ratio with spatial resolution.

Labyrinths in all the transmission lines reduce neutron streaming outside the vacuum vessel and bioshield. Vacuum windows of fused quartz directly bonded to metal structures, and inclined at the Brewster angle for the appropriate polarisation, provide robust, low mm-wave loss, pressure boundaries.

4.22.2.8.1 Electron Cyclotron Emission (ECE) Diagnostic

(WBS 5.5.F.01)

The ECE system consists of two collection antennas in an equatorial port plug, a transmission line set, and spectrometers for analyzing the emission. Two antennas are used staggered vertically to give access to the core for a variety of plasma heights near the nominal plasma centre height and to allow four instruments to be coupled. For each antenna, there are built-in calibration hot sources at the front end. The sources can be intermittently viewed through a reflective shutter.

The radiation from each antenna is transmitted to the port cell using wide-band corrugated waveguide with suitable mechanisms to take up machine movements. There, the signal is split into X and O mode components using a polarising beam splitter contained in an extension of the secondary vacuum. The signals are transmitted to the diagnostic hall through a secondary vacuum window using a dedicated corrugated waveguide open to the pit pressure. In the diagnostic hall, the signal is divided between two survey spectrometers and two fixed multichannel spectrometers.

4.22.2.8.2 Main Plasma Reflectometer

(WBS 5.5.F.02 & 09)

The reflectometer for the main plasma provides information on the density profile and density perturbations. In order to provide coverage of the full profile, three sub-systems are necessary: (i) an extraordinary mode (X-mode) launch system, reflecting off the upper cutoff on the low-field side which provides measurements of the SOL profile, (ii) an ordinary (O-mode) system to provide the inboard and outboard density profile in the gradient region, and (iii) an X-mode system reflecting off the lower cutoff and launched from the high field side to provide the core profile.

The low field side system uses similar arrangements to the ECE system without the need for a calibration source. There are twelve waveguide runs between the port and the diagnostic hall to accommodate the various functions of the system. The high field side system construction is very similar to that of the Plasma Position reflectometer (below). However, certain of the lines can be brought to suitable locations in the diagnostic hall.

4.22.2.8.3 Plasma Position Reflectometer

(WBS 5.5.F.03)

The plasma position reflectometer is designed to act as a stand-by gap measurement, in order to correct or supplement the magnetics for plasma position control, during very long (>1,000 s) pulse operation, where the position deduced from the magnetic diagnostics could be subject to substantial error due to drifts.

To meet the requirements for accuracy of the location of the gaps, it is necessary to measure the density profile to a density comparable to, or exceeding, the separatrix density.

As for the high field side of the main reflectometer with which it shares one antenna pair, the antenna pairs (about 14 mm tall) are mounted to view the plasma between blanket modules. Special slots are required in the modules adjacent to these antennas. Radiation is routed to them using small bore waveguides. The waveguides are brought out through two of the upper ports to the sources and detectors installed in the pit.

4.22.2.8.4 Divertor Reflectometer

(WBS 5.5.F.04)

The divertor reflectometer measures the profile of the electron density in the divertor region. The system will operate in the cm to mm-wave domain (15 - 300 GHz) reaching densities of 10^{21}/ m^3. At least two transmission bands are required to cover this range.

Provision is made in the port for 20 waveguides, allowing up to five sightlines using separate transmit and receive waveguides over two bands. The waveguides are embedded in the cassette sides using suitably sized grooves. The antennas project close to the divertor target and dome structure components and will require some cooling as well as separate support structures.

4.22.2.8.5 Divertor Interferometer

(WBS 5.5.F.10)

This system is designed to complement the Divertor Reflectometer, providing the line average density over the same five sightlines but located in a different port. It uses a similar access path within the cassette but with discrete optical elements rather than mm-wave guides. A two-colour infra-red system is expected to be suitable.

4.22.2.9 Plasma-Facing and Operational Diagnostics (WBS 5.5.G)

A range of diagnostics will be installed to aid the protection and operation of the tokamak. Several diagnostics will be dedicated to monitoring the condition of the high heat flux components in the main chamber and the divertor. Other systems include Langmuir probes, pressure gauges and residual gas analyzers.

4.22.2.9.1 Wide-Angle Camera(IR/TV)

(WBS 5.5.G.01 & 10)

The principal high heat flux protection diagnostic will be a wide-angle camera system giving views of the in-vessel components (including parts of the divertor) in the IR and visible wavelength ranges. Combining several cameras achieves high coverage (75%) of the area of the first wall. One or more views of the equatorial system will also be used for the measurement of runaway electrons.

The first element of the system is a metal mirror and the image is transmitted through a rigid labyrinth to a flattening array immediately before the vacuum window. From here the image is transmitted by lenses to CCD cameras mounted on the shielded side of the biological shield plug.

4.22.2.9.2 Thermocouples

(WBS 5.5.G.02)

Thermocouples are installed to monitor the temperature of a small sample of the divertor targets and port BSMs. They will provide information on divertor and blanket power load unaffected by systematic errors in emissivity, and act as references for the G01 and G10 systems. They will also give the location of the strike point to confirm the position indicated by magnetics.

4.22.2.9.3 Pressure Gauges

(WBS 5.5.G.03)

Small, fast pressure gauges, are installed in certain instrumented divertor cassettes. Slower gauges are installed in the pumping ducts. Their function is to monitor operation of the divertor.

4.22.2.9.4 Residual Gas Analysers (RGAs)

(WBS 5.5.G.04)

RGAs will be used to monitor the quality of the machine vacuum at each of the four pumping ducts and in the Neutral Beam pumping line.

4.22.2.9.5 Infra-Red Thermography (Provisional)

(WBS 5.5.G.06)

If installed, the dedicated divertor IR thermography system would provide surface temperature measurements of both divertor target plates in a poloidal plane with good spatial and temporal resolution. It employs mirrors under the cassette dome, and an optical transmission line to the port cell, with fibers coupling the signal to the instrument.

4.22.2.9.6 Langmuir Probes

(WBS 5.5.G.07)

Langmuir probes are used for local measurements of plasma parameters and as attachment / detachment indicators. They are attached to the edge of sample divertor targets in the cassette gap.

4.22.2.9.7 Erosion Monitor (Provisional)

(WBS 5.5.G.08)

A system is under investigation to monitor divertor erosion at one toroidal location. The system is envisaged to have similar access to the IR thermography system and map the target profile with a time-of-flight (optical radar) technique, similar to that used in commercial systems.

4.22.2.9.8 Dust Monitor (Provisional)

(WBS 5.5.G.09)

Provision is made in the space and cabling allocation for some type of dust monitor to measure at the level of the machine floor between the cassette gaps. The requirements for dust measurememts are still under development and so it is not possible, at this stage, to define the measurement system.

4.23 Test Blanket (WBS 5.6)

Currently, 2006, the Test Blanket effort is not yet incorporated into the ITER Programme.
Staff and resources will be included when a framework agreement is put in place.

4.23.1 Functional Requirements

The test programme includes small test articles and modules that can be inserted in equatorial ports. The potential concepts include:

- water-cooled solid breeder;
- helium-cooled solid breeder;
- self-cooled liquid lithium;
- water/self-cooled lithium-lead;
- helium/self-cooled molten salt blankets..

The test blanket modules provide their own first wall, shielding and thermal control to appear to the vessel and surrounding blanket modules as a normal port plug. They also provide their own services (cooling and tritium recovery) and connection to site services for heat rejection and tritium environmental control. They conform to tritium inventory limit requirements and can be remotely repaired in the hot cell.

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

Blanket module testing will be done whenever significant neutron fluxes become available To ensure that test blanket modules will be compatible with tokamak operation they must be installed as early as possible before beginning of the DT operation.

As a rule the TBMs will be changed once per year in accordance with ITER operational schedule. Some modules may need additional replacements, which will be synchronised among the Parties and with machine operation. Target replacement time must be less than 1 month for all TBM's ports together.

4.23.2 Configuration

4.23.2.1 Overall

TBD – This topic is still to be investigated.

