ITER EDA DOCUMENTATION SERIES No. 23

ITER
COUNCIL PROCEEDINGS: 2001

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2001


ITER

ITER EDA DOCUMENTATION SERIES NO. 23

International Thermonuclear Experimental Reactor
(ITER)

Engineering Design Activities
(EDA)

ITER
COUNCIL PROCEEDINGS: 2001

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2001

ITER COUNCIL PROCEEDINGS: 2001
IAEA, VIENNA, 2001
IAEA/ITER EDA/DS/23

Printed by the IAEA in Vienna

November 2001

FOREWORD

Development of nuclear fusion as a practical energy source could provide great benefits. This fact has been widely recognized and fusion research has enjoyed a high level of international co-operation. Since early in its history, the International Atomic Energy Agency has actively promoted the international exchange of fusion information.

In this context, the IAEA responded in 1986 to calls at summit level for expansion of international co-operation in fusion energy development. At the invitation of the Director General there was a series of meetings in Vienna during 1987, at which representatives of the world's four major fusion programmes developed a detailed proposal for co-operation on the International Thermonuclear Experimental Reactor (ITER) Conceptual Design Activities (CDA). The Director General then invited each interested Party to co-operate in the CDA in accordance with the Terms of Reference that had been worked out. All four Parties accepted this invitation.

The ITER CDA, under the auspices of the IAEA, began in April 1988 and were successfully completed in December 1990. The information produced within the CDA has been made available for the ITER Parties and IAEA Member States to use either in their own programmes or as part of an international collaboration.

After completing the CDA, the ITER Parties entered into a series of consultations on how ITER should proceed further, resulting in the signing of the ITER EDA (Engineering Design Activities) Agreement on July 21, 1992 in Washington by representatives of the four Parties. The Agreement entered into force upon signature of the Parties, with the EDA conducted under the auspices of the IAEA.

As the original six-year EDA Agreement approached a successful conclusion, the Parties entered into a series of consultations on how future steps could be taken toward decisions on construction. A provisional understanding was reached that the EDA Agreement should be extended by three years to enable the Parties to complete their preparations for possible construction decisions. By the time of the expiration of the original EDA Agreement, the EU, JA and RF Parties had agreed to extend the Agreement while the US Party, complying with Congressional views, did not participate beyond an orderly close out activity ending in September 1999.

The ITER Engineering Design Activities were successfully terminated on 21 July 2001.

As part of its support of ITER, the IAEA is pleased to publish the documents summarizing the results of the Engineering Design Activities.

CONTENTS

DOCUMENTS OF THE ITER COUNCIL MEETING HELD IN TORONTO

DOCUMENTS OF THE ITER COUNCIL MEETING HELD IN VIENNA

INTRODUCTION

Continuing the ITER EDA, two further ITER Council Meetings were held since the publication of ITER Documentation Series No. 20, namely the ITER Council Meeting on 27 – 28 February 2001 in Toronto, and the ITER Council Meeting on 18 – 19 July 2001 in Vienna. That Meeting was the last one during the ITER EDA.

This volume contains records of these Meetings. Together with the twenty-two previous volumes in the ITER EDA Documentation Series, on:

- ITER EDA Agreement and Protocol 1 (DS/1)
- Relevant Documents Initiating the EDA (DS/2)
- ITER Council Proceedings: 1992 (DS/3)
- ITER Council Proceedings: 1993 (DS/4)
- ITER EDA Agreement and Protocol 2 (DS/5)
- ITER Council Proceedings: 1994 (DS/6)
- Technical Basis for the ITER Interim Design Report,
  Cost Review and Safety Analysis (DS/7)
- ITER Council Proceedings: 1995 (DS/8)
- ITER Interim Design Report Package and Relevant Documents (DS/9)
- ITER Interim Design Report Package Documents (DS/10)
- ITER Council Proceedings: 1996 (DS/11)
- ITER Council Proceedings:1997 (DS/12)
- Technical Basis for the ITER Detailed Design Report
  Cost Review and Safety Analysis (DDR) (DS/13)
- ITER Final Design Report, Cost Review and Safety Analysis (FDR) and
  Relevant Documents (DS/14)
- ITER Council Proceedings: 1998 (DS/15)
- Technical Basis for the ITER Final Design Report, Cost Review
  and Safety Analysis (FDR) (DS/16)
- ITER Council Proceedings: 1999 (DS/17)
- ITER-FEAT Outline Design Report (DS/18)
- Technical Basis for the ITER-FEAT Outline Design (DS/19)
- ITER Council Proceedings: 2000 (DS/20)
- Final Report of the ITER EDA (DS/21)
- Summary of the ITER Final Design Report (DS/22)

it presents essential information on the evolution of the ITER EDA.

ITER Council Meeting, 27–28 February 2001, Toronto

DOCUMENTS
OF THE
ITER COUNCIL MEETING
27 – 28 February 2001, Toronto

RECORD OF DECISIONS
ITER COUNCIL MEETING
27 28 February 2001, Toronto

1.1 The Council accepted participation as attached (**ROD Attachment 1**). The Council was informed about the nomination of a new Member from JA, Mr. T. Sugawa, and a new Member from the EU, Dr. A. Mitsos. The Council expressed its thanks to Mr. M. Konaka and to Dr. J. Routti for their valuable contributions to ITER during their terms as IC Members. The Council noted the designation of Mr. M. Drew to succeed Dr. E. Canobbio as EU CP, and the designation of Dr. P. Barabaschi as the Director's point of contact. The Council recorded its thanks to Dr. E. Canobbio, on the occasion of his retirement, for his consistent efforts for ITER.

1.2 The Council adopted the Agenda (**ROD Attachment 2**).

1.3 The Council took note of the presentations from Dr. H. Kishimoto on the successful completion of the Explorations, and from Dr. J.-P. Rager on the ITER International Industry Liaison Meeting held in Toronto in November 2000.

1.4 Having noted statements of Parties' status in particular concerning the readiness to start negotiations and the progress toward site offers, the Council encouraged the Parties to pursue preparations toward future implementation of ITER along the general lines proposed in the Explorer's final report.

1.5 Following the discussion on the transition from the EDA to the CTA, the Council noted the readiness of the RF and EU Parties to instruct specified current JCT members to remain at their places of assignment after the end of the EDA.

2.1 The Council took note of the Director's Status Report (**ROD Attachment 3)**.

2.2 The Council noted with satisfaction that meetings with the Director of the ITER Parties' Designated Safety Representatives had started and commended the progress toward achieving timely licensing processes with a good common understanding. The Council noted with appreciation the Director's view that no difficulties of principle in the licensing approach had been identified during the informal discussions with the regulatory representatives. The Council stressed the importance of providing appropriate information on safety, environmental, and waste management aspects of ITER for wide external consideration.

3.1 Taking note of the TAC Report and recommendations **(ROD Attachment 4**), the Council accepted the Draft Summary ITER Final Design Report (**ROD Attachment 5**). The Council commended the Director with the support of the JCT, the Home Teams, and Industrial Participants for their efforts to enable preparation of the Draft ITER Final Design Report in due time.

3.2 The Council noted with appreciation the conclusion of the TAC that "*ITER-FEAT is now ready for a decision on construction*" and the achievement of the 50% cost reduction target set in 1998. Recognising the enhancement of the project in terms of the strengthening of the Physics Basis and the accuracy of the cost evaluation, the Council stressed the importance of maintaining the momentum of the Project.

3.3 The Council considered the TAC obligations fulfilled and expressed its thanks to TAC for its consistent contribution in providing independent assurance of the scientific and technical coherence of the Project.

3.4 With regard to the TAC comments on the need after the end of the EDA for 1) leadership and a focus for further physics and technology R&D progress, and 2) the ongoing provision of independent technical advice, the Council recommended to the Parties to take notice of the relevant TAC recommendations in preparing the framework of their negotiations.

3.5 The Council agreed to transmit the Draft Summary ITER Final Design Report and supporting technical basis to the Parties for their consideration and domestic assessment with a view to providing comments to the Director within the middle of April. This will lead inter-alia to the synoptic Summary of the ITER Final Design Report suitable and available for wide distribution under the responsibility of the Director.

4.1 The Council took note of the MAC Report and Advice (**ROD Attachment 6**) and accepted its recommendations including, in particular, recommendation 5e on the proposed establishment of an ad-hoc body for the exercise of continuing joint administrative responsibilities that cannot be completed within the duration of the EDA.

4.2 Following the MAC recommendation, the Council requested the IC Chair to send a letter to the IAEA in order to discharge the US agent from Joint Fund responsibilities.

5. The Council took note of the report of the CP's.

6.1 The Council approved further tasks for MAC (**ROD Attachment 7**).

6.2 The Council approved further tasks for the CP's (**ROD Attachment 8**).

8. Upon invitation by the IAEA to meet in Vienna, the Council agreed to meet there on July 18-19, 2001. The Council also underlined the need to commemorate the conclusion of the EDA and recognise the forthcoming developments of the Project in a way that would confer, for such an occasion, the appropriate external visibility to the Project. The Council took note with appreciation the RF invitation to celebrate the completion of the EDA in Moscow in May 24-25, 2001 and asked the CP's to consult on finalising arrangements and developing a programme.

9.1 The Council approved this IC-Toronto ROD.

9.2 The Council approved the IC-Toronto Minutes.

IC-Toronto 2001
ROD Attachment 1

ITER COUNCIL MEETING
Toronto, 27-28 February 2001
List of Attendees

EU	Dr. U. Finzi	IC Member	European Commission
	Dr. J-P. Rager	Expert	European Commission
	Mr. M. Drew	CP	European Commission
	Dr. K. Lackner	Expert, HTL EU	
	Dr. D. Church	Expert	Canadian Dept. of Foreign Affairs
	Dr. J. Campbell	Expert	Canadian Dept. of Natural Resources
	Dr. P. Barnard	Expert	ITER Canada
	Dr. D. Dautovich	Expert	ITER Canada
	Dr. M. Stewart	Expert	ITER Canada
JA	Mr. T. Sugawa	IC Member	MEXT
	Dr. M. Yoshikawa	IC Co-Chair, MAC Chair	JAERI
	Mr. M. Nakamura	Expert	MEXT
	Mr. Y. Ando	Expert	MOFA
	Dr. H. Kishimoto	Expert	JAERI
	Dr. T. Tsunematsu	Expert, HTL JA	JAERI
	Dr. H. Takatsu	CP	JAERI
RF	Acad. E. Velikhov	IC-Chair	Kurchatov Institute
	Dr. Yu. Sokolov	IC Member	MINATOM
	Dr. O. Filatov	Expert, HTL RF	Efremov Institute
	Dr. L. Golubchikov	CP	MINATOM
	Dr. A. Evgrashin	Expert	MINATOM
	Dr. A. Kolotukhin	Expert	Ministry of Finance
ITER	Dr. R. Aymar	Director	
JCT:	Dr. Y. Shimomura	Deputy to the Director	
	Dr. P. Barabaschi	PC-w/D	
IC:	Dr. V. Vlasenkov	Secretary	
TAC:	Prof. M. Fujiwara	TAC Chair	
	Dr. T. Fukuda	Secretary	
MAC:	Dr. Y. Miura	Secretary	

IC-Toronto 2001
ROD Attachment 2

ITER COUNCIL MEETING

Toronto, 27-28 February
AGENDA

1. Meeting Opening and Arrangements
   Attendance
   Information on Completion of Explorations
   Information on Meeting of Parties' Industries
   Exchange of information on Parties' Status
   Adoption of Agenda

2. Director's Status Report

3. Draft Final Design Report
   3.1 Director's Presentation
   3.2 TAC Report and Recommendations

4. MAC Report and Advice

5. CP's Report

6. Further Tasks
   6.1 MAC
   6.2 TAC
   6.3 CP's

7. Other Business

8. Next Meeting(s) – Date(s) and Place(s)

9. Approval of the ROD and of the Minutes

IC-Toronto 2001
ROD Attachment 3

ITER EDA STATUS

report by the Director

1 This note summarizes the progress made in the ITER Engineering Design Activities in the period between the Moscow ITER Meeting in June 2000 and March 2001.

Overview

2 The Project has focused on drafting the Plant Description Document (PDD), which will be in July 2001 the Technical Basis for the ITER Final Design Report (FDR) and its related documentation in time for the ITER review process. The preparations have involved continued intensive detailed design work, analyses and assessments by the Home Teams and JCT who have co-operated closely and efficiently.

3 The main technical document has been completed in time for circulation, as planned, to TAC members for their review at TAC-17 (19-22 February 2001). Some of the supporting documents, such as the Plant Design Specification (PDS), Design Requirements and Guidelines (DRG1 and DRG2), Plant Safety Requirement (PSR) are also available for reference in draft form. A synoptic summary paper of the PDD for the Council's information is available as a separate document..

4 A new documentation structure for the project has been established as described in Annex 1. This document hierarchical structure facilitates the entire organization in a way allowing better change control and avoiding duplications. The initiative was intended to make this documentation system valid for the construction and operation phases of ITER.

5 As requested, the Director and the JCT have been assisting the Explorations to plan for future joint technical activities during the Negotiations, and to consider technical issues important for ITER construction and operation for their introduction in the draft of a future joint implementation agreement.

6 As charged by the Explorers, the Director has held discussions with the Home Team Leaders in order to prepare for the staffing of the International Team and Participants Teams during the Negotiations (Coordinated Technical Activities, CTA) and also in view of informing all ITER staff about their future directions in a timely fashion.

7 At the 18th IAEA Fusion Energy Conference (Sorrento 4-10 October 2000) the results of the ITER EDA were presented in one special oral session and one poster session. In total 31 papers were selected, the authorship of which includes JCT, Home Teams and ITER Physics Expert Groups Members.

8 The 3rd ITER International Industry Liaison Meeting was held in Toronto in November 2000. During this venue the Director proposed some ideas for the ITER construction procurement scheme and associated project management and discussed them with representatives from Industries from the EDA Parties as well as from the US. A copy of the Director’s presentation at Toronto is provided for information as a separate document.

9 One important element of the work was the completion by the Parties' industries of costing studies of about 85 "procurement packages," each representing a potential real procurement contract for an ITER component. The results, after analysis and evaluation by the JCT have provided the basis for a JCT “evaluated cost estimates” Report for all packages (Business Confidential) which was presented

during a one week meeting at Garching (29 Jan-2 Feb) to an Ad Hoc Group of Parties' costing Experts.

10 A meeting of the ITER Test Blanket Working Group (TBWG has been held in October 2000. The group has continued its activities during the period of extension of the EDA with a revised charter on the co-ordination of the development work performed by the Parties and by the JCT leading to a coordinated test programme on ITER for a DEMO-relevant tritium breeding blanket. This follows earlier work carried out during the EDA, which formed part of the 1998 Final Design Report. A concise summary of the meeting is attached in Annex 2.

Safety

11 A meeting was held in October 2000 with the ITER Parties' Designated Safety Representatives of regulatory bodies from Europe, Canada, Japan and Russia. This discussion was devoted to achieve a consensus on a set of safety principles and environmental criteria and on design requirements in order to provide all Parties expected to participate in the ITER joint implementation with an acceptable level of safety according to their own standards.

12 Each potential host country/party presented a summary of its licensing approach for ITER and the licensing process that may be followed. No fundamental difficulties have been identified and foreseen along the path toward a full authorization for construction.

13 During the meeting it was also agreed that it is required to maintain the process continuity and future commitments throughout the discussions with the regulatory authority in preparation for the licensing process. An ITER Legal Entity should exist to submit the application for the construction authorization. It is mandatory to have a competent 'Design Authority' (responsible for the design) to support dialogue with regulatory authority to take into account in the design regulatory requirements, for continuity of the safety organization, and for their transfer to the ITER Legal Entity that eventually obtains the licenses.

14 It has been the Director's understanding that the Partners will need to rely on the Design Authority to give assurance of the ability of the design to fulfill the requirements, based on the dialogue with the regulatory authority in each Host Country (Party), in particular to obtain a full understanding of the risk that the licensing process might be unduly extended.

R&D Progress

15 The charging experiments of the CS Model Coil and the CS Insert Coil were carried out until August 2000 with excellent performance results achieved, fulfilling all the goals of the technology development of the CS Model Coil, namely, maximum magnetic field of 13 T, operation current of 46 kA, fast ramp-up rate of 0.4 T/s and ramp-down rate of –1.2 T/s from 13 T and a 10,000 cyclic test. With this success of the CS Model Coil project, developing the fabrication technology of the ITER CS coil and validating its engineering design, we are now ready technically to initiate construction of the ITER CS coil with confidence.

16 Fabrication of the TF Model Coil was completed and delivered to FZK at Karlsruhe to be assembled with a LCT coil and to be tested at the TOSCA test facility. With this achievement one of the main objectives of the project has been reached namely the demonstration of the feasibility of ITER relevant coil manufacturing processes and quality assurance methods. The first charging test is expected by mid-June.

17 The preparation of the review paper titled “Review of the ITER Technology R&D” was pursued. The whole program of the ITER technology R&D and achievements covering the seven large R&D and nine key R&D areas are summarized in a special issue of the Journal Fusion Engineering and Design. The manuscript was submitted to the publisher at the end of January and the printing is expected before the end of the EDA.

Joint Central Team and Support

18 The status of the Team at the start of February 2001 is summarised in Table 1 below. There has been small changes in the number of staff and their distribution among Parties: one Canadian left Naka and ITER, one European moved from Naka to Garching and another one left Garching and ITER; three additional RF members have joined the project at Naka and one left Naka and ITER.

Table 1: JCT - Status by Joint Work Site and Party at 1 February 2001

	Garching	Naka		Total
by Site	45[1]	50		95[1]
	EU	JA	RF	
by Party	37[1]	33	25	95

[1] includes two Canadians provided through the Canadian association with the EU Party.

19 The JCT numbers have been supplemented by VHTP’s (~3-4 PPY from RF, and ~4-5 PPY from EU, in average per year) and other temporary attachments to the JCT.

Task Assignments

20 Tables 2 and 3 below summarise the status of R&D and Design Task Agreements. More details and commentary are presented in the specific papers to MAC. Table 2 covers the numbers of Task Agreements over the entire period of the ITER EDA. Table 3 summarises the cumulative total values of task agreements concluded to date.

Table 2. Number of Task Agreements (cumulative)

TA Status	**R&D** Number	**Design** Number
Task Agreements committed (EU,JA,RF)	**643**	**539**
Task Agreements completed	486	432
Task Agreements on-going	157	107
US (to 7/99)	*173*	*162*

Table 3. Task Agreements Summary Values per Party

Party	R&D (IUA)	Design (PPY)
EU	232,341	293.04
Japan	224,815	267.98
Russia	92,953	230.45
US(to 7/99)	*108,023*	*170.71*
Total	658,132	962.18

Joint Fund

21 Following notification from the JA Party that it had completed the procedure to confirm the validity of the ITER EDA extension, it has been possible for the ITER Council to regularise outstanding Joint Fund matters including, approval of the Accounts for 1998, the Revised Budget for 1999, the Accounts for 1999 and the Budget for 2000. The Revised 1999 Budget and the Budget for 2000 have been adopted in accordance with Article 3 of the Joint Fund financial rules. The Director has made the formal calls for contributions. The Parties are requested to complete payments of any sums outstanding in respect of 1999 and 2000.

22 Following the approval of the 1999 Accounts, the USDOE has confirmed the final charges incurred by the US Agent (SAIC) in finalising its Joint Fund activities. It is therefore now possible to proceed with the discharge of the Agent by requesting the IAEA to send a letter, whereupon SAIC will remit to the IAEA Trust Fund the small balance of Joint Fund monies, that remains in its account.

23 The ITER Director has requested the current Joint Fund Agents to submit their accounts for 2000 promptly so as to assist planning of an orderly wind-up of the Joint Fund arrangements at the end of the EDA duration. Initial indications are that the total unspent appropriations at the end of 2000 amounted to about $1.3M of which some $0.2M represented unspent appropriations carried forward from 1999. In accordance with Article 5.8 of the ITER Joint Fund rules, unspent appropriations from the 2000 budget are carried forward into 2001.

24 The ITER Council is invited, after MAC advice, to consider the overall policy and arrangements for winding up the Joint Fund at the end of the EDA, including proposals on the possible disposition and use of residual Joint Fund property and cash. As agreed at the ITER Meeting in Moscow, 29-30 June 2000, the issues of the unspent appropriations at the end of 2000 and of the provisional Joint Fund Budget for Jan-Jul 2001 will be addressed in the context of the policy conclusions.

25 The RF Joint Fund Agent is pursuing support contracts which provide an efficient contribution to the JCT design effort . In 2000 contracts to the value of $494,750 have been concluded with four Institutes. The list of contracts concluded after June 2000 is attached in Annex 3.

ITER Physics

26 The seven ITER Physics Expert Groups are in full operation and the arrangements for continued interaction with US fusion scientists on generic issues of tokamak physics are proceeding smoothly. A new framework, called International Tokamak Physics Activity (ITPA), is being planned under the auspices of IFRC of

IAEA. It was proposed that the new framework after July 2001 should have a structure and membership similar to ITER Expert Groups with additional participation of US physicists.

Table 4: High Priority Physics Research Areas

	Research Areas	Issues
*	Finite-β effects in H-mode	Tolerable ELMs ($\delta W/W<2\%$) with good confinement alternate to type-I ELMs (e.g. type II, Type III+core confinement) Stabilisation of neoclassical islands at high β and recovery of β
*	Plasma termination and halo currents	Runaway electron currents: production and quenching, e.g. at low safety factor
*	Sol and divertor	Achievement of high n_{sep} and relation of $n_{sep}/\langle n_e\rangle$ in ELMy H-modes, especially at high n and δ Carbon Chemical sputtering, re-deposition and deuterium retention/cleaning methods
*	Core confinement	Non dimensional scaling and identity experiments; effect of finite β and flow shear Determine dependence of τ_E upon shaping, density peaking etc.
*	H-mode power threshold	H-mode accessibility in ITER-FEAT , Data scatter
*	Good H-mode confinement at high n	Confinement degradation onset density; its dependence on aspect ratio, shape and neutral source
*	Pedestal physics	Scaling of pedestal properties and ELMs Effects of plasma shape on pedestal and ELMs MHD stability analysis of transport barrier
	Internal transport barrier properties	ITB power thresholds vs n, B, q, Te/Ti, Vrotation etc. for strong reversed shear ($q_{min}>3$), moderate reversed shear($q_{min}>2$, and weak shear ($q_{min}>1$). Compatibility with impurity exhaust and divertor Accessiblity of ITBs in reactor scale devices at low toroidal rotation, $Ti/Te \approx 1$, and flat density profile, etc.
	Resistive Wall Mode	RWM analysis and experimental verification
	Heating/CD, Steady State	Development of steady state scenarios :active current and pressure control Active control of LHCD coupling Assess fast particle effects (EPMs and ITBs)
	Diagnostics	Continue assessment of possible methods for measurement of q(r) and search for new approaches Continue study of First Mirrors especially effects of deposition and possible mitigating methods Assess impact of RIEMF on magnetic measurements and perform improved measurements on prototype magnetic coils Complete determination of measurement requirements for divertor target and divertor plasma parameters (in collaboration with the Divertor Expert Group), and complete assessment of probable performance of proposed diagnostic methods
* : relevant to main scenario (ELMy H)		

27 The priorities for physics research in 2001 as set forth by the ITER Physics Committee, held on 14 October 2000, are shown in Table 4. The main objectives are to strengthen further the physics basis for the inductive Q = 10 operating scenario and to explore further and clarify scenarios for new modes of operation that could be used to approach steady-state operation.

Annex 1

ITER DIRECTOR Memo

TO: All staff

FROM: R. Aymar

SUBJECT: **New arrangements for future ITER documentation**[i]

DATE: 18 October, 1999

According to the EDA Agreement requirements of what should be achieved, and following the experience acquired in writing project documents up to 1998, the design documentation will be somewhat reorganised. The idea is that this documentation system should remain valid for construction and operation, but for obvious reasons its volume should be streamlined compared with the FDR global documentation. The annual series of reports to the ITER Council will remain, but its Technical Basis will be reduced in volume. The system is hierarchical, avoids duplication, and allows control of changes at all levels. A single person is clearly responsible for each document.

By July 2001 we will have to deliver to the Parties an FDR and all supporting documentation electronically. Therefore please ensure that all documents you refer to are made available electronically (at least in pdf) as part of the document package.
This includes reports from the Home Teams. *In addition, as a separate electronic item, please include with your document any background memos not available electronically from another source that are relevant but not explicitly referred to*. The objective is to have a complete set of documentation of the project without reference to other sources.

In the following, the word "system" is used to mean a set of functionally related elements. The word "process" is used to describe the dynamic behaviour of one or more systems.

The main features of the new ITER documentation are as follows (see figure):

- A top level *Plant Design Specification (PDS)* document, where externally imposed essentially design-independent requirements at the highest level are defined, including safety principles and criteria.

- Design Requirements and Guidelines Level 1 (DRG1) deals with the requirements and specifications above the system level (DRF1 Table of contents is attached). This includes not only system-wide requirements but also interfaces or specifications affecting the design of more than one

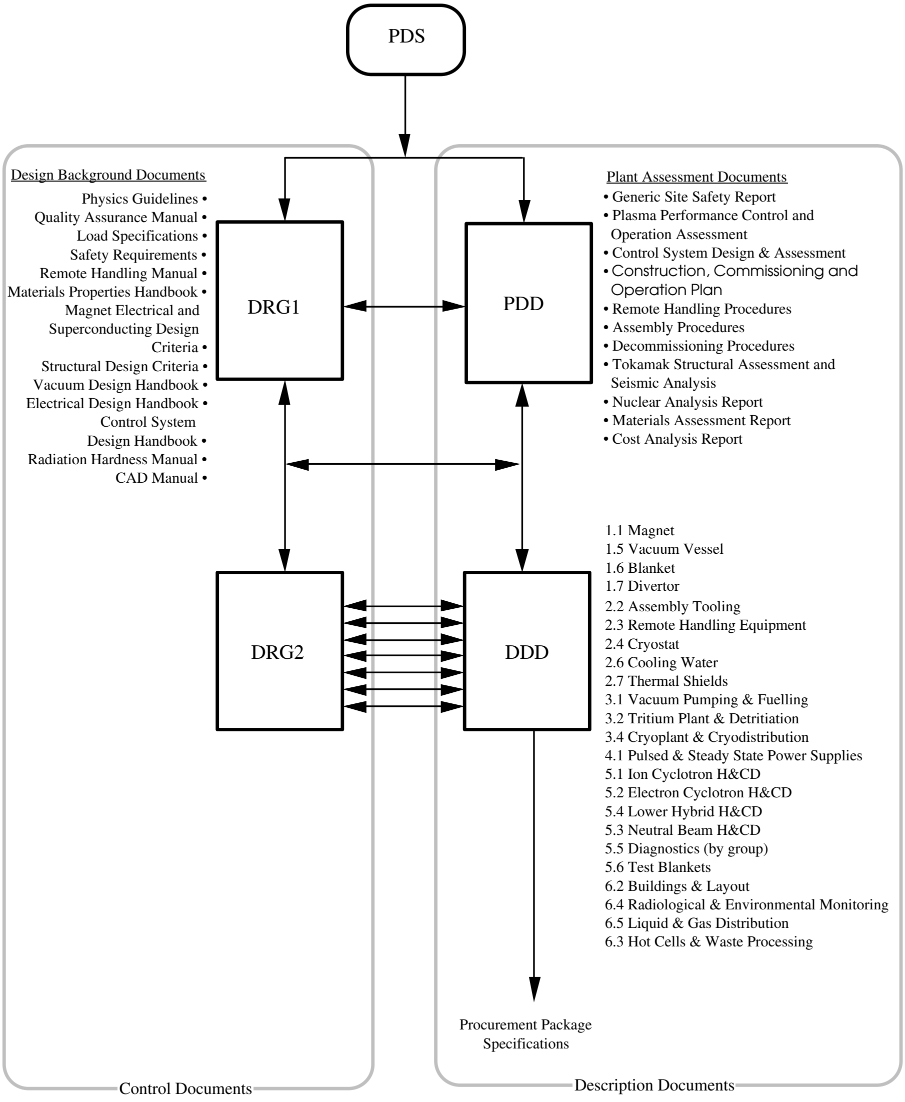
---

[i] This is a proposed scheme. It will be reviewed and modified as necessary. Names of responsible individuals will be established accordingly.

single system. DRG1 identifies the functional and physical interfaces between two systems and refers to any document and drawings defining the interface in more details It effectively includes overall “configuration drawings” More detailed "Design Background" documents are annexed.

- Design Requirements and Guidelines Level 2 (DRG2) defines in one document the boundaries of each system and deals in more detail with the functions, requirements and specifications at the system level. The system division is identical to that of the DDDs.
- Design Description Documents (DDDs) are one per system. Their normalised format is attached.
- The Plant Description Document (PDD) is the global plant description. It summarises the design based on the details in the DDDs, gives an overview of major plant processes which usually involve more than one system, summarises plant level assessments, and overall planning. The latter items are described in more detail in "Plant Assessment" document annexes.
- The new arrangement of documents will require the DDD TO to be aware of the PDS and the annexes to the DRG1, since they provide the design frame in general (and sometimes even in specific) terms. The designer will have to adhere only to the requirements listed in the relevant section of DRG2, and be knowledgeable about the content of DRG1 and adhere to any requirements there, or try to get them changed if they drive that part of the design well away from the optimum.

Overall hierarchy of ITER documentation



DRG1 - Table of contents

DDD Table of Contents

Each DDD will have the following format:

1. **Engineering Description**

1.1 System Description

Description of the overall design of the complete system hardware and internal processes in terms of all the functions and requirements in DRG2, based on the conclusions from performance analyses in section 2. This includes instrumentation and control, that part of all external interfaces that will be procured as part of this system, as well as assembly, commissioning, maintenance and decommissioning (relying heavily on PDD annexes on these topics).

1.2 Component Description

Description of the design of the component hardware or subsidiary internal processes, referring where necessary to subsidiary functions and requirements in DRG2, or subsidiary conclusions from performance analyses in section 2, where these are not covered in section 1.1. The emphasis is to be on drawings, with the minimum necessary text.

1.3 Procurement Packaging

Relationship e.g. by list of subcomponents, to procurement packages. Description of type of procurement packages used, including how responsibilities are to be shared. Manufacturing processes to be followed. Description of the tests to be conducted to assure the system quality. Acceptance tests, and equipment outside the scope of the procurement package that must be available for them if performed on site. List of all key "line items" to be considered in cost of each procurement package. Timescale for procurements.

2. **Performance Analysis**

Analyses backing the above design, summarised in sufficient detail for understanding, and making full use of references covering the details. The analyses should be described in approximate order of importance and must include an FMEA. Analyses should be reported for normal and off-normal loading conditions, and the absolute limiting operation should also be predicted. The analyses should elaborate and underpin but not repeat more "lumped" analyses conducted at the process level.

Typical analyses include:

- System analyses
- Tradeoff analyses of design options (only covering "current" alternatives)
- Structural Analyses
- Seismic Analysis
- Thermohydraulic analyses
- Thermomechanical analyses
- Electromagnetic analyses
- Analysis of repair procedures
- Failure modes and effects analysis (FMEA)
- etc.

Appendices

A List of Drawings

B List of References

ANNEX 2

ITER-FEAT Test Blanket Working Group (TBWG)

6 Month Report for the second Half of 2000

Introduction

The ITER Test Blanket Working Group (TBWG) has continued its activities during the period of extension of the EDA with a revised charter on the co-ordination of the development work performed by the Parties and by the JCT leading to a co-ordinated test programme on ITER for a DEMO-relevant tritium breeding blanket. This follows earlier work carried out during the EDA, which formed part of the Final Design Report.

Whilst the machine parameters for ITER-FEAT have been significantly revised compared to the FDR, testing of breeding blanket modules remains a main objective of the test programme and the development of a reactor-relevant breeding blanket to ensure tritium fuel self-sufficiency is recognised as a key issue for fusion. Design work and R&D on breeding blanket concepts, including co-operation with the other Contracting Parties of the ITER-EDA for testing these concepts in ITER, are included in the work plans of the Parties.

During the second half of 2000 there was one meeting of the TBWG in October, which was preceded by a two day working meeting of Experts, in which interface issues between the blanket test modules and the ITER machine were addressed, covering remote handling, port interface, ITER services, integration in the pit, tritium plant and tokamak cooling water systems (TCWS) vault.

Summary of Activities

The key focus for the October meeting and the key action addressed during the period was the completion of the TBWG Report for the period of extension of the EDA, in which the work performed by the Parties and by the JCT, related to blanket R&D and to the test programme, is described. A first working draft was completed before the end of the year. The boundary conditions for testing and the requirements imposed on the test blanket modules (TBM`s) have been determined by the ITER-JCT.

Three adjacent main horizontal ports have been allocated for blanket testing, numbers 1, 2 and 18. A Design Description Document (DDD) has been produced for each of the ports used. All common features, including frame details and a remote handling description, are included in the DDD for port 1. Features specific to a particular port are described in the port specific DDD. A co-ordinated blanket test programme has been developed and is described. Each of the Parties has provided detailed descriptions of the TBM`s. Interface issues, auxiliary systems and safety considerations are also described in the report.