4.24 Buildings and Layout (WBS 6.0,6.1,6.2,6.3)

Substantial review and site adaptation is underway (Oct 2006).
A thorough update of this section will take place in 2007

4.24.1 Functional Requirements

The tokamak building provides for the following:

1 assembly of the machine, with the surrounding buildings and systems organised to permit approach to the tokamak from both north and south, and with the tokamak buildings and site arranged to allow future construction for tokamak repair or decommissioning;
2 a convenient and a safe operating environment with the minimum size and cost;
3 radiologically controlled areas enclosed in a contiguous boundary with ventilation and cleanup systems.

All buildings must house, support, protect, control access to, provide suitable environmental conditions for, and provide services to the components, systems, personnel and operations which are located or operate within them.

The site layout provides the space necessary to house and construct the tokamak and all of its auxiliary systems and services.

Table 4.24-1 ITER Buildings and Structures

WBS #	Building Number on Site Layout Drawing	Complete Building Name	Short Building Name
6.2.A.01	11	Tokamak Building	Tokamak Building
	14	Tritium, Vacuum, Fuelling and Services Building	Tritium Building
	74	Diagnostic Hall and RF Fast Discharge Resistors and Capacitors	Diagnostic Building
6.2.A.04	13	Laydown, Assembly and RF Heating Building	LA & RFH Hall
Note that the **Tokamak Building (11)**, the **Tritium Building (14)** and **Diagnostic Building (74)** share the same basemat and, collectively, are referred to as the **Tokamak Complex**			
6.2.B	21	Hot Cell Building	Hot Cell Building
6.2.G.01	23	Low Level Radwaste Building	Radwaste Building
6.2.G.02	24	Personnel Access Control Building	Personnel Building
6.2.E.01.1	32	North Magnet Power Conversion Building	N-MPC Building
6.2.E.01.2	33	South Magnet Power Conversion Building	S-MPC Building
6.2.E.04	34	NB Power Supply Building	NB PS Building
6.2.E.05	36	Alternating Current Distribution Building	AC Distribution Building
6.2.L.01	41	Emergency Power Supply Building	EPS Building
6.2.L.03	LC1 - LC12	Electrical Load Centres	Load Centres
6.2.J.01	52	Cryoplant Compressor Building (and PF Coil Fabrication Building 1)	Cryoplant Compressor Building
6.2.J.02	51	Cryoplant Coldbox Building (and PF Fabrication Building 2)	Cryoplant Coldbox Building
6.2.M	61	Site Services Building	Site Services Building
	67	Hot Basin and Cooling Towers	Cooling Towers
	68	Cooling Water Pumping Station	Pumping Station
6.2.H	72	Laboratory/Office Building	Lab/Office Building
6.2.K.01	71	Control Building	Control Building
6.2.K.02	73	Personnel and Vehicle Access Control Gatehouse	Gatehouse
6.2.S	not applicable	Utility Tunnels and Service Structures	various

ITEM	Value in Hectares
Total site area within outer fence	25.31
High security area	25.41
Power Supply complex incl. NBH	0.80
Heat sink complex	0.63
Laboratory office, control building and parking area	4.81
Future expansion and storage area	2.76
Utility company area	0.83
Area for construction	Area to be decided

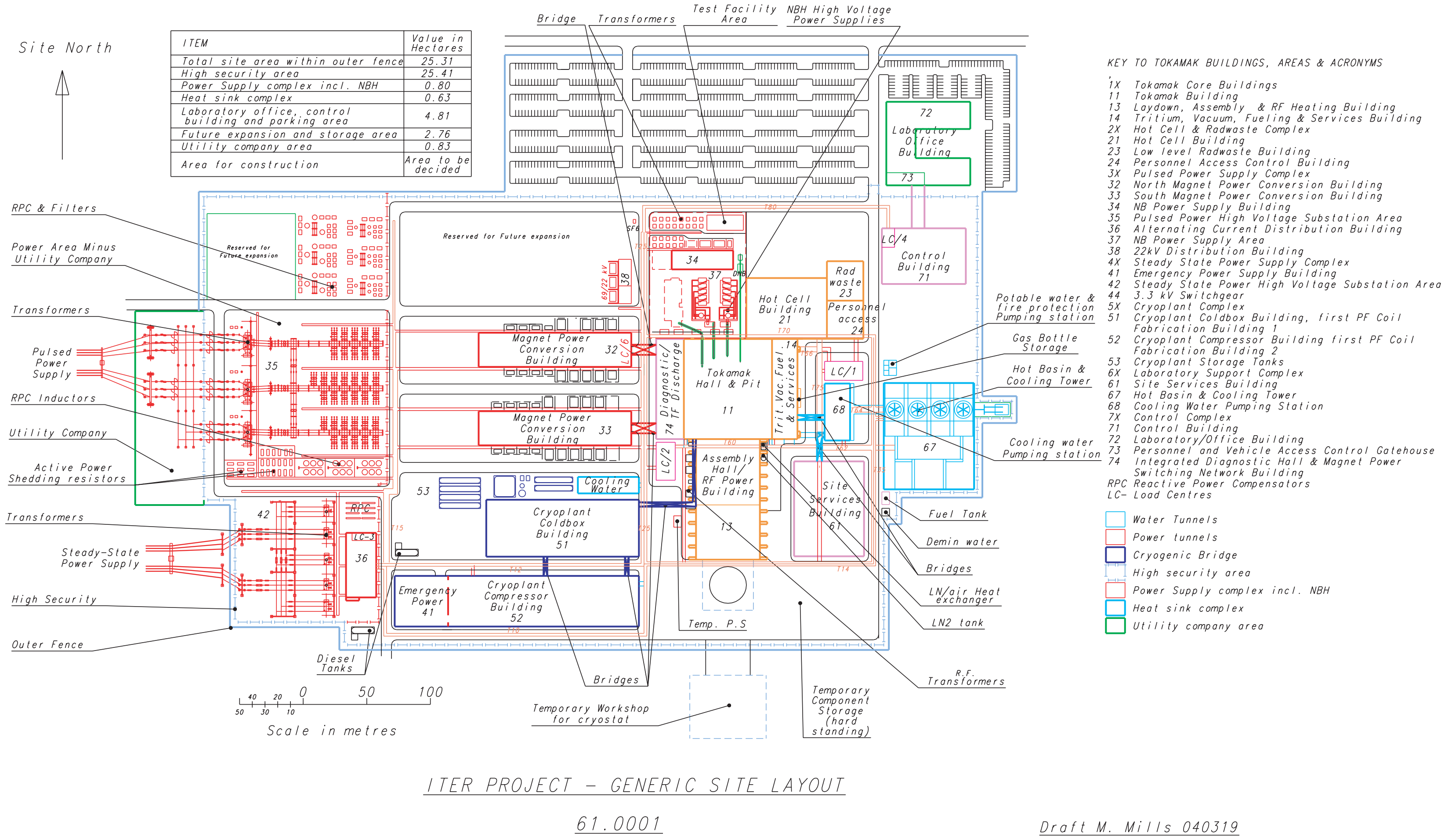


Figure 4.24-1 ITER Site Layout (Generic)

4.24.2 Configuration

4.24.2.1.1 Site Arrangement

Before site selection, the generic site is arranged so that electrical services enter from the west, cooling systems are located on the east, personnel-related functions and waste management functions are located on the north (these directions are for identification purposes only).

The generic site layout developed is shown in **Figure 4.24-1**, and correspondingly, the major buildings and structures are arranged as listed in Table 4.24-1.

The buildings are categorized into radiologically controlled buildings and conventional buildings, and these design features are presented in the following sections.

To the maximum extent possible, the design of systems, buildings, and the site, is such that future additions in system capacity are not precluded.

4.24.2.1.2 Site Fences

The ITER site is enclosed by two fence systems. The outer fence encompasses all the land area under control of the ITER Organisation. The inner fence surrounds a high security area and prevents unauthorised entry by persons or vehicles.

All ITER buildings and structures except, the laboratory office building, and the pulsed and steady state switchyards, are inside the high security area.