It is concluded in the report that for the blanket test programme, the three ports allocated offer adequate space for testing of up to six different types of breeding blanket. Most of the important breeding blanket functions can be investigated with the parameters of ITER-FEAT and the machine operational scenario offers a stepwise increase in performance, to which the overall blanket test programme can be synchronised. The test strategies differ for the different blanket concepts, resulting in sequential testing involving different test modules, which may be optimised for single or combined effects testing. The main effects to be studied relate to electro-magnetics, neutronics, thermo-hydraulics, thermo-mechanics and tritium breeding/extraction/migration.
Key aims of the tests are code validation and verification of design assumptions based on calculations and out-of-pile or in-pile fission reactor experiments. However, for some testing parameters considerable extrapolations have to be made from the ITER conditions to those of a potential fusion reactor. This is in particular the case for the first wall heat load and for the pulse length. It is desirable to obtain the longest possible single pulse length. Furthermore, due to the limited neutron fluence available in ITER, materials studies related to neutron irradiation will have to be conducted in fission reactors and in an accelerator based intense 14 MeV neutron source, such as IFMIF. The ITER remote handling and maintenance scheme will permit the exchange and servicing of the blanket test modules. Space available in the hot cell is limited and must be taken into consideration for operations of longer duration. Finally, operational safety during blanket testing can be assured at all times on the basis of the safety principles adopted for ITER. Accident hazards are generally minimised by material inventories.

Overall, the Report concludes that, from the data presented, a high level of confidence in a successful test programme on the ITER-FEAT machine for a DEMO-relevant tritium-breeding blanket is justified. It is recommended that effort be maintained on this area of R&D in the future.

The Report, together with revised DDD’s for the blanket concepts will be completed during the first half of 2001. A final TBWG Meeting is scheduled for mid May 2001 in Moscow.

Annex 3 RF Design support contracts - 2000

Ref.No.	Title/Dates/Volume	Institute	Financial status
13-00	**Thermal Creep in Tungsten Flat Tile Target** Start - 1 August 2000 Final - 28 February 2001	Efremov Institute	**$11,250** 1 Stage
14-00	**Heat Sink Fatigue Lifetime/CuCrZr Minimum Properties** Start - 1 July 2000 Final: 30 Dec. 2000	Efremov Institute	**$3,750** 1 Stage
15-00	**Lifetime of the Neutron Irradiated CFC&W Armoured Vertical Target** Start - 1 July 2000	Efremov Institute	**$11,250** 1 Stage
16-00	**Beryllium Dust on Hot Surfaces** Start - 1 July 2000	Efremov Institute	**$3,750** 1 Stage
17-00	**Electro-magnetic Analyses** Start - 1 July 2000 Final - 30 December 2000	Efremov Institute	**$3,750** 1 Stage
18-00	**Hypervapotron Analysis** Start - 1 July 2000 Final - 28 February 2001	Efremov Institute	**$15,000** 2 Stages
19-00	**Electromagnetic, Structural, Thermal and Hydraulic Analyses for the VV** Start - 1 June 2000 Final - 30 November 2000	Efremov Institute	**$48,000** 1 Stage
20-00	**Study of the ITER-FEAT correction coils** Start - 21 August 2000 Final - 21 December 2000	Efremov Institute	**$12,000** 1 Stage
21-00	**Physics Design** Start - 1 August 2000 Final - 20 July 2001	Kurchatov Institute	**$60,000** 4 Quarters
			Total from June to December 2000: **$168,750.00.** From January to June (12 contracts):**$326,000**

IC-Toronto 2001
ROD Attachment 4

ITER TECHNICAL ADVISORY COMMITTEE
19-22 February, 2001
ITER Garching Joint Work Site, Germany

REPORT OF THE TAC-17 MEETING

1. Introduction and background

After the TAC-16 meeting held in June 2000, intensive work to elaborate a detailed design document, including the cost estimate based on the analysis of procurement packages, has been performed by the JCT and Home Teams, and the Draft Final Design Report of ITER-FEAT was made available by the Director at the end of January 2001.

According to the decision taken at the ITER Meeting held in Moscow on 30 June 2000, TAC was charged to conduct a "review of the Draft Final Design Report of ITER-FEAT." Accordingly TAC-17 was convened to assess the ability of the self-consistent overall design both to satisfy the technical objectives previously defined and to meet the cost limitations, in order to provide technical advice to the ITER Council on 27 and 28 February 2001 in Toronto.

Eleven TAC members, eleven invited TAC experts and three Home Team Leaders participated in the review. The Joint Central Team staff gave a total of eleven presentations related to the description of the above report.

TAC members have agreed to provide their input to the development of the Detailed Design Document in time for the completion of the EDA documentation in July.

2. Overall assessment and key recommendations from TAC

(1) TAC appreciates the substantial progress made in physics and engineering design activities throughout the process of elaborating the Outline Design Report, Progress Report and Draft Final Design Report, which has led to the establishment of a design basis. TAC hereby greatly acknowledges the dedicated effort and the intensive design work made by the Director, JCT and Home Team members.

(2) TAC has high confidence that in the inductive mode of operation ITER-FEAT can meet its objectives of extended burn at a power multiplication of Q•10 for the reference operating scenario ($I_p = 15$ MA, $P_{aux} = 40$ MW). This scenario provides adequate margins to achieve the objectives and is robust against the principal operational boundaries. The approach to ignition (Q•50) can be explored at higher plasma currents (~ 17 MA), consistent with the engineering design, with a sufficient margin based on existing physics databases. With the addition of non-inductive current drive, it has the flexibility to establish hybrid scenarios with Q ~ 5 and pulse durations of up to 1500 s, as a route to establishing steady-state operation. The capability to investigate scenarios aimed at demonstrating full steady-state operation with the ratio of fusion power to input power for current drive of at least 5 has been confirmed in some detail.

(3) TAC notes that the engineering design has made good progress since the last TAC meeting and that the results achieved in all R&D areas validate the ITER-FEAT engineering design.

In particular, the vacuum vessel (VV) structural integrity has been validated by the results of VV structural analyses on single and combined load cases for the $I_p = 15$ MA standard operation. Operation in the range 15-17 MA offers one route to allow studies of performance with higher Q. TAC agrees to the proposed addition of local reinforcement of the vessel to accept the rare occurrence of high load conditions in 17 MA operation with full confidence.

(4) TAC is pleased to note that considerable effort has been made by the JCT and the Home Teams to assess the safety of ITER and acknowledges the conclusion of the informal meeting of the Parties' designated safety representatives that the overall approach of ITER safety, based on the deployment of the defence-in-depth and the ALARA principles, appears to be compatible with the licensing requirements of the Parties.

(5) The investment cost for the construction of ITER-FEAT has been estimated by the JCT to be 49.2% of the cost of the 1998 design of ITER. TAC considers this figure to be credible. Some R&D in order to optimise manufacturing in view of potential cost reductions should be continued.

(6) The focus of the international activities on Physics R&D should continue, together with close coordination with technology R&D including safety studies, in support of ITER and to further enhance the prospects for the development of fusion as an attractive future energy source.

Conclusion:

TAC considers that the proposed design is based on a firm physics and engineering basis, satisfying the ITER objectives and cost limitations. The proposed design gives confidence in the ITER-FEAT physics and engineering performance and in the attainment of the envisaged technological goals of the project. ITER-FEAT is now ready for a decision on construction.

Recommendations by TAC:

(a) TAC recommends that to take ITER forward under the new arrangements there is a need for strong leadership and a focus for co-ordinated physics design and coherent technology activities. TAC believes that this is essential to ensure that ITER-FEAT fully benefits from the international physics and technology programmes, thereby enhancing its performance, flexibility and reliability.

(b) A technical review body should be established which comprises the present range of disciplines as represented for example in TAC. This would bring together the physics expert groups, the physics committee (under the International Tokamak Physics Activity), and the technology (and materials) R&D programmes of the Parties, together with the possibility of the participation of third parties.

3. Progress in physics R&D and assessment of the physics progress in PDD

3.1 Objectives

TAC confirms that ITER-FEAT is robust to meeting its objectives of extended burn in inductive operation with a power amplification of greater than 10 and has sufficient margin to explore the approach to ignition. There is good flexibility to explore hybrid scenarios with a long pulse duration and scenarios aimed at demonstrating steady-state operation with the ratio of input power to current drive power of at least 5.

3.2 Progress in Physics R&D

Since the last TAC meeting, there has been substantial progress in the Physics R&D, in particular with the demonstration of good confinement at high normalised densities in JET with pellet refuelling from the high field side, impurity seeding and by operation at the ITER-

FEAT triangularity. This significantly confirms the operating window and performance capability of ITER-FEAT.

The energy loss from Edge Localised Modes (ELM's) leads to peak power loads on the divertor targets which can give rise to rapid erosion. A range of techniques to reduce the amplitude of the ELM's has been explored and a variety of scenarios has emerged which offers the potential for extended divertor lifetime and yet maintains the basic ITER-FEAT objectives of sustained burn with an energy amplification of Q~10.

There has been good progress with the further development of scenarios aimed at steady-state operation with reduced power thresholds for the formation of internal transport barriers and the required confinement enhancement, but so far for limited pulse duration and normalized beta.

The control of seed islands which can lead to neoclassical tearing modes (NTM's) has allowed significant pressure increases and augurs well for further performance enhancement. The direct stabilisation of NTM's has also been achieved but has not yet resulted in equivalent increases in plasma pressure.

3.3 Assessment of the physics progress in the PDD

TAC notes with pleasure the progress made in the PDD on a variety of modelling issues sought in its last report. Significant enhancements in the power amplification can be achieved as a result of density peaking from high field side pellet fuelling, based on results from existing devices, preferential ion heating and a reduction in the ratio of ion to electron thermal conductivity, leading to greater robustness in achieving the objectives.

The inductive basic scenario of Q•10, hybrid and steady-state scenarios have been shown to be consistent with the divertor requirements. Results on the exhaust of helium ash indicate that it is acceptable for the basic and hybrid scenarios. For the steady-state scenarios the rate of helium exhaust depends on the degree of magnetic shear reversal. More complete 2D divertor modelling for the steady-state scenario shows that a lower effective charge and normalised beta are possible at the presently achieved confinement enhancement. The results from current experiments involving internal transport barriers are now approaching the requirements for ITER-FEAT. Nevertheless, the database and scaling (with varying degrees of shear reversal) for the steady-state scenarios need to be expanded, particularly to longer pulse durations.

Basic physics models are leading to improved understanding and predictions for the edge pedestal and the scaling and structure of edge localised modes, though a more focussed, co-ordinated effort in this area in theory, modelling and experiments, is called for. This is leading to improved predictions of divertor power handling and erosion.

As carbon has been selected for the lower part of the divertor target in the strike point region, there is increasing focus and co-ordinated efforts on the development of techniques to minimise the retained tritium in the vessel and effect its removal.

Disruption characterisation has improved, which should lead to better techniques for mitigation and avoidance.

Fusion power control techniques have been modelled for the different scenarios indicating a requirement for good diagnostic measurements for real-time control.

Extensive analysis of the poloidal field control system has been performed, including its ability to respond to transient events, and it is clearly capable of providing equilibrium control over the full range of scenarios foreseen.

The control of NTM's has been analysed in some detail and the prospects for suppression within the power capability of the ECRH system are good.

Calculations of the fast particle losses due to toroidal field ripple in the presence of ferromagnetic inserts lead to acceptable losses. The effect of MHD activity and scenarios with high central safety factor on fast particles and Alfvén modes are being examined theoretically and experimentally.

3.4 Key findings

TAC encourages the further development of techniques to reduce the amplitude of ELM's under the proposed new framework with its focus on physics R&D.

The heating and current drive systems have been adapted to the ITER-FEAT design and the issues of power coupling, pulse length and reliability are being addressed, but there is a need to focus more co-ordinated effort on this area to ensure that required performance becomes available on the appropriate time scale.

A comprehensive set of diagnostics is proposed which should provide good capability for machine protection and control and for physics studies. There is an ongoing R&D programme particularly on diagnostic in-vessel components, but more effort in the Parties is needed to take the design tasks forward.

3.5 Conclusion

TAC endorses the physics analysis presented in the Plant Description Document as fully reflecting the current state of physics R&D and as a firm basis as an input to a construction decision.

4. Assessment of engineering design

4.1 Magnet and supporting structures

The reference design for both, the TF coils and the CS, has been worked out in further detail. The study to confirm the feasibility of the required accuracy on the wedged surfaces at the inboard legs of the TF coils, recommended by the TAC at the last meeting, has been carried out with satisfactory results. TAC, however, notes, that R&D to validate the choice of the material for the pre-compression ring of TF coils would be valuable.

TAC supports that for the CS conductor, beside the reference solution (Incoloy square jacket with reinforcement) two alternative designs (Ti/SS jacket and pure SS jacket) are being considered for which R&D is ongoing in Japan and EU respectively. For the cost evaluations the design option with the intermediate cost, i.e. the pure SS jacket, has been chosen.

4.2 Vacuum vessel

At the highest current foreseen, $I_p = 17$ MA, in the most demanding load cases the stress exceeds the allowable level at some localized places of the vacuum vessel (VV) for the presently assumed specifications of halo current and plasma current quench time. Compliance with the allowable level can be recovered by local reinforcement of the VV and the blanket module connections to the VV. Improved experimental characterization of disruption quench time scales and halo current distribution can also contribute to the establishment of adequate margins for structural integrity in the extreme load case.

4.3 Review on R&D

The CS model coil project (L1) has provided very good results since the last TAC meeting, confirming the applied design and manufacturing principles. It will be completed by testing the TF conductor insert coil provided by the RF home team by the end of the EDA.

The TF model coil project (L2): The coil manufacture has been successfully completed, confirming the feasibility of all fabrication processes. Installation of the coil in the test facility is proceeding. Tests will start in June 2001 and will continue after the end of the EDA.

TAC already recommended at the last meeting several new R&D tasks which even after the EDA phase should be carried out in preparation of the construction phase and which can result in cost and design optimizations. The TAC noted with satisfaction that some of these tasks are already proceeding in the Parties e.g.,

- Investigations on the alternatives for the Incoloy jacket of the CS conductor,
- Improved fabrication techniques for the radial plates of the TF coil windings, (The reduced costs have already been taken into account for the ITER cost evaluation.)
- R&D for the NbTi conductors. Strands are being provided by the RF and the EU will fabricate a conductor and an insert coil, for testing preferably in the CS model coil test facility.

5. Assessment of overall plant design, safety and cost

5.1 Overall plant design including the heating and current drive systems

The plant layout for the generic site has been designed in view of minimizing the area covered by the buildings and of optimizing the hydraulic, fluidic and electrical interconnections in view of reducing the costs.

The tritium system takes into account the existing operating experience in similar plants and the design has been focused to minimize the mobilisable inventory and to ensure tritium confinement, as well as to accommodate the ITER operational requirements and flexibility. Dynamic simulation of the system response has shown possible operation schemes with a tritium inventory near the project target of 450 g.

Atmosphere and water detritiation systems have been also developed meeting the objective of minimizing the normal discharges and accidental releases of radioactive materials and tritium.

Considering the different plants, the power supplies and the water cooling system are properly designed. In particular the water cooling system has been rationalised by reducing the number of independent systems. This has a positive impact in reducing the operator radiation exposure during maintenance operations.

The cryogenic plant is designed to absorb the thermal loads generated by the ITER cryogenic components. The modular nature would allow the system (four equal modules) to cope with unforeseen demands.

The design of hot cell and remote handling equipment has been based on the R&D results from full-scale mockups.

Four heating and current drive systems have been studied (LH, ICRH, ECRH, Neutral Beams), including integration of the detailed interface structures into the vacuum vessel ports. An initial choice of ICRH,ECRH and Neutral Beams has been made and will have to be confirmed by the results of the ongoing experimentation and R&D. The development of the new, more advanced components of these systems is underway in the Parties. TAC considers that a more intense effort should be conducted with a focus on the development of the 1 MeV Neutral Beam Injector and of the 1 MW, 170 GHz gyrotron for the Electron Cyclotron System, both for continuous duty, in order to achieve the system performance needed for operation.

5.2 Safety

A wide range of safety analysis and assessment has been performed including the international activities on safety R&D, code development for safety analysis and its verification against experiments and benchmark calculation.

It is also noted that the classification of components and subsystems and the event categories have been simplified through the safety assessment and iteration with the design development. The safety classification is implemented in specific design rules at the component level.

Attention has been called to the possible uncertainties in activation calculations and the need to optimise shielding.

The assessment provides confidence that the operation of ITER will be safe for the workers and for the public even considering hypothetical events. An aspect that still requires effort and experimentation is the problem of tritium retention, during operation, and the build up of tritium inventory in the first wall. Wastes represent a sensitive issue in view of the operation and decommissioning of ITER. The amount of radioactive waste has been calculated. Continuing attention must be given to the nature and characteristics of the materials and the methods to be used to deal with tritiated wastes and to separate the different components of mixed wastes, with a view to minimising waste production.

Recommendation:

TAC recommends and encourages an informal dialogue between ITER design team and the safety contact persons of the potential host parties in order to identify common safety issues and relevant solutions prior to a licensing application. TAC expects that all technical information will be made available through the Generic Site Safety Report, in particular with respect to potential needs arising from the licensing viewpoint, Occupational Radiation Exposure, nature and tritium content of radiological waste and separation of mixed wastes, during and after operation.

5.3 Cost

TAC appreciates the contribution of Home Teams and notes that the cost estimate for the components is in reasonable agreement with the Parties' own evaluation supported by industry.

The human resources indicated for the construction phase appear reasonable.

APPENDIX

List of the TAC members present and in agreement with this report: Prof. M. Fujiwara (Chair)

Dr. R. Andreani
Dr. J. Jacquinot
Prof. P. Komarek
Dr. D. Robinson
Prof. A. Shikov

Prof. S. -I. Itoh
Dr. K. Soda
Prof. V. Kuchinski
Prof. V. Shishkin
Prof. V. Smirnov

IC-Toronto 2001
ROD Attachment 5

DRAFT

Summary ITER Final Design Report (July 2001)

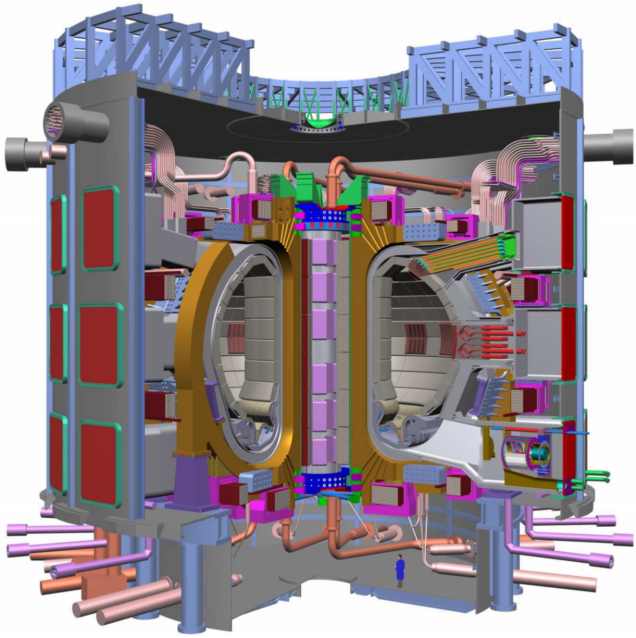


Table of Contents

1 Introduction

This document presents a summary of the technical basis for the ITER Final Design Report (presently in Draft form). This technical basis is itself supported by detailed technical documentation which is composed of "living" internal documents whose content evolves with progress in the design details.

The Final Design Report is the culmination of collaborative design and supportive technical work by the ITER Joint Central Team (JCT) and Home Teams (HTs) under the terms of the ITER EDA Agreement.

The ITER Parties, working through a special working group (SWG) constituted under the terms of the ITER EDA Agreement, reviewed and compared two possible strategies for meeting the programmatic objective of demonstrating the scientific and technological feasibility of fusion, based on:

- an ITER-like machine, capable of addressing both scientific and technological issues in an integrated fashion, and
- a number of complementary lower cost experiments each of which would specialise on scientific/technological issues.

With regard to the second strategy, the SWG[1] found that the complex non-linear interactions between α-particle heating, confinement barriers and pressure and current profile control, and their compatibility with a divertor can be addressed only in an integrated physics/technology step such as an ITER-type experiment, capable of providing long burn in conditions in which α-particles are the dominant source of plasma heating. A satisfactory understanding of these physics/plasma/technology interactions is essential to any reactor-oriented fusion development programme. Moreover the SWG expressed the unanimous opinion that the world programme is *"scientifically and technically ready to take the important ITER step."* This viewpoint was subsequently endorsed by the Parties through the ITER Council.

The SWG specified technical guidelines and objectives fitting the programmatic background of the above conclusions, which were also endorsed by the Parties. In summary the revised performance specifications for ITER, adopted by the parties in June 1998, require:

- to achieve extended burn in inductive operation with $Q \geq 10$, not precluding ignition, with an inductive burn duration between 300 and 500s, a 14MeV average neutron wall load $\geq 0.5 \mathrm{MW/m^2}$, and a fluence $\geq 0.3 \mathrm{MW/m^2}$.
- to aim at demonstrating steady state operation using non-inductive current drive with $Q \geq 5$.

In addition, the device should:

- use as far as possible technical solutions and concepts developed and qualified during the EDA, and
- cost about 50% of the direct capital cost of the 1998 ITER Design, and particular attention should be devoted to cash flow.

---

[1] SWG report to the ITER Council on Task #2 Result, ITER Meeting 10-3-1999 ROM Attachment 5

To identify designs that might meet the revised objectives, a task force involving the JCT and the HTs met during 1998 and 1999 to analyse and compare a range of options for the design of such a device. Using system codes to consistently relate the main plasma parameters, physics, engineering design constraints and costs, representative options spanning an appropriate range of aspect ratio were selected for further elaboration and consideration. This led, at the end of 1999, to a single configuration for the ITER design with parameters considered to be consistent with limitations and cost targets, yet fully meeting the objectives with sufficient margins.

In January 2000, the ITER Meeting (Tokyo) *"accepted the ITER-FEAT Outline Design Report, taking note of the TAC Report and recommendations and agreed to transmit the report to the Parties for their consideration and domestic assessment"*. The Parties assessments were overwhelmingly positive in their endorsement of the outline design, and the process of assessment by the Parties offered the opportunity to further tune the design to find the best compromise to simultaneously match the aims of each Party. The design was subsequently approved by the governing body of ITER in Moscow in June 2000, recognising it as a single mature design for ITER consistent with its revised objectives.

With the further development, summarised in this report, on the technical basis for the final design of ITER, the result is a robust design which confers a high degree of confidence that it will meet its objectives. All information needed to take a decision on construction is now available. After the completion of Explorations, the Parties negotiators should agree on a preferred site to allow specific site adaptation, and a text for the construction agreement ready to sign.

The progress towards the proposed design rests on:

1. physics understanding: the ITER Physics Basis[2] (IPB) plus new results of voluntary R&D;
2. R&D results in technology[3] development since 1992, which have provided qualified solutions by testing models after their manufacture: they have demonstrated feasibility through clearly identified manufacturing processes;
3. a consensus across Parties on safety principles and criteria for limiting consequences to the environment, and results of analysis on all possible, even hypothetical, accidents with regard to their consequences.
4. a cost target: a cost analysis has been established by industries of all Parties for manufacturing which is probably not yet fully optimised towards a reduced cost; this would be the outcome of "manufacturing R&D", needed anyway to achieve reliable production.

The key requirements to achieve Q >10 in inductive pulsed mode of operation according to the IPB can be summarised as:

1. a plasma current sufficient to provide adequate plasma energy confinement,
2. a large enough plasma density and a plasma energy confinement, good enough to achieve $Q \geq 10$,

---

[2] Nuclear Fusion 39 (1999) 2137-2664

[3] To be Published in Fusion Engineering and Design in July 2001. A short review is attached at the end of this document.

3. a reliable power exhaust and impurity control in a single-null divertor configuration, while at the same time considering the limits imposed by various instabilities on plasma design parameters such as safety factor, normalised beta, elongation, triangularity, and He ash impurity content after transfer of α-energy to the thermal plasma.

With regard to steady state operation modes, the data presently in hand does not possess the coherence across the present experiments which is required to make it the design basis for nominal performance. However, there does not appear to be any crucial conflict regarding designs based on H-mode physics to exploit whatever operational modes future progress will establish, if efficient and flexible current drive systems are available with enough power.

2 Design Overview

ITER is a long pulse tokamak with an elongated plasma and a single null poloidal divertor (Figure 2.1-1 to Figure 2.1-5 and Table 2.1-1 to Table 2.1-3). Nominal inductive operation produces a fusion power of 500 MW for a burn length of 400 s.

The major components of the tokamak are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside a toroidal vacuum vessel. The magnet system comprises toroidal field coils, a central solenoid, external poloidal field coils, and correction coils. The centring force on toroidal magnets is reacted by the central solenoid. The TF coil cases are used to support the external PF coils. The vacuum vessel is a double-walled structure supported on the toroidal field coil. The magnet system together with the vacuum vessel and internals are supported by gravity supports, one beneath each sector.

Inside the vacuum vessel, the internal, removable components, including blanket modules, divertor cassettes, port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics sensors, absorb most of the radiated heat from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The divertor exhausts the helium ash and limits the concentration of impurities in the plasma. The other vessel internals are chosen so that they do not contribute unacceptably to the concentration of impurities in the plasma. The shielding blanket design does not preclude its later replacement on the outboard side by a breeding blanket.

The heat deposited in the internal components and the vessel is rejected to the environment via the tokamak cooling water system (comprising individual heat transfer systems) which is designed to preclude releases of tritium and activated corrosion products to the environment. Some parts of these heat transfer systems are also used to bake and hence clean the plasma-facing surfaces inside the vessel by releasing impurities. The tokamak is housed in a cryostat, with thermal shields between the hot parts and the magnets, and support structures which are at cryogenic temperature.

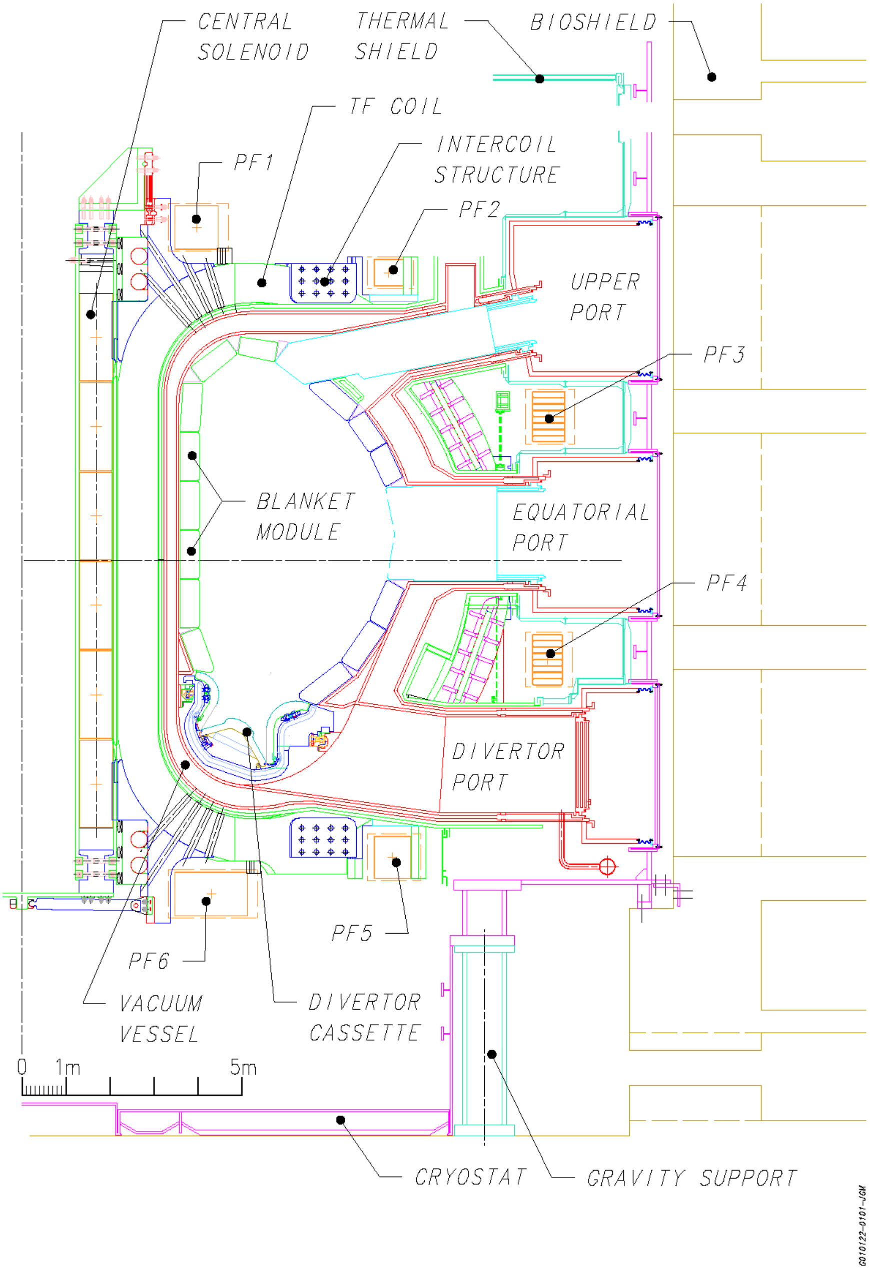


Figure 2.1-1 ITER Tokamak Cross-section

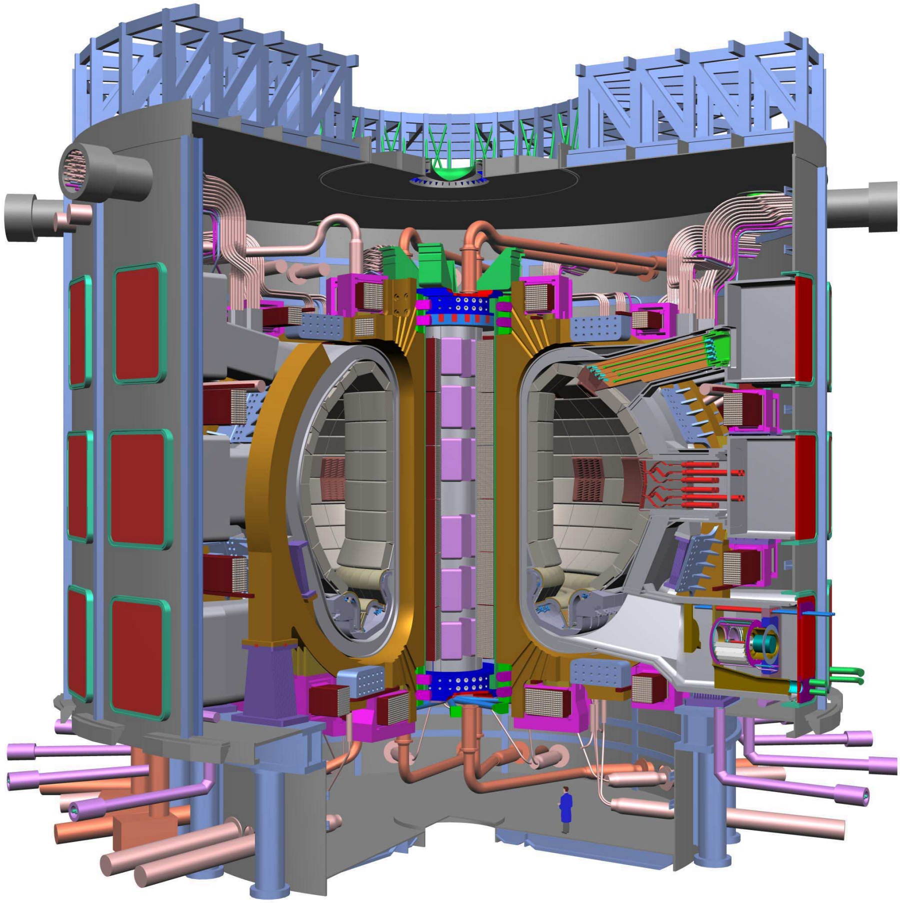


Figure 2.1-2 ITER Tokamak Cutaway

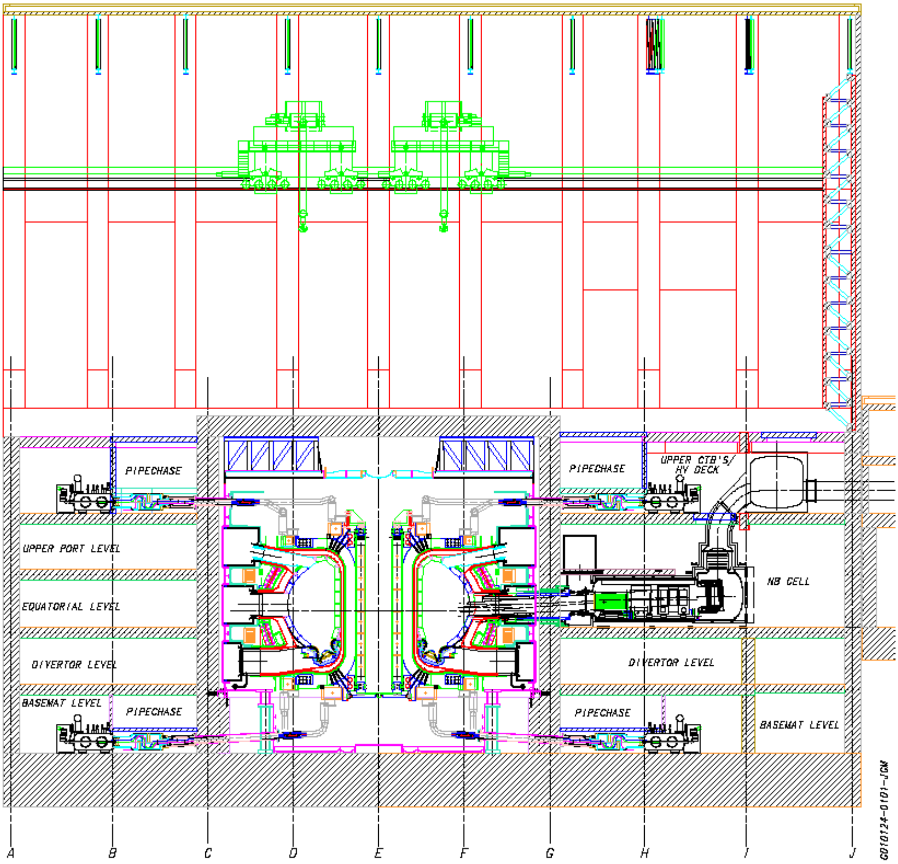


Figure 2.1-3 Cross-section NS Through the Tokamak Building

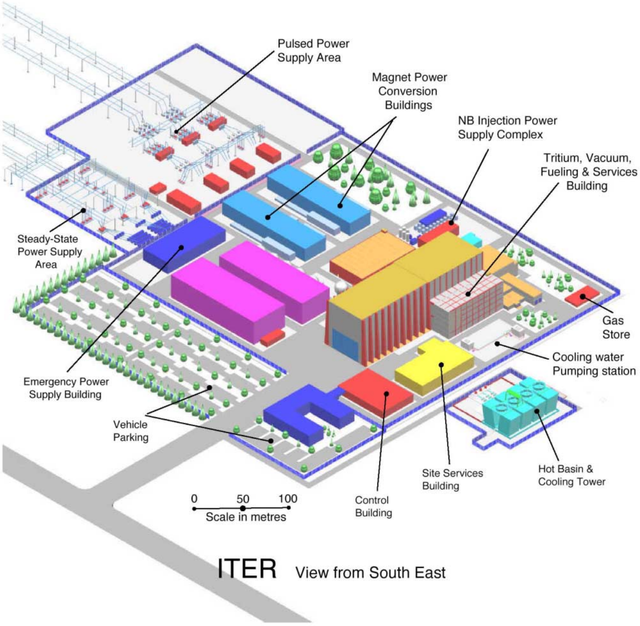
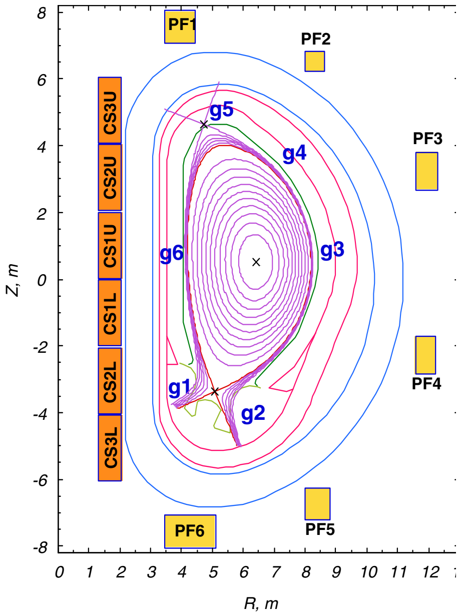


Figure 2.1-4 Generic ITER Site View

Figure 2.1-5 ITER Nominal Plasma Configuration



The tokamak fuelling system is capable of gas and solid hydrogen pellet injection. Low-density gaseous fuel will be introduced into the vacuum vessel chamber by a gas injection system. The plasma will progress from electron-cyclotron-heating-assisted initiation, in a circular configuration touching the limiter, to an elongated divertor configuration as the plasma current is ramped up. After the current flat top (nominally 15 MA for inductive operation) is reached, subsequent plasma fuelling (gas or pellet) together with additional heating for ~ 100 s leads to a high Q DT burn at 500 MW. With non-inductive current drive from the heating systems, the burn duration can be extended to ~ 3600 s, or longer. In inductive scenarios, before the inductive flux available is consumed, the burn is terminated by reducing the fuelling to rampdown the fusion power, followed by current rampdown and plasma termination. The inductively driven pulse has a nominal high energy multiplication burn duration of 400 s, and the pulse repetition period may be as short as 1800 s. Plasma control is provided by the poloidal field system, and the pumping, fuelling (D,T and impurities such as N_2, Ar) and heating systems, based on feedback from diagnostic sensors.

With regard to safety and licensing issues, the current design focuses on confinement as the overriding safety function of equipment, other functions being recognised as being required to protect confinement. A "lines-of-defence" methodology is used to obtain the required level of safety while balancing the functional requirements of systems and components. The number and quality of the lines of defence then depend on the inventory at risk.

Successive barriers are provided for tritium (and activated dust). These include the vacuum vessel, the cryostat, active air conditioning systems, with de-tritiation and filtering capability in the building. Confinement and effluents, normal as well as accidental, are filtered and detritiated, in such a way that their release to the environment is as low as reasonably achievable (ALARA).

Table 2.1-1 Main Plasma Parameters and Dimensions

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

Table 2.1-2 Main Engineering Features of ITER

Superconducting toroidal field coils (18 coils)	
Superconductor	Nb$_3$Sn in circular stainless steel (SS) jacket in grooved radial plates
Structure	Pancake wound, in welded SS case, wind, react and transfer technology
Superconducting Central Solenoid (CS)	
Superconductor	Nb$_3$Sn in square Incoloy jacket, or in circular Ti/SS jacket inside SS U-channels
Structure	Pancake wound, 3 double or 1 hexa-pancake, wind react and transfer technology
Superconducting poloidal field coils (PF 1-6)	
Superconductor	NbTi in square SS conduit
Structure	Double pancakes
Vacuum Vessel (9 sectors)	
Structure	Double-wall, welded ribbed shell, with internal shield plates and ferromagnetic inserts
Material	SS 316 LN structure, SS 304 with 2% boron shield, SS 430 inserts
First Wall/Blanket (421 modules)	(Initial DT Phase)
Structure	Single curvature faceted separate FW attached to shielding block which is fixed to vessel
Materials	Be armour, Cu-alloy heat sink, SS 316 LN str.
Divertor (54 cassettes)	
Configuration	Single null, cast or welded plates, cassettes
Materials	W alloy and C plasma facing components Copper alloy heat sink, SS 316 LN structure
Cryostat	
Structure	Ribbed cylinder with flat ends
Maximum inner dimensions	28 m diameter, 24 m height
Material	SS 304L
Tokamak Cooling Water System	
Heat released in the tokamak during nominal pulsed operation	750 MW at 3 and 4.2 MPa water pressure, ~ 120°C
Cryoplant	
Nominal average He refrig. /liquefac. rate for magnets & divertor cryopumps (4.5K)	55 kW / 0.13 kg/s
Nominal cooling capacity of the thermal shields at 80 K	660 kW
Additional Heating and Current Drive	
Total injected power	73 MW initially, 100 MW nominal maximum
Candidate systems	Electron Cyclotron, Ion Cyclotron, Lower Hybrid , Negative Ion Neutral Beam
Electrical Power Supply	
Pulsed Power supply from grid	
Total active/reactive power demand	500 MW / 400 Mvar
Steady-State Power Supply from grid	
Total active/reactive power demand	110 MW/ 78 Mvar

Table 2.1-3 Heating and Current Drive Systems

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

2.1 Operation Scenarios and Operation Phases

The design of ITER needs to be able to cope with various operation scenarios. Variants of the nominal scenario are therefore considered for extended duration plasma operation, 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 the accommodation of "advanced" plasma operation based on active control of plasma profiles by current drive or other non-inductive means.

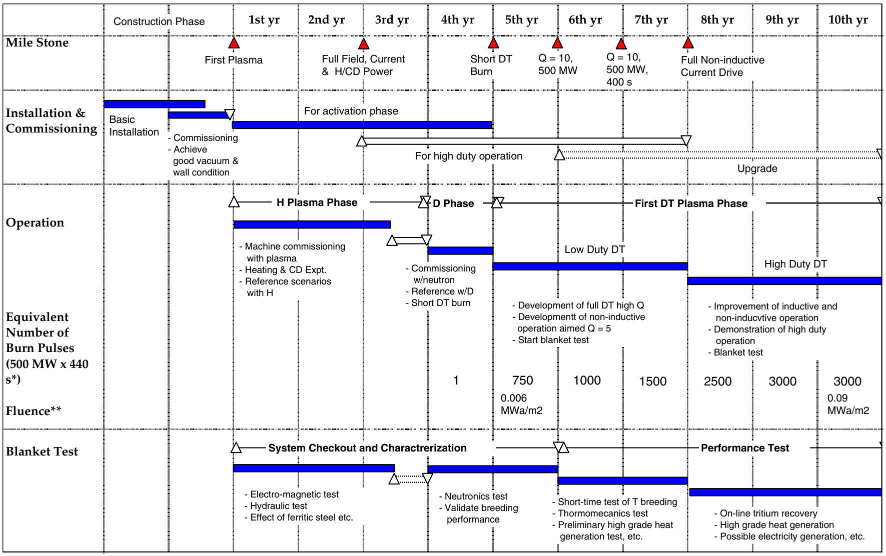
Four reference scenarios are identified for design purposes. Two alternative scenarios are specified for assessment purposes to investigate if and how plasma operations will be possible within the envelope of the machine operational capability assuming a reduction of other concurrent requirements (e.g. pulse length).

<u>Design scenarios</u>

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

<u>Assessed scenarios</u>

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



* The burn time of 440 s includes 400 s flat top and equivalent time which additional flux is counted during ramp-up and ramp-down

** Average Fluence at First Wall (Neutron wall load is 0.56 MW/m2 in average and 0.77MW/m2 at outboard midplane.)

Figure 2.1-1 Initial Operation Plan

The ITER exploitation during its lifetime is divided into successive phases.

H Phase

This is a non-nuclear phase, mainly planned for full commissioning of the tokamak system in a non-nuclear environment where remote handling is not mandatory. The discharge scenario of the full DT phase reference operation can be developed or simulated in this phase. The peak heat flux onto the divertor target will be of the same order of magnitude as for the full DT phase. Characteristics of electromagnetic loads due to disruptions or vertical displacement events, and heat loads due to runaway electrons, will be basically the same as those of the DT phase. Studies of the design-basis physics will significantly reduce the uncertainties of the full DT operation.

Some important technical issues cannot be fully tested in this phase because of smaller plasma thermal energy content and lack of neutrons and energetic alpha-particles.

The actual length of the hydrogen operation phase will depend on the merit of this phase with regard to its impact on the later full DT operation, in particular on the ability to achieve good H mode confinement with large enough plasma density.

D Phase

Characteristics of deuterium plasma behaviour are very similar to those of DT plasma except for the amount of alpha heating. Therefore, the reference DT operational scenarios, i.e., high *Q,* inductive operation and non-inductive steady state operation, can be simulated. Since tritium already exists in the plasma, fusion power production at a significant power level for a short period of time without fully implementing cooling and tritium-recycle systems, which would be required in the subsequent full DT phase, could therefore also be demonstrated. By using limited amounts of tritium in a deuterium plasma, the integrated commissioning of the device is possible. In particular, the shielding performance can be checked.

DT Phases

During the first phase of DT operation the fusion power and burn pulse length will be gradually increased until the inductive operational goal is reached. Non-inductive, steady-state operation will also be developed. DEMO-relevant test blanket modules will also be tested whenever significant neutron fluxes are available, and a reference mode of operation for that testing will be established.

The second phase of full DT operation, beginning after a total of about ten years of previous operation, will emphasise improvement of the overall performance and the testing of components and materials with higher neutron fluences. This phase should address the issues of higher availability of operation and further improved modes of plasma operation. Implementation, and the programme, for this phase should be decided following a review of the results from the preceding three operational phases, and assessment of the merits and priorities of programmatic proposals.

A decision on incorporating tritium breeding during the course of the second DT phase will be decided on the basis of the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance. Such a decision will depend on the R&D completed during the first phase.

3 Plasma Performance

According to the conclusions of the ITER Physics Basis, from broadly based experimental and modelling activities within the fusion programmes of the ITER Parties, the regime retained for nominal inductive operation of ITER is the ELMy-H-mode confinement regime.

In this regime, plasma turbulent heat conduction across the magnetic surfaces drops dramatically in a thin transport barrier layer just inside the magnetic separatrix. This layer is commonly observed to undergo successive relaxations called edge localized modes (ELMs). The interest in ELMy H-modes follows from experimental observations that show that this mode reduces transport throughout the plasma core. The standard working hypothesis, supported by many observations, is that H-mode occurs when the power transported across the separatrix (P_{loss}) exceeds a threshold value (P_{L-H}).

From the statistical analysis of confinement results obtained in all previous devices, an expression of the energy confinement time has been established as a function of plasma parameters, verified through three orders of magnitude, and expressed as

$$\tau_{E,th}^{IPB98(y,2)} = 0.0562 H_H I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_x^{0.78} \qquad \text{(rms err. 0.145)}$$

where the units are s, MA, T, 10^{19}m^{-3}, MW, m and amu, and where H_H is a scalar which can be used to represent either how close the actual value observed in one experiment is from the average, or a level of inaccuracy. This expression will only be valid in H-mode, that is when $P_{loss} > P_{L-H}$, where

$$P_{\text{L-H}} = 2.84 M^{-1} B_{\text{T}}^{0.82} \bar{n}_{\text{e}}^{0.58} R^{1.00} a^{0.81} \qquad \text{(rms err. 0.268)}$$

where the units are MW, amu, T, 10^{20}m^{-3}, m.

3.1 ITER Plasma Current and Size

Assuming $P_{loss} > P_{L-H}$, and using the previous expressions for τ_E and P_{L-H}, one can derive the relationship between the plasma parameters and the capability to achieve a given value of Q = P_{fusion}/P_{aux}, which can be formulated approximately as

$$\left[\frac{H_H I_{MA} \frac{R}{a}}{X} \right]^3 = \frac{Q}{Q+5}$$

with X ~ 50-60, a slowly varying function of parameters. This relation provides an easy basis for I_{MA} = 15MA, R/a = 3.1 if Q = 10, X = 55 and H_H = 1.

Expressing now I_{MA}.R/a = 5.B_T.a/q.f, where f is a function of aspect ratio, increasing with triangularity, δ, and mostly with plasma elongation κ, it is obviously important to increase the value of f, decrease the value of q, and compromise between B_T and size.

However, there are limiting values for f and q: too large an elongation provides a condition where the vertical stability of plasma position cannot be assured practically, and q below 3 is limited by the occurrence in a large volume near the plasma axis of “sawtooth relaxation” (an instability which periodically destroys the confinement in this volume) and there is an increasing susceptibility to instability as q=2 is approached.

A large aspect ratio allows a larger value for B_T on the plasma axis for a given maximum B on the TF coil conductor (B~l/R), an important constraint for superconductors, but access to the plasma for heating and maintenance becomes limited by smaller openings, and plasma shape control becomes more difficult. A compromise is therefore needed.

3.2 Plasma Confinement Extrapolation

Experiments have shown that once an ideally stable equilibrium is assured by externally applied shaping fields, the plasma response to auxiliary heating and fuelling is governed by the spontaneous appearance of a fine scale turbulence.

Profiles of plasma parameters (Figure 3.2-1) are the consequence of transport properties which are governed by turbulence, the characteristic scale of which is much smaller than the device size. The physical processes prevailing depend on dimensionless variables, built from density, temperature and magnetic field values, mainly ρ^* = ion gyration radius/minor plasma radius, β = plasma pressure/magnetic pressure and ν = collisionality.

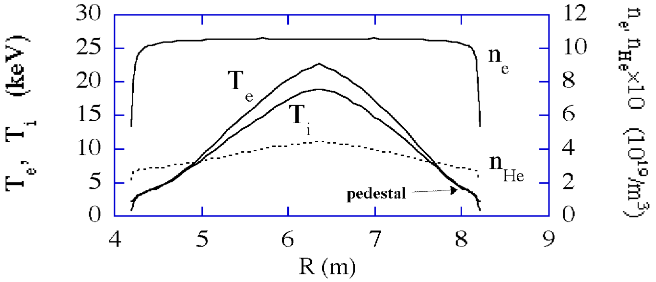


Figure 3.2-1 Profiles of Electron Temperature (T_e), Ion Temperature (T_i), Electron Density (n_e), Helium Density (n_{He})

In this respect, experiments having identical non-dimensional parameters, but differing magnetic field, density and temperature, have been shown to have the same non-dimensional energy confinement time defined by $\Omega_{ci}.\tau_E$, (where Ω_{ci} is the plasma ion cyclotron frequency). Therefore, present experiments have been used to simulate ITER discharges, which reduces the problem of extrapolation to that of a single parameter ρ^*.

Accordingly, codes have been built to model plasma evolution, in particular ITER. These codes take into account the magnetic configuration in detail, assuming constant density and temperature on a magnetic surface, adjusting the thermal diffusivity in such a manner that the global energy confinement time computed by the code is constrained to be equal to the global scaling relation, and adapting its spatial profile to provide temperature profiles close to those observed in ITER demonstration discharges.

These results, confirmed by all the previous experiments of very different size, provide confidence in the performance of confinement ITER will achieve.

3.3 H Pedestal and ELMs

The transport barrier which occurs just inside the magnetic separatrix in H-mode provides a thin layer, where the pressure increases sharply (a large radial gradient is established) and at its inner edge a pedestal is formed where density and temperature values serve as boundary conditions for the core profiles. Pedestal temperatures can be very important if the core temperature gradient is constrained (a fact not always observed in present experiments) to lie near a marginal upper value. Even if this is not the case, the energy content of the pedestal is generally not negligible compared to the core energy content (in fact about one third). Moreover it has a scaling that differs from the core global scaling, and is not definitely agreed yet.

ELMs appear as a pseudo-periodic relaxation of the pressure gradient, due to an instability which depends on the detailed shape of the magnetic surface near the separatrix (under the global influence of the separatrix curvature variation, triangularity and magnetic shear). As the ELMs' periodicity becomes smaller, their amplitude increases and the energy removed from the pedestal by each ELM becomes large and, as it is deposited onto the divertor targets, leads to their strong erosion. The physical phenomena involved are not understood in quantitative terms at present.

3.4 Internal Confinement Barrier

In some conditions (not completely understood or controlled until now), another confinement barrier might occur inside the core, and limit even more the turbulent heat conduction across the plasma. This barrier again requires for its existence a threshold in the power crossing it. It provides a steep pressure gradient and occurs usually in a region where the magnetic shear is very weak (as at a minimum of q). This internal barrier, if its existence and stability can be controlled on a long time scale, will lead to better confinement performance, which can be the more interesting the larger can be its radius. In addition, because a toroidal current is driven by the pressure gradient (the "boot-strap" current), this internal barrier is considered an important feature for possible steady-state tokamak operation, where the toroidal current, driven by non-inductive drive methods from auxiliary power, has to be minimised.

3.5 β limits; Non-axisymmetric Perturbations; Islands

Because fusion power production scales as β^2B^4, there is an incentive to operate at the highest value of β allowed by plasma stability. For simple, monotonic q profiles, characteristic of inductive operation, the MHD stability limits $\beta_N = \beta/(I/aB)$ to values $< 4\, l_I \simeq 3.5$.

However, numerous experiments have shown the appearance of modes, no longer rigorously axisymmetric, which change the topology of the magnetic field in the vicinity of low order rational magnetic surfaces (q = 1.5, 2). These modes lead to islands which grow from a small seed width to a much larger width when β_N reaches values much lower than 3.5. The observed limit in present experiments is around 2.5 but it may decrease with ρ^* by a factor ~ 3. Nevertheless, the stabilisation of these "neoclassical tearing modes" (NTMs) has been achieved by a localised plasma current addition, driven from electron cyclotron waves on the specific magnetic surfaces, a method to be applied in ITER.

The existence of small-amplitude non-axisymmetric error fields produced by residual asymmetries in the magnetic coil positions, can lead to the development of large magnetic islands, again on low order rational magnetic surfaces, and subsequently to disruptions. Therefore, these error fields should be eliminated by appropriate correction coil currents which produce a controlled small amplitude helical field.

Disruptions are abrupt uncontrolled events, involving a rapid cooling of the plasma. Growing, large amplitude, islands overlap and lead to complete ergodisation of the magnetic field lines, thus a large heat flow occurs along field lines to the boundary walls, cooling the plasma and leading to an influx of impurities through interaction with the walls. This is followed by a rapid decrease of the plasma current; its decay time depends on the amount of losses through impurity radiation, the faster decay corresponding to the lower plasma temperature. Simultaneously, electrons can be accelerated to large energies by the electric field associated with the decrease in current in these low temperature plasmas, and lead to large runaway currents, if the confinement of energetic electrons is not limited by the magnetic fluctuations which may remain from the previous field ergodisation phase.

3.6 Divertor and Power Exhaust

The magnetic field configuration in Figure 2.1-5 shows closed nested magnetic surfaces with increasing inside volumes from the plasma axis until a separatrix occurs, outside of which magnetic surfaces are open. The particles diffusing out of the plasma through the separatrix flow along the field lines until they hit a target, localising the plasma contact with the wall to a large distance from the plasma along field lines (a few times the torus major circumference).

Along these field lines, the power flow is very high and if it were not for the possibility of volume power losses, the power density on the target (even taking into account its inclined position and the flux expansion due to a smaller B_p) would be too large for the capability of heat removal and the surface material temperature. This power should remains below 10 MW/m^2 on average. With no power losses, the temperature gradient along the field lines remains small, the pressure constant, and the plasma temperature at the target very high: this is the so-called “attached plasma” divertor operation, not acceptable for ITER.

On the contrary, if the plasma density is high enough at the separatrix, the possibility of radiation losses from impurities, and from ionisation of a large neutral density built in front of the target, provides a new situation, the so-called “detached plasma”. Towards the divertor target, the pressure along field lines decreases, the plasma density increases significantly and the plasma temperature at the target becomes very low (a few eV): the power crossing the separatrix becomes distributed by radiation (and charge exchange neutrals) onto the much larger surface of the divertor side walls, and the power density to the divertor target can remain inside reasonable limits.

In the latter conditions, the impurities, which are removed from the target by erosion and ionised by the plasma, contribute to the radiation losses, and thus to the decrease of the plasma temperature. Because this erosion increases with the particle energy impinging on the target, the process in itself may be self-regulating, as modelled in the case of a carbon target material. Moreover, these impurities are mostly stopped from flowing upwards along the field lines and entering the plasma by the hydrogen flow towards the target. This is the main function of the divertor, to screen the plasma from impurities due to plasma-wall interaction.

Another function of the divertor is the control of plasma density, and in particular the removal of the helium reaction product, the density of which should remain small (a few % of the electron density) in order not to dilute the reacting ions D and T, for a given electron density.

These helium particles are born at 3.5 MeV. They should become thermalised in the plasma at some keV, and provide, mostly by interaction with the plasma electrons, the heat source needed (in addition to the auxiliary heating power) to keep the plasma temperature constant by compensating for its power losses. This helium ash should not be lost to the boundary at high energy, through the action of specific instabilities or because of a large ion gyration radius and too large a magnetic field ripple (along field lines) due to the discreteness of the TF coils. This last source is minimised in ITER by ferromagnetic inserts, installed in the shadow of each TF coil, while the first one should not be detrimental in ITER according to the present understanding (with the probable values of He pressure and its gradient).

Due to the flat density profile and the small scale turbulence present in the core, the He ions when created in the plasma volume are driven to the boundary, then flow into the divertor, where the high neutral particle density allows an easier pumping at high pressure ($\simeq$ 1 Pa).

Together with the He, a large flow of D and T are pumped, and the plasma density is the result of this outwards flow against an inward flow coming from gas fuelling near the separatrix and/or solid (D or T) pellet periodic injection (a few tens of mm^3 at a few Hz). These pellets should be able to reach the plasma core, well inside the H-barrier: it will be the source of particles, which will allow a density gradient to be built; where there is no source, the density is flat.

It is generally observed that the plasma density is experimentally limited on average across the plasma width by the so-called Greenwald limit ($n_{GW} = I_{MA}/\pi a^2 \times 10^{20}/m^3$). The fusion power being quadratic with the density, it is important, if possible, to provide a peaked density profile, for a given average.

4 Functional Role of Systems

The preceding tokamak physics issues are linked with the hardware systems necessary to be installed in ITER, and with their functional requirements and implementation.

4.1 Magnets

The plasma is confined and shaped by a combination of magnetic fields from three origins: toroidal field coils, poloidal field coils and plasma currents. Aiming in ITER at steady-state operation, all the coils are superconducting; copper coils would require too large an electric power to be acceptable for ITER as well as for a future reactor.

4.1.1 Toroidal Field Coils

The nested magnetic surfaces are able to confine a plasma pressure equivalent to a few atmospheres, with a density 10^6 times smaller that in the atmosphere ($n = 10^{20}/m^3$, $T \approx 10$ keV). For an average β of 2-3%, the toroidal magnetic field value on the plasma axis is 5.3T, which leads to a maximum field on the conductor ≤ 12 T. Because of this high field value, Nb_3Sn is used as superconducting material, cooled at 4.5K by a flow of supercritical helium at ~ 0.6 MPa. The total magnetic energy in the toroidal field is around 40 GJ, and leads to very large forces on each coil which are restrained by a thick steel case to resist circumferential tension (≈ 100 MN) and by constructing a vault with the inboard legs of all 18 coils (the large centripetal forces are due to the 1/R variation of the toroidal field). The compressive stress levels inside this vault are large, and therefore the side surfaces of each coil should match one another as perfectly as possible.

The coils are linked together (Figure 4.1-1) around their lengths (except to provide access to the vacuum vessel through ports) by specific bolted structures, and by two compression rings made of unidirectional fibre glass, which provide an initial radial force on each coil (2 x 30 MN) in the absence of the current in the coils.

This very robust assembly is provided to resist the toroidal forces induced by interaction of the TF coil current with the transverse poloidal fields from plasma and poloidal field coils. These forces produce a distribution of torque around the axis of revolution proportional to the magnetic flux crossing unit length of the TF coil (the net torque is thus 0). These local forces are pulsed, and therefore fatigue is a concern for the highly stressed structural steel of the coils. These forces, due to the highly shaped plasma, are largest across the inboard coil legs, mostly at their curved parts, where they are resisted by the friction between coil sides (under high compression) and by specific keys.

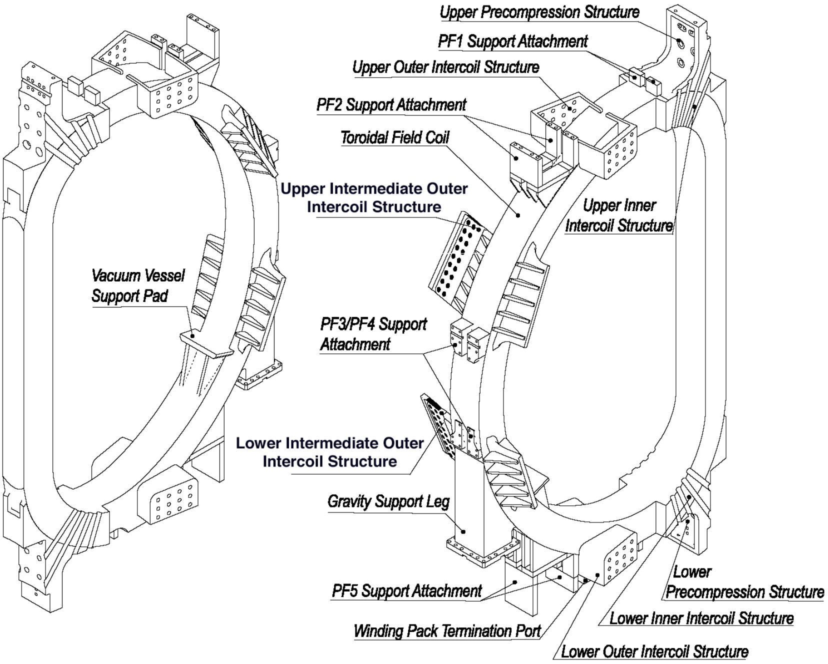


Figure 4.1-1 TF Coil Structure

4.1.2 Poloidal Field Coils

The plasma shape is controlled by the currents distributed inside the six modules of the central solenoid (CS) and the six large coils placed outside the TF coils. All these axissymetric coils use superconductors cooled by a flow of supercritical helium at 4.5K and 0.6 MPa.. Nb_3Sn is used in the CS modules, NbTi in the PF coils, where the maximum field value is lower than 6T.

The magnetic configuration provided by these currents is such that the plasma toroidal current will experience a vertical force as soon as its centre is displaced vertically, and this force will increase with the displacement: the plasma with its elongated shape is in a vertically unstable equilibrium.

Stabilisation of the plasma vertical position can be achieved in the following way. First, any plasma movement, associated with small changes of its energy content, induces eddy currents in any axisymmetric conducting surface surrounding the plasma, i.e. the two-shell vacuum vessel, which reacts to slow down the displacement. These conducting surfaces are shaped in order that the current distribution can provide a neutral equilibrium position, near the plasma centre of gravity, for most expected plasma energy changes, so as to minimise the sources of instabilities.

Second, using a feedback position control system, the currents in the 4 large poloidal field coils will be changed by feeding them with a fast power supply , in an antisymmetric way, across the plasma equatorial plane. These changes provide an additional radial magnetic field and lead to a restoring force on the plasma towards its controlled position.

Moreover, the plasma shape can be similarly feedback controlled, by an appropriate action on each coil voltage by its own distinct power supply. The gaps (Figure 2.1-5) between the plasma boundary and the walls are measured at six critical positions, and brought back to a prescribed value after an excursion due to a plasma internal change (loss of or change in current distribution/internal inductance, l_I, loss of plasma thermal energy).

In the inductive scenario, the plasma current is generated by the magnetic flux change inside the toroidal plasma annulus. This flux swing is largely realised by the CS coil, which will see a complete inversion of field from +13.5 T to - 12 T in the central modules. The flux consumption during plasma initiation and current increase will be reduced by providing a few MW of plasma heating through electromagnetic waves, in order to secure a sufficient flux variation available to sustain the current flat top during at least 400s at a plasma current of 15 MA.

After the plasma current is set up inductively, a non-inductive scenario can follow, in which the plasma current flat top is extended towards steady state by driving the current externally. This will be achieved by high energy (l MeV) beams of neutral D^0 injected at a small angle to field lines, which become ionised along their path across the magnetic field. This can also be achieved by toroidally propagating electromagnetic waves (at ion and electron cyclotron frequencies, or at the lower hybrid frequency), in addition to the “bootstrap” current linked to the plasma radial pressure gradients. The preceding sources of different radial current distributions deposit large amounts of power at specific locations in the plasma, and this has to be done in a way compatible with the necessary plasma pressure profiles and their allowable rates of change. A complete scenario for steady state operation in ITER with Q=5 is yet to be consistently developed. Nevertheless, the non-inductive current drive systems provided in ITER should be able to accommodate the steady state operational requirements (for over 2000 s).

4.1.3 Error Field Correction Coils

As mentioned previously, the need to correct imperfections in the magnetic field symmetry, due to the imperfect positioning of the TF, CS and PF coil currents, requires the use of “correction coils”, able to provide a helical field of a few 10^{-5} times the TF value. The Fourier components of toroidal and poloidal modes are $n = 1$ in the toroidal direction, and a distribution between $m = 1$, 2 and 3 in the poloidal direction. These coils are composed of 3 sets of six coils, around the torus, above and below the TF coils. The same coils can be used to stabilise possible resistive-wall modes, which happen to have the same geometry as the error fields to be corrected, but a much faster time variation. These coils counteract the MHD instabilities which are not stabilised by the conductive walls, on the longer time scale associated with the wall resistance.

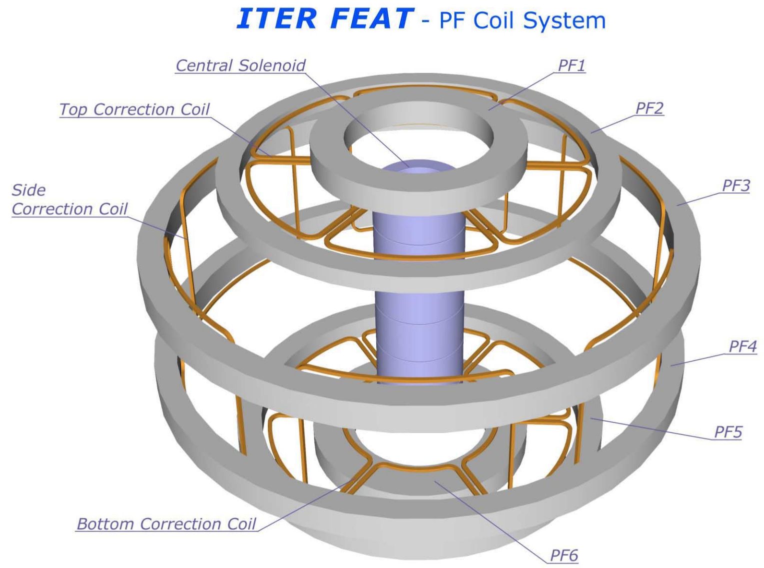


Figure 4.1-2 ITER Error Field Correction Coils

4.1.4 Superconducting Coil Protection

The superconductor of all coils should be protected against local overheating, should the coil current continue to flow after a local transition from superconducting to normal conducting state due to an off-normal local energy dump. In this case, after identification of a resistive voltage across the coil terminals increasing with time, an external resistor is switched in, dumping rapidly a large part of the coil magnetic energy. The time constant of this fast emergency discharge should be small, in order to minimise the energy dissipated into the coil and to limit its local temperature increase, but there is a minimum value for this time constant due to the maximum voltage through the coil terminals and the induced current (and related forces) in conducting material magnetically coupled with the coil. One example of this limit comes from the forces applied to the vacuum vessel due to the large poloidal current induced in the vessel shells by the fast discharge of all TF coils.