During operation, all access (personnel and vehicular) to the ITER high-security area of the site is through the single access control gate and gatehouse (73), located on the north-east side of the site, outside the security fence.

4.24.2.1.3 Site Communications

A site-wide internal communication system is provided in buildings, including telephone connections, public address system, and appropriate warning systems for plant emergency, crane movement, fire, radiation monitoring, access control, etc. In addition there will be dedicated intercommunication systems between the control room and important plant locations and access control points.

4.24.2.1.4 Fire Prevention and Mitigation

Building materials and installed components make extensive use of non-flammable or low-flammable materials, such as concrete, steel, and fire retardant cable insulation, etc. Nonetheless, the buildings and structures provide fire detection, alarm, and mitigation systems commensurate with the occupancy and fire risk loading.

There will be an on-site incident response team which will be able to carry out fire-fighting duties. This team will be supplemented by off-site emergency services as required.

4.24.3 Radiologically Controlled Buildings

These buildings include the tokamak building (11), tritium building (14), hot cell building (21), low level radwaste building (23), diagnostics building (74), and personnel access control building (24).

The radiologically controlled buildings are currently designed and constructed in accordance with American Concrete Institute (ACI)-349 (or equivalent), and ACI-318 (or equivalent) for specified seismic conditions, and all the quality assurance and inspections contained therein, plus any additional requirements specified by the ITER QA program.

4.24.3.1 Tokamak Complex

The tokamak complex (11, 14, 74) consists of three buildings; tokamak building, tritium building and diagnostic building. It is a large reinforced concrete structure with dimensions of approximately 112 m x 79 m. The complex shares a common rectangular basemat to minimize possibilities of relative displacement due to seismic events in any of the large number of pipes and ducts that run between them. Seismic isolators can give it an SL-2 earthquake tolerance significantly greater than the generic site design basis of 0.2 g peak horizontal and vertical ground acceleration.

A plan (at EL 0.00) view and two elevation views of tokamak complex are shown in Figures 4.24-2 through 4, respectively.

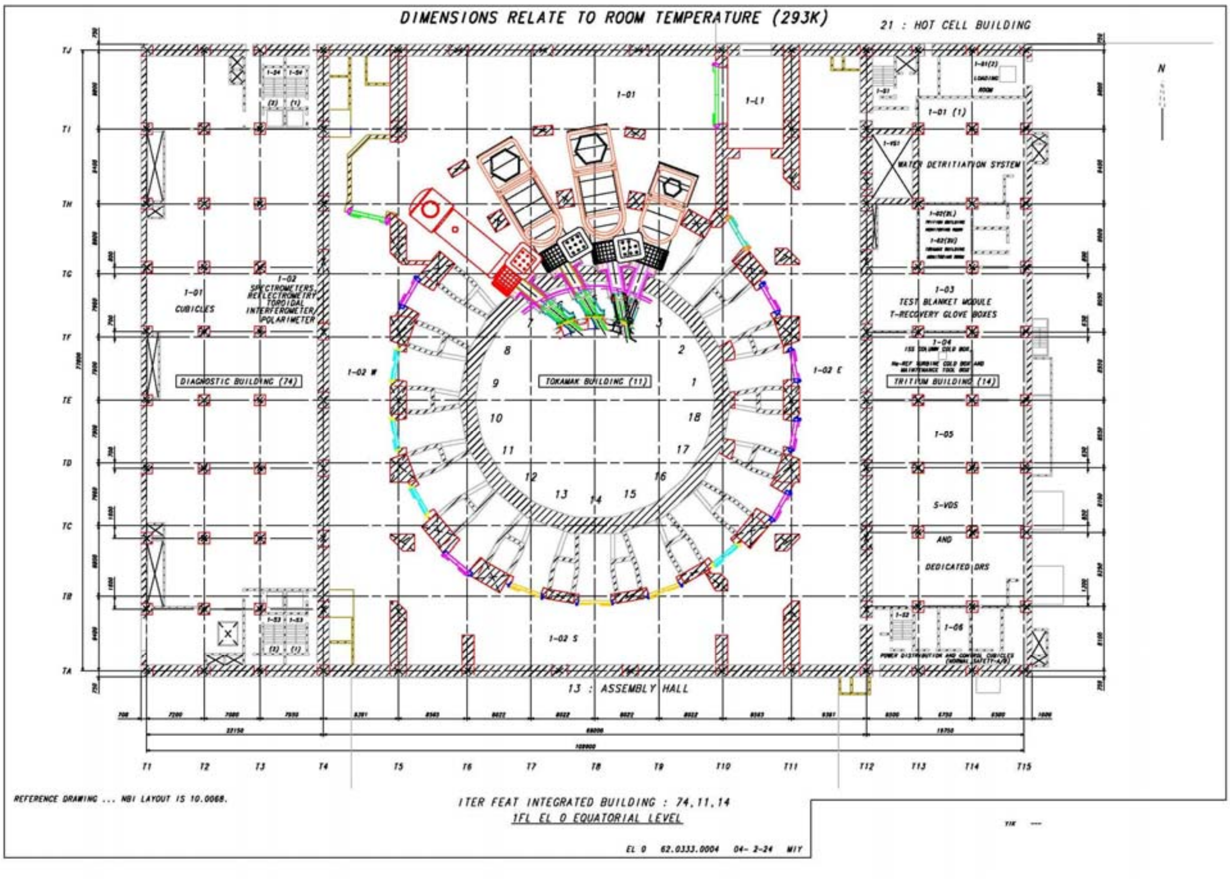


Figure 4.24-2 ITER Tokamak Complex Horizontal Cross Section

(http://ftp.itereu.de/cad/pdf/62.0333.0004.2D.0405.W---N.pdf)

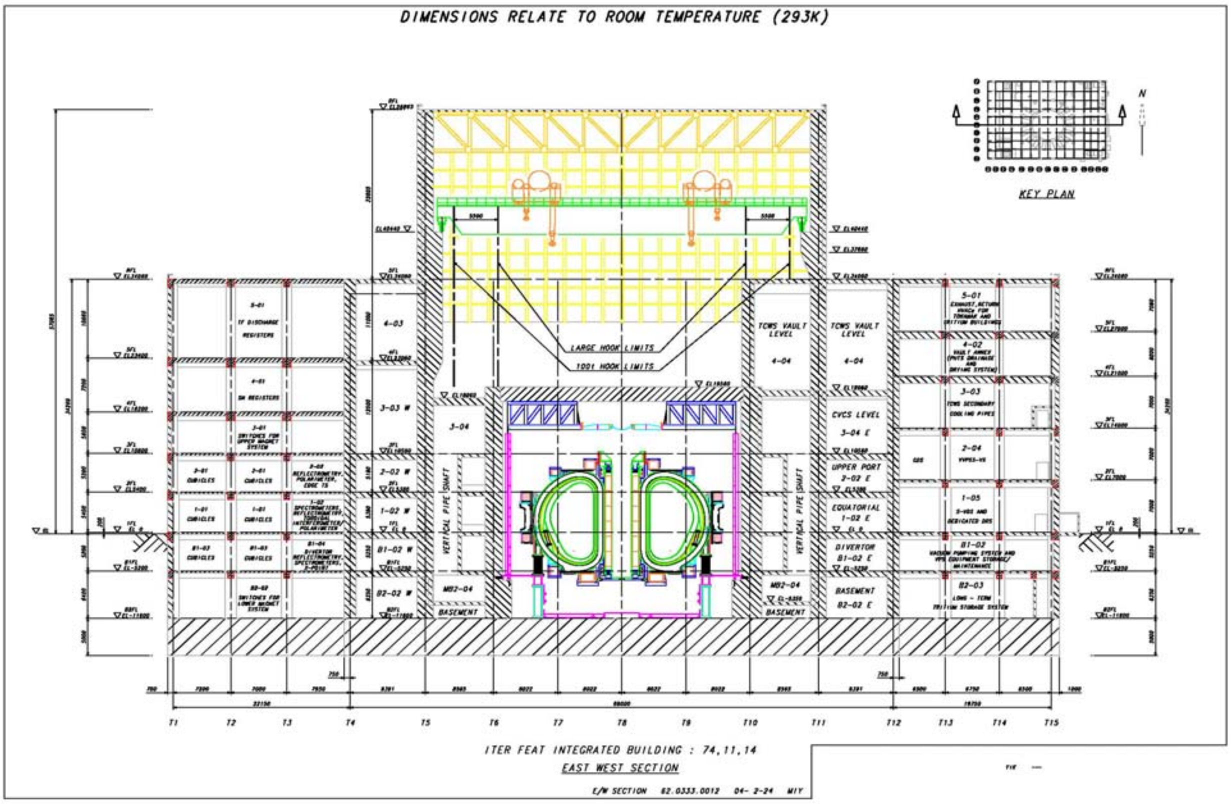


Figure 4.24-3 ITER Tokamak Complex Vertical Cross Section

(http://ftp.itereu.de/cad/pdf/62.0333.0012.2D.0405.W---N.pdf)