In addition, all these coils should be protected against the heat coming from their surroundings. Therefore, a large cryostat puts all the coils in a vacuum good enough to limit convection, and a thermal shield, cooled at about 80 K by a flow of helium, is provided between the coils and hot parts to shield against their radiation. The geometry of this thermal shield is obviously complex, but the avoidance of radiation hot spots is compulsory to limit the already large amount of power to be removed from the coils at 4.5K. This permanent heat load (13.5 kW) due to radiation, and conduction through supports, adds to the non-ideal efficiency of the circulation pumps feeding the supercritical helium in each coil.

4.1.5 Superconducting Coil Cryogenic Cooling

In addition, there is a large pulsed heat load on the coils from two origins: the neutron flux produced by the fusion reaction and attenuated by the blanket and vessel shields, and eddy currents induced by any field change in the coil superconductor and steel cases during the operational scenario of the plasma pulse, and even more during a disruption ending the pulse.

However, the cryogenic plant, producing the cold helium through expansion of a high pressure flow in turbines with brakes, is essentially a steady state system. Therefore, between the coils and the cryogenic plant, an energy storage should be present to buffer the large pulsed loads, and to control the transfer time of this energy to the cryogenic plant.

The energy storage is mainly provided by the large steel mass of the TF coil cases, and by the temperature variation of the liquid helium bath which cools the supercritical helium flow through heat exchangers. The extra energy dumped into the coils at 4.5 K during a nominal pulse amounts to 19 MJ; and a plasma disruption can add a further 14 MJ. Due to the assumed duty cycle, the time average load on the cryogenic plant (all users) amounts to 41 kW.

4.2 Vessel and In-vessel Systems

4.2.1 Neutron Shielding

The 14 MeV neutrons, i.e. 80% of the fusion power produced, escape from the plasma. This power should be transferred to a water coolant, and subsequently to the environment, by their collisions with the materials present around the plasma (mostly steel and water) in the blanket modules and in the vacuum vessel. The neutron power, not absorbed in these two shields, will be dumped in the TF coil structure at very low temperature, and should be absolutely minimised.

In addition to inelastic collisions, the neutrons will be absorbed by some nuclei, which will become activated and later radiate energetic γ rays according to their specific properties. Neutrons, not absorbed in the radial thickness of the blanket, VV and magnets, or leaking through gaps, will be absorbed outside and induce activation in the cryostat, a process which should be limited as far as possible, to allow human access, in case of an unexpected need for repair. Therefore, the shielding thickness (and attenuation efficiency by optimising the volume ratio between steel and water) has been carefully chosen, and its variation along the poloidal length optimised, to match the above two goals.

The thickness distribution between blanket and vessel comes from the need to allow rewelding of the vessel inner shell until the ITER end of life. This requires a low enough (around 1 appm) helium content (due to n, α reactions) in the vessel steel material. Accordingly, the blanket thickness is set at 45 cm, and gaps maintained as small as practicable.

4.2.2 Blanket Modules

The blanket is divided into two parts: a front part separable from a back one (Figure 4.2-1). The back part of around 30 cm thickness is a pure shield made of steel and water. The front part, the “first wall”, includes different materials: 1 cm thick Be armour, 1 cm Cu to diffuse the heat load as much as possible, and around 10 cm of steel. This will become the most activated and tritium-contaminated component of ITER. It could be in contact with the plasma in off-normal conditions, and thus can suffer damage from the large heat locally deposited, and may have to be repaired or changed.

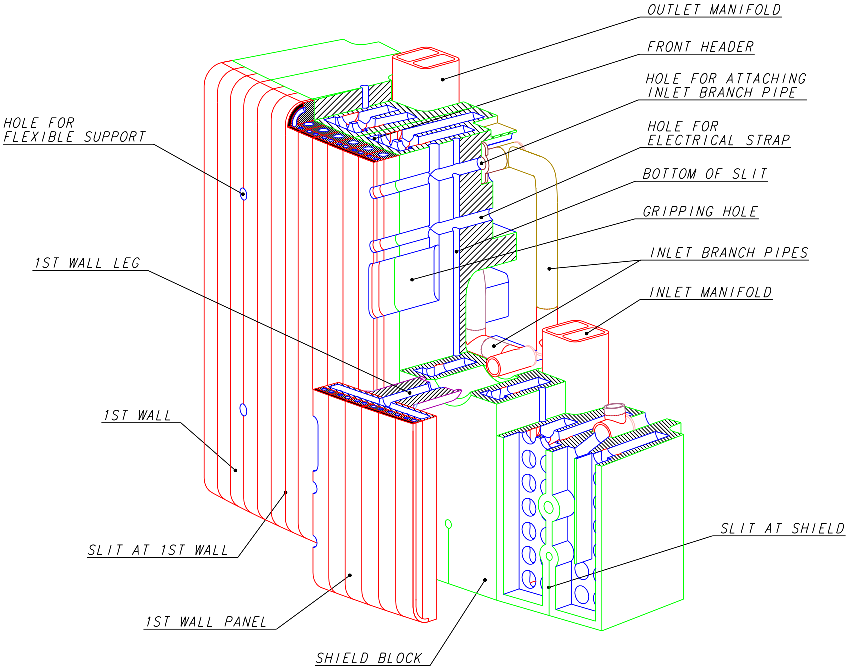
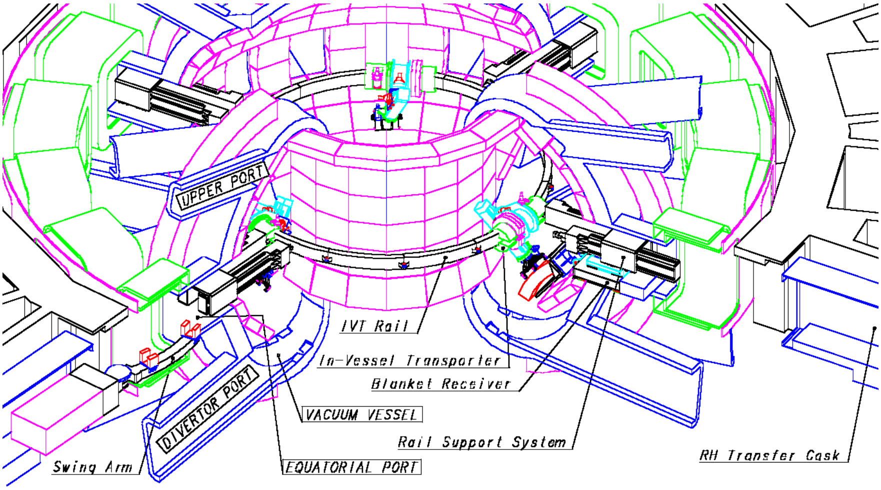


Figure 4.2-1 Blanket Module

4.2.3 Blanket maintenance

In order to allow a practical method of maintenance, this blanket has to be built in modules (~ 420 in total) with a maximum weight of 4.5 t (each with about 1.5 m^2 facing the plasma). Each module is attached to the vessel by 4 flexible links stiff radially but pliant against toroidal or poloidal motions. This flexibility is required because, across the blanket thickness, the absorbed power density decreases sharply and, whereas the water cooling redistributes the heat progressively toward a uniform temperature, at the end of the pulse the front part becomes necessarily the colder part. Thus the blanket module suffers an alternating bowing effect during each plasma pulse. The toroidal and poloidal loads (i.e. during a disruption) acting on the module are therefore reacted by additional key elements provided with sufficient compliance gaps.

The maintenance of a blanket module is done by first removing it from the vessel. A vehicle, equipped with an end effector, moves along a toroidal rail which is deployed toroidally along the vessel centreline, from a specific cask attached to an equatorial port door of the vessel (see Figure 4.2-2). The end effector is able to cut the connection to the water pipe feeders, to unbolt the module, and to bring it to an equatorial maintenance door. There it will be transferred into a cask, and from there, to the hot cell for repair or replacement. The cask operates by docking and undocking automatically to the ports of the vessel and of the hot cell, avoiding any atmosphere contamination to the environment. The same procedure is used for removal of any equipment installed in any equatorial or upper port of the vessel, i.e. heating launcher, diagnostics, or tritium breeding test blanket.



From Drawing 23.182.1

Figure 4.2-2 Blanket Maintenance Concept

4.2.4 Divertor

The same modularity philosophy and maintenance procedure are used for the divertor. The cassettes (54 in total) are removed from the vessel at three lower ports, to which they are brought by a toroidal mover on specific rails, to which the cassettes are finally attached.

Besides providing shielding of the vessel, the modular cassettes (Figure 4.2-3) support the divertor targets, and very high heat flux components, built with high conductivity armour of carbon fibres and tungsten.

These armour materials can be eroded by the plasma particles, mostly during short pulses of high heat loads, associated with ELMs or plasma disruptions. This erosion process not only will call for replacement from time to time of the worn out divertor targets, but also will create dust, and in particular tritiated carbon dust. Studies are going on to define the best way for removal of this dust, mostly to limit the tritium inventory inside the vessel, and the possibility of metallic dust (Be, W) reaction with hot water during an accidental in-vessel water leak, which could lead to hydrogen formation.

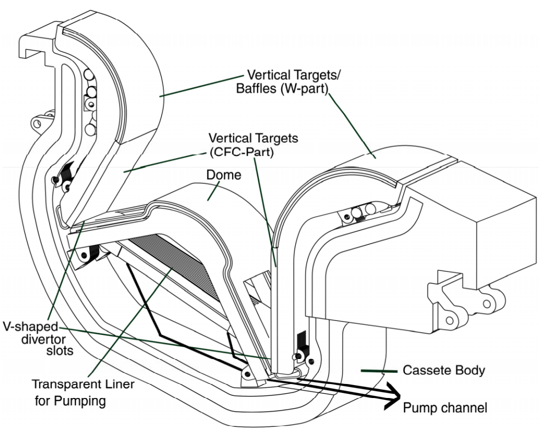


**Figure 4.2-3
Divertor Cassette**

4.2.5 In-vessel Component Water Cooling

Each divertor cassette is separately cooled by water, with feeder pipes connecting to the manifold outside the vessel and cryostat. Two, sometimes three blanket modules are similarly fed by separate pipes installed on the plasma side of the inner shell of the vacuum vessel. This arrangement leads to handling a large number of small size pipes, but allows the identification of possible water leaking modules or cassettes from tests done from outside the cryostat, a crucial procedure to be able to localise the leaks in vacuum.

The pressurised coolant water input is maintained around 100°C permanently; the output temperature during a pulse at nominal fusion power will be around 150°C. At the end of a pulse, control valves allow to short circuit the large heat exchanger to the heat rejection system, in order not to cool the in-vessel components below 100°C. During standby, the coolant flow will be reduced, using a different pump, to 10% of the flow during the pulse, and the flow in the heat rejection system reduced to 25% of its normal value. From time to time, after a maintenance period, the pressurisation will be increased, and the water coolant used to heat and bake the in-vessel components to 240°C.

4.2.6 Cryogenic Pumps

At the divertor port level, cryogenic pumps operating at 4.5 K are installed, which have the capacity to pump hydrogenic atoms as well as helium by adsorption and condensation. These pumps are equipped with a closing valve. The pumping performance can be varied and the condensed gases removed by heating the pumping panels to 80 K and pumping them with a roughing pump when the valve is closed. For long plasma pulses, this procedure will be carried out in sequence through all the installed cryogenic pumps, one after another, in order to limit the amount of hydrogen in each pump below the deflagration level in case of an accidental ingress of oxygen, which corresponds to pumping 200 Pam^3s^{-1} of DT for 450 s.

4.2.7 Vacuum Vessel

The vacuum vessel has a multiple role, namely:

- to provide a boundary consistent with the generation and maintenance of a high quality vacuum, necessary for limiting impurity influx into the plasma;

- to support the in-vessel components and their resultant mechanical loads;
- to participate in shielding against neutrons, and to remove the corresponding power during a pulse, and moreover to remove the after-pulse decay heat of all in-vessel components in case of there being no other coolant available;
- to provide a continuous conductive shell, fitting tightly to the plasma, for plasma stabilisation;
- to provide all access to the plasma through ports, for diagnostics, heating systems, pumping, water piping, etc…;
- to provide the first confinement barrier for tritium and activated dust with a very high reliability.

All these functions are essential, and they require a very robust vessel mechanical design analysed for stresses in all possible normal and accidental conditions. The vessel is built with two shells linked by ribs and fitted with shielding material.

To ensure cooling reliability, two independent water loops are used which can remove by natural convection the decay heat from all in-vessel components (if they are not cooled directly). The vessel water temperature is maintained at 100°C (at 200°C during baking of the in-vessel components), limiting to 50°C its difference with the in-vessel component cooling temperature.

4.2.8 Vacuum Vessel Pressure Suppression

In the case of a water pipe rupture inside the vessel, the pressure will increase, but be limited below 0.2 MPa by the opening of rupture disks and communication with a large enclosed volume located above the vessel and half-filled with water, in which the steam will be condensed (the vacuum vessel pressure suppression system - VVPSS). Simultaneously, liquid water condensed in the vessel will be driven into drain tanks located at the bottom of the tokamak building.

4.3 Mechanical Loads and Machine Supports/Attachments

Part of the technical difficulties in the ITER design are due to the large mechanical loads which are applied to the various components.

The mechanical loads acting on ITER fall into four categories.

1 Inertial loads which are due to gravitational and seismic accelerations,
2 Kinetic pressure loads due to coolant pressure and atmospheric pressure,
3 Thermal loads,
4 Electromagnetic loads, which are usually a strong design driver, either static (as in TF coils) or dynamic, acting on the magnet and on all conducting structures nearby due to fast or slow transient phenomena such as plasma disruptions and VDEs.

The support scheme of tokamak components must be designed to minimise the reaction of each support to loads on the component. In interconnecting components, a proper load path must be chosen to maximise the stiffness associated with the load path itself. The support hierarchy is schematically drawn in Figure 4.3-1. All core components of the machine are attached to the TF coil cases.

The support schemes for the magnet and the vacuum vessel must allow for their changes in temperature from the time of assembly, i.e. the radial shrinkage of the magnet and radial growth of the vessel, and provide adequate resistance to seismic and disruption forces. All supports of the machine core are flexible in the radial direction and stiff in all others.

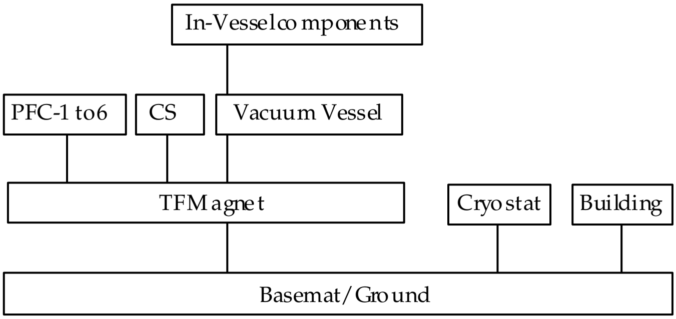


Figure 4.3-1 Schematic of Supports Hierarchy

4.3.1 Seismic Loads

Earthquakes simultaneously produce vertical and horizontal random ground motions which are statistically independent; the horizontal ones having the most important consequences. Even if the ground peak horizontal acceleration is a small fraction of gravity, a seismic event is in fact, in many cases, the most demanding loading condition, in particular for the interface structures (e.g. supports). Under horizontal excitations with a relatively broadband spectral content in the range 1-10 Hz, resonances occur in component motions. The tokamak global structure exhibits then oscillatory modes which involve horizontal shearing as well as rocking motions

In addition, the relative distance between components must be maintained (between vessel, TF coils, thermal shield etc…). Normalised seismic conditions of 0.2 g ground acceleration (at high frequency, 33 Hz) have been applied to ITER and found acceptable. In case of larger values, the use of horizontal seismic isolators below the building basemat has been shown to be effective at lowering the peak acceleration to acceptable values, at the expense of larger displacements.

4.3.2 Electromagnetic loads

The restraint of the loads occurring on the TF coils, either static in-plane from the toroidal field itself, or out-of-plane cyclic due to their interaction with the poloidal field crossing them, and the consequence of a fast emergency TF energy discharge on the VV stress level, which provides a limit to operating conditions, have been described above.

Other important electromagnetic loads are associated with transient phenomena which are consequences of changes in plasma current, internal energy or position. They act on the poloidal field coils and all conductive structures close to the plasma (blanket modules, divertor cassettes, VV).

For slow transients (time scales longer than those associated with these structures), there are negligible induced currents, and thus no net force on the PF/CS magnet assembly as a whole. In each PF and CS coil, vertical forces are reacted through the TF coil structure (the shortest path) and radial ones by the development of a toroidal hoop stress inside each coil.

For fast transients, such as plasma disruption or loss of vertical position control (vertical displacement event, VDE), significant currents are induced in conducting structures, and their interaction with the toroidal or poloidal magnetic field develops significant forces and stresses.

In the case of disruption, the load severity is the larger the shorter is the assumed current quench duration (lowest plasma temperature after the thermal quench). In the case of a VDE, load severity will depend on how large is the plasma displacement across the destabilising poloidal field, without a decrease of the plasma current. Again, in all these cases the forces developed between the coils and the vessel are restrained through the shortest path through the TF coil structure, taking advantage of the direct link between these components.

Detailed numerical studies, under conservative assumptions, of all these important events, have led to the following conclusions:

- the plasma control system will be capable of maintaining the plasma vertical position for all nominal plasma disturbances including minor disruptions; therefore VDEs should occur only during a major disruption or a failure of the control system;
- during a major disruption, the plasma will move inward radially and upward vertically, but vertical forces will be much smaller than for a downward VDE, which could occur only in the absence of control; if a “killer pellet” can be used, it will trigger plasma quench early during its motion, and thus limit the loads.

The global mechanical structure of ITER is strong enough to resist the most conservative assumed case: a combination of loads which can occur simultaneously, being trigged by one single event, i.e. an earthquake for example.

4.4 Fuel Cycle

For delivering 500 MW of total fusion power, about 0.1 g of tritium will be burnt every 100 s. But, if the maximum throughput of fuelling and pumping are used to satisfy the divertor operational conditions, more than 25 g of tritium will be injected into the vessel during the same 100 s, and removed by the cryogenic pumps.

The need is obvious to process the pumped gases on line, to remove impurities and separate the tritium, and to store it for recycling. This fuel cycle is shown in Figure 4.4-1. It includes first a permeator to separate impurities from hydrogen in line with the pumping exhaust. Then, the impurity flow is processed before sending it to atmosphere, with an ALARA (as low as reasonably achievable) content of tritium. The hydrogen flow is processed to separate the different masses, by isotope separation through cryogenic distillation. This part of the plant is optimised to minimise the tritium inventory as far as possible, compatible with the isotope separation ratio required (not very high) and the global throughput. For nominal pulses (< 450 s), the fuel cycle does not operate as a steady state, online system. The outlet stream of hydrogen isotopes from the permeator feeds a buffer storage tank, before being processed on a longer timescale by the isotope separation system. For longer pulses, on the contrary, steady state operation could be reached using a direct feed from the permeator output stream.

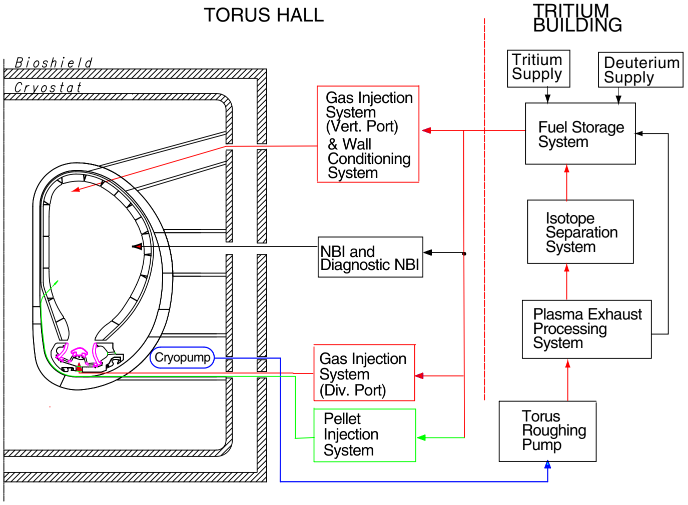


Figure 4.4-1 Block Diagram of Fuel Cycle

Segregation of tritium-containing equipment in separated structures, with limitation of the local inventory and robust confinement barriers, is appropriate for safety reasons. The storage of D_2, DT and T_2 is achieved in many parallel canisters, and adsorbed on ZrCo beds, which can deliver rapidly the required flow for plasma fuelling. Their tritium content is measured by calorimetry with around 1% accuracy.

Tritium accountability in the fuel cycle is an important issue, because a part of the tritium injected in the plasma can remain in the vessel trapped by co-deposition with carbon dust. This tritium in-vessel inventory should remain below a ceiling, fixed presently at 450 g, and measures for dust removal are under study, as already mentioned.

In addition, a small amount of tritium, adsorbed on all in-vessel surfaces, will be progressively desorbed, and recovered partially in oxidised form, from detritiation systems installed to limit the tritium content in the in-vessel atmosphere during maintenance, or in the hot cell atmosphere during component repair. The resulting tritiated water will be processed to reinject the tritium into the fuel cycle.

4.5 Tokamak Building

The buildings should provide the volumes and controlled atmosphere required for ITER assembly and operation. In addition, the tokamak building is important for its contribution to safety, by the following means.

First, a biological shield of borated concrete is provided around the cryostat to limit the radiation levels outside the pit to values insignificant for the activation of components, even if human presence will not be allowed, during plasma pulses.

Second, its role is essential as a further confinement barrier (even containment in this case), forming two concrete leak tight vaults around the neutral beam injectors and the water cooling system, or even as a third confinement barrier in the tritium building (the metallic equipment inside glove boxes provide the first and second barriers in this case).

Third (Figure 4.5-1), there is a differential pressure maintained in the different zones around the tokamak, according to the risk of being contaminated by an accidental release of tritium or activated material during operation or maintenance. In this way, the atmosphere should leak only from smaller to larger contamination levels. These differential pressures are maintained by the air conditioning system. When a release is detected in an area, this area will be isolated from the others and its atmosphere will be filtered and detritiated before being sent to the environment. This policy explains the design arrangement of a separate cell around each vessel port access, especially justified during the maintenance procedure of removing components from the vessel. In case of an accidental release, the normal air conditioning system is shut down, and a large detritiation process becomes active.

In addition, the concrete walls provide appropriate shielding against emission from activated components, during their automatic transport via cask from one vessel port to the hot cell (and back) through the galleries.

The very robust structure of the tokamak and tritium buildings groups the interdependent components within these buildings on a common basemat to better react seismic conditions. If the actual site has much harsher conditions than the generic site used in the design, the common basemat will be put on isolators and the acceleration amplification suffered by the components above will be maintained at the accepted design level, as indicated previously.

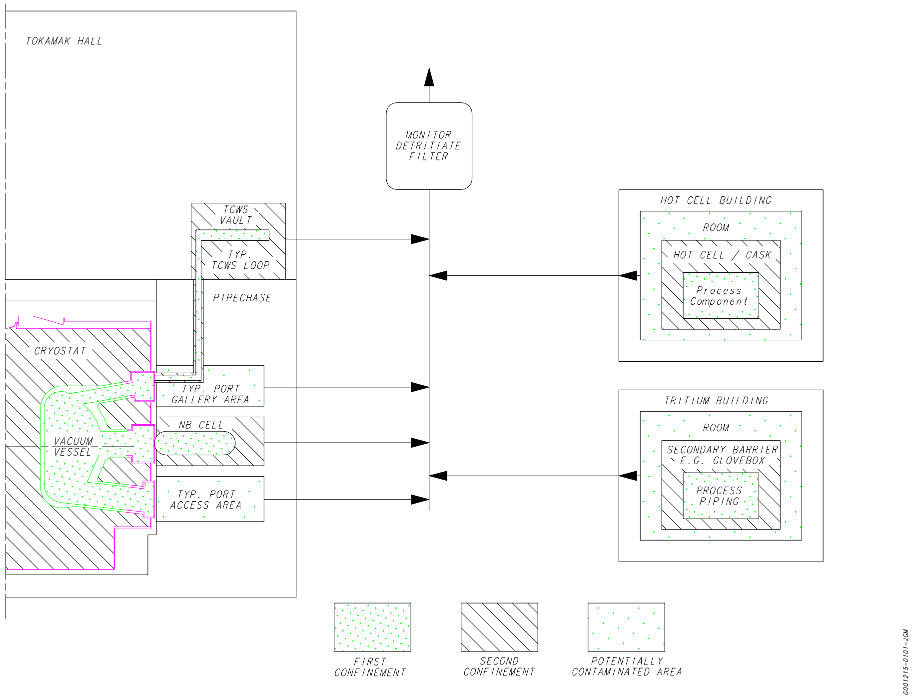


Figure 4.5-1 Schematic of Confinement Approach illustrating Successive Confinement Barriers that are Available

4.6 ITER Plant Operation and Control

Compared to today's experiments, the need for ITER to operate with a burning plasma under stationary conditions for more than 400s, while handling about 0.5 GJ of plasma thermal and magnetic energy, poses quite challenging and new constraints in the design of hardware that affects the control of plasma operation. The typical waveforms of a standard driven burn plasma pulse are shown in Figure 4.6-3.

Therefore, one of the most important objectives of plasma operation and control in ITER is the protection of tokamak systems against the normal and off-normal operating conditions, from safety and protection of investment viewpoints. Reliable control of the fusion burn conditions and provision for its rapid termination under off-normal conditions are crucial.

To support plasma operation, plant systems must be efficiently and reliably operated. In particular, the fuel cycle needs special attention in order to manage and control the tritium inventory within the system. In the water cooling system, the control of activated corrosion products and tritium content is very important. Although the basic mode of tokamak operation is pulsed, there are certain resulting requirements and constraints for operation of the tokamak supporting plant systems, which are intrinsically more steady state.

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 a large amount of light impurity injections.

Control of plasma parameters can be characterised by three basic attributes of closed loop control systems – diagnostics, control algorithms and actuators as shown in Figure 4.6-1. The control algorithm is typically a proportional/integral/derivative (PIDS) feedback scheme. There are, however, alternate algorithms designed with more sophisticated optimisation procedures.

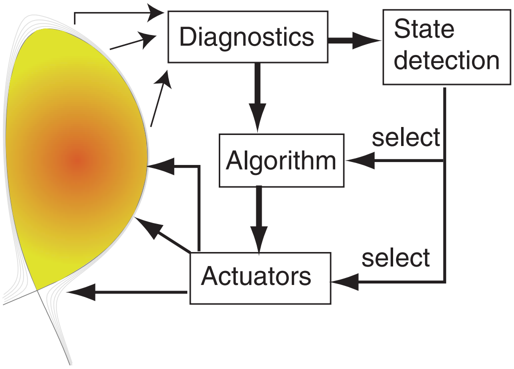


Figure 4.6-1 Plasma Feedback Control

Figure 4.6-1 introduces the concept of plasma state cognisance, and state-dependent control actions. Here the change of the plasma state can dynamically modify the control algorithms and choice of control actuators so as to more optimally control the overall plasma response. The implementation of state-cognisant control gives the control system a certain degree of autonomy. It will ultimately lead to a highly dynamic and state- and scenario-phase dependent 'expert system'.

A plasma control matrix for ITER to relate control actions or actuators and controllable parameters is shown in Figure 4.6-2. The vertical organization of the matrix reflects the division of the plasma control system into the four hierarchical categories mentioned above, namely scenario, magnetics, kinetics and fast shutdown.

PLASMA CONTROL MATRIX

● = Major direct effect

◎ = Appreciable secondary effect

○ = Possible secondary effect

blank = No appreciable effect
or not applicable

		Control Action (Controllable Parameter or System)																		
		Scenario and Magnetics						Fueling and Exhaust						Auxiliary Heating and Current Drive (options)						Shutdown
	Measurable Quantity or Attribute to be Controlled	TF field (static, 0-5.3 T)	PF currents	PF voltages	Prefill pressure (D_2)	Startup EC (ω, P, t, Δt)	Error field compensation current	DT fueling (gas, into SOL)	DT fueling (shallow pellet, ~1 km/s)	Impurity fueling (He,Ne,Ar... to SOL)	DT divertor fueling (gas)	Impurity divertor fueling (Ne, Ar, ...)	Pumping speed (0-300 m^3/s)	NBI power (0-max)	ICH power (0-max, ω, mode)	ECH power (0-max, launch angle)	FWCD (power, radial location)	ECCD (power, radial location)	LHCD (power, $n_\parallel$)	Shutdown Pellet
1: Scenario 2: Magnetics	Plasma current, q_{edge}	●		●																
	Plasma shape (R, a,			●																
	Plasma shape (FW gaps)			●																
	IC coupling impedance			●				◎		◎	○								○	
	Plasma current initiation		●	●	●	●														
	Locked mode susceptibility		◎				●							●						
3(a): Core Kinetics	Plasma density							●	●	●	●	○	○	○						
	Fusion power							●	●	◎	●	◎	◎	○	○	○	○			
	He fraction									◎		○	●	◎	◎	◎	◎	◎	◎	
	Core D/T ratio							●	●	●										
	Core impurity fraction									●		◎								
	Core radiation fraction									◎		◎	●	◎	◎	◎	◎	◎	◎	
	Core plasma rotation (f_{rot})													●						
	W_{th} or β_N (at given P_{fus})	●						○	○					◎	◎	◎	◎	◎	○	
	Axial safety factor q(0)													◎			○	○	○	
	Current profile j(r)	●												◎			●	●	●	
	Sawtooth period	◎												◎			◎	◎	○	
3(b): Edge Kinetics	ELM period, magnitude			○				●	◎	●										
	n_{edge}							●	◎	◎			◎						◎	
	SOL flow							●	◎	◎			●							
	SOL radiation fraction									●		◎								
3(c): Divertor	Divertor power input							◎	◎	●	◎	●	◎	◎	◎	◎	◎	◎	◎	
	In-divertor radiation (x,y)									◎		●	◎							
	Target plasma (n,T)							●	◎	●	◎	◎	◎							
	Target power or temp.			○				●	◎	●	◎	●	●	◎	◎	◎	◎	◎	◎	
	Divertor neutral pressure			○				◎	◎	●		●	●							
	Divertor He fraction							◎	◎	●	◎	●	◎							
4: Shutdown	Fast P_{fus} and I_p shutdown																			●

Figure 4.6-2 ITER Plasma Control Matrix

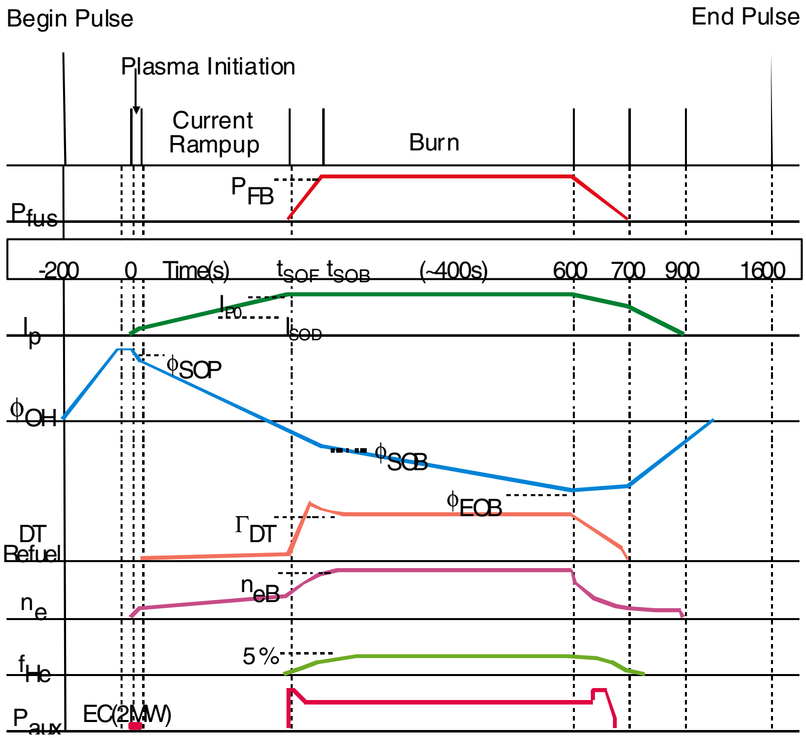


Figure 4.6-3 Waveforms for Standard Driven-burn Operation Scenario

4.6.1 ITER Plant Control System

The ITER plant operation is controlled and monitored by the “Command Control and Data Acquisition and Communication” (CODAC) system. The CODAC system consists of a centrally positioned supervisory control system (SCS) and sub-control systems dedicated to each plant subsystem under the supervision of the SCS.

The SCS provides high level commands to plant subsystems, and monitors their operation, in order to achieve integrated control of the entire plant. An interlock system, sometimes in parallel with the CODAC system, ensures plant-wide machine protection, as well as personnel protection, in case of off-normal events. It monitors operational events of the plant, and performs preventative and protective actions to maintain the system components in a safe operating condition. The interlock system is also hierarchically structured and has individual interlock subsystems which are dedicated to each plant subsystem under the central supervisory interlock system.

The integrated control of the entire ITER plant will be achieved by the CODAC, and the interlock system. A concept of the ITER control system is schematically shown in Figure 4.6-4.

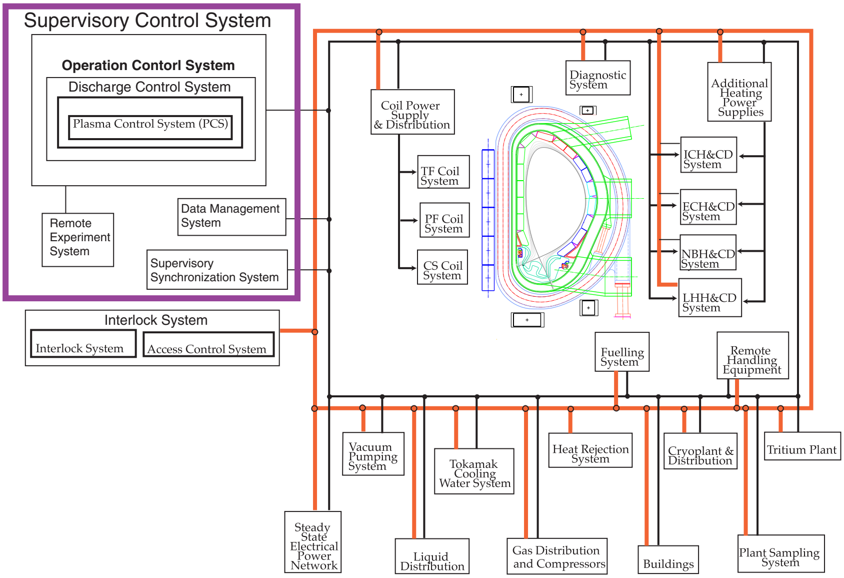


Figure 4.6-4 ITER Plant Control System

5 Construction, Commissioning and Decommissioning Plans

5.1 Introduction

The planning schedule for supply, construction/assembly, commissioning and decommissioning set out below depends on a number of assumptions detailed in the following. As the design progresses, decisions reached by the Parties may confirm or alter the assumptions that have led to its present status. The actual plan will therefore depend on the licensing procedure, as well as the organization and arrangements that will be put in place for the procurement/construction commissioning.

The construction agreement is expected to be signed at the end of 2002 or the beginning of 2003 following formal negotiations. The ITER legal entity (ILE) will be established after ratification of the agreement within each Party. This organisation will start the formal regulatory procedure and procurement process for the long lead-time items. The regulatory approval process, however, will remain speculative until a site is formally selected. As the site proposals are received before or at a sufficiently early stage of negotiations, it will be possible to assess the time needed for licensing in the various possible host Parties and the effects on the overall schedule. Since the start of the actual construction on the site depends upon when the licence to construct is issued by the regulatory authority, dates in the construction and commissioning plan are, therefore, measured in months from a start date ("T = 0") defined as the date at which the actual construction work of excavation for the tokamak buildings is started.

Furthermore, the following assumptions pertain at T = 0.

- Informal dialogue with regulatory authorities should be established and should orientate the technical preparation toward a licence application with a view to solving the major technical issues prior to establishment of the ILE. Documents required for the formal regulatory process are assumed to be prepared before the ILE exists, so as to allow the ILE to begin the formal regulatory process immediately after the establishment of the ILE.
- Procurement specification of equipment/material for the longest lead-time items and critical buildings are assumed to be finalized during the co-ordinated technical activities (CTA).
- Procurement sharing is assumed to be agreed among the Parties during the CTA so as to permit the placing of all contracts at the appropriate time.
- The construction site work starts immediately at T = 0. It is assumed that site preparation has been started sufficiently early by the host Party so as not to place constraints on the start of construction.

5.2 Overall and Summary Schedule

The overall schedule that leads up to the first hydrogen plasma operation is shown in Figure 5.2-1. It represents a reference scenario, which is a success-oriented schedule of procurement, construction, assembly and commissioning of ITER, based on the assumptions. The detailed construction schedule is developed to correspond to each procurement package specified for the cost estimate (see later). The schedule for each package includes procurement specification preparation, bid process, vendor's design (if appropriate), manufacturing (if appropriate), transport to site (if appropriate), installation and commissioning.

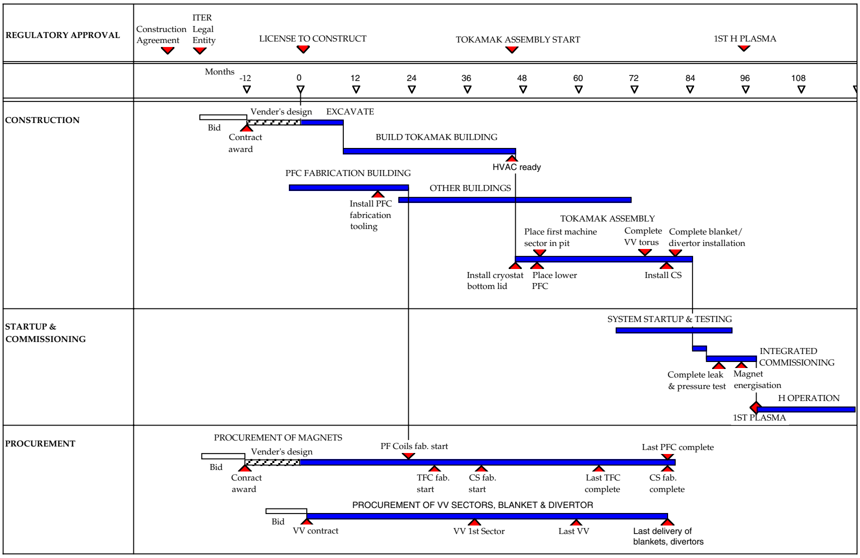


Figure 5.2-1 Overall Schedule up to First Plasma

5.3 Construction and Procurements

5.3.1 Procurement Assumptions

The lead-times for the different components of ITER vary widely. Also some items – including the buildings, parts of the cryostat, magnets, and vacuum vessel - are logically on the critical path irrespective of their time schedule, whereas others can be delayed until an ultimate date when they will become part of the critical path. In addition, cash flow may pose constraints in conflict with the need to make procurements at a date compatible with smooth planning of implementation. For the purposes of evolving this schedule, procurement is assumed to occur such that systems/components are delivered just in time, i.e. at the latest time, on the critical path, for assembly and installation/construction, in accordance with the construction logic. In a second evolution of this plan, some items could be moved earlier in the schedule to gain same margins, which would remove some items from the critical path line. However if cashflow peaks caused by critical items are too high, only an extension of the overall schedule is possible as a solution.

Another important assumption is that the placing of purchase orders is allowed on the establishment of the ILE, even before the licence to construct is awarded (T = 0), in order to allow vendor's design and tooling preparation for the critical lead-time components and buildings. In reality, for non-safety-related items (e.g. magnets), manufacturing can even be started before the granting of a construction licence, provided it is clear one will eventually be granted. For safety-related items, however, construction can only start after the license to construct is issued, if necessary through a second contract. Documentation required for purchasing the various items will be completed in time for the scheduled procurement. The procurement bid process is assumed to take typically six to twelve months from the release of tender documents to industries to the awarding of contracts.

5.3.2 Buildings and License to Construct

The critical path of the plan is the regulatory licensing procedure and the construction of the tokamak buildings. In order to start excavation immediately at T = 0, the design of the complex must be complete by then. The contract for the vendor's preparation of the complex, thus has to be awarded at least twelve months before. Considering a period for the procurement bid process, the tender documents have to be released 18 to 24 months before T = 0. If the establishment of the ILE occurs 24 months before T = 0 and the license process can be completed in this period, the construction period defined from the establishment of the ILE to first hydrogen plasma discharge is ten years. If the regulatory process takes more than two years, the construction period becomes longer.

Excavation is to be completed within 9 months. Installation of the large pipes of the heat rejection system that are below grade will follow. The tokamak buildings must be functional, including cranes and HVAC system, by the end of 45 months from T = 0 in order to allow the timely start of tokamak assembly.

Two cryoplant buildings are to be built and serve dual purposes, as the PF coils fabrication buildings in the early stage, and cryoplant cold box and compressor buildings later. To maximise the time available for PF coil manufacture and to allow the cryoplant to be used for NB injector stand-alone commissioning as soon as possible, the construction of these buildings is also started at T = 0, or sooner for buildings not related to safety/licensing.

5.3.3 Procurement of Long Lead-Time Items

The tokamak building is ready for machine assembly at month 46. In order to start the pre-assembly of TF coils and vacuum vessel sectors in the assembly hall, the first two TF coils at least must be delivered by month 45 and the last two by month 62. It is essential to purchase an initial quantity of Nb_3Sn conductor to train the lines for the forthcoming serial strand production and to award the contract for the design of the TF coils manufacturing prior to T = 0, as soon as the ILE is established. This should be possible if (1) a sufficient number of contractors in the world have been qualified and trained, (2) the detailed specifications for manufacturing TF coils are fully available by the time of the signing of the construction agreement, and (3) procurement sharing will be agreed among the Parties before the signing of the construction agreement. This allows the call for tender to be issued immediately after establishment of the ILE.

Most of the PF coils are too large to consider their transfer from the factory to the site (unless both factory and ITER site have deep water access). Thus, fabrication on site is likely to be unavoidable. To save cost, the cryoplant buildings are used. However, the two smallest coils, PF1 and PF6, may be fabricated at an off-site factory. The lower PF coils, PF5 and PF6 must be ready to be placed at the bottom of the pit at the beginning of the tokamak assembly starting in month 46. The other coils are stored until the time of installation.

Nine (40º) sectors are shipped to the construction site and pre-assembled with the TF coils and vacuum vessel thermal shield (VVTS) in the assembly hall before being installed into the pit. Three VV sectors are simultaneously at different stages of manufacture. The manufacturing time for one sector including the welding of a port stub extension is 21 months. Another 3 months are taken into account for shipping and inspection for acceptance at the site. The first sector should be available by month 45 to start machine assembly. The manufacturing time for the 9 sectors is 42 months in total. The last sector should be accepted at month 62.

5.4 Tokamak Assembly

The assembly procedure is grouped in five activities, lower cryostat assembly, TF/VV/VVTS (machine 40° sector) sub-assembly in the assembly hall, integrated TF/VV/VVTS assembly in the pit, in-vessel component assembly after completion of the torus and establishment of the magnetic axis, and ex-vessel component assembly. It takes two and a half years to complete the assembly of the vacuum vessel torus with TF coils and VV thermal shields. About a year is needed to assemble the in-vessel components and ex-vessel components simultaneously after completion of the torus and establishment of the magnetic axis.

The tokamak assembly starts with the installation of the bottom lid of the cryostat, placed on the basemat of the pit at month 45. The lower cryostat cylinder is then installed. For manufacturing of the bottom lid 18 months are needed at the factory and a further 6 months for construction work on the site before installation. Early procurement is required, namely within nine to twelve months after T = 0. The upper cylinder and top lid of the cryostat will be fabricated to match the tokamak assembly process. The completion of the cryostat with the installation of the top lid is at month 84.

Drain tanks and the main pipes for the primary heat transfer loops and heat rejection system are placed on the basemat level of the pit during the construction of the tokamak building and in the surrounding service tunnels. Thus these components have to be procured early.

5.5 Commissioning Plan

5.5.1 Individual Sub-System Test

Testing of each individual plant subsystem has to start immediately when permitted by their delivery and by the corresponding assembly work. This testing is the last phase of procurement for each individual subsystem. Individual plant subsystem tests will be done by simulating interfaces with other systems or by using dummy loads or bypasses. Links with the CODAC supervisory control system are compulsory at this time. Each individual subsystem must be ready before the next phase of integrated testing with other subsystems. Subsystems not needed in an early phase will be commissioned in parallel with operation.

A more complex example is that of the remote handling (RH) equipment. Many RH tools, especially transporters, will be tested on mock-ups during the construction phase. Some complete RH techniques will be used as part of the initial installation either because it is more efficient or for demonstration purposes. Therefore, the major RH equipment will be installed and commissioned before first plasma.

5.5.2 Integrated Commissioning up to the First Plasma Discharge

There will be the need for adequate testing of controls and interfaces between subsystems. CODAC is designed to permit testing of the complete system in the absence of one or more of the sub-subsystems. This one year of integrated testing requires that all the key subsystems have been successful in their individual tests.

All systems are tested to the extent possible without plasma. This includes the following major items.

- Vacuum leak and pressure test of vessel and cryostat
- Hydraulic test and baking
- Magnet cooldown (40 days)
- TF coil energisation
- Pulsing without plasma and wall conditioning

At the end of this phase the following will be achieved:

- readiness of operation of the tokamak machine (e.g., vacuum, baking, sufficiently high toroidal magnetic field (4 T) to match ECH&CD for start-up and discharge cleaning, and more than 50% of coil current in all PF coils);
- readiness of all subsystems, including the additional heating and current drive system, start-up diagnostics set, and fuelling (except the tritium system), which are needed for H plasma operation; initial test blanket modules (or blanking plugs) have been installed and commissioned; some subsystems may be completed later (as indicated above).

5.5.3 Commissioning after First Plasma

After the first plasma there will be further integrated commissioning over about four years leading to full operation in DT. The first 2.5 years of operation without DT is defined as a "pre-nuclear commissioning phase" and "nuclear commissioning" (about one year) will be done by using DD discharges with limited amounts of tritium.

5.6 Decommissioning Plan

It is assumed that the ITER organization at the end of operation will be responsible for starting the machine decommissioning through a de-activation period after which the facility will be handed over to a new organization inside the ITER host country. It is therefore necessary to provide a feasible and flexible plan for the decommissioning of the ITER machine and associated active components. The plan is based on the rationale of resources and equipment usage optimization, and takes into account the statutory occupational radiological exposure (ORE) limits. The plan provides a framework for the organisation to decide when and how to implement the ITER facility dismantling, depending on priorities applicable at the time. Flexibility is provided by the use of two separate phases.

During the first phase, the machine will, immediately after shutdown, be de-activated and cleaned by removing tritium from the in-vessel components and any removable dust. Also, any liquid used in the ITER machine systems will be removed (no component cooling will be further required) and processed to remove activation products prior to their disposal. De-activation will include the removal and safe disposal of all the in-vessel components and, possibly, the ex-vessel components. The main vacuum vessel may be prepared for dismantling by the cutting of the inner vessel wall. The ITER de-activation will also provide corrosion protection, for components which are vulnerable to corrosion during the storage and dismantling period, if such corrosion would lead to a spread of contamination, or present unacceptable hazards to the public or workers. These activities will be carried out by the ITER organization using the remote handling facilities and staff existing at the end of operation. At the end of phase 1, the ITER facility will be handed over to the organization inside the host country that will be responsible for the subsequent phase of decommissioning after a dormant period for radioactive decay.

The plan does not include the dismantling of the buildings and of the non-active components (except, when applicable, for the ex-vessel components), or the disposal of wastes from decommissioning. Outside of the pit, the re-use or scrap value of components is higher than the cost of dismantling them.

Table 5.6-1 Summary of the ITER Decommissioning Plan

PHASE	ACTIVITY	DESCRIPTION	DURATION
1	De-activation	1 Removal of mobilizable tritium and dust from the machine using available techniques and equipment. Removal and de-activation of coolants. 2 Classification and packaging of active, contaminated and toxic material. 3 Removal of all the in-vessel components. OPTION 1: removal of ex-vessel components (if not done in phase 2).	~ 5 years
	The ITER facility is handed over to an organization inside the host country		
Radioactivity decay period		1 The vacuum vessel radioactivity is left to decay to a level which allows the extraction of vessel sectors into the tokamak building (during phase 2) for size reduction and disposal. 2 No site activities are required except security and monitoring.	As required
2	Final Dismantling and Disposal	1 Removal of vacuum vessel sectors and their size reduction by remote/semi-remote operations. OPTION 2: removal of ex-vessel components (if not done in phase 1) 2 Classification and packaging of active, contaminated and toxic material	~ 6 years

6 Cost Estimates

6.1 Resources Required for ITER Construction

6.1.1 Cost Estimating Approach for ITER Construction

The approach to cost estimating for the construction of ITER is based on the presumption that ITER will be constructed as an international joint project in which the participants' (Parties') contributions will mainly be specific systems or components contributed directly to the project ("in kind").

The main objective of ITER cost estimates is to provide a realistic and sufficient basis for ITER Parties to make their decisions on the scope of their involvement and to select the desirable systems for them to manufacture. The estimates have been developed from the engineering designs following a "bottom-up" approach which emphasises physical estimates (such as labour hours, material quantities, physical processes, etc.) so as to ensure that the data are comprehensive and coherent and provide a basis for evaluating results from different Parties.

In considering the ITER cost estimate, it is important to recognize that:

- economic conditions in the Parties vary widely over time and these changes are not necessarily or adequately reflected in relative monetary exchange rates;
- domestic industrial practices, contracting policies and labour costs for manufacturing prototypes depend on Parties and are not reflected in relative currency exchanges (only valid for goods for which a worldwide market exists);
- the overall ITER management approach and specific procurement and contracting practices have not been determined and the host party has not been selected;
- the aggregate costs that will be incurred in constructing ITER will depend greatly on how the responsibilities for specific components are distributed between the different participants, and on the procurement policies pursued by each.

6.1.2 Basic Data — Procurement Packages for Cost Estimation

In order to elicit the basic data for ITER cost estimates, about 85 "procurement packages" have been developed for the elements of the project work break-down structure (WBS), each defined at a level consistent with a plausible procurement contract. Each package comprises comprehensive information, including the functional requirements, detailed designs, specifications, interfaces and other relevant data that would be needed by potential suppliers in order to prepare for contract quotations, e.g. the proposed split of responsibilities between supplier and customer and the necessary QA arrangements.

In some areas the packages provide for the possibility of splitting contracts between several suppliers in case more than one Party might wish to participate in a specific area. In some others, this splitting between suppliers in more than one Party is necessary in order to produce the volume required because of the limited capacity of each supplier (e.g. superconductor strand, shield/blanket modules, divertor targets).

Industrial companies or large laboratories with relevant experience were invited, through the Home Teams, to generate, from the procurement packages, their best estimates of the likely current costs of supply, assuming that all data necessary to support procurement would be available on schedule. To allow review and comparative evaluation of the estimates, the participants were requested to provide detailed supporting data and detailed descriptions of the potential deliverables and processes, in the format of standard ITER Cost Estimation Workbooks.

The information thus generated offers a comprehensive database for cost analysis, comparison and evaluation.

However, the cost estimates provided by each Party are not intended to be the lowest values which could be obtained in this Party, keeping the same technical specifications, since they were not the result of competitive estimation and tendering.

6.1.3 Evaluation of Cost Estimates

The JCT has analysed the results of the procurement package studies in consultation with Home Teams concerned, with the objective of deriving “evaluated estimates” which distil the database to a consistent set of project cost estimates.

The final result is a complete set of “evaluated cost estimates” for building ITER, expressed in IUA, which is robust to currency fluctuations and domestic escalation rates and which can be used by the Parties jointly and individually in reviewing their options and the possible budgetary effects of participating in ITER construction [1 IUA = $ 1000 (Jan 1989 value)].

Data expressed in terms of physical quantities can be readily compared directly, after confirming the definition of the terms. Data expressed in money values needs to be converted into common terms for the purposes of comparison and analysis.

Financial data have first been converted from current costs to the established reference date for ITER cost estimates of Jan 1989, using standard inflation factors for each Party (Table 6.1-1).

This methodology has already been applied for the ITER 1998 design cost estimates.

It is not assumed that the 1989 exchange rates between currencies is better than those at any other date. Even, exchange rates between two currencies and inflation rates in their two countries do not sometimes vary in a coherent way (examples are Canadian and US Dollars, British Pound and Euro).

Nevertheless, looking backwards in time, using this approach the Parties have their own appreciation of these old rates and can apply at once the correction factor they feel appropriate (because of labour cost, industrial practices or any other reason) to quantify directly their potential contribution in their current money.

Moreover, a constant reference unit is the most appropriate tool to cost a project which extends in time over many decades in an international framework.

Table 6.1-1 Exchange and Escalation Assumptions

Currency	Conv. Rates to US$ 2000 (for comparison only)	Conversion Rates to Euro (Fixed)	Conversion Rates to US$, (January 1989)	Escalation Factors for Reference Years					Conversion Factor 2000 to IUA (x 10^{-3})
				1989	1997	1998	1999*	2000*	
US $	1.000		1.000	1.00	1.32	1.35	1.37	1.39	1.392
Can $	1.507		1.207	1.00	1.21	1.22	1.24	1.25	1.509
Euro	1.036	1.0000	0.876	1.00	1.37	1.41	1.44	1.46	1.279
DM		1.9558							
FF		6.5596							
Lira		1936.3							
UK Pound	0.634		0.585	1.00	1.42	1.47	1.51	1.55	0.906
Yen	107		128	1.00	1.13	1.14	1.15	1.16	0.148
Rouble	28.34		39.45	1.00	1.00	1.00	1.00	1.00	39.45

Note: * Extrapolated data
References [1] : International Financial Statistics Yearbook, International Monetary Fund, 1999, [2] : Eurostat Yearbook, Edition 98/99.

In order to emphasise the relative costs of the different systems/components, the following method has been used to derive from the basic data a JCT cost estimate, consistent across the whole ITER plant:

- world market prices have been used where they exist, for instance for standard materials or items of equipment;
- unified standard labour costs have been established which reflect (see Table 6.1-1):
    - rates in IUA/khour established for the main categories of labour, averaged across skill levels and Parties;
    - the different levels of manufacturing support costs, including facility, equipment, overhead and profit, for different types of industrial works and working conditions (e.g. staff in their existing home facilities, on site, or in a facility charged as an itemised cost).

After analysis of the proposals from the different Parties for amounts of tooling, material quantities and labour hours to manufacture each identified item, the JCT has established its own assessed numbers for these cost driving elements and, using the above-mentioned lists of standard materials prices and labour costs, has derived an "evaluated cost estimate" for each identified item.

Summing all items per procurement package, the JCT provides through this methodology a normalised cost for each package, the most credible given the present uncertainties on ITER construction management and procurement. For reasons described above, when applying to each package the conversion factor from IUA to the present value of currency in one Party, the result should not be expected to match the cost incurred by this Party to provide the given component.

Table 6.1-2 ITER Labour Rates

The cost estimates prepared for the EDA Final Design Report include a number of parameters which need to be normalised. A large table was established to provide a set of labour rates for both field construction and shop manufacturing operations. These rates were used for all estimates for both home and field manufacturing operations as well as site installations. Established in IUA /khours, these rates are assumed constant in time (which means that, converted in any Party's currency, they will follow the inflation rate of that Party).

Labour costs are composed of labour wage rates and support costs:

- *The Labour Wage rates are the Hourly Wages each worker earns plus fringe benefit packages including employer's contribution to taxes, health benefits, vacation benefits if any, and related employer-paid labour cost items.*
- *Support costs, also presented as an hourly rate, include the cost of use of the manufacturing facility and equipments, ship supervision, management, consumables, other overheads, and profit.*

For some commodities, such as for the TF or PF coils, in which Facilities, Special Tooling and Equipment are estimated separately, the support costs have been decreased accordingly. For those items to be fabricated on site (in the same Country as Home), it is assumed that the support costs will be the same as at Home, but that a premium (0.005 IUA) is added to the hourly wage.

The Wage Rates were determined by using an average of wage rate cost data from the U.S. ("R.S. Means, 30 City Average, January 1997"), Japan ("Chingin Jijyo", Factory labour direct cost and "Seskisan Shiryo", construction labour rate), and EU (Eurostat). All rates include 5% casual overtime. Manufacturing labour cost is based on a composite of Highly Skilled, Skilled, and Semi-Skilled Labour. Site Installation Labour is based on a composite of Craft Labour.

From these values of 1997 using conversion factors to IUA (according to inflation rate in each Party), one can note that in average:

- *the normalised labour wages are 20% above the European values and 20% below the Japanese ones;*
- *the normalised support cost per hour is equal to the European value and 10% below the Japanese one; it amounts to a value between about 2 (for welding) and 3 (for machining) times the normalised labour wage, and only about 1.25 (for welding) and 1.80 (for machining) if the special equipment is estimated separately.*

A few global values for Labour rates are mostly used in the ITER "estimated cost" (in IUA/khour):

- *for Engineering, a mix of professional designers (90), professional engineers (67), CAD and procurement technicians (38), in average 67-73,*
- *for machining including full support (67) or limited support (48),*
- *for welding including full support (55) or limited support (40),*
- *for QA and testing, (45),*
- *for assembly/installation, (35 to 45, depending on the amount of professional support).*

For civil work, the costing is done through the use of a table of commodity rates for all activities referred to the "unit measurement" of the quantity of the commodity used (m^3, m^2, t,etc..). They include material and labour amounts as well as specific support per unit.

6.1.4 Conclusions on the Approach to the ITER Construction Cost

The approach to develop a JCT evaluated cost estimate for ITER construction, expressed in IUA tries, to the extent possible, to remove variations in costs that are due to differences in estimating practices by the different Parties, and to exchange rate fluctuations. This means that although ITER costs for each item reflect the same "value" in two Parties, they will not, when expressed in these Parties' currencies, necessarily correspond to each other through exchange rates at any date.

The approach provides fair and consistent relative costs for the different ITER systems and components. The Parties can, jointly, appreciate in advance the relative contributions (in percentage) that each might make to building ITER and, individually, estimate from the underlying physical data the absolute costs (in their own currency) that each might expect to incur in providing specific components, inside their contribution "in kind", by applying their own appropriate conversion factor to IUA.

This detailed "evaluated cost estimate" aids the "design to cost" and, later, the "manufacture to cost" approaches by which design/process changes are made to maintain costs within the budgeted amount because itemized quantities and manufacturing man-hours are clearly defined.

6.1.5 Cost Estimates Summary

6.1.5.1 Component/system "Evaluated Cost Estimate"

The evaluated cost estimates for ITER construction in kIUA, summarised by procurement packages, are presented in the Table "ITER Cost Estimate Summary". A much larger "Business Confidential" document provides for each procurement package the JCT detailed evaluation. The table also indicates, for each package, which of the Parties provided data, what is its direct capital cost, its percentage of the total capital investment, the cost of spares and of deferred items to be supported by operation funds, and their comparison with the relevant numbers for the ITER 1998 design.

As recognised by Experts from the four Parties, who have reviewed the details of the component/systems costing which support these estimates, they are not intended to be the lowest values which could be obtained, keeping the same technical specifications, since they were not the result of competitive estimation and tendering within each Party.

Globally for ITER construction they represent the most credible cost estimates, given the present uncertainties on ITER Construction Management, Siting and Cost sharing: a global value inside which one can be confident to be able to build ITER.

In addition, there is room to achieve substantial savings in some areas already identified, and more might be found through the needed and expected industry feedback on design to optimise manufacturing processes.

6.1.5.2 Siting and Construction Costs

The present ITER design, and its “evaluated cost estimate”, follows the “ITER Site Requirements and Site Design Assumptions”, which have been approved by the ITER Council[4]. Site specific adaptations of the design may induce changes in the cost of some systems; they will be analysed during the Coordinated Technical Activities (CTA), when potential sites are characterised.

Similarly, the present design is consistent with codes, and standards which have been defined inside the project. These rules are coherent but are not identical to those of any specific Party, even if they do not contradict them. Regulatory bodies from potential host country may request application of different and specific design rules or quality assurance measures. This can induce cost variations to be analysed again during the CTA.

The “ITER Site Requirements and Site Design Assumptions” describe a list of Host responsibilities, for which the project bears no cost, and which include in summary:

- infrastructure for industrial support of ITER and socio-economic provisions,
- land for the ITER site,
- high quality (potable), and raw water, treatment for sanitary and industrial sewage,
- supply to steady state electrical power network and a tie line capable of large pulse power for magnet and plasma heating,
- off site fire protection equipment and personnel,
- receipt of waste of all types, generated from ITER operation and decommissionning

6.1.5.3 Procurement Scheme

Each procurement package relies on a single dominant technology, and therefore mainly leads to a single industrial contract and avoids the need for a large amount of subcontracts. On the other hand, the possibility of sharing the work to be done for a package between more than one industry was always considered, even if this splitting leads to an (acceptable ?) increase in cost. This method provides for the ITER Parties the possibility, if they wish, to contribute in all fusion specific technologies. For example, the following list offers a possible splitting of procurement orders:

---

[4] PDS Chapter 4

- the Nb_3Sn and NbTi conductors (355 kIUA in total) should be provided by contributions from all Parties; even more, industries in the Parties should be encouraged to prepare for a larger production capability, in particular in Russia for niobium of the required quality;
- two suppliers are proposed to share equally the manufacturing of the TF coils windings (117 kIUA in total);
- one or two equal suppliers for the procurement of TF coil mechanical structures (168 kIUA in total).

In these cases, it is thought appropriate that one contractor will have the responsibility to develop the design for the special toolings, which will be duplicated for use by the second contractor. The two sets of tools are anyway necessary to meet an acceptable manufacturing schedule. Therefore, this sharing possibility between two Parties' industries leads to only a small increase of cost due to a small loss of labour efficiency.

- one manufacturer for the central solenoid (31 kIUA);
- one manufacturer for the PF coils, (50 kIUA), assumed to work on the ITER site and thus from the Host;
- one manufacturer for coil Feeders (41 kIUA): a complex area of superconductor and cryogenic technologies and instrumentation;
- at most, three equal industrial partners for the vacuum vessel manufacture (155 kIUA in total), and one or two more to produce the port structures (75 kIUA in total), if they agree on cooperation in the engineering of the specific process toolings;
- no difficulty should be met to share between different industries the delivery of more than 420 blanket modules (143 kIUA in total);
- the high flux components for the divertor are assumed to be provided by three, or more industries (65 kIUA in total);
- a few of the packages require only one responsible supplier per package (joined by industrial subcontractors): assembly of the machine, (50 kIUA), reinforced concrete buildings (312 kIUA), the steel frame buildings (69 kIUA) – all activities on ITER site and thus for the Host industries – the cryostat manufacturing (76 kIUA), the cooling water piping (60 kIUA), and the steady state electrical power network (40 kIUA) which all require also a large amount of work on site;
- all other packages should be procured through more than one supplier; particular ones provide for heating and current drive systems and diagnostics, and are discussed below.

In summary, the procurement of the machine core (1465 kIUA) could be possibly split into about 40 different contracts of 40 kIUA in average. The procurement of Auxiliaries and Heating and CD systems, except the Concrete Building (800 kIUA) could be split into 40 different contracts of 20 kIUA on average.

6.1.5.4 Heating and Current Drive Systems

The ITER requirements for plasma heating and current drive cannot be satisfied by a single method. Four have been envisaged (NB, IC, EC and LH), which could provide ITER with flexibility of operation. The machine design makes possible the installation of three NB lines, each assumed to provide 16.5 MW of neutral deuterium atoms at 1 MeV. All RF heating systems have been designed to provide 20 MW to the plasma per port. All these systems costs have been estimated per port of injection.

Provisions in the layout of equipments are such that heating and current drive systems procurement can be staged, to reach a maximum of 50 MW of NB (three lines), and 40 MW for any of the three RF methods: in all possible scenario, installed power cannot provide more than $\simeq$ 130 MW and more than 110 MW available at the same time.

The present status of R&D results in any of those heating methods has not achieved the level required to be confident in their assumed performance and therefore in their availability at the start up of ITER operation. An R&D effort in all Parties, more extended and more efficient, is absolutely needed with the highest priority.

For the time being, it is assumed that the start up scenario will use two NB lines (33 MW, 96 kIUA), one equatorial port equiped with an IC launcher (20 MW, 32 kIUA)) and 20 MW of EC power (77.5 kIUA) which can be delivered either through one equatorial port or through 3 upper ports (for NTM stabilisation).

6.1.5.5 Diagnostic System

ITER requires a comprehensive diagnostic system for monitoring in real time the conditions of the different machine components and for measuring the value of the key plasma parameters, in order either to control ITER operations or to increase the understanding in physical phenomena.

The diagnostic system comprises about forty individual measurement systems. The responsibility for design and procurement of these specific systems should be shared by the Laboratories of the Parties which aim at participating in ITER operations through their physicists. The Central ITER Team cannot bear alone this responsibility, even if it should specify all interfaces with the machine and in particular the responsibility of all “generic” packages.

The cost of the diagnostic system has been estimated through:

- seven diagnostic specific procurement packages: A: Magnetic diagnostics – B: Neutron systems – C: Optical systems – D: Bolometry – E: Spectroscopic systems – F: Microwave systems – G: Operational systems.
- and six “generic” packages: N01: In vessel services - N03: Diagnostic port plugs and first closure flanges - N04: Port interspace structures and second closure flanges - N05: Divertor diagnostic components - N06: Ex vessel services - N07: Windows assemblies.

In addition, the cost of the Diagnostic Neutral Beam has been obtained together with the Heating Neutral Beam cost. Other related costs are for Blanket Shield Modules for Diagnostics Ports and the Test Stand in the Hot Cell, which are included in other packages.