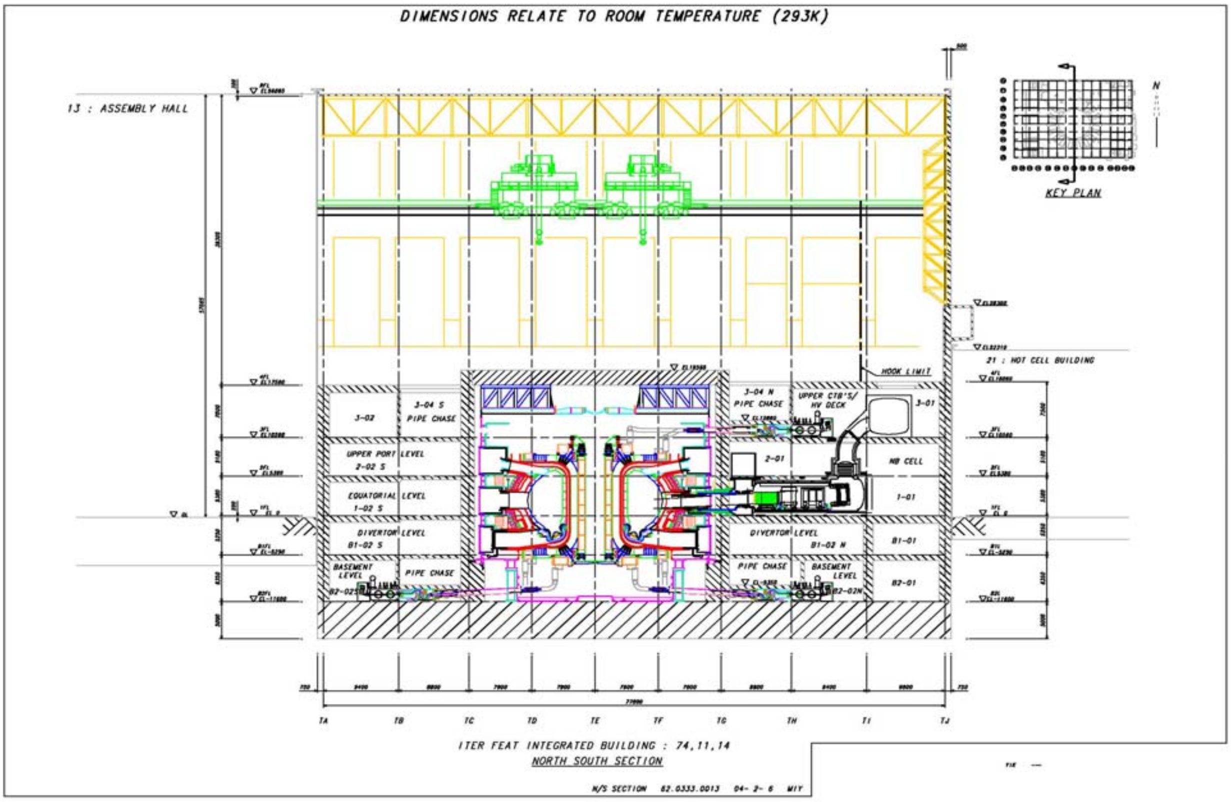


Figure 4.24-4 ITER Tokamak Complex Full Vertical Cross Section Including Crane Housing

(http://ftp.itereu.de/cad/pdf/62.0333.0013.2D.0405.W---N.pdf)

Zoning for radiation hazards is defined in PDS (**Table 3-3** "Area Classification and Radiation Access Zones" and above in Table 3.1-12 Radiological zoning according to total doses), and the various areas in the tokamak building and in the tritium building are assigned in accordance with the radiation zone classifications as shown in Table 4.24-2**.**

Table 4.24-2 Radiation Zone Classifications *5

Building, Room Designation, Area Designation	**DT Pulsed Operation**	**Maintenance after Shutdown**	**Transfer of Irradiated Components**	**RH Class III e.g. Coil Replacement**
Tokamak Building				
Cryostat Bottom	D *1	D	D	D
Basemat Galleries	D *1	B	B	B
TCWS Drain Tanks	D *1	B	B	B
TCWS Drain Tanks Upper Level	D *1	B	B	B
Lower Pipe Chase, Upper Pipe Chase	D *1	B/C *2	B/C *2	B/C *2
Divertor Galleries, Equatorial Galleries, Upper Port Galleries	D *1	B	D/B *3	B
NBI Cell	D *1	C	D/C *3	C
TCWS Vault (TCWS CVCS Area & TCWS equipment)	D *1	B/C *2	B/C *2	B/C *2
HV Deck & CTB Area N , CTB Area S, Cryodistribution Area W	D *1	B	B	B
VV Suppression Tank	D *1	B	B	B
Crane Hall	D *1	A	A	C/D
All Stairwells	D *1	B	B	B
Tritium Building				
Manifold Glove Box, Vacuum Pump Sets, Front End Permeater, Detritation Purification	B *4	B *4	B *4	B *4
Vault Annex	D *1	B/C *2	B/C *2	B/C *2
All Stairwells	B	B	B	B
The Other Rooms	B	B	B	B

Notes:

*1 This area is inaccessible to workers during tokamak operation due to various operating hazards, including radiation and magnetic fields.

*2 Radiation fields in areas related to the TCWS will be determined by activated corrosion products in TCWS equipment, and will vary within the vault with proximity to the equipment and other factors.

*3 Accessibility is changed depending where the cask is.

*4 These rooms require 500 mm thick concrete walls and slabs as local shielding.

*5 Zoning (A to D) for radiation hazards is defined in PDS (Table 3-3 "Area Classification and Radiation Access Zones").

(1) Tokamak Building

The tokamak building houses the vacuum vessel, the in-vessel components, the superconducting magnet system, the cryostat and thermal shields. The tokamak building is designed to permit assembly and operation of a tokamak with 18 sectors with radial ports at three vertical levels.

The general architectural structure is arranged around the cryostat. The floor levels, radial walls and pillar positions are generally identical and directly related to port access and the remote handling cask docking/transport system. The space availability for the cask transportation including rescue operation is under studied in accordance with the updated layout design.

The tokamak building is arranged in four major volumes:

- pit (inside the bioshield and accessible with the tokamak building cranes)
- port cells (a compartment to envelop the area outside the bioshield in front of each port at three levels)
- galleries (extending to the building boundaries)
- crane hall.

The tokamak building accommodates the transportation routes for the components removed from the dedicated ports to the building high capacity (100 t) cargo lift. The lift has access from each floor of tokamak building and connected with the hot cell building at the equatorial (grade) level and upper level. The buildings provide radiation shielding for transporting the components in the passageway.

The tokamak crane height is about 20 m above the slab on the top of cryostat lid. There will be two, independent, identical 750 t capacity bridge cranes, each equipped with 2 trolleys with a 375 t capacity main hoist and a 100 t auxiliary hoist. The crane rail spacing is 44.82 m. Loads up to 1,500 t (including jigs) can be raised.