Costs of providing the diagnostic system are divided into several components. The cost of diagnostics required for initial operation (startup) is determined separately from the cost of those that are not required until the DT operation (deferred). Included in the startup costs is the cost of the in-vessel and inter-space equipment, and interfaces, for the deferred systems which would otherwise be expensive and time consuming to install later. Further, the diagnostic costing identifies items which would have to be provided during construction under control of the central ITER Team (in-machine items) and those which could be provided by the ITER Parties directly (ex-machine items). Examples of the former could be in-vessel services and wiring, while examples of the latter could be specialised lasers and spectrometers. As in previous costing exercises, the cost of the specialist diagnostic effort to support the design, procurement and implementation of the diagnostic systems is costed separately in PPY: part of this effort will be required early to prepare for consistent plans and interfaces with the central ITER Team.

The procurement of diagnostics for the startup requires 121 PPY from laboratories and 118 kIUA of which 72.2 kIUA are under direct design control of the central ITER team; deferred diagnostics will require in addition 106 PPY and 42.3 kIUA.

6.1.6 ITER Direct Capital Cost

Taking account of the previous assumptions, the total “JCT estimated capital investment” for ITER amounts to 2,755 kIUA; in addition the cost of spares and items needed only a few years after start of operation (full DT operation) amounts to 258 kIUA and is deferred to be supported by operating funds. The present investment cost is only 49.2 % of the previous estimate for the ITER 1998 design, which amounted to 5603 kIUA and 302 kIUA deferred.

6.2 Construction Management and Engineering Support

An estimate of the cost of construction management and support cannot be done without assumptions on the future organisation to execute the construction and the manner of contracting and managing contracts for procurement.

6.2.1 Assumptions

For this purpose, it is assumed that the ITER Legal Entity (ILE), which will be responsible for the management of ITER during its whole life time, will provide a direct and effective line of accountability by incorporating all actors in a single management entity, including:

- an International Team at the ITER site which will have the overall responsibility to meet the project objectives and to ensure the design continuity and coherence.
  For this goal, it will :
    - define the technical specifications for procurement packages, the general QA rules and contracting rules,
    - control, analyse and decide upon design changes or deviations,
    - maintain databases on R&D and manufacturing results,
    - integrate all aspects of ITER: technology and physics - safety and licensing - assembly - CODAC and diagnostics - cost and schedule.

- a National Team, as part of the ILE in each Party, which will manage and follow up the technical content of the procurement contributed by the Party, when the financial and legal contents of the relevant contracts are being taken care of by a Domestic Agency

For this goal, each National Team will :

- adapt technical specifications to its national usages and ensure engineering at detailed level,
- implement QA,
- assure technical control of each domestic supplier contract by a permanent presence in suppliers' premises, and a schedule control by accepting contractual payments according only to work progress.

With these assumptions it is clear that the size of the International Team can be deduced approximately from its functions, but the size of each Party National Team will depend on the level of the Party's contribution to ITER construction, on these specific packages in its contribution, and on the specific national practices in contract management.

Presently, when the respective choices of the ITER Partners in matters of procurement packages are unknown, a global approximation can only be given of the manpower necessary to follow up all the procurement packages, expressed in professional and technical manyears, integrating all National Teams, and taking no account of the possible splitting of procurement contracts between Parties and its relevant increase in manpower for their follow-up.

6.2.2 Estimated Personnel Cost During Construction

To exercise its responsibilities, the International Team will probably include a core management group and a few technical groups, in charge of physics, safety, engineering, assembly, etc.. These groups should be able to ensure technical continuity with the EDA and CTA, and, as construction approaches its end, these groups, suitably increased by personnel from the National Teams who have followed procurements, will eventually be involved in the integrated commissioning and start up of operation of the facilities. The number of professionals of this International Team can thus range from about 80 at the beginning of construction, to about 200 towards the end.

The support personnel for the core management (mostly administrative) and for the technical groups (mostly CAD) is expected to be in number equal to the number of professionals.

Following the assumed schedule (seven years of construction beginning two years after establishment of the ILE – the one year of integrated commissioning is assumed to be the first year of operation) the global manyears for the International Team during construction amounts to 840 PPY and a similar number for support personnel.

The global estimate of professionals and support personnel (clerical, technicians and CAD) for the different National Teams to follow up all procurement contracts in all Parties amounts to about 960 PPY and twice this number for support personnel. The number of professionals is assumed to be about 120-140 in average for six years and decreasing during the last three years of construction, for probably being transferred to the site to participate in the components installation and commissioning.

Assuming the annual cost of one professional and one support staff to be 150 IUA and 75 IUA respectively, the cost estimate for the International Team during construction until the start of ITER operation (integrated commissioning of the whole machine) amounts to 189 kIUA and the global estimate for all the National Teams during the same period amounts about 288 kIUA. Again, these costs are normalised and global, including all Parties; they might not be representative for the specific conditions of each Party. Excluded from the previous numbers are those relevant to diagnostic procurement referred to above.

6.2.3 Possible R&D Cost During Construction

In addition to personnel costs, a certain amount of R&D during construction should always be considered. As already mentioned, R&D for all heating and current drive methods is required with high priority.

The EDA has provided the principle qualification of design solutions to be implemented in ITER. Nevertheless, during the manufacturing of components, proposed process improvements and design changes or unexpected difficulties could require new tests.

Moreover, to achieve in industrial production reliable results and good efficiency in the manufacturing of a large amount of high technology components (e.g. superconductor strands, high heat flux components, etc…), it is probably more efficient to launch, at the chosen industrial firm, a manufacturing R&D before contracting the global procurement. Along a similar line, it is conceivable that some Nb_3Sn coils (TF and CS) should be tested at their operational cryogenic temperature to confirm quality, even if presently the cost/ benefit ratio of technical results to be expected from these tests does not appear high enough.

It is therefore prudent to expect a spending in R&D of 60-80 kIUA during ITER construction.

6.3 Resources for ITER Operation

6.3.1 Project Manpower Costs

Manpower costs of permanent staff on site and cost of extra manpower brought in from time to time to aid in maintenance of the device are costed assuming an average level of 200 professionals and 400 support staff (clerical, technicians and CAD), at 150 IUA and 75 IUA respectively per year; thus the annual personnel cost is about 60 kIUA.

The permanent professional and support staff above are expected to operate and maintain the facility, and support the experimental programme, for example diagnostics, or installation and testing of the blanket modules. However, visitors to the site to conduct experiments (experimentalists or theoreticians) are not included in the manpower cost: their costs are assumed borne by the Parties. This is consistent with including the cost of diagnostics in the construction or operating costs, but not the Laboratories staff having the responsibility of their procurement.

6.3.2 Energy Costs

Electric power costs, which not only include the power required for pulse operation, but also must cover energy consumption during various levels of standby/maintenance of the machine, depend on aspects of the load time profile and on the characteristics of the national electricity network of the host site. Whereas electric peak power to be delivered, and the average power to be made available over a certain time period, place a premium on the cost of energy consumed, the consumption dominates the cost in any well-provisioned site. Thus the electricity costs can be estimated knowing the steady state power levels required in the various operating states, and the fraction of time spent in those states, superimposing the integral of the pulsed power demand when in the plasma operating state.

A typical scenario is shown in Table 6.3-1. The steady state power loads are representative values over the whole life of the plant, but are somewhat uncertain. Similarly the time devoted to maintenance (or conversely to burn/dwell) as also difficult to specify at this stage. For this reason ranges are given, with the implications noted. A unit cost of 0.05 IUA/MWh is used.

Table 6.3-1 Key Features of Electricity Cost Calculation

	Low	Nominal	High
Steady state power loads			
Plasma operating state (POS)	100	100	100
Short-term standby (STS)	70	80	90
Short-term maintenance (STM)	50	60	70
Long-term maintenance (LTM)	30	35	40
Time usage (lifetime average)			
In LTM	0.25	0.375	0.5
In STM	0.5	0.375	0.25
Average annual cost (kIUA)	24.7	29.1	34.0

The most sensitive parameters of this calculation are shown above. The results are not sensitive to the average burn time, or to the details of the operation stages over the years, since the integral burn time is constrained to deliver the nominal fluence. Typical yearly cost variations from the average, at various operation stages, are below 5 kIUA.

6.3.3 Fuel Costs

The ITER plant must be operated, taking into account the available tritium externally supplied. The net tritium consumption is 0.4 g/plasma pulse at 500 MW burn with a flat top of 400 s. During the first 10 years of ITER operation, the total burn duration at 500 MW is planned to be about 0.15 years. The total consumption during the first 10 years is 4.7 kg. The typical tritium receipt, consumption, and site inventory during the first ten years are given in For commissioning the tritium plant, several tens of grams of tritium will be needed. This will be carried out in parallel with hydrogen plasma operation. A small amount of tritium (< 0.01 g) will be burned during the deuterium plasma operation.

The maximum total tritium transportation per year will be about 1.2 kg/year in the first 10 years. Assuming a 50 g transport tritium container, there will be two shipments every month.

Table 6.3-2. During ITER operation, all tritium will be supplied by external sources.

For commissioning the tritium plant, several tens of grams of tritium will be needed. This will be carried out in parallel with hydrogen plasma operation. A small amount of tritium (< 0.01 g) will be burned during the deuterium plasma operation.

The maximum total tritium transportation per year will be about 1.2 kg/year in the first 10 years. Assuming a 50 g transport tritium container, there will be two shipments every month.

Table 6.3-2 Receipt, Consumption and Site Inventory of Tritium

Year	Receipt (kg)	Consumed (kg)	Site Inventory (kg)
1	0.0	0.0	0.0
2	0.0	0.0	0.0
3	0.1	0.0	0.1
4	0.8	< 0.01	~ 0.9
5	0.8	0.3	~ 1.4
6	0.8	0.4	~ 1.8
7	0.8	0.6	~ 2.0
8	1.0	1.0	~ 2.0
9	1.2	1.2	~ 2.0
10	1.2	1.2	~ 2.0

Fuel costs include deuterium and tritium burnt during operation, plus that lost by decay of the inventory (taken as 2 kg) during plant operation. The deuterium cost is negligible at 2 IUA/kg. There is no market for tritium for the quantities required, and thus tritium may have little or no monetary value. Nevertheless a largely hypothetical 10 kIUA/kg for tritium purchase is used. The total tritium received on site during the first 10 years of operation, according to Table 6.2.5.1., amounts to 6.7 kg. The total consumption of tritium during the plant life time may be 16 kg to provide a fluence of 0.3 MWa/m^2 in average on the first wall; this corresponds, due to tritium decay, to a purchase of about 17.5 kg of tritium. This is well within, for instance, the available Canadian reserves.

Therefore the fuel costs are in average 6.7 kIUA per year during the first ten years, and probably 11.5 kIUA per year after.

6.3.4 Capital Improvement, Spare Parts, Maintenance Costs

Analysis of the expected capital costs to keep the ITER facility in the required effective state has shown that very different ratios (annual cost against the initial investment) should be considered for the different systems, going from almost 0% (e.g. magnets) to 10% (for RF power generators and diagnostics), up to 15% (e.g. computers).

Considering all systems, the required maintenance cost amounts to 2.5% of the initial investment, about 70kIUA per year. To this cost should be added investments deferred initially to operation costs and the cost of replacement of the divertor high heat flux components (possibly five times during the plant life time). This leads to a total of 90kIUA per year in average during operation.

6.3.5 Conclusion: Average Annual Operation Costs

In summary, the ITER average annual operation costs amount to about 60 kIUA for the personnel permanently on site, 30 kIUA for the energy consumption, 8 kIUA for the tritium purchase and 90 kIUA for spare parts, maintenance and improvements, i.e. a total average per year of 188 kIUA. Again this value will depend on the ITER site, mostly through the electricity cost (assumed to be 0.05 IUA/MWh), and on the specific arrangement between the Parties on how to support the personnel cost.

6.4 Decommissioning Costs

The technical implementation and schedule have been detailed according to a credible option. Even if the Host Party may consider other options, the global cost to be borne should not change drastically as long as a full dismantling of the machine is not required before the vacuum vessel activity has decayed substantially.

The ITER facility, because of the remote maintenance implemented during operation, offers initially most of the tools, procedures, and even trained staff, to accomplish the decommissioning operations. This capacity is an essential element in keeping their cost down.

The manpower estimate is based on the requirements for the dismantling of the main active parts of the ITER facility only. The non-active parts are not considered, because their residual values are probably higher than their dismantling costs.

For the technical operations and their schedule in two active phases described in Chapter 6.3, the estimated integrated work force over 11 years amounts to about 2,800 manyears. The average cost per manyear is rated at 90 IUA as a mix of different staff categories. In addition, it is assumed that new hardware may be required to replace a few aging tools, or to enhance the hot cell, and radwaste processing efficiency. For this purpose, one third of the manpower cost is put as an AFI, as observed in previous experience.

Other costs (dependent on the Host country) are not included in the present estimate:
- radwaste disposal,
- components and facilities salvage value after dismantling where applicable (e.g. materials below “clearance”),
- non-active parts dismantling and salvage value,
- site restoration,
- financing-related costs, if spending is made at a later stage.

Under the assumptions and limitations listed above, the estimated cost for decommissioning amounts to 250 kIUA for manpower costs and 85 kIUA for possible hardware costs.

6.5 Summary

For reference, a summary of the cost estimates for all phases of ITER plant lifetime is set out in Table 6.5-1 and compared with estimates for the 1998 design. This summary is subject to all the qualifications and considerations outlined above.

In particular the construction cost estimates:
- assume the agreed Site Requirements and Site Design Assumptions, and are thus valid for a generic site,
- include the policy for additional heating and diagnostics procurement,
- exclude costs associated with site hosting, notably, the provision of land, off-site facilities and all service supplies up to the boundary fence,
- exclude items deferred beyond the start of operation.

The average yearly operation cost estimates include permanent staff costs on site to operate and maintain the facility, but exclude the cost of visitors (physicists) to conduct the experimental programme of ITER.

Table 6.5-1 Summary of ITER Cost Estimates

	ITER-FEAT (1) kIUA	Ratio 1/2 %	ITER/1998 Design (2) kIUA
Construction costs			
A) Direct capital cost	**2755**	49.2	**5603**
B) Management and support	**477**	61.2	**780**
C) R&D during construction	**60-80**	≃ 50	**150**
Operation costs (average/year)			
A) Permanent personnel	60	66	90
B) Energy	≃ 30	50	≃ 60
C) Fuel	≃ 8	40	≃ 20
D) Maintenance/improvements	~ 90	50	≃ 180
Total	**188**	54	**350**
Decommissioning cost (total)	**335**	110	**≃ 300**

MAC Report and Advice

MAC Meeting
23 February 2001
Garching

MAC Membership and Meeting Participation:

Participation was accepted as attached in the **MAC Garching February 2001 R&A Attachment 2**.

Agenda:

The Agenda was adopted as attached in the **MAC Garching February 2001 R&A Attachment 1**.

ITER EDA Status

MAC Garching February 2001, R&A-1:

a) MAC noted the Status Report presented by the Director set out in the **MAC Garching February 2001 R&A Attachment 3**.

b) MAC appreciated the efforts of the Director, Joint Central Team, Home Teams and Industrial participants to enable the draft Technical Basis for the ITER Final Design Report to be completed in due time.

Work Program

MAC Garching February 2001, R&A-2:

a) MAC took note of the Task Agreements Status Summary and compiled list of Task Agreements per Party in the **MAC Garching February 2001 R&A Attachment 4**.

b) MAC took note of four new R&D Task Agreements of which credit is not more than 500 IUA per task in Section 2 (1) in the **MAC Garching February 2001 R&A Attachment 5**.

c) MAC took note of six new Design Task Agreements including VHTP for which credit is not more than 500 IUA or 2.5PPY per task in Section 2 (2) in the **MAC Garching February 2001 R&A Attachment 5.**

d) MAC reviewed and supported the modifications of Task Agreements since MAC Moscow June 2000 of which credit changes are more than 500 IUA or

2.5PPY, or more than 20% as presented in Section 1 A (1) and (2) in the **MAC Garching February 2001 R&A Attachment 6**. This review and support constitute Council approval according to the IC-5 ROD 4 (c) 9.

e) MAC took note of the modifications of Task Agreements since MAC Moscow June 2000 of which credit changes are not more than 500 IUA or 2.5 PPY, or not more than 20% as set out in Section 1 B (1) and (2) in the **MAC Garching February 2001 R&A Attachment 6**.

f) MAC took note of cancellation of Task Agreements as presented at Section 2 (1) and (2) in **MAC Garching February 2001 R&A Attachment 6**.

Review Joint Fund

MAC Garching February 2001, R&A-3:

a) MAC noted that the indicative balance of 2000 budget appropriations left at the end of 2000 appears adequate to cover likely needs for expenditure till the end of EDA on the assumption that all Parties pay their outstanding Joint Fund contributions.

b) Having noted the Director's statement on the indicative amounts of unspent 2000 budget appropriations, MAC recommends to the Council to approve the following proposed budget transfers in order to meet the expected patterns of expenditure in the first half of 2001:
   1) in the budget allocation of the EU Agent, $ 60,000 from Other Expenses to Travel and Subsistence;
   2) in the budget allocation of the RF Agent, $ 10,000 from Other Expenses to Travel and Subsistence.

c) MAC noted the status of the US agent for the Joint Fund as attached in the **MAC Garching February 2001 R&A Attachment 7**.

d) MAC recommends to the ITER Council to make adequate procedures to discharge the US agent from Joint Fund responsibilities and to request the Director to inform ITER Council Members when the discharge of the US agent is completed.

Review the Director's Proposal for ITER Meetings

MAC Garching February 2001, R&A-4:

a) MAC reviewed and supported the schedule of ITER Meetings as set out in **MAC Garching February 2001 R&A Attachment 8**. In accordance with the IC-3 ROD 5.2, this review constitutes IC endorsement of the proposals for meetings.

b) MAC noted that the seven ITER Physics Expert Groups are in full operation and the arrangements for continued interaction with US fusion scientists on generic issues of tokamak physics are proceeding smoothly.

Arrangements for termination and wind-up of the EDA

MAC Garching February 2001, R&A-5:

Completion of Task Agreements

a) MAC recommends that Parties should undertake to complete EDA Task Agreements and to share the results in accordance with normal EDA practice. For specific Tasks that might not be completed before the end-date of the EDA, each Party concerned should designate a person responsible for pursuing its Tasks to completion following normal EDA practice. MAC recommends to the ITER Council to endorse this approach as a proposal to the Parties for appropriate action under Section 5 of Protocol 2 in the EDA Agreement.

R&D hardware and facilities

b) MAC took note of the mode of disposition and associated cost sharing for R&D hardware and facilities as set out in **MAC Garching February 2001 R&A Attachment 9**.

Initial discussions on ways to handle data produced on facilities constructed during the ITER EDA and that would be operated beyond the end of EDA

c) MAC recommends to the ITER Council to support the proposal on ways to handle data produced on facilities constructed during the ITER EDA and that would be operated beyond the end of EDA (**MAC Garching February 2001 R&A Attachment 10**) and, accordingly, recommends to the ITER Council to ask each of the Parties, by the next MAC meeting, to designate a person who is responsible for the supervision of both disposition and utilization of the R&D hardware and facilities.

Joint Fund Assets

d) MAC recommends to the ITER Council to approve valuation procedures for ITER Joint Fund property at the end of the EDA (**MAC Garching February 2001 R&A Attachment 11**).

e) MAC recommends to the ITER Council:
   1) to endorse the proposals (**MAC Garching February 2001 R&A Attachment 12**) towards disposition of the ITER Joint Fund, including the establishment, according to Section 5 of Protocol 2 to the EDA Agreement, of an ad hoc body for the exercise of continuing joint responsibilities for the winding-up of the EDA which cannot be completed within the duration of the EDA;

   2) to request the Parties, assisted by the Director and acting through the MAC-CPs, to make necessary preparations to implement the above proposals, including preparation of specific actions/documents to be approved/adopted, following MAC review, at the final meeting of the ITER Council.

Future Meeting:

MAC tentatively decided that the next MAC Meeting would be held in conjunction with the final ITER Council Meeting at the same place.

MAC Report and Advice:

MAC approved the present MAC Garching February 2001 REPORT and ADVICE to the ITER Council.

IC-Toronto 2001
ROD Attachment 7

MAC FURTHER TASKS

1. **Review Work Program**
   - Status of Task Agreements.

2. **Review Joint Fund**
   - Consolidated annual Accounts of the 2000 ITER Joint Fund.

3. **Arrangements for termination and wind-up of the EDA**

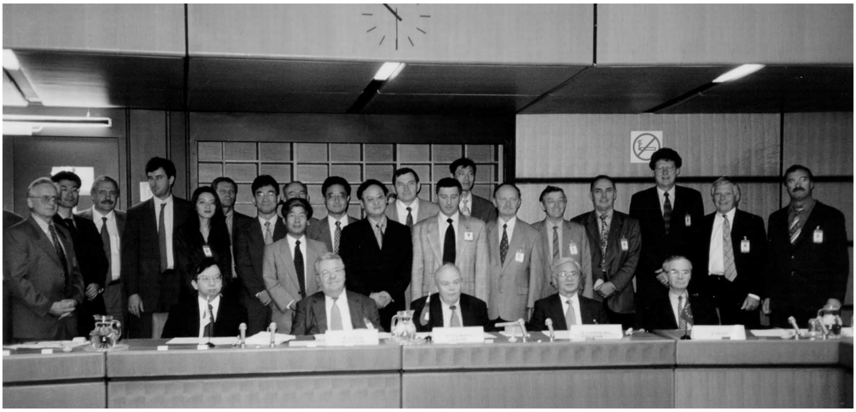
   3.1 R&D hardware and facilities
   - Confirm designation by the Parties of a person for each of the (four) Parties who is responsible for the supervision of both disposition and utilization of the R&D hardware and facilities;
   - Confirmation and revision, if necessary, of its disposition of commingled R&D hardware components after the EDA as agreed in MAC Garching February 2001 R&A Attachment 9.

   3.2 Joint Fund
   - Evaluation of remaining values of Joint Fund property as of July 2001;
   - Provisional assessment of unspent funds as of July 2001;
   - Provisional allocation by Party of Joint Fund assets – property and funds;
   - Preparations for implementation of the proposals (MAC Garching February 2001 R&A Attachment 12) towards disposition of the ITER Joint Fund (to be reported by MAC-CPs). The MAC-CPs should maintain close liaison with the Director in the coming months concerning the disposition of the ITER Joint Fund.
   - Confirm establishment by the Parties of an ad hoc body for the exercise of continuing joint responsibilities for the winding-up of the Joint Fund which cannot be completed within the duration of the EDA.

IC-Toronto 2001
ROD Attachment 8

CPs' FURTHER TASKS

1. To facilitate inter-Parliamentary contact on ITER;
2. In consultation with the Director, to prepare a draft of the Final Report of the EDA for approval at the final IC Meeting;
3. To update and circulate the Tentative Sequence of Events;
4. To co-ordinate preparations of final meetings of the EDA Phase of ITER;
5. To continue exchanging information on matters of public image.



ITER Council Meeting, 18–19 July 2001, Vienna

DOCUMENTS
OF THE
ITER COUNCIL MEETING
18 – 19 July 2001, Vienna

RECORD OF DECISIONS

ITER EDA Council Meeting
Vienna, 18 – 19 July 2001

1. The Council accepted participation as attached **(ROD Attachment 1).**

2. The Council adopted the Agenda **(ROD Attachment 2).**

3. It was a common view of the Parties that the ITER EDA objectives have been entirely accomplished. The Parties representatives commended the IC Chair, IC Co-Chair and the ITER Director for their efforts toward the successful completion of the EDA.

4. The Council took note of the Director's Status Report **(ROD Attachment 3).**

5. The Council noted with great satisfaction the "ITER EDA Technical Activities Report" presented by the Director **(ROD Attachment 4)** and, having heard the positive views of the Parties based on their in-depth domestic assessments, and following discussions, approved, for transmission to the Parties, the Director's report on the ITER design as summarized in the "Summary of the ITER Final Design Report" *).

6. The Council expressed its appreciation to the Director, the Joint Central Team and the Home Teams for fulfilling the task set for them and for the quality and depth of the work completed.

7. The Council took note of the MAC Report and Advice, and accepted its recommendations **(ROD Attachment 5).** With regard to the recommendations on winding-up the EDA, the Council:
    a. took note of the establishment by the Parties of an Ad Hoc Group for the exercise of continuing joint responsibilities for the winding up of the ITER EDA Joint Fund **(ROM Attachment 6)**;
    b. requested the IC Chair to send letters to the IAEA and to the Joint Fund Agents, informing them of the establishment of the arrangements for winding up the Joint Fund and requesting their cooperation with the Ad Hoc Group;

8. The Council took note of the nomination by the Parties (Mr. J.P. Rager for the EU, Mr. S. Matsuda for Japan, and tentatively Mr. Y. Sokolov for the Russian Federation) as senior level persons who should be responsible jointly to resolve issues that are not in the terms of reference of, or that could not be resolved by the responsible persons and the Ad Hoc Group referred in MAC Vienna July 2001 R&A Attachment 08.

9. The Council considered MAC obligations fulfilled and expressed its sincere thanks to MAC Chair and MAC members for their consistent contribution to administration of ITER during the EDA.

---

*) This Report is part of ROD Attachment 4. It is, however, not included in this publication since it was published by the IAEA in November 2001 as a separate book (*Summary of the ITER Final Design Report,* no. 22 of the ITER Documentation Series).

10. The Parties representatives thanked the IAEA for its vital role in the progress of ITER and in the development of fusion in general for the benefit of all mankind.
11. The Council approved the Final Report of the ITER EDA **(ROD Attachment 7).** The Council asked the Contact Persons to complete any necessary editorial work on the document for publication.
12. The Council expressed its acknowledgments to the Contact Persons for their contribution to the success of the EDA.
13. The Council developed a Press Release **(ROD Attachment 8).**
14. The Council asked the Chair and Co-Chair in consultation with the Parties, to send letters of recognition to all individuals who have contributed to the success of the EDA.
15. The Council approved this Record of Decisions.

IC Vienna ROD
Attachment 1

ITER EDA COUNCIL MEETING
Vienna, 18 – 19 July 2001
LIST OF ATTENDEES

EU	Dr. U. Finzi	IC Member	European Commission
	Dr. J-P. Rager	Expert	European Commission
	Mr. M. Drew	Expert, CP	European Commission
	Dr. K. Lackner	Expert, HTL EU	Max Planck IPP
	Dr. J. Campbell	Expert	Canadian Dept. of Nat. Res.
	Dr. P. Barnard	Expert	Iter Canada
	Dr. M. Stewart	Expert	Iter Canada
	Dr. H. Sulimma	Expert	Iter Canada
	Dr. R. Hemmings	Expert	Iter Canada
	Mr. F. Potter	Expert	Govmt. of Prov. of Ontario
JA	Mr. T. Sugawa	IC Member	MEXT
	Dr. M. Yoshikawa	IC Co-Chair, MAC Chair	JAERI
	Mr. A. Nakanishi	Expert	MEXT
	Mr. H. Michigami	Expert	MOFA
	Ms. K. Inui	Expert	MOFA
	Mr. K. Suzuki	Expert	MOFA
	Mr. K. Takeda	Expert	Cabinet Office
	Dr. H. Kishimoto	Expert	JAERI
	Dr. T. Tsunematsu	Expert, HTL JA	JAERI
	Dr. Y. Okumura	Expert, CP	JAERI
	Mr. M. Bourgeot	Interpreter	
RF	Acad. E. Velikhov	IC-Chair	Kurchatov Institute
	Dr. V. Korzhavin	Expert	MINATOM
	Prof. V. Smirnov	Expert	Kurchatov Institute
	Dr. N. Cheverev	Expert	MINATOM
	Dr. L. Golubchikov	Expert, CP	MINATOM
	Dr. K. Chernov	Expert	MINATOM
	Dr. O. Filatov	Expert, HTL RF	Efremov Institute
ITER	Dr. R. Aymar	Director	
JCT:	Dr. Y. Shimomura	Deputy to the Director	
	Dr. P. Barabaschi	PC-w/D	
IC:	Dr. V. Vlasenkov	Secretary	
MAC:	Dr. Y. Miura	Secretary	
IAEA:	Dr. D.D. Sood	Representative	
	Dr. T. Dolan	Representative	
Closing Ceremony:	Dr. M. ElBaradei	IAEA DG	
	Dr. W. Burkart	IAEA DDG NA	
	Dr. U. Knueppel	Head of Delegation of EU	
	Dr. N. Abe	Res. Rep. Of Japan	
	Dr. B.D. Kvock	Dep. Res. Rep. Of RF	
	Dr. I. Goldman	Attaché, US Permanent Mission	

IC Vienna ROD
Attachment 2

ITER EDA Council Meeting

18 – 19 July 2001, Vienna, Austria

AGENDA

Meeting opening
Meeting Arrangements and Attendance
Adoption of Agenda

1. Parties' Status

2. Director's ITER EDA Status Report

3. Director's Report on ITER EDA Technical Activities
    - Director's presentation
    - Parties' reports on their domestic assessments
    - Council approval of the Director's Report on the ITER Design

4. MAC Report and Advice
    - Work Programme
    - Joint Fund Accounts for 2000
    - Arrangements for termination and wind-up of the EDA

5. Final Report of the ITER EDA
    - Decision of the Council

Closing Ceremony
    IAEA DG
    IC Chair
    ITER Director

6. Press Release

7. Approval of Record of Decisions

IC Vienna ROD
Attachment 3

ITER EDA STATUS

report by the Director

1 This note summarizes the progress made in the ITER Engineering Design Activities in the period between the Toronto ITER Council Meeting in February 2001 and July 2001.

Overview

2 The Project has concentrated on the final completion of the entire ITER documentation structure (ITER Final Design Report) in time for the end of the EDA and on the basis of the comments received from the Parties' domestic assessments. The synoptic summary paper of the Final Design Report has also been finalized for distribution to the ITER Council.

3 The top-level technical documents (Plant Design Specifications, Plant Description Document and annexes, Design Requirements and Guidelines Level 1 and annexes) have been completed and a Compact Disc has been prepared for distribution to the Parties at this ITER Council meeting. The remainder of the documentation as well as drawings are also nearly completed and will be delivered to the Parties shortly.

4 The Director has continued to hold discussions with the Home Team Leaders in order to prepare for the integrated organization of the International Team and Participants Teams during the Negotiations (Coordinated Technical Activities, CTA).

R&D Progress

5 The fabrication of the TF Insert was completed at the RF and delivered to JAERI in May 2001. It is now being installed in the CS MC test facility and its testing is expected to start in September. The tests of the TF Insert will be completed by the end of this year. Early next year, the Nb_3Al insert will be installed and tested in the test facility. The whole program is expected to be completed by July 2002.

6 The TF Model Coil has been installed in the TOSCA test facility, and tests will start in July 2001. The testing will be continued throughout the end of this year.

7 During the nine years of the EDA, eight hundred fifteen R&D Task Agreements have been developed and performed successfully under collaboration among the JCT and Parties. The achievements of the R&D tasks are summarized in the ITER Technology R&D and it will be published in July 2001 in the journal, Fusion Engineering and Design (ref. ITER Technology R&D, ITER JCT and Home Teams, Fusion Engineering and Design 55 (2001) 97 – 358). The R&D results support the design of ITER and give confidence in its feasibility. International collaboration effectiveness was also tested during these R&D efforts with a very constructive outcome.

Safety

8 The 2nd meeting of the ITER Parties' Designated Safety Representatives took place in Tokyo, 21-22 May 2001. Discussions focused on progress in preparation for siting including regulatory aspects, scope of regulatory examination and required documentation, and quality assurance, codes and standards to be applied to ITER. It was agreed that the international nature of ITER must be respected which will require a level of understanding by all project participants of the regulations and process to be applied to ITER at any site. The scope of the generic safety approach and the Generic Site Safety Report appears to be a reasonable basis for potential host teams to prepare for the regulatory application tailored to the specific site, and the need was confirmed for site specific interaction between the Design Authority and those potential host teams.

9 Also at this meeting, it has been agreed that proper activities for quality assurance (QA) must be executed, referring to ISO standards and other well-recognised standards. This QA programme needs to cover all interfaces between the ILE and its regulator for safety important items and between the ILE and its suppliers for all items. Further meetings were proposed.

Joint Central Team and Support

10 The status of the Team at the start of July 2001 is summarised in Table 1 below. From the last ITER Council meeting, there have been small changes in the number of staff and their distribution among Parties: one Canadian left Garching and ITER being replaced with another one in Naka, three Japanese left the JCT (one from Garching, two from Naka) with an additional one joining the team in Naka.

Table 1: JCT - Status by Joint Work Site and Party at 1 July 2001

	Garching	Naka		Total
by Site	43[1]	50		93[1]
	EU	JA	RF	
by Party	37[1]	31	25	93

[1] includes two Canadians provided through the Canadian association with the EU Party.

11 The JCT numbers have been supplemented by VHTP’s (~3-4 PPY from RF, and ~4-5 PPY from EU, in average this last year) and other temporary attachments to the JCT.

12 The estimated cumulative PPY effort on site to 1 July 2001 is shown in Table 2 below by JWS and by Party.

Table 2: PPYs JCT on-site to 1 July 2001

	Garching	Naka	SD		Total
by Site	350	383	266		999
	EU	JA	RF	US	
by Party	335	293	185	186	999

13 The Visiting Home Team Personnel scheme continues to function well as a means to enhance JCT/Home Team interaction and to offer some flexibility. As noted at IC-10, some VHTPs are being assigned by the RF in lieu of further JCT secondments. They are therefore accorded zero PPY credit and are noted in the RF contribution to JCT resources.