Within the tokamak building, certain areas make up a confined volume.
These are:

1 the TCWS vault;
2 the CVCSs area;
3 the TCWS vault annex;
4 the NB cell;
5 the port cells;
6 the vertical pipe shafts;
7 the upper and lower pipe chases;
8 the lower pipe chase sump;
9 the tokamak pit.

These are designed to withstand the maximum internal pressure at accident and to keep lower operating pressure than that of the surrounding areas by ventilation and cleanup systems, as shown in Section 3.1.3.1 (Confinement Objectives and methodology)

The biological shield wall thickness of 2m is essentially for structural reasons, and more than satisfies the radiation shielding defined in section 3.8" Neutron & Radiation Loads – Shielding". The result of detailed study is shown in "G 73 DDD 2 04-04 W 0.1 Nuclear Analysis Report".

(2) Tritium Building

The tritium building houses the tritium plant subsystems such as storage and delivery system, tokamak exhaust processing system, hydrogen isotope separation system, atmosphere detritiation system, water detritiation system, tritium plant analytical system, and ventilation systems including HVAC, depression system, local air cooling system, etc). The descriptions of the tritium plant are provided 4.14.2.1.1 “Tritium Plant Overall Arrangement”.

The tritium building is located adjacent to the tokamak building to minimize the length of vacuum and tritium lines and its size is approximately 21 m x 79 m. The bottom floor of the tritium building is at the same elevation as the tokamak building bottom floor, but has five additional floors which do not correspond with those in the tokamak building.

(3) Diagnostic Building

The diagnostic building houses several systems; cubicles for diagnostic, diagnostic equipments, test laboratory, shielded neutron test area, counterpulse capacitors, PF and TF discharge resistors and HVAC system.

The diagnostic building with seven floors is located adjacent to the tokamak building and the floor levels of the five lowest levels are matched to the tokamak building to accommodate the routes of diagnostic equipment and pulsed power supply busbar. The busbars run at basement and third floor at the north and south ends of the building from the tokamak building to ~~a~~ two outside bridges. The size of the diagnostic building is approximately 23 m x 79 m.

The diagnostic building is a non-radiological control building, except neutron test area, and has personnel and truck access independent from the access to the tokamak building.

4.24.3.2 Hot Cell Building

The hot cell building (HCB, 21) is designed in compliance with the new layout of the hot cell facilities described in 4.8. However note that this design is still under review (2005) with the Participating teams – ref DCR-35 and TCM-16.

The hot cell building is located North-East relative to the tokamak building to allow the transfer of Vacuum Vessel components to and from the tokamak building. It consists of a rectangular reinforced concrete structure, organized on four levels, with a footprint approximately 70m x 62m. The hot cell building provides a shielded and controlled area equipped for the refurbishment/testing of in-Vessel components, including space for their temporary storage and/or preparation for disposal as radwaste. It accommodates also the systems and services required to support its operations: cranes, RH tools, RH tools' repair and storage, active dust filtration system, ventilation and atmosphere clean-up system (linked to the tritium building services). Space for RH equipment testing facility, control rooms and cask storage is also provided.

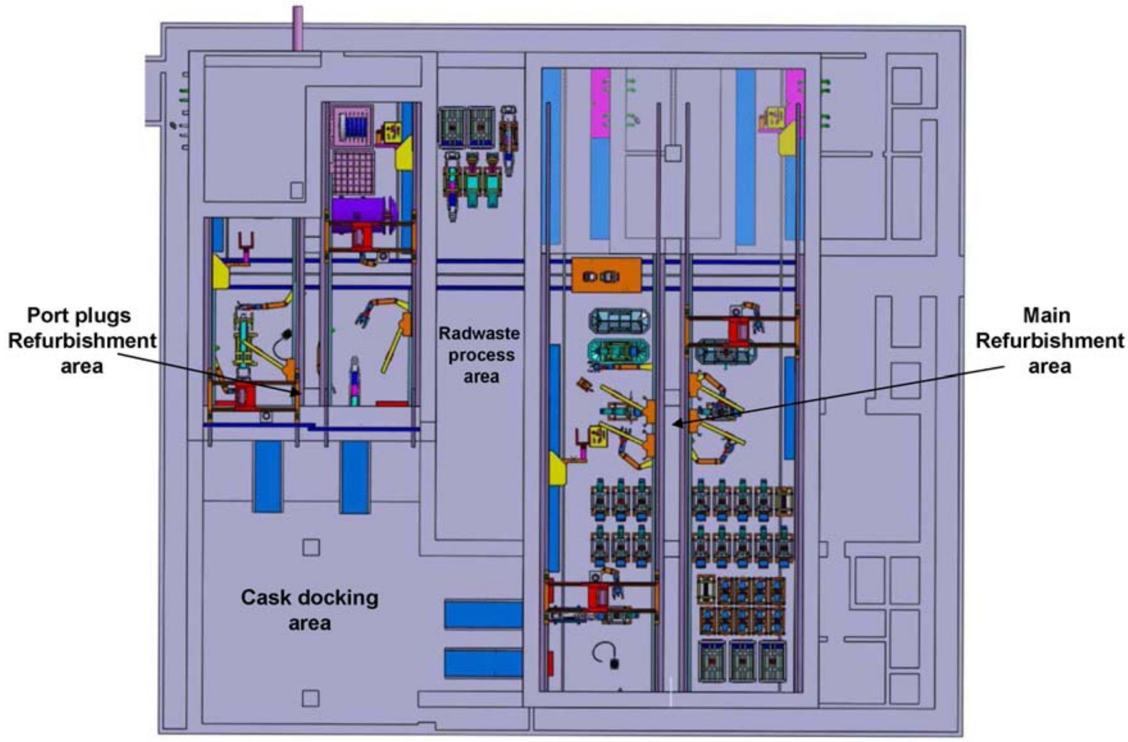


Figure 4.24-5 Plan view of the refurbishment area in the hot cell building.

The HCB is available during the initial installation phase of the tokamak in-vessel components, to provide a pre-assembly, beryllium-controlled area, and a facility for loading components into transfer casks.

4.24.3.3 Low Level Radwaste Building

The low level radwaste building (23, which abuts the eastern wall of the hot cell building and the north wall of the access control/personnel building) is a reinforced concrete structure. It houses liquid and solid waste processing systems and the facility for packaging and storing the low level waste prior to its transfer to an approved off-site facilities.

The low level radwaste building also contains drain tanks and systems for processing mildly contaminated water from floor drains from radiologically controlled buildings, active laboratories, PHTS equipment drains, sinks, showers and laundry drains.

A door is provided between the hot cell building and the radwaste building. This feature provides a route for low-level solid radwaste produced in the tokamak building to be transferred to the radwaste building. Additionally, this feature provides a route for the delivery of new large components (e.g. transfer cask, etc.) from the outside.

4.24.3.4 Personnel Access Control Building

The personnel access control building (PACB, 24) is a rectangular steel frame building contiguous to the tokamak, the hot cell, and the low level radwaste buildings in order to provides the single, controlled pathway for personnel access to the radiologically controlled buildings.

4.24.3.5 Contamination Control

The radiologically controlled buildings are divided into areas based on their potential contamination ranging from "white" (uncontaminated) through "green" and "amber" to "red" (with different degrees of airborne and surface contamination - see §3.1.5).

The design of HVAC systems ensures appropriate pressure and flow gradients within the buildings so that air flows from areas of lowest probability of contamination towards areas of higher probability. The HVAC air flow is passed through appropriate filters, and, if necessary, through the detritiation systems, before being released through the plant exhaust.

The HVAC system is specifically designed to respond to potential off-normal events. In locations that may be involved with such events due to loss of vacuum or coolant leakage, the HVAC systems are equipped with high-efficiency particulate filters. Areas where the release of elemental tritium or tritiated water is possible are equipped so that the exhaust flow can be passed through a vent detritiation system (VDS).

Floor drainage from radiologically controlled areas is collected in tanks where it can be monitored and, if necessary, treated before it is released to the environment.