Table 3 JCT - Cumulative VHTP effort (PPY) from July 1998 to July 2001

	EU	JA	RF	US (to 7/99)	Total
Garching	4.95	0	3.0	-	7.95
Naka	4.17	5.5[1]	3.5	0.0	13.17
San Diego (to 3/1999)	0		0.0	-	0
Total	9.12	5.5	6.5	0	21.12

Notes:

1 includes two visiting scientists

Table 4 PPY credits for VHTP tasks up to July 1998

EU	JA	RF	US	Total
16.29	2.61	0.5	2.11	21.51

Task Assignments

14 The status of existing Task Assignments is set out in a separate Paper to the current Meeting. The total resource contributed by three Parties for the period July 1992- July 2001 are 549,094 IUA for technology R&D and 771.81 PPY for design tasks. Including the US contribution up to July 1999, The total become 657,284 IUA and 942.52 PPY.

Table 5 R&D Task Agreements Summary per Party

Party	R&D resources 7/1992 - 7/2001 (IUA)	Design resources 7/1992 – 7/2001 (PPY)
EU	231,841	290.68
Japan	224,650	267.38
Russia	92,603	213.75
Total	549,094	771.81
US*	108,190	170.71
Grand Total	657,284	942.52

* US contribution up to 7/'99

15 The task status is summarised as of 13 July 2001. The six hundred Technology R&D Task Agreements have been completed or are to be completed with receipt and/or acceptance of their Final Reports. For Design Task Agreements, the works have been completed and their results were incorporated to the ITER Final Design Reports although some final reporting is still outstanding. An overall total of five hundred thirty one have been completed and concluded. Tables 6 below summarises the status of R&D and Design Task Agreements. More details and commentary have been presented in specific papers to MAC. Table 6 covers the numbers of Task Agreements over the entire period of the ITER EDA.

Table 6. Number of Task Agreements (cumulative)

	R&D Number	Design Number
Task Agreements committed (EU,JA,RF)	642	531
Task Agreements completed	594	531
Final reporting to be completed	48	0
US (to 7/99)	*173*	*162*

Joint Fund

16 Discharge of the US Joint Fund Agency: the IAEA had confirmed receipt from the US agent SAIC of the residue of Joint Fund currency remaining after completion of its close-out activities, in accordance with the Council's decision on recommendation of MAC (IC Toronto ROD 4.2). The procedure to discharge SAIC from its responsibilities as ITER Joint Fund Agent in the USA is complete.

17 Detailed information on the JF has been provided to MAC under a separate paper.

18 Summarizing the Joint Fund from January 1994 to the end of the EDA, the Parties have contributed $14.56M in total, of which the three current ITER Parties each contributed $3.93M and the USA contributed $2.97M.

19 The RF Joint Fund Agent is pursuing support contracts which provide an efficient contribution to the JCT design effort. In 2001 the technical scope of contracts has been defined with four Institutes to the value of $667K. They will be implemented according to the funds available. The list of contracts concluded or planned after January 2001 is attached in Annex 1.

ITER Physics

20 The seven ITER Physics Expert Groups are still in full operation and the arrangements for continued interaction with US fusion scientists on generic issues of tokamak physics focussed by ITER are proceeding smoothly. A new framework, called International Tokamak Physics Activity (ITPA), is being planned under the auspices of IFRC of IAEA. It was proposed that the new framework after July 2001 should have a structure and membership similar to ITER Expert Groups with additional participation of US physicists.

Annex 1 RF Design support contracts - 2001

Ref.No.	Title/Dates/Volume	Institute	Financial status
01-01	**Magnet Design** Start - 1 January 01 Final Rep.- 20 July 01	Efremov Institute	**$66,000**
02-01	**Mechanical Design of components of ITER NB injectors** Start - 1 December 00 Final Rep. - 30 June 01	Kurchatov Institute	**$24,000**
03-01	**Steady State Electrical Power Network** Start - 1 January 01 Final Rep. - 30 June	VNIPIET	**$7,500**
04-01	**VV Thermal Shields** Start - 1 Jan.01 End - 30 June 2001	Efremov Institute	**$37,500**
05-01	**Capability of Correction Coils to Reduce Error Fields and to Stabilize Resistive Wall Mode** Start - 1 Jan. 01 End - 30 June 01	Efremov Institute	**$12,000**
06-01	**Effect of Vacuum Vessel Ferromagnetic Inserts on Toroidal Field Ripple** Start - 1 Jan. 01 End - 30 June 01	Efremov Institute	**$3,000**
07-01	**3D Magnetic Analysis for NB System** Start - 1 Jan 01 End - 30 June 01	Efremov Institute	**$12,000**
08-01	**Cold Trap Design** Start - 1 Jan 2001 End - 20 July 2001	Efremov Institute	**$15,750**
09-01	**Dust Deposition in Groove on the Machine First Wall** Start - 1 January 2001 Final - 30 June 2001	Efremov Institute	**$3,000**
10-01	**Divertor Cassette Gas Seals** Start - 1 January 2001 Final - 30 June 2001	Efremov Institute	**$3,750**

11-01	**Divertor Cassette Dynamic Analysis** Start - 1 January 2001 Final - 20 July 2001	Efremov Institute	**$16,500**
12-01	**Fatigue Life and Crack Propagation of Irradiated Heat Sink** Start - 1 January 2001 Final - 20 July 2001	Efremov Institute	**$6,000**
13-01	**Effect of residual stresses on the performance to W to CuCrZr joint** Start - 1 January 2001 Final - 20 July 2001	Efremov Institute	**$9,000**
14-01	**Vacuum Pumping and Fuelling** Start - 1 January 2001 Final - 30 June 2001	Kurchatov Institute	**$7,500**
15-01	**Design and Analysis for the Shield Blanket Module and Limiter** Start - 1 February 2001 Final - 18 May 2001	ENTEK	**$19,500**
16-01	**Vacuum Vessel Design, Analysis and Assessment** Start - 1 January 2001 Final - 20 July 2001	Efremov Institute	**$79,500**
17-01	**Structural Analysis of Torus Cryopump** Start - 1 March 2001 Final - 30 June 2001	Efremov Institute	**$7,500**
18-01	**CuCrZr Minimum Properties and Fatigue Lifetime for Primary FW and Limiter** Start - 1 Febr. 2001 Final - 30 April 2001	Efremov Institute	**$10,500**
19-01	**Physics Design** Start - 1 April 2001 Final - 20 July 2001	Kurchatov Institute	**$90,000**
20-01	**Study of the ITER-FEAT TF ripple, error and correction coils** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$18,000**
21-01	**Steady-State Power Distribution** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$15,000**

22-01	**Diagnostic Neutral Beam Injector** Start - 1 April 2001 Final - 20 July 2001	Kurchatov Institute	**$15,000**
23-01	**Magnet Design Tasks** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$89,850**
24-01	**Thermal Shields** Start - 1 March 2001 Final - 30 June 2001	Efremov Institute	**$19,500**
25-01	**Cryoplant and Cryodistribution** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$18,000**
26-01	**Pellet Flight Tube Modelling** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$9,000**
27-01	**Design Assessment of the Baffle Region Modules** Start - 1 April 2001 Final - 20 July 2001	ENTEK	**$7,500**
28-01	**Coolant Flow Hydraulics of Module** Start - 13 April 2001 Final - 18 May 2001	ENTEK	**$6,000**
29-01	**ITER Tokamak Seismic Analysis** Start - 1 April 2001 Final - 20 July 2001	Efremov Institute	**$9,000**
30-01	**Diagnostic Design** Start - 1 March 2001 End - 20 July 2001	Kurchatov Institute	**$30,000**

IC Vienna ROD
Attachment 4

ITER EDA TECHNICAL ACTIVITIES REPORT

Report by the Director

Introduction

July 2001 marks the end of the nine-year period of the ITER Engineering Design Activities (EDA) Agreement. Under the terms of the Agreement the ITER Parties undertook to conduct jointly the EDA to produce a detailed, complete, and fully integrated engineering design of ITER and all technical data necessary for future decisions on the construction of ITER. This report provides a compendium of the results of the technical work performed in the frame of the Agreement over that period.

Evolution of the ITER Design

During the first six years of the joint work, originally foreseen under the ITER EDA Agreement, a design for ITER has been developed fulfilling the overall programmatic objectives and detailed technical objectives as well as within the cost target adopted by the ITER Parties in 1992. During this period the progress of the design and supporting technical activities has been presented through the ITER Council to the Parties in a number of major milestone reports.

When they accepted the 1998 report, the ITER Parties, anticipating the Agreement to extend the period of the EDA by three additional years[1] and recognising the possibility that they might be unable, for financial reasons, to proceed to the construction of the then foreseen device, established a Special Working Group (SWG)[2], and charged it to "*propose technical guidelines for possible changes to the detailed technical objectives and overall technical margins, with a view to establishing option(s) of minimum cost still satisfying the overall programmatic objective of the ITER EDA Agreement*".

In reporting on the this task, the SWG[3] proposed revised guidelines for Performance and Testing Requirements, Design Requirements, and Operation Requirements, noting that *"preliminary studies ... suggest that the direct capital costs of ITER can be reduced significantly by targeting the less demanding performance objectives recommended..."* and expressing the view that *"these less demanding performance objectives will satisfy the overall programmatic objectives of the ITER Agreement*

---

[1] Text of the Agreement extending the EDA Agreement, IC 1998 proceedings, IAEA ITER EDA documentation series no. 15, Vienna 1998, p102

[2] SWG Charter, ibid, p108

[3] ITER Special Working Group Report to the ITER Council on Task #1 Results, ibid, p148

even though these performance objectives are necessarily less than those that could be achieved with the present [1998] *design."*

Furthermore, the SWG expressed the unanimous opinion that the world programme is *"scientifically and technically ready to take the important ITER step"*. The Parties through the ITER Council subsequently endorsed this viewpoint.[4]

On the basis of the revised performance specifications adopted by the ITER Council in June 1998[5], system studies indicated a domain of feasible design space, with aspect ratios in the range 2.5 to 3.5 and a major radius around 6 m, able to meet the technical guidelines and objectives, with a shallow cost minimum across the aspect ratio range.

To provide a basis for rigorous exploration and quantification of the issues and costing, representative design options that span an appropriate range of aspect ratio and magnetic field were selected for further elaboration and more comprehensive consideration. A task force involving the JCT and the HTs met during 1998 and 1999 to analyse and compare them.

The Task Force recommendations were instrumental in developing consensus on the criteria and rationale for the selection of major parameters and concepts as the precursor to converging and integrating the various considerations into a single coherent outline design which is described in the annexed 'Summary of the ITER Final Design Report - July 2001'.

In January 2000, the ITER Meeting[6] *"accepted the ITER-FEAT Outline Design Report, taking note of the TAC Report and recommendations and agreed to transmit the report to the Parties for their consideration and domestic assessment"*. The Parties assessments were overwhelmingly positive in their endorsement of the outline design, and the process of assessment by the Parties offered the opportunity to further tune the design taking into account their recommendations. The governing body of ITER subsequently approved the design in June 2000 (Moscow ITER Meeting), recognising it as a single mature design for ITER consistent with its revised objectives.

In February 2001, the ITER Council (Toronto) "*accepted the Draft Summary ITER Final Design Report"* and *"agreed to transmit the Draft Summary ITER Final Design Report and supporting technical basis to the Parties for their consideration and domestic assessment*". The subsequent domestic assessment, which took place in all Parties, led to comments, in April 2001, that have been considered in the design and supporting documentation as summarized in the annexed report.

Resources used during the ITER EDA

Under the terms of the Agreement, work on the ITER design has been shared among the personnel resources of the Joint Central Team situated at three Joint Work Sites, up to the beginning of 1999, and two Joint Work Sites subsequently, supported by

---

[4] "ITER Council Proceedings: 1999", ITER Documentation Series No 17, IAEA, Vienna, p11

[5] ITER Final Design Report, Cost Review and Safety Analysis, "ITER Council Proceedings: 1998", ITER Documentation Series No 15, IAEA, Vienna, p148

[6] "ITER Council Proceedings: 2000", ITER Documentation Series No 20, IAEA, Vienna, p7

Host Organisations and the Home Teams of each of the Parties. Validating technology research and development was carried out by the Home Teams.

The Tables below summarise the estimate of the Parties' main EDA resources established for planning purposes early in the EDA and at the beginning of its extension in 1998[7] as well as the actual resources, in total and per Party, expended in each of the main categories during the entire EDA.

Table 1 Planned and Expended Total Resources during the EDA

	Design Resources (PPY)		Technology R&D (kIUA)
	JCT	HT's TA	
Planning estimates	1236	870	925
Resources expended	999	943	658

Table 2 Design and R&D Resources Expended during the EDA

Party	JCT Design Resources	HT Design resources (PPY)	R&D resources (kIUA)
EU	335	291	232
Japan	293	267	225
Russia	185	214	93
Total	813	772	550
US*	186	171	108
Grand Total	999	943	658

* US contribution up to 7/'99

Physics Activities

The Parties' work on ITER Physics has drawn on inputs from within each of the Parties' basic fusion physics programmes. The contributions were provided on a voluntary basis focused and coordinated through a network of seven Expert Groups under the general coordination of the ITER Physics Committee. The ITER physics framework has proven to be a most effective means for focusing on key issues of fusion physics and for coordinating and collating results finally leading to the joint conception of the "ITER Physics Basis"[8]. In this way the ITER physics programme has been a valuable stimulus to progress in all the main areas of magnetic fusion physics.

Local Support from Host Parties

---

[7] "ITER Council Proceedings: 1998", ITER Documentation Series No 15, IAEA, Vienna, p104

[8] "ITER Physics Basis", Nuclear Fusion, Volume 39, Vienna 1999, pages 2137-2664.

From the beginning of the EDA up to early 1999 the Joint Central Team has been located at three Joint Work Sites: Garching (Germany), Naka (Japan) and San Diego (USA). The number of sites was reduced to two with the closure, in early 1999, of the San Diego Joint Work Site. In accordance with the Agreement the Host Parties have provided support to the work sites as defined by the ITER Council consistent with the needs of the JCT. This support has included office accommodation, design office staff and equipment, information technology and administrative support. In addition the Host Parties have provided the necessary support for Workshops and other multi-Party meetings undertaken in the ITER EDA frame.

ITER Joint Fund

The EDA Agreement provides for a Joint Fund to be established to support certain common expense, under financial rules to be established by the ITER Council. The ITER Joint Fund has been in operation since 1994 in the form of an IAEA Trust Fund. Under the agreed financial rules, the Director has been responsible for proposing the Joint Fund budget for each year and for implementing it as approved by Council through designated Joint Fund Agents in each of the Parties. The Joint Fund has been supported by the ITER Parties who have made equal contributions during their active participation in the EDA. From January 1994 to the end of the EDA, the Parties have contributed \$14.56M in total, of which the three current ITER Parties each contributed \$3.93M and the USA contributed \$2.97M.

Significant uses of the Joint Fund have included official travel and subsistence of the ITER Director and senior members of the JCT, items of information technology intended for common purposes at the Joint Work Sites and complementary to those provided under Host Support, specific special common services, and agents' costs of Joint Fund Administration.

One aspect of the special common services supported by the Joint Fund has been design contracts undertaken in specialist design institutes in the Russian Federation in support of and directed by the JCT. Such work, which is quite separate from Design Tasks undertaken by the RF Home Team, has provided a valuable and efficient supplement to JCT effort in certain areas of design work.

Results of the EDA

As required by the Agreement, a total of nine years of technical work under the ITER EDA agreement has resulted in a design that constitutes a complete description of the ITER device and of its auxiliary systems and facilities.

The design has been validated by wide-ranging physics and engineering work, including detailed analyses, experiments in existing fusion research facilities, and dedicated technology developments and tests. Unavoidable uncertainties remain in the extrapolation of performance from current experience to the ITER size and parameters; these can only be fully resolved through experiments at ITER scale. Within the ITER framework, the Parties conduct well-focused physics investigations that strengthen further the physics database, reduce the ranges of uncertainty in extrapolation and explore wider options for possible ITER operation.

The major validating technology R&D projects[9] are now in their concluding stages for the purpose of confirming performance and understanding operating margins. Some activity is foreseen to continue beyond July 2001 in order to broaden operating experience. Assessment of results to date indicates, from an engineering and technology point of view, that the design is feasible, that it can be manufactured to specifications and that it will be capable of meeting its operating objectives.

A comprehensive generic site safety and environment report (GSSR) has been prepared which indicates that ITER will meet the objective of demonstrating the safety and environmental potential of fusion power. The GSSR is a generic report which has been developed so as to be readily extended to meet the specific regulatory documentation needs of possible host Parties. Such an extension remains to be done, in consultation with the relevant authorities of potential host Parties.

A planning schedule for ITER construction and commissioning, operation and decommissioning has been developed and a comprehensive set of project cost estimates has been established, based on studies by the Parties' industries of procurement packages for the supply or fabrication and assembly of all the ITER systems/components. Given appropriate preparations, construction is expected to last about 8 years from the start of on-site construction until first plasma. Total costs for constructing ITER according to the present design are estimated at 2755 kIUA for direct capital costs plus 477 kIUA for construction management/support and ~70 kIUA of R&D during the construction period. Yearly operating costs average about 188 kIUA.

In summary, the annexed final version of the synoptic "Summary of the ITER Final Design Report" *(not included in this publication, but published separately by the IAEA in the ITER Documentation Series, as no. 22)* together with the entire body of supporting ITER documentation and CAD drawings, addresses the Parties comments and marks the achievement of the full technical scope of activities indicated in the ITER EDA Agreement, with a final design which meets the programmatic objective defined in the Agreement and satisfies detailed scientific, technical and costing objectives set by the ITER Council in 1998.

The ITER co-operation, in combination with the continuing general progress in fusion research, has brought its Parties and the world fusion development programme to the point at which they are technically ready and able to proceed to construction, thus bringing to successful fruition the Parties' efforts, investments and aspirations to date. By enabling, in a single device, full exploration of the physics issues as well as proof of principle and testing of key technological features of possible fusion power stations, ITER will provide the integration step necessary to establish scientific and technical feasibility of fusion as an energy source.

The Parties now have at their disposal, in accordance with the purpose of the ITER EDA Agreement, a well founded and robust ITER design that confers a high degree of confidence that it will meet its objectives. All technical data necessary for future decisions on the construction of ITER are now available.

---

[9] Y. Shimomura for the ITER Central Team and Home Teams, ITER Technology R&D, Fusion Engineering and Design 55 (2001), 97 - 358.

MAC Report and Advice

MAC Meeting
16 July 2001
Vienna

MAC Membership and Meeting Participation:

Participation was accepted as attached in the **MAC Vienna July 2001 R&A Attachment 2 *)**.

Agenda:

The Agenda was adopted as attached in the **MAC Vienna July 2001 R&A Attachment 1**.

ITER EDA Status

MAC Vienna July 2001, R&A-1:

a) MAC noted the Status Report presented by the Director set out in the **MAC Vienna July 2001 R&A Attachment 3**.

b) MAC noted that the procedure to discharge SAIC from its responsibilities as ITER Joint Fund Agent in the USA has been completed.

c) MAC expressed its great appreciation to the Director and the members of JCT and Home Teams for their continuous efforts to have performed the ITER Engineering Design Activity smoothly and successfully.

Work Program

MAC Vienna July 2001, R&A-2:

a) MAC reviewed and supported the modifications of Task Agreements since MAC Garching February 2001 of which credit changes are more than 500 IUA or 2.5PPY, or more than 20% as presented in Section 1A (1) in the **MAC Vienna July 2001 R&A Attachment 4**. This review and support constitute Council approval according to the IC-5 ROD 4 (c) 9.

---

*) Attachments not included in this publication

b) MAC took note of the modifications of Task Agreements since MAC Garching February 2001 of which credit changes are not more than 500 IUA or 2.5 PPY, or not more than 20% as set out in Section 1B (1) in the **MAC Vienna July 2001 R&A Attachment 4**.

c) MAC reviewed and supported the modifications of the 2000-2001 CTA for Design to the EU, JA and RF as presented in Section 1C Table 3, 4 and 5, respectively in the **MAC Vienna July 2001 R&A Attachment 4.** This review and support constitute Council approval according to the IC-5 ROD 4 (c) 9.

d) MAC reviewed and supported the cancellation of Task Agreements as presented at Section 2 (1) and (2) in **MAC Vienna July 2001 R&A Attachment 4**. This review and support constitute Council approval according to the IC-5 ROD 4 (c) 9.

e) MAC took note of the status and summary of Task Agreements and compiled list of Task Agreements per Party in the **MAC Vienna July 2001 R&A Attachment 5**.

f) MAC noted with pleasure the publication of "ITER Technology R&D" (Vol. 55, 2001, pages 97-358) to Fusion Engineering and Design and commended those involved in preparing the documents for timely publication.

Review Joint Fund

MAC Vienna July 2001, R&A-3:

a) MAC reviewed consolidated accounts for the ITER Joint Fund Budget of 2000 as presented by the Director (**MAC Vienna July 2001 R&A Attachment 6 (MAC-V06 July 2001)**) with supporting detailed information.

b) On the basis of the information provided, MAC recommends the ITER Council to approve the consolidated annual accounts of the ITER Joint Fund for 2000 and to approve the utilization of unspent funds from the 1999 budget against Joint fund expenditure in 2001.

c) MAC noted that each Party has exercised appropriate oversight of the funds provided to the Joint Fund Agent in its territory.

Arrangements for termination and wind-up of the EDA

MAC Vienna July 2001, R&A-4:

Final Reporting of Task Agreements

a) MAC recommends the ITER Council to approve the final reporting procedure to apply to the specific Tasks for which final report is not available before the end-date of the EDA (**MAC Vienna July 2001 R&A Attachment 7**).

b) MAC recommends to the ITER Council to request the Parties to complete and distribute the outstanding final reports listed in the tables (**MAC Vienna July 2001 R&A Attachment 7**).

c) For this purpose, MAC recommends the ITER Council to take note of the designation by each Party of a person who is responsible for ensuring the preparation and distribution of final reports for the specific Task Agreements. The names of the responsible persons are set out in "Responsible for the final reports for the Task Agreements" of **MAC Vienna July 2001 R&A Attachment 8**.

R&D hardware and facilities

d) MAC took note of the revised tables on disposition and associated cost sharing for R&D hardware and facilities as set out in **MAC Vienna July 2001 R&A Attachment 9**.

e) MAC proposes to the ITER Council to ask the Parties to confirm their intention to make available the assets and facilities established under ITER EDA Task Agreements. Any decision on disposal or other use shall be taken only after consultation with the other Parties.

f) MAC recommends the ITER Council to take note of the designation by the Parties of persons who are responsible for the supervision of both disposition and utilization of the R&D hardware and facilities. The names of the responsible persons are set out in "Responsible for the supervision of both disposition and utilization of the R&D hardware and facilities" of **MAC Vienna July 2001 R&A Attachment 8**.

Joint Fund

g) MAC recommends the ITER Council to approve the estimate of the depreciated values at each Joint Work Site of items of ITER Joint Fund property procured before the end of 2000 (**MAC Vienna July 2001 R&A Attachment 10**).

h) MAC recommends the ITER Council to take note of the provisional status of the ITER Joint Fund at the end of the ITER EDA (**MAC Vienna July 2001 R&A Attachment 11**).

i) MAC recommends the ITER Council to propose the establishment by the Parties of the ad hoc group for the exercise of continuing joint responsibilities for the winding-up of the ITER EDA Joint Fund (**MAC Vienna 2001 R&A Attachment 12**).

j) MAC recommends the ITER Council to endorse the proposed specific responsibilities for the ad hoc group set out in the annex of **MAC Vienna 2001 R&A Attachment 12** with respect to the charge (MAC Garching February 2001 R&A Attachment 12).

k) The names of the member for the ad hoc group are set out in "Ad hoc group for the exercise of continuing joint responsibilities for the winding-up of the ITER Joint Fund" of **MAC Vienna July 2001 R&A Attachment 8**.

l) MAC recommends the ITER Council to initiate the procedure according to the MAC Garching February 2001 R&A Attachment 12. The draft letters are attached to **MAC Vienna 2001 R&A Attachment 12**.

Other Matters

m) MAC recommends to the ITER Council to propose to the Parties to establish a mechanism for jointly addressing other matters as and when they arise in the course of orderly termination.

Acknowledgements:

At the time of closing the last MAC Meeting, MAC expressed its acknowledgements to the guidance of the ITER Council and the contributions of the Director and the members of Joint Central Team in facilitating the successful performance of MAC tasks.

MAC Report and Advice:

MAC approved the present MAC Vienna July 2001 REPORT and ADVICE to the ITER Council.

ITER-EDA Joint Fund Winding-up Ad Hoc Group

Decision of the Parties on proposal of the ITER Council

The **ITER-EDA Joint Fund Winding-up Ad Hoc Group** (the **Group**) is hereby established with the charge to supervise, on behalf of the Parties to the ITER EDA Agreement, implementation after 21st July 2001 of the arrangements to wind up the ITER EDA Joint Fund**,** according to the MAC Report and Advice (MAC Garching February 2001 R&A-5 e)) approved by the ITER Council (IC Toronto ROD 4.1). Specific responsibilities with respect to this charge are set out in the Annex.

In undertaking its activities, the Group shall take due account of the provisions of the ITER EDA Agreement and decisions of the ITER Council as they relate to its charges.

The Group shall end its work when the Parties will have:

1. endorsed its final report on the termination of the Winding-Up of ITER EDA Joint Fund;
2. agreed the final distribution of Joint Fund Assets among the Parties and been notified of its implementation;
3. been notified of the approval of the Final Accounts, and the consequent discharge of the Parties' Agents and of the (former) ITER Director;
4. been notified by the IAEA of closure of the IAEA Trust Fund for the ITER EDA.

Attached are draft letters on ITER Council requests to IAEA and Agents regarding Post-EDA winding-up arrangements of the Joint Fund.

ANNEX

ITER-EDA Joint Fund Winding-up Ad Hoc Group
Specific responsibilities for the winding up and closure of the ITER Joint Fund.

The Group, acting by unanimity for and on behalf of the Parties, shall make an orderly termination of the Joint Fund established under Article 12 (3) of the ITER EDA Agreement. In particular, taking due account of the provisions of ITER Joint Fund Rules, the Group shall:

1. make requests to the IAEA concerning the transfer of funds held in the IAEA Trust Fund to the Joint Fund Agents as defined under Section 2 of the ITER Joint Fund Financial Rules (" the Agents");
2. give instruction to the Agents concerning, inter alia, the authorisation of expenditures and payments, the preparation of final accounts and the disposition of items of Joint Fund Property and residual holdings of joint fund cash;
3. approve the final consolidated accounts of the Joint Fund and consequent discharges of the (former) ITER Director and, through the IAEA, of the Agents;
4. issue a request to the IAEA for the closure of the IAEA Trust Fund for ITER.

The Group may authorise the (former) ITER Director to act on its behalf and subject to its directions in respect of items 1 and 2 above.

DRAFT
Letter from IC Chair Velikhov - IAEA DG ElBaradei

The ITER Council has held its last meeting under the ITER EDA Agreement and was pleased to confirm that the Parties to the Agreement have jointly achieved the Purpose of the Agreement. In so doing they have also demonstrated a capacity to meet the challenges, both technical and organisational, of successfully undertaking a demanding large-scale project at the boundaries of fusion science and technology in an international co-operative framework.

At the request of the ITER Council, I have the honour to convey to you the expression of the Parties' gratitude to the IAEA for the generous assistance that the Agency has given to the ITER process from its inception and, in particular, during the course of the ITER EDA Agreement.

Whilst the technical activities under the ITER EDA Agreement are essentially completed, some administrative matters for winding up the Joint Fund remain to be finalised. For this purpose, on proposal of the ITER Council, the ITER Parties have established the **" ITER-EDA Joint Fund Winding-up Ad Hoc Group "** (the Group), which is charged to supervise, on behalf of the Parties, implementation after 21st July 2001 of the arrangements to wind up the ITER EDA Joint Fund. I attach a copy of the Parties' decision on this matter.

The Group’s work will be the winding up and closure of the ITER Joint Fund arrangements. For this purpose the Group will take over former functions of the ITER Council and Director as they relate to management and operation of the Joint Fund.

May I please ask for the co-operation of the IAEA with the Group in bringing this aspect of the ITER EDA collaboration to a timely and proper conclusion.

I am writing also to the ITER Joint Fund Agents to advise them of the Group’s establishment and charge.

Yours sincerely

Acad. E Velikhov
ITER Council Chairman

DRAFT
Letter from IC Chair Velikhov - ITER Joint Fund Agents

As you may know, the ITER EDA Agreement terminates on 20th July 2001. The ITER Council has held its last meeting under the ITER EDA Agreement and was pleased to confirm that the Parties to the Agreement have jointly achieved the Purpose of the Agreement.

At the request of the ITER Council, I have the honour to convey to you the expression of the Parties' gratitude to the ITER Joint Fund Agents for their efficiency and co-operation in implementing this important aspect of the ITER EDA Agreement.

Whilst the technical activities under the ITER EDA Agreement are essentially complete, some administrative matters for winding up the Joint Fund remain to be finalised. For this purpose, on proposal of the ITER Council, the ITER Parties have established the " **ITER-EDA Joint Fund Winding-up Ad Hoc Group "** (the **Group**), which is charged to supervise, on behalf of the Parties, implementation after 21st July 2001 of the arrangements to wind up the ITER EDA Joint Fund.

The Group's work will be the winding up and closure of the ITER Joint Fund arrangements. For this purpose the Group will take over former overall responsibilities of the ITER Council and Director for the management and operation of the Joint Fund, for instance, in requesting the IAEA to transfer of funds to the Agents, in authorising payments by the Agents and in giving instructions to the Agents on the preparation of final accounts and the disposition of items of Joint Fund Property and residual Joint Fund cash. I attach for your information a copy of the decision of the Parties on this matter.

May I please ask for the co-operation of [name of Agent] with the Group in bringing this aspect of the ITER EDA collaboration to a timely and proper conclusion.

I have also written to the IAEA to advise them of the Group 's establishment and charge.

Yours sincerely

Acad. E Velikhov
ITER Council Chairman

IC Vienna ROD
Attachment 7

FINAL REPORT
of the
ITER ENGINEERING DESIGN ACTIVITIES

Prepared by the ITER Council

CONTENTS

Preface

This is the Final Report by the ITER Council on work carried out under the terms of the Agreement among the European Atomic Energy Community, the Government of Japan, the Government of Russian Federation and the Government of the United States of America on Cooperation in the Engineering Design Activities (EDA) for the ITER. The ITER EDA were conducted under the auspices of the International Atomic Energy Agency (IAEA).

In accordance with Article 16 of the ITER EDA Agreement, this reports follows the approval by ITER Council of the Director's report on the ITER design.

1. Introduction

Thermonuclear fusion is one of the very few options, potentially acceptable from the environmental, safety and economic points of view, to provide energy over the long term for a growing world population. World-wide research efforts have brought the leading programmes to the threshold of conditions that might be expected in a fusion reactor. Together with advances in fusion science arising from a range of smaller experiments and the development of necessary fusion technologies, the fusion programme at the world level has now achieved a level of knowledge that allows to address, in the next step on the development path, the challenge of exploring the physics of a burning plasma in an experimental device that incorporates the key features of fusion technology in reactor-relevant conditions and thus to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes. ITER was conceived to meet this objective.

The history of co-operation on ITER began in 1985 when government leaders in summit meetings called for more substantial international co-operation in order to increase the efficiency and minimize the cost of fusion power development. In response to the summit initiatives, experts from the then European Community, Japan, the then Soviet Union, and the United States, were invited by the Director-General of the IAEA to develop "Terms of Reference Concerning Conceptual Design Activities (CDA) for an International Thermonuclear Experimental Reactor". The CDA began in 1988 after the four Parties formally accepted the invitation from the Director-General to participate, in accordance with the terms of reference, under the auspices of the IAEA. The CDA were successfully completed in 1990.

After completing the CDA, the four Parties entered into negotiations on how ITER should proceed further, resulting in the ITER EDA Agreement [1] under the auspices of the IAEA. The Agreement was signed on July 21, 1992 in Washington by representatives of the four Parties and entered immediately into force. Joint work during the six years period originally set for the Agreement led, by July 1998 to the development of a design for ITER that was adjudged to fulfill the overall programmatic objective and the detailed technical objectives and cost target as originally set.

With the approaching end of the original period set for the EDA, the Parties negotiated an agreement extending the EDA period for three years which was signed by three Parties, Euratom, Japan and the Russian Federation. The USA agreed to participate for one additional year. At that time the Parties, recognising the possibility that they might be unable, for financial reasons, to proceed to the construction of the then foreseen device, adopted a less demanding set of detailed technical objectives [2] that would still meet the overall programmatic objectives but at a significantly reduced cost (about 50%). The EDA extension period was primarily devoted to developing a design under the revised technical objectives set in 1998, as well as completion of planned technology R&D projects. The results of these activities have been reflected in the report of the ITER design presented by the ITER Director and approved by the ITER Council [3].

The terms of the ITER EDA Agreement allowed for the participation of other countries in the joint work of the Parties. Accordingly, Canada participated in the EDA as an associate of Euratom and Kazakhstan participated through the Russian Federation.