4.24.3.5.1 Personnel Access

Access to the potential contamination areas is through air-locks which provide for radioactive contamination control. The hot cell building, the low level radwaste building and the tritium building provide for periodical truck access.

Access is normally forbidden under certain operational conditions as follows:

1 tokamak galleries (the areas below the crane hall of the tokamak building) during tokamak operation (including baking);
2 tokamak crane hall during tokamak operation, due to the intensity of the magnetic field;
3 parts of the hot cell building where radioactive components have been accepted;
4 tokamak galleries and cask docking area of the hot cell, and the passageway between these buildings, when casks transfer radioactive components.

4.24.3.5.2 Emergency Evacuation

There are at least two independent emergency evacuation routes from every working area on each floor or basement of each of the buildings, including the tokamak building, leading to the evacuation stairways and to an outside exit. Evacuation exits lead directly to, or through a lobby to, a street or to an outdoor area that gives safe access to a public way. Also, roofs occupied by components require exits as floors. The detailed design is required to meet the regulations of the host country.

4.24.4 Conventional Buildings

The conventional buildings consist of Laydown, Assembly and RF Heating building, Magnet Power Conversions buildings, AC Distribution building, Cryoplant Compressor and Cold Box buildings and Site Service building. The main functions of these buildings are as follows.

4.24.4.1 Laydown, Assembly and RF Heating Building

During construction and the machine assembly phase, the laydown, assembly, and RF heating building (LA & RFH building, #13) will be used for assembling the ITER machine. For this, it is adjacent of south side of tokamak complex because the crane hall of the tokamak building and the LA & RFH building form a continuous crane bay necessary for tokamak assembly and maintenance. . After the tokamak is assembled, additional internal walls and floors will be added to install the RF power heating and current drive systems.

4.24.4.2 Magnet Power Conversion Buildings

The magnet power concversion buildings (MPC building, 31 and 32) are used to house the rectifier and power smoothing equipment that converts AC power coming from the pulsed power switchyard to DC power for the superconducting magnet system. The building also provide a space for AC/DC converters, dummy loads, local cubicles, inter-space reactor, grounding switch, disconnecters and busbars.

4.24.4.3 AC Distribution Building

The alternating current (AC) distribution building (36) is part of the steady state power supply distribution system which supplies ac power to all plant electric loads, except those supplied by the pulsed power distribution system.

4.24.4.4 Cryoplant Compressor and Coldbox Buildings

The Cryoplant Compressor Building (52) and the Cryoplant Coldbox Building (51) are scheduled to be built earliest in the construction phase since they will be used initially for large PF coil fabrication in accordance with the current project construction plan. After the PF coil fabrication, the buildings will be converted for use of the cryoplant buildings to house the cryoplant equipment such as helium compressors and cold boxes. The reference project master schedules are defined in the "Construction, Commissioning and Operation Plan (COP)" annex to this document. However note that this design is still under review (2006) with the Participating teams.

4.24.4.5 Site Service Building

The site services building provides space for industrial support systems including compressed and breathing air, chilled water, demineralised water and hot water treatment, which is used primarily for space heating, and some electrical power distribution.

4.24.4.6 Others

In addition to the above buildings, the following buildings and structures are included in the generic site layout.

- Neutral Beam power supply buildings
- Electrical load centres
- Cooling towers
- Cooling water pumping station
- Utility tunnels
- Service structures
- Laboratory/office building
- Control building

4.25 Radiological and Environmental Monitoring (WBS 6.4)

(This section will be reviewed in early 2007, when effort and a responsible officer is available.)

4.25.1 Functional Requirements

In accordance with the fundamental approach for radiation protection including a zoning plan given in the PDS, the personnel radiation monitoring and protection system (PRM & PS) provides radiological monitoring to assist in protection of personnel from penetrating (ionizing) radiation including tritium hazards. The PRM & PS also provides routine data on the radiological state of the plant throughout ITER project operating life. The function is accomplished by a combination of sub-systems, which include fixed and portable radiation/contamination monitors working in conjunction with a dosimetry and bioassay system. Personnel radiation protection is provided by protective equipment such as bubble suits and breathing air systems (covered under liquid and gas distribution systems), and health physics (HP) control and records centre.

The environment monitoring system (EMS) provides informatiom on the environmental impact of ITER operations as necessary to ensure compliance with environmental regulations. The function is accomplished by a combination of fixed and portable environmental monitors working in conjunction with a sampling and inspection program.

4.25.2 Configuration

The PRM & PS consists of the following subsystems: fixed area monitoring system, portable monitoring system, personnel dosimetry system, personnel bioassay system, personnel protective equipment and HP control and records center. The PRM & PS subsystems shall be designed to meet the monitoring and protection functions so as to ensure that that plant worker doses, other ITER personnel and the general public do not exceed ITER guidelines of occupational exposure given in PDS, or in the event of accidental exposure that the dose can be estimated.

4.25.2.1.1 Radiation Monitors - Location

In the radiologically controlled buildings tritium monitors are provided at potential discharge points, at ductwork close to ventilation fans, close to extraction ducts, and in large rooms. All rooms have a facility whereby the air inside can be checked for tritium contamination before personnel entry.

Gamma monitors are located in the tokamak building, the radwaste building, and the hot cell. These are sited in such a way as to measure the maximum dose in a particular room. The PRM & PS is designed such that the gamma dose inside the room can be checked from outside.

Neutron monitors are located in the tokamak building only, and are sited to measure the maximum dose in a given region.

Rooms and volumes where personnel go infrequently do not need fixed monitors.

Radioactive particulates and beryllium are monitored by air sampling through appropriate filters which are subsequently analysed.

The HP control and records centre is located in the personnel building and monitors and records all measurements and alarm all conditions and operations.

4.25.2.1.2 Radiation Monitors - Operation

All monitors are capable of operating at their place of installation independently of other systems and services. The time of autonomous operation in the event of power failure should be specified for each location and an appropriate uninterruptible power supply (UPS) provided for each instrument or group of instruments.

The monitors, control and alarm systems that shall function during and after an incident shall be connected to the emergency power supply systems and their signal lines shall be separated from each other.

Monitors will have internally-triggered audio and visual alarms as appropriate.

The monitors, controls and alarms should not be accessible except to authorised personnel. In particular the ability to switch off or otherwise disable monitors and alarms will only be possible for authorised personnel, and alarms can only be muted where an audible alarm may interfere with corrective measures (e.g. control room).

Personnel will be trained to react appropriately to the alarm.

The status of all monitors will be monitored centrally via hard-wired links.

Monitors will fail safe (i.e with alarm on and signal sent to central monitoring station).

Data will be acquired from each monitor every 20 seconds and archived in full. The monitors will not be capable of being set centrally, but will be set and reset locally by authorised personnel. The monitors will be interrogated centrally.