2. ITER EDA Objectives

The purpose of the ITER Agreement is to produce a detailed, complete, and fully integrated engineering design of ITER and all technical data necessary for future decisions on the construction of ITER. Such design and technical data shall then be available for each of the Parties to use either as part of an international collaborative programme or in its domestic programme.

2.1 Programmatic Objectives

The overall programmatic objective of ITER is to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes.

ITER would accomplish this objective by demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat-flux and nuclear components required to utilize fusion energy for practical purposes.

2.2 Technical objectives

Detailed technical objectives to determine the best practicable way to achieve the programme objectives were adopted in 1992 by the ITER Council and served as the focus for design work for the first six years of the EDA Agreement. In 1998 revised detailed objectives were adopted by reducing the size of device as well as the capital cost, while maintaining the programmatic objectives and by taking account of the progress in physics and technology R&D. Detailed technical objectives adopted by ITER Council in 1998 are summarized in Table 2-1.

Table 2-1 Summary of Detailed Technical Objectives

Plasma Performance:

- to achieve extended burn in inductively driven plasma operation with a ratio of fusion power to auxiliary power injected into the plasma $Q \geq 10$ with an inductive burn duration between 300 and 500 s,
- to aim at demonstrating steady state operation using non-inductive current drive with $Q \geq 5$,
- controlled ignition should not be precluded.

Engineering Performance and Testing:

- demonstrate availability and integration of essential fusion technologies,
- test components for a future reactor,
- test tritium breeding module concepts; with a 14MeV neutron average power load on the first wall ≥ 0.5 MW/m^2 and fluence 0.3 ≥ MWa/m^2,
- the option for later installation of a tritium breeding blanket on the outboard of the device should not be precluded.

Operation requirements:

- the operation anticipated over an approximately 20 year period should address the issues of burning plasmas, steady-state operation and improved modes of confinement, and testing of blanket modules.

3. Scope of the ITER Engineering Design Activities

The EDA Agreement called for the Parties to conduct jointly the following EDA:

(a) to establish the engineering design of ITER including

(i) a complete description of the device and its auxiliary systems and facilities,
(ii) detailed designs with specifications, calculations and drawings of the components of ITER with specific regard to their interfaces,
(iii) a planning schedule for the various stages of supply, construction, assembly, tests and commissioning of ITER together with a corresponding plan of human and financial resources requirements, and,

(iv) specifications allowing immediate calls for tender for the supply of items needed for the start-up of the construction of ITER if and when so decided;

(b) to establish the site requirements for ITER, and perform the necessary safety, environmental and economic analyses;
(c) to establish both the proposed programme and the cost, manpower and schedule estimates for the operation, exploitation and decommissioning of ITER;
(d) to carry out validating research and development work required for performing the activities described above, including development, manufacturing and testing of scalable models to ensure engineering feasibility; and,
(e) to develop proposals on approaches to joint implementation for decisions by the Parties on future construction, operation, exploitation and decommissioning of ITER.

4. Organization and Resources of the EDA

The EDA were directed and managed by the ITER Council (IC), responsible for overall direction of the EDA and exercising overall supervision of its execution, and the ITER Director, responsible for direction and coordination of the performance of technical activities which were shared among an international Joint Central Team and Home Teams in each of the Parties. The Technical Advisory Committee (TAC) and Management Advisory Committee (MAC) advised the IC on technical and on administrative matters, respectively. Joint Work Sites for the Joint Central Team were set up at three sites, San Diego (US), Garching (EU), and Naka (JA). Activities at the three sites were supported by the respective Parties. In December 1998, the San Diego site was closed following the decision of the USA to withdraw from the project.

The total resources for the conduct of the EDA activities are:

- Joint Central Team amounting to 999 PPY (Professional Person Year) with an additional contribution by the Visiting Home Team Personnel of 43 PPY.
- 531 Design Tasks curried out by the Home Teams amounting to 943 PPY equivalent.
- 815 Technology R&D Tasks carried out by the Home Teams amounting to 658 kIUA [1].
- Physics R&D carried out by the Parties on a voluntary basis using existing plasma devices.
- Joint Fund amounting to 14.56 M$US for the entire EDA period.

---

[1] One IUA is an ITER Unit of Account and is equal to 1000US$ of January 1989.

5. Engineering Design

The ITER design is based on scientific knowledge and extrapolations derived from the operation of world tokamaks over the past decades and on the technical know-how flowing from the fusion technology R&D programmes around the world. The design has been validated by wide-ranging physics and engineering work, including detailed analyses, specific experiments in existing fusion research facilities and dedicated technology developments and tests.

The main parameters and characteristics follow directly from the detailed objectives and cost target set in 1998. Safety and environmental characteristics reflect a consensus among the Parties on safety principles and design criteria for minimising the consequences of ITER operation for the environment and the results of analysis on all postulated events and their consequences.

ITER is a long pulse tokamak with elongated plasma shape and single null poloidal divertor. The nominal inductive operation produces a DT fusion power of 500 MW for a burn length of 400 s, with the injection of 50 MW of auxiliary power.

The major components of the tokamak are the superconducting toroidal and poloidal field coils which magnetically confine, shape and control the plasma inside a toroidal vacuum vessel. The magnet system comprises toroidal field coils, a central solenoid, external poloidal field coils, and correction coils. The toroidal field coil windings are enclosed in strong cases as well as the external poloidal field coils. The vacuum vessel is a double-walled structure. The magnet system together with the vacuum vessel and internals are supported by gravity supports.

Inside the vacuum vessel, the internal, replaceable components, including blanket modules, divertor cassettes, and port plugs such as the limiter, heating antennae, test blanket modules, and diagnostics modules, absorb the radiated heat as well as most of the neutrons from the plasma and protect the vessel and magnet coils from excessive nuclear radiation. The heat deposited in the internal components and in the vessel is rejected out of the vessel by means of the tokamak cooling water system.

The entire tokamak is enclosed in a cryostat, with thermal shields between the hot components and the cryogenically cooled magnets.

During plasma start-up, low-density gaseous fuel will be introduced into the vacuum vessel chamber by the gas injection system. The plasma will progress from a circular configuration to an elongated divertor configuration as the plasma current is ramped up. As the current develops (nominally up to 15 MA), subsequent plasma fuelling (gas or pellets) together with additional heating leads to a high energy gain burn and finally to a controlled burn with a fusion power of about 500 MW. With non-inductive current drive from the heating systems, the burn duration is envisaged to be extended up to 3000 s. In inductive scenarios, before the inductive flux available has been fully used, reducing the fuelling rate so as to slowly ramp-down the fusion power terminates the burn. This phase is followed by plasma current ramp-down and finally by plasma termination. The inductively driven pulse has a nominal burn duration of 400 s, with a pulse repetition period as short as 1800 s.

With regard to safety, the current design focuses on confinement as the overriding safety function, other functions being recognised as being required to protect confinement. Successive barriers are provided for tritium and activated dust. These include the vacuum vessel, active air conditioning systems, with de-tritiation and filtering capability in the building confinement. Effluents are filtered and detritiated in such a way that their release to the environment is as low as reasonably achievable.

The main parameters of ITER are shown in Table 5-1. Figure 5-1 shows a cutaway view of the ITER device inside the cryostat while Figure 5-2 shows a vertical cross section of the tokamak.

Table 5-1 Main Parameters of ITER

Total Fusion Power	500 MW *(700 MW)*
Q — fusion power/additional heating power	≥10
Average 14MeV neutron wall loading	≥0.5 MW/m^2
Plasma inductive burn time	≥ 400 s
Plasma major radius (R)	6.2 m
Plasma minor radius (a)	2.0 m
Plasma current (I_p)	15 MA *(17 MA)*
Toroidal field at 6.2 m radius (B_T)	5.3 T

Note: The machine is capable of plasma current up to 17 MA, with the parameters shown in parentheses, within some limitations on other parameters such as pulse length.

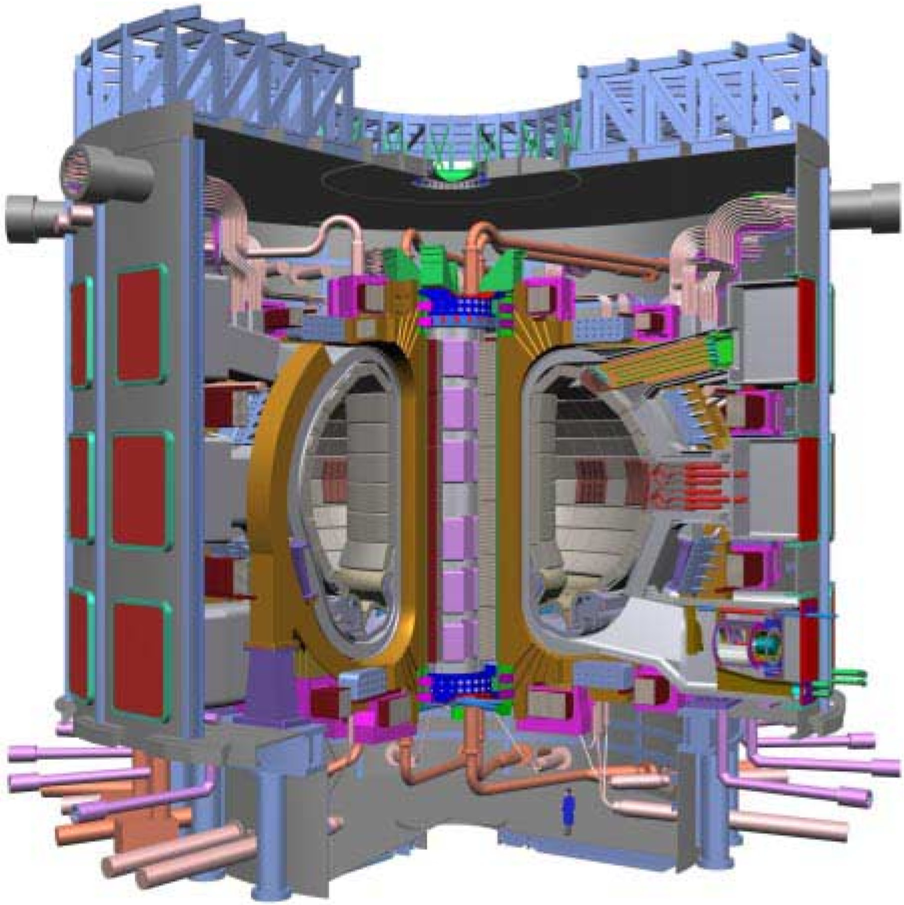


Figure 5-1 View of the ITER device inside the cryostat

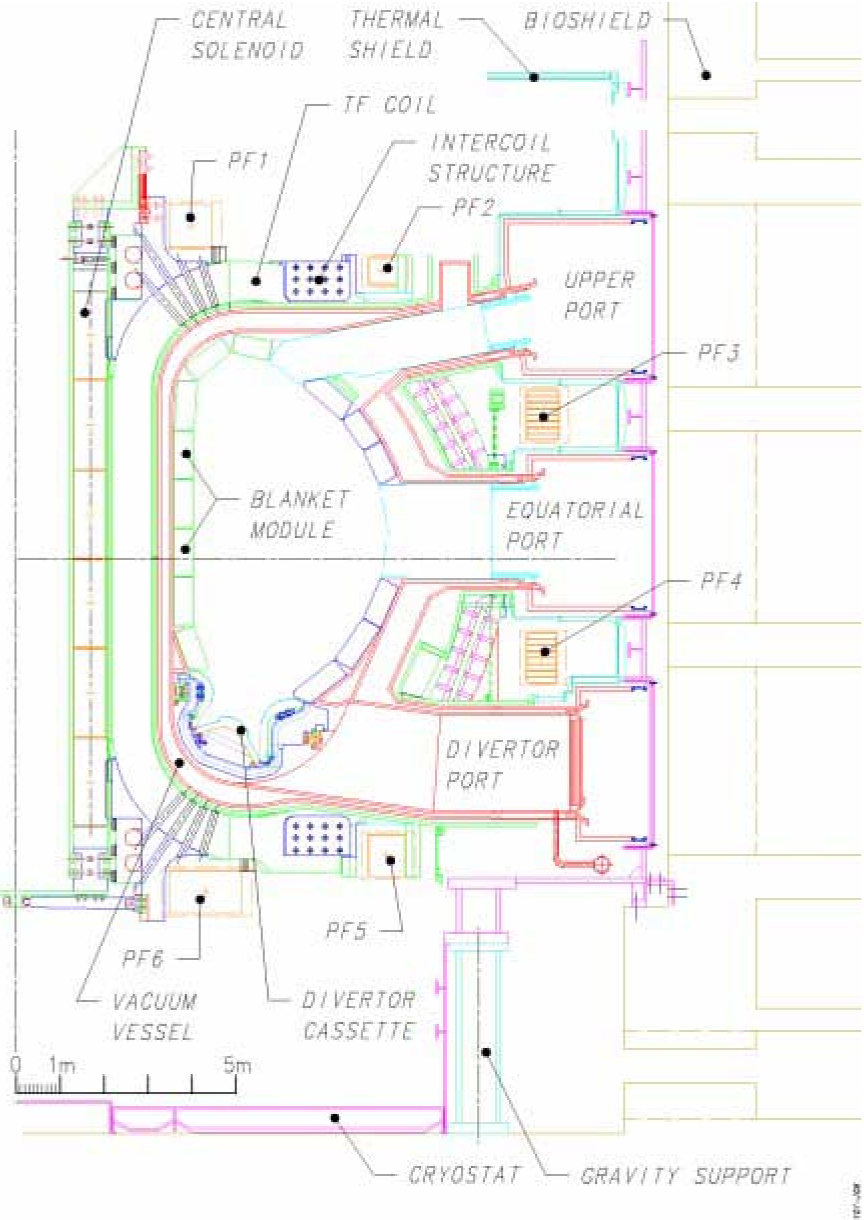


Figure 5-2 Tokamak cross section

6. Validating Research and Development

The overall design philosophy for ITER has been to use established approaches and to validate their application through wide-ranging physics and engineering work. This included detailed analyses, experiments in existing fusion research facilities, and dedicated technology developments including fabrication and test of full scale or scalable models of key components [4]. Unavoidable uncertainties remain in the extrapolation of performance from current experience to the ITER size and parameters; these can only be fully resolved through experiments at ITER scale.

On a voluntary basis, the Parties have conducted well-focused physics investigations that strengthen further the physics database, reduce the ranges of uncertainty in extrapolation and explore wider options for possible ITER operation. The ITER Physics R&D [5] has been carried out by seven Expert Groups, under the coordination of the ITER Physics Committee, as given in Table 6-1.

Table 6-1 Seven Expert Groups on ITER Physics R&D

1994 – 1999	1999 – 2001
Confinement and Transport Physics	Transport and Internal Barrier Physics
Confinement Database and Modeling	Confinement Database and Modeling
Edge Database and Modeling	Edge and Pedestal Physics
Scrape-Off Layer and Divertor Physics	Scrape-Off Layer and Divertor Physics
Disruptions, Equilibrium Control and MHD	MHD, Disruptions and Control
Fast Particles, Heating and Current Drive	Energetic Particles, Heating, and Steady State Operation
Diagnostics	Diagnostics

The design also addresses demanding technical challenges including:

- the unprecedented size of the superconducting magnets and structures,
- high neutron flux and high heat flux in the first wall,
- extremely high heat flux in the divertor,
- remote handling for maintenance and interventions in an activated tokamak structure,
- equipment unique to fusion reactors such as fuelling and pumping heating and current drive systems and diagnostics.

Major technology R&D projects therefore have been undertaken for the purpose of confirming performance and understanding operating margins. Assessment of results to date indicates, from an engineering and technology point of view, that the design is feasible, that it can be manufactured to specifications, and that it will be capable of meeting its operating objectives. In particular, seven large projects, undertaken jointly by the Parties have demonstrated the feasibility, performance, and maintainability of the main engineering components and systems of ITER. About three quarters of the R&D resource were concentrated to these seven projects as shown in Table 6-2.

In addition to the seven large R&D projects, development of components for fuelling, pumping, tritium processing, heating/current drive, power supplies, plasma diagnostics as well as safety related R&D have significantly progressed.

Table 6-2 ITER Technology R&D Areas and Resource Allocation

R&D Area	%
Magnets	27.9
Vacuum Vessel	5.3
Blanket and First Wall including Materials	16.3
Divertor & PFC including Materials	15.1
In-vessel Remote Handling	11.3
Subtotal	*75.9*
Fuelling & Pumping & Tritium System	5.3
Heating & Current Drive	7.9
Diagnostics	2.5
Safety Related R&D	3.4
Miscellaneous	5.0
Total	*100.0*

Note: Total resources: 658 kIUA.

7. Safety and Environmental Assessments

A central goal of ITER is to demonstrate the safety and environmental potential of fusion as an energy source.

A consensus across the Parties on safety principles and criteria for minimising the consequences to the public and the environment from ITER operations has been attained, based on internationally recognised safety criteria and radiological limits following ICRP and IAEA recommendations, and in particular on the concept of defense in depth and the As Low As Reasonably Achievable (ALARA) principle. Safety-related design requirements have been established and assessments have been or are being made to evaluate the success in meeting these requirements in the facility, system, and component design.

Comprehensive safety and environmental assessments of the ITER design and operation have been completed under generic site assumptions. The conclusion indicates that ITER will meet the objective of demonstrating the safety and environmental potential of fusion power.

The ITER Council considers that the safety assessments show that ITER can be constructed and operated without undue risk to health and safety and without significant environmental impact. The analyses and assessments completed with the involvement of experts from all Parties offer a well-developed technical basis for applications to the regulatory authorities of potential host countries for approval to construct and operate ITER.

8. Site Requirements

The ITER Council endorsed a set of ITER Site Requirements and Site Design Assumptions[6] The site requirements are firm in the sense that reasonable reconfiguration of the plant design will not result in a less demanding set of requirements. The requirements cover such matters as:

- land area and geotechnical characteristics,
- water supply and sanitary and industrial sewage capacity,
- heat sink capability,
- energy and electrical power capability for steady state loads,
- transport and shipping capacity.

The requirements have been set so as not to preclude the possibility of ITER construction in the territory of any of the ITER Parties.

The site design assumptions are a set of assumptions that have been made to carry out the ITER design until a siting decision is reached. These assumptions form some of the bases for the ITER construction cost estimate and schedule. They are not compulsory site requirements, but are guidelines for designers to follow.

Requirements for public safety and environmental considerations are, by their nature, site sensitive. Until a site is determined, the ITER plant design process works to a set of safety and environmental assumptions, which are expected to approximate the actual requirements.

The Site Design Assumptions have a wider and more detailed scope than the Site Requirements, and include additional considerations of infrastructure, external hazards and accident initiators and the regulatory environment. In general, the site design assumptions were selected so that the design would not be significantly invalidated by actual site deviations from the assumptions. Deviations from the site design assumptions by the actual ITER site may require design and/or construction modifications. Such modifications are expected to be feasible but they may have an impact on the cost estimate and the construction schedule.

9. Proposed Schedule and Estimates of Manpower and Cost

9.1 Proposed Schedule

The ITER project, after the signing of Joint Implementation Agreement will consist of three phases; construction phase, operation phase and decommissioning phase. It will take about ten years for construction including licensing procedure. Twenty years of ITER operation are envisaged to achieve the project goal. The decommissioning phase comprises, de-activation and clean-up, cooling-down, as well as dismantling and disposal.

Procurement, Construction/Assembly, and Commissioning
The overall schedule for procurement, construction/assembly, and commissioning that leads up to the first hydrogen plasma operation is shown in Figure 9-1. It represents a reference scenario developed on a number of assumptions. The actual schedule that can be realised will depend on the site specific licensing procedure, as well as the organization and arrangements under which ITER would be built.

Since the start of the actual construction on the site depends upon when the license to construct is issued by the regulatory authority, dates in the construction and commissioning plan are measured from the date at which the actual construction work for the tokamak buildings is started. The critical path of the plan starts with the regulatory licensing procedure and construction of the tokamak buildings and, in parallel, the manufacture of magnets and vacuum vessel. If the license process can be completed within 24 months, the construction period defined from the start of the regulatory process to first hydrogen plasma discharge is ten years.

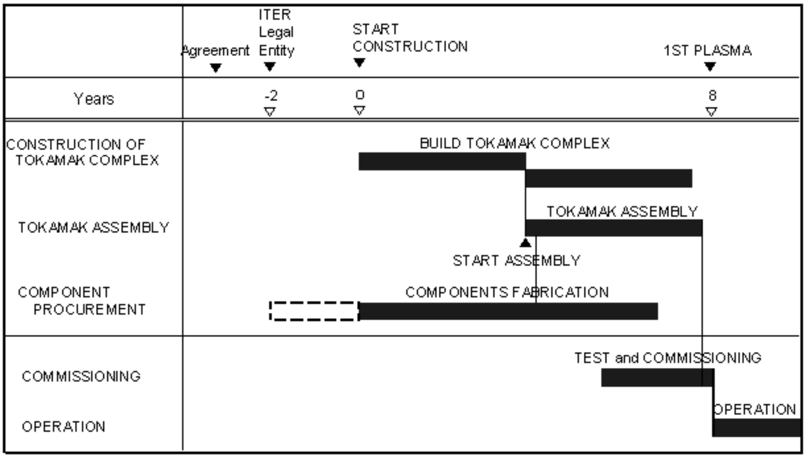


Figure 9-1 Overall ITER Construction Schedule

Operation and Exploitation

After completion of assembly, and following individual sub-system tests, there will be the need, for about one year, for adequate testing of controls and interfaces between subsystems. Thereafter, plan for operations foresees initial phases of hydrogen and deuterium operation in order fully to commission and characterise the machine and to simulate operations in anticipation of the nuclear conditions of full deuterium and tritium (DT) operation.

Two main phases of DT operation are then foreseen. During the first phase, the fusion power and burn pulse length will be gradually increased until the inductive operational goal is reached. Non-inductive, steady state operation will also be developed. Reactor relevant blanket modules will also be tested whenever significant neutron fluxes will be available.

The second phase of full DT operation, beginning after a total of about ten years of previous operation, will emphasise improvement of the overall performance and the testing of components and materials with a higher neutron fluence. This phase will address the issues of higher availability and further improved modes of plasma operation. The implementation and the programme for this phase will be decided on the basis of the results from the preceding operational phases and an assessment of the merits and priorities of programmatic proposals.

Decommissioning

Two separate phases are considered for decommissioning. During the first phase, the machine will, immediately after shutdown, be de-activated and cleaned by removing tritium and activated dust from the vacuum vessel. In-vessel components will also be removed from the vessel and prepared for long term storage according to applicable regulations. These activities will be carried out over a period of about five years using the remote handling facilities and staff in place at the end of operation. After the first phase, the plan foresees a period for radioactive decay before a final dismantling and disposal over a period of about six years.

9.2 Estimated Manpower

The estimate of manpower required for the operation, exploitation and decommissioning of ITER will depend, inter alia, on the organisational structure established for implementing ITER. It has been assumed that an international ITER Project entity will be established with its staff to take responsibility for jointly implementing ITER on behalf of the Parties. There would then be a site-based international team complemented by staff located in each of the Parties who would be responsible to manage and follow the technical content of the procurements provided by the Party. Each would also establish appropriate domestic arrangements to manage the financial and legal aspects of its contributions.

On this basis, the global man-years of the international ITER entity during the construction phase amount to 1800 PPY for professionals and 2760 PPY for support personnel.

Manpower estimates of permanent staff on site during the operation phase indicate an average level of 200 professionals and 400 support staff (clerical, technicians and CAD operators). These staff are expected to operate and maintain the facility, and support the experimental programme. In addition it is expected that the Parties will send visitors to the site to conduct specific experiments on ITER. No estimates have been made of the additional numbers concerned in this way in the experimental exploitation of ITER.

9.3 Cost Estimates

Construction Costs

In order to isolate the cost estimating process from fluctuating economic factors such as variations in exchange rates and domestic inflation rates, cost figures for ITER are expressed in terms of a normalised "ITER Unit of Account (IUA)", where 1IUA is defined as the equivalent purchasing power of US$1000 in January 1989. The conversion factor of 1 IUA to each currency in 2000 is: 1 IUA= 1.39 kUS$, 1.28 kEuro, 148 kYen, and 39.5 kRouble.

In order to provide data for the cost estimates on as realistic a basis as possible, the cost structure of ITER was broken down into 85 "procurement packages" each at a level representing a plausible procurement contract, which were then presented to the Parties industries and large laboratories with relevant for detailed quantified studies. The results of these studies constitute a broadly-based comprehensive database for cost analysis, comparison and evaluation.

The evaluated cost estimate for ITER construction is presented in Table 9-1. The total direct capital cost of ITER amounts to 2,755 kIUA. In addition the cost of spares and items needed only a few years after start of operation (for full DT operation) amounts to 258 kIUA.

Site-specific adaptations of the design may induce changes in the cost of some systems. Similarly, the present design is consistent with codes and standards, which have been defined inside the project. These rules are coherent but are not identical to those of any specific Party, even if they do not contradict them. Regulatory bodies from the host country may request application of different and specific design rules or quality assurance measures. This may also induce cost variations.

Based on the estimated manpower as described above, and assuming the annual cost of one professional and one support staff member to be 150 IUA and 75 IUA respectively, the global cost estimate for the professional and support personnel in the ITER team amounts to 477 kIUA for ten years until the start of ITER operation

In addition to personnel costs, some R&D during construction will still be necessary. For instance, R&D for all heating and current drive methods is required with high priority. Although the EDA has provided the principle qualification of design solutions to be implemented in ITER, during the manufacturing of components, proposed process improvements and design changes or unexpected difficulties could require new tests. It is therefore prudent to expect a spending in R&D of 60-80 kIUA during ITER construction.

Manpower costs of permanent staff on site based on the manpower estimates of an average level of 200 professionals and 400 support staff (clerical, technicians and CAD), amount to about 60 kIUA/year. The costs of Parties' staff visiting the project for specific experiments are assumed to be borne by the Parties concerned.

In addition to personnel costs, 30 kIUA is estimated for the energy consumption, 8 kIUA for the tritium purchase and 90 kIUA for spare parts, maintenance and improvements, i.e. a total average per year of 188 kIUA.

Table 9-1 Summary ITER Direct Capital Cost Estimates

	Direct Capital Cost (kIUA)	Percentage of Total
Machine Core		
Magnet Systems	762.1	28%
Vacuum Vessel	230.0	8%
Blanket and Divertor	241.2	9%
Other Machine Core	231.5	8%
Machine Core, subtotal	**1464.8**	**53%**
Auxiliaries		
Buildings	380.3	14%
Power Supplies & Distribution	214.7	8%
Cryoplant and Cooling Water System	131.5	5%
Other Auxiliary Systems	189.7	7%
Auxiliaries, subtotal	**916.2**	**33%**
Heating and Current Drive	**205.7**	**7%**
Diagnostics, Control and Data Acquisition	**168.0**	**6%**
Grand Total	**2754.7**	**100%**

Decommissioning Costs

Because of the remote maintenance implemented during operation, the ITER facility will offer most of the tools, procedures, and trained staff, to accomplish the decommissioning operations. This capacity is an essential element in keeping down the estimated cost of decommission. Moreover the estimate is based on the requirements for the dismantling of the main activated parts of the ITER facility only. The non-active parts are not considered, because their residual values are probably higher than their dismantling costs.

Other costs (dependent on the Host country) are not included in the present estimate:

- radwaste disposal,
- components and facilities salvage value after dismantling where applicable,
- non-active parts dismantling and salvage value,
- site restoration.

Under the assumptions and limitations listed above, the estimated cost for decommissioning amounts to 250 kIUA for manpower costs and 85 kIUA for possible hardware costs.

Summary

For reference, a summary of the cost estimates for all phases of ITER plant life is set out in Table 9-2.

Table 9-2 Summary of ITER Cost Estimates

	Cost (kIUA)
Construction costs	
Direct capital cost	2755
Management and support	477
R&D during construction	60-80
Operation costs (average/year)	
Permanent personnel	60
Energy	≃ 30
Fuel	≃ 8
Maintenance/improvements	~ 90
Total	188
Decommissioning cost	335

10. Proposals on Approaches to Joint Implementation

A Special Working Group in terms of the EDA Agreement Protocol 2 was set up by the ITER Council in order to develop proposals on approaches to joint implementation for decisions by the Parties on future construction, operation, exploitation and decommissioning of ITER.

The central feature of the proposals is the conclusion of an International Agreement on Joint Implementation of ITER under which the Parties involved would jointly establish and support an ITER legal entity (ILE) which would have the charge, the structure, the authority, and means to implement the project on behalf of the Parties.

The terms of the International Agreement would establish the Parties' political commitment to realising the aims of the project and provide the means, both directly and through subsidiary instruments, necessary to enable ITER to come to fruition. To this end the Agreement and its related instruments would address, inter alia:

- the organisational and managerial structure, including the means by which the Parties exercise the governance and share overall responsibility for the Project
- the place of the ITER site and relations of the project with the host authorities
- the distribution among the Parties of responsibilities for cost sharing and technical contributions to the Project.

Following successful conclusion of the work of the Special Working Group [7], the three current ITER EDA Parties (Euratom, Japan and the Russian Federation) conducted during 2000 non-committal, pre-negotiation discussions, to prepare for negotiations concerning joint implementation of ITER.

11. Conclusions

Upon completion of the ITER Engineering Design Activities, the ITER Council's final conclusions are as follows:

1. The objectives of the ITER EDA Agreement have been fully met: the Parties have at their disposal a complete, detailed and mature design for ITER, with a supporting body of validating analysis and R&D and other technical information, which meets the detailed technical objectives and cost objectives set for it, including those relating to safety and environmental considerations.

2. The ITER co-operation has served to focus the fusion research efforts of the Parties to a common goal and has established a joint capability to undertake successfully tasks that might be beyond the financial or technical capacity of individual Parties.

3. ITER would enable, in a single device, full exploration of the physics issues as well as proof of principle and testing of key technological features of possible fusion power stations. It would provide the integration step necessary to establish scientific and technical feasibility of fusion as an energy source.

In light of these conclusions, the ITER Council, recognising the social importance of the realization of fusion energy:

- considers ITER as the essential tool to achieve this goal,
- affirms a shared single vision of ITER and of the means to realize it,
- considers that the fusion programme at the world level is now scientifically and technically ready to take the important ITER step, and
- reconfirms a common desire to promote construction of ITER through international co-operation.

At a time of increasing global pressure on energy resources and global environmental concerns, the time is ripe to undertake the next step in the development of fusion energy. This will establish fusion as an option for large scale energy supply with intrinsic safety and environmental benefits in the long term.

The ITER Council therefore recommends to the Parties to take the necessary steps to realise a Joint Implementation of ITER as the next step in the development of fusion as a source of energy for peaceful purposes.

References

[1] ITER EDA Agreement and Protocol 1. ITER EDA Documentation Series No 1. IAEA Vienna, 1992. ITER EDA Agreement and Protocol 2. ITER EDA Documentation Series No 5. IAEA, Vienna, 1994. Text of the Agreement Extending the EDA Agreement. ITER Council Proceedings: 1998. ITER EDA Documentation Series No 15. IAEA, Vienna, 1999.

[2] ITER Special Working Group Report on Task 1 Results. ITER Council Proceedings: 1998. ITER EDA Documentation Series No 15. IAEA, Vienna, 1999.

[3] Summary of The ITER Final Design Report, July 2001. ITER Council Proceedings: 2001. To be published.

[4] ITER Technology R&D, Fusion Engineering and Design. Vol. 55 (2001) 97-358.

[5] ITER Physics Basis, Nuclear Fusion, Vol. 39, No. 12 (1999).

[6] ITER Council Proceedings : 2000. ITER EDA Documentation Series No 20. IAEA, Vienna, 2001.

[7] Report of the ITER SWG-P2 on Joint Implementation of ITER, ITER Council Proceedings : 2000. ITER EDA Documentation Series No 20. IAEA, Vienna, 2001.

IC Vienna ROD
Attachment 8

PRESS RELEASE

ITER EDA Council Meeting

Vienna, 18-19 July 2001

Upon the successful completion of the ITER Engineering Design Activities the ITER Council, the governing body of this venture, with representatives of the European Union, Japan, and the Russian Federation, assembled for its last meeting on July 18 and 19 at the Vienna IAEA headquarters.

ITER is an international fusion energy research and development project, being developed through a unique international collaboration, with the goal of taking the next major step in the development of fusion energy as a safe and sustainable energy source for our planet.

Under the auspices of the IAEA, Conceptual Design Activities took place between 1988-1990 followed by Engineering Design Activities that began in 1992 and are now completed. Delegations of the Parties together with representatives of Canada have already initiated preparatory discussions which are anticipated to lead to the joint implementation of the ITER project. Joint coordinated technical work is now underway to define potential sites' specific design adaptations. Following the choice of a site and the commitment by the ITER participants of suitable funds, the construction phase (about 10 years) may start. This would be followed by an operation phase lasting roughly 20 years. ITER will be one of the world's largest scientific and technical collaborations with the involvement of industry.

The demanding technical challenges of the project and its international collaborative nature have led to the breaking of new technical ground in fusion science and engineering. In addition to the technical results, the project has demanded and enabled new modes of closer collaboration between countries in pursuing this large international project.

It was the common view of the ITER Council that, at a time of increasing global pressure on energy resources and global environmental concerns, the time is ripe to undertake the next step in the development of fusion energy. This will establish fusion as an option for large scale energy supply with intrinsic safety and environmental benefits in the long term.