Table 4.25-1 Monitoring and Detection Equipment

Type of equipment	Basic reason for equipment	Range of sensitivity
Whole body detector	For the detection of gamma radiation in the whole body	Standard for application
Area gamma radiation monitor	For the detection of gamma radiation in the area	0.5 μSv/hr to 10mSv/hr
Tritium-in-air area monitor	For the detection of total tritium in the air	0.1 to 10000 DAC *(to be corrected to VDO)*
Tritium-in-air duct monitor	For the detection of total tritium in the air	0.1 to 10000 DAC *(to be corrected to VDO)*
Area neutron monitor	For the detection of neutrons in the area	0.5 μSv/hr to 10 mSv/hr
Portal monitors	For the detection of beta/gamma contamination in the control area	Per Standards
Friskers	For the detection of beta/gamma contamination on surfaces in the area	Per Standards
Portable radiation monitor	For the detection of beta/gamma in the area	0.5 μSv/hr to 1 mSv/hr
Portable tritium-in-air monitor	For the detection of total tritium in the area	0.1 to 10000 DAC *(to be corrected to VDO)*
Portable neutron monitor	For the detection of neutrons in the area	0.5 μSv/hr to 1 mSv/hr
Radioactive particulate monitor	For the detection of beta/gamma contamination on samples taken in the area	10 kBq/m^3 – 100 MBq/m^3
Hi-range gamma monitor	For the detection of gamma radiation in the area	up to 1 Sv/hr
Tritium-in-air HVAC trigger monitors	For the detection of total tritium in the air ducts and to send a signal to the HVAC control system for appropriate action.	10 to 100000 DAC *(to be corrected to VDO)*
Surface smear counter	For the sampling of beta/gamma contamination on swipe samples taken in the area	> 1 Bq/cm^2
Personal electronic visit dosimeter	For the detection of accumulated beta/gamma dose for-personnel in the area	1 μSv to 10 mSv
Personal dosimeters	For the detection of accumulated beta/gamma dose for-personnel in the area	1 μSv to 10 mSv
Personal dosimeters	For the detection of accumulation of neutron dose for personnel in the area	1 μSv to 10 mSv
Air activation products monitor	For the detection of beta/gamma in air in the area	10 kBq/m^3 – 100 MBq/m^3
Aqueous tritium sampler	For the detection of HTO in water samples	> 1 kBq/L
Aqueous activation products	For the detection of beta/gamma in water samples	TBD
Portable aqueous sampler	For the detection of beta/gamma in water samples	TBD
Portable aqueous sampler	For the detection of HTO in water samples	> 10 kBq/L

Note that the range of sensitivity should be finalized in accordance with the safety requirements and regulations. The type of personal dosimeters are to be selected and approved as legal dosimeters according to French regulations.

Table 4.25-2 The ITER Environmental Monitoring Program

Sample	Frequency	Sampling Location	Analyses
Air	Continuously sampled; analysed monthly Continuously sampled; analysed monthly	One HT/HTO discriminating sampler at each of approx. 10 environmental monitoring sites. High volume samplers at each of 10 environmental monitoring sites	H-3 (tritium) Gamma spectrometric analysis – particulate and iodine [iodine at only 1 of 10 sites]
Background radiation	TLD dosimeters, changed quarterly	Three TLDs at each of 10 environmental monitoring sites	Integrated quarterly gamma dose
Precipitation	Continuously sampled; composite of buckets from each environmental monitoring station analysed quarterly	One bucket at each of 10 environmental monitoring sites	H-3 in water; gross beta in water; gross beta on sample bags
Milk	Monthly	From at least 2 farms within 10 km of the site	H-3 [individual sample] C-14 [individual analysis] Gamma spectrometric analyses for radioactive particulates and corrosion products
Drinking water	Sampled twice daily, composite analysed weekly Composite analysed monthly	From drinking water supplies to neighbouring communities From drinking water supplies to neighbouring communities	H-3 Gross beta; Gamma spectrometric analyses for radioactive particulates and corrosion products
Fish	Annually	Near station outfall	Gamma spectrometric analysis; H-3
Surrounding water bodies	Annually	Various locations, TBD	H-3; gross beta
Active drain water	Weekly	Hold-up tanks	H-3, gross beta, gamma spectrometric analysis

Note: the above monitoring programme may require site-specific finalisation in accordance with the safety requirements and regulations.

Table 4.25-3 Fixed Monitor Numbers and Locations

Building	Tritium	Gamma	Neutron	Particle
Tokamak	39	40	6	
Laydown, Assembly, RF (13)	3	3		
Tritium (14)	36	11		
Hot Cell (21)	14	24		20
Radwaste (23)	1	17	1	1
Personnel (24)	4	5		
Diagnostic (74)	6	9	3	
Other (including ventilation/ discharge monitoring	18	4		10
Total	121	113	10	31

4.26 Liquid and Gas Distribution (WBS 6.5)

(This section will be updated in 2007 along with the Site Adaptation)

4.26.1 Functional Requirements

The functions of the liquid and gas distribution system is to distribute potable and fire-fighting water, remove sanitary and industrial sewage, to supply steam and demineralized water, to supply compressed and breathing air, and to provide nitrogen, helium and special gases.

4.26.2 Configuration

4.26.2.1 Overall

4.26.2.1.1 Potable Water System

The potable water system is sized to accommodate 1000 persons on-site. A reserve tank located in the site services building provides for 3 days use. 10% is reserved for fire protection. The potable water system does not feed the radiologically controlled buildings.

4.26.2.1.2 Fire Protection System

The fire protection system is designed with redundancy to permit maintenance. Failure to deliver on demand is less than 10^{-2}. The system can resist earthquake SL-0. Each outdoor fire hydrant (fireplug) covers a 40 m radius, so all buildings are covered by at least one hydrant.

4.26.2.1.3 Hot water flow and return

The hot water flow and return system uses boilers located in the site services building. Two separate thermally isolated pipes are used to feed the radiologically controlled and the other buildings.

4.26.2.1.4 Demineralised Water System

The demineralised water system supplies deionised water for process purposes or for makeup of the ITER closed cooling loops. Te demineraliser is located in the site services building.

4.26.2.1.5 Compressed Air System

Separate systems are planned for the following groups of buildings: tokamak + tritium + diagnostics, hot cell + radwaste + personnel, cryoplant + emergency power supply, site services + control, magnet power supply. All taps in the distribution systems for compressed air have check valves to isolate them if the pressure applied to the tap exceeds the system pressure.

4.26.2.1.6 Breathing Air System

A single separate centralised system is provided for breathing air, to support respiration and cooling in protective suits, and to fill respirators for inadequately ventilated areas. The main equipment is located in the site services building, and each user building supplied through a separate distribution system.

4.26.2.1.7 Bottled Gases

Except for the crogenic plant, nitrogen and helium are delivered through a local distribution network from the site services building bottled gas plant. Any other specialised gases are to be provided by the user local to the need through dedicated systems.

Table 4.26-1 Parameters of Plant Services

System or Service	Description	Parameter value	Comments
Potable water	based on supplying 1,000 people	~380 m^3/day at ~0.8 MPa	
Fire-fighting water	a pressurised, plant-wide system	3 m^3/min at ~ 1.3 MPa	
Hot water	supplies and distributes hot water at 80 ^{0}C and 0.5 MPa to components and systems (including HVAC) which require auxiliary heating	Two boiler units 3 MW x 2	
Demineralized water	water for process purposes and makeup to cooling systems	30 kg/h at 0.2 ~ 1 MPa	1.0 µMhocm^{-1} with dissolved oxygen and chlorine concentrations less than 0.1 ppm.
Sanitary sewage	for a site population in operation of 1,000 discharged to an off-site receiver pipeline	Approximately 120 m^3/day	15 day (~2,000 m^3) hold-up
Industrial sewage	discharged to an off-site receiver pipeline	200 m^3/day	excludes storm drains
Instrument Air	clean, oil-free and dry air for instruments	~670 m^3/h at 0.7 MPa	See Table 4.26-3
Breathing air	oil-free compressors, non-cycling refrigerated dryers operating in parallel, coupled to air receivers	~380 m^3/day at ~0.8 MPa	~5 m^3 of buffer tanks in tokamak, tritium, hot cell and radwaste buildings
Special gases		Various	See Table 4.26-2

- The figure of 360 m^3/h is for 30 suited operators. Experience shows that this requires a major overhead in terms of support personnel and equipment. Therefore it is recommended to scale this figure down to well below 10 suited workers.

Table 4.26-2 Nitrogen, Helium and Special Gas Demands

Gas	Supply Capacity	Purity Specification
Industrial Nitrogen	~ 50 m^3 as liquid at 0.8 MPa	< 0.1% impurity
SF_6	6,000 kg initial fill; makeup at 60 kg/month	Industrial grade
Ultra-Pure Nitrogen	5 std m^3/month	Ultra-high purity
Ultra-Pure Helium	5 std m^3/month	Ultra-high purity
Deuterium	5 std m^3/month	Ultra-high purity
Hydrogen	25 std m^3/month	Laboratory Grade
Argon	10 std m^3/month	Laboratory Grade
Neon	10 std m^3/month	Laboratory Grade

Table 4.26-3 Compressed Air Station Capacities

Station	Buildings Served	Instrument Air Capacity std m^3/h *	Location of compressor
1	Tokamak Building	350	Site Services bldg.
2	Tritium Building	100	Site Services bldg
3	Hot Cell Building	100	Site Services bldg
4	Cryoplant Building	120	Cryoplant bldg
5	Emergency Power Supply Building	Included in (4)	Cryoplant bldg
6	Auxiliary Buildings	TBD	TBD
7	Site Services Building	TBD	Site Services bldg

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5 Influence matrix between content and WBS elements

Interface management at ITER is one of the most complex subjects that the ITER Project Management Tools will need to handle. A preliminary interface chart has been constructed for the top level interactions. This is shown in the figure below. The new Design Office software of Catia-5 and Enovia have interface management links. The Primavera P5 Planning and Scheduling platform is oracle data based and available to link into an interface management program. The Diagnostics Engineering group have similarly constructed a spread sheet for "Diagnostic Port Structure Interfaces with other Tokamak Systems" (IDM, ITER_D_22F7HD). An active "Interface Control" tool is being developed to link these elements.

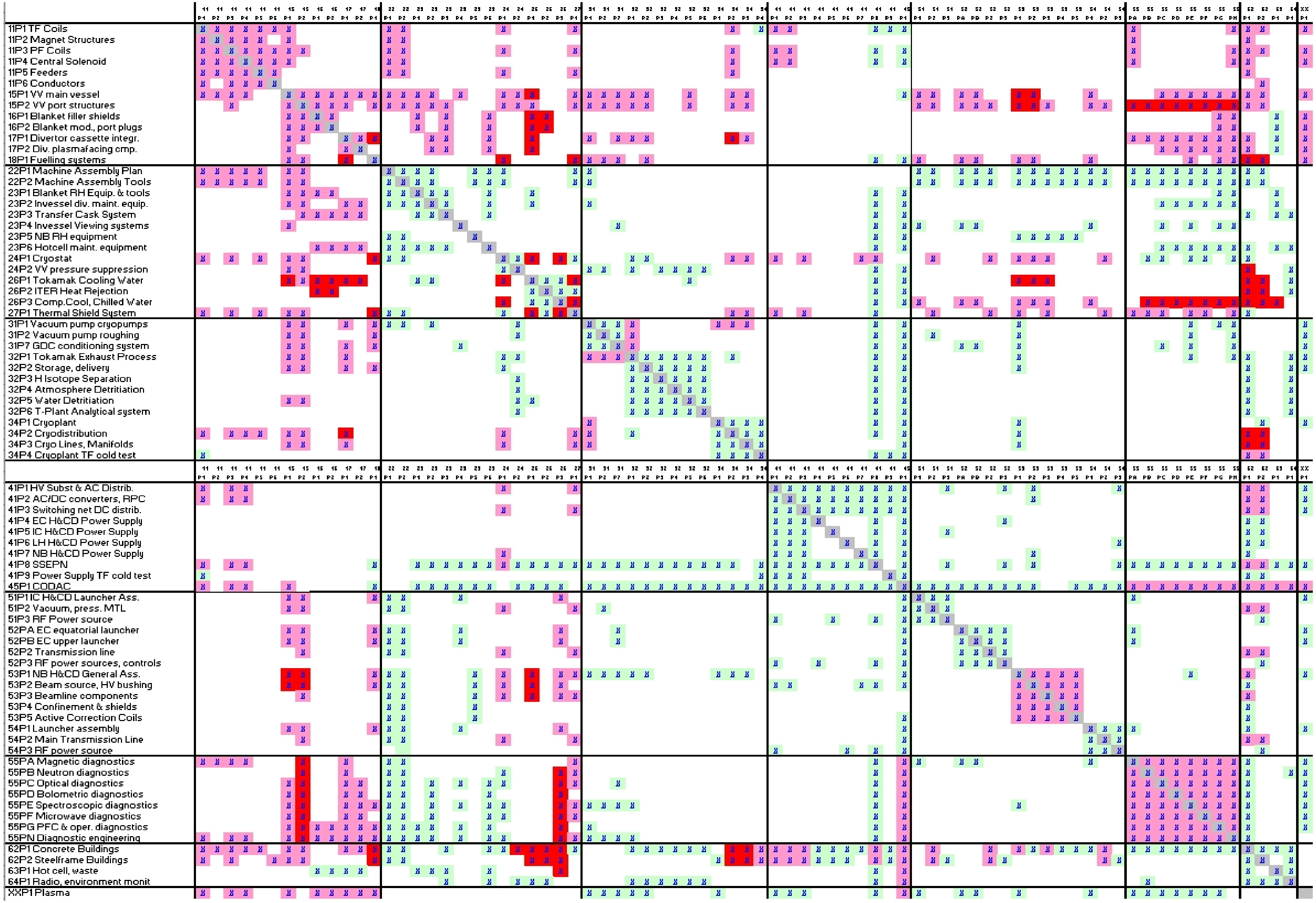
In the following an influence matrix between the elements of this document and the various WBS systems are defined. In this way the configuration management process will be able to identify the parts of the Project that must sign off modifications to this document.

Figure 5.1 ITER Interfaces between ITER Elements

Interfaces between ITER elements

To access the files describing the interfaces between the ITER documents click on the cross.

Colour coding:	
partner interface	X
normal interface	X
complex interface	X
very complex interface	X


11P1 TF Coils
11P2 Magnet Structures
11P3 PF Coils
11P4 Central Solenoid
11P5 Feeders
11P6 Conductors
15P1 VV main vessel
15P2 VV port structures
16P1 Blanket filler shields
16P2 Blanket mod., port plugs
17P1 Divertor cassette integr.
17P2 Div. plasmafacing cmp.
18P1 Fuelling systems
22P1 Machine Assembly Plan
22P2 Machine Assembly Tools
23P1 Blanket RH Equip. & tools
23P2 Invessel div. maint. equip.
23P3 Transfer Cask System
23P4 Invessel Viewing systems
23P5 NB RH equipment
23P6 Hotcell maint. equipment
24P1 Cryostat
24P2 VV pressure suppression
26P1 Tokamak Cooling Water
26P2 ITER Heat Rejection
26P3 Comp.Cool, Chilled Water
27P1 Thermal Shield System
31P1 Vacuum pump cryopumps
31P2 Vacuum pump roughing
31P7 GDC conditioning system
32P1 Tokamak Exhaust Process
32P2 Storage, delivery
32P3 H Isotope Separation
32P4 Atmosphere Detritiation
32P5 Water Detritiation
32P6 T-Plant Analytical system
34P1 Cryoplant
34P2 Cryodistribution
34P3 Cryo Lines, Manifolds
34P4 Cryoplant TF cold test
41P1 HV Subst & AC Distrib.
41P2 AC/DC converters, RPC
41P3 Switching net DC distrib.
41P4 EC H&CD Power Supply
41P5 IC H&CD Power Supply
41P6 LH H&CD Power Supply
41P7 NB H&CD Power Supply
41P8 SSEPN
41P9 Power Supply TF cold test
45P1 CODAC
51P1 IC H&CD Launcher Ass.
51P2 Vacuum, press. MTL
51P3 RF Power source
52PA EC equatorial launcher
52PB EC upper launcher
52P2 Transmission line
52P3 RF power sources, controls
53P1 NB H&CD General Ass.
53P2 Beam source, HV bushing
53P3 Beamline components
53P4 Confinement & shields
53P5 Active Correction Coils
54P1 Launcher assembly
54P2 Main Transmission Line
54P3 RF power source
55PA Magnetic diagnostics
55PB Neutron diagnostics
55PC Optical diagnostics
55PD Bolometric diagnostics
55PE Spectroscopic diagnostics
55PF Microwave diagnostics
55PG PFC & oper. diagnostics
55PN Diagnostic engineering
62P1 Concrete Buildings
62P2 Steelframe Buildings
63P1 Hot cell, waste
64P1 Radio, environment monit
XXP1 Plasma