ITER

ITER EDA DOCUMENTATION SERIES No. 18

International Thermonuclear Experimental Reactor
(ITER)

Engineering Design Activities
(EDA)

ITER-FEAT
OUTLINE DESIGN REPORT

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2001

ITER-FEAT OUTLINE DESIGN REPORT
IAEA, VIENNA, 2001
IAEA/ITER EDA/DS/18

Printed by the IAEA in Vienna
March 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.

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

CONTENTS

INTRODUCTION

In July 1998 the ITER Parties were unable, for financial reasons, to proceed to construction of the ITER design proposed at that time, to meet the detailed technical objectives and target cost set in 1992. It was therefore decided to investigate options for the design of ITER with reduced technical objectives and with possibly decreased technical margins, whose target construction cost was half that of the 1998 ITER design, while maintaining the overall programmatic objective.

To identify designs that might meet the revised objectives, task forces involving the JCT and Home Teams met during 1998 and 1999 to analyse and compare a range of options for the design of such a device. This led at the end of 1999 to a single configuration for the ITER design with parameters considered to be the most credible consistent with technical limitations and the financial target, yet meeting fully the objectives with appropriate margins.

This new design of ITER, called "ITER-FEAT" was submitted by the ITER Director to the ITER Parties as the "ITER-FEAT Outline Design Report" (ODR) in January 2000, at their meeting in Tokyo. The Parties subsequently conducted their domestic assessments of this report and fed the resulting comments back into the progressing design. The progress on the developing design was reported to the ITER Technical Advisory Committee (TAC) in June 2000 in the report "Progress in Resolving Open Design Issues from the ODR" alongside a report on Progress in Technology R&D for ITER. In addition, the progress in the ITER-FEAT Design and Validating R&D was reported to the ITER Parties. The ITER-FEAT design was subsequently approved by the governing body of ITER in Moscow in June 2000 as the basis for the preparation of the Final Design Report, recognising it as a single mature design for ITER consistent with its revised objectives.

This volume contains the documents pertinent to the process described above. More detailed technical information relevant to the ITER-FEAT Outline Design is contained in a separate book, "Technical Basis for the ITER-FEAT Outline Design", published by the IAEA, as No. 19 in the ITER EDA Documentation Series. That publication contains two documents: "Technical Basis for the ITER-FEAT Outline Design" and "Progress in Resolving Open Design Issues from the ODR".

ITER-FEAT Outline Design Report

ITER-FEAT — Outline Design Report
Report by the ITER Director

1.0 Background and Introduction

Six years of joint work under the ITER EDA agreement yielded, by July 1998, a mature design for ITER as presented in the ITER Final Design Report, Cost Review and Safety Analysis (FDR)[1] (the 1998 ITER design), supported by a body of scientific and technological data which both validated that design and established an extensive knowledge base for designs for a next step, reactor-oriented tokamak experiment. The 1998 ITER design fulfilled the overall programmatic objective of ITER - to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes - and complied with the detailed technical objectives and technical approaches, and cost target adopted by the ITER Parties at the start of the EDA.

When they accepted the FDR report, the ITER Parties, 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 with two tasks:

- 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,* and
- to *provide information on broader concepts as a basis for its rationale for proposed guidelines, and articulate likely impacts on the development path towards fusion energy.*

In reporting on the first 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 even though these performance objectives are necessarily less than those that could be achieved with the present* [1998] *design."* Consequently, the ITER Council adopted the recommended revised guidelines and asked the Director *"to continue efforts with high priority toward establishing, option(s) of minimum cost aimed at a target of approximately 50% of the direct capital cost of the present design with reduced detailed technical objectives, which would still satisfy the overall programmatic objective of ITER.*"[4].

---

[1] ITER Final Design Report, Cost Review and Safety Analysis, IC-13 ROD Attachment 6
[2] IC-13 ROD Attachment 10
[3] ITER Special Working Group Report to the ITER Council on Task #1 Results, EIC-1 ROD Attachment 1
[4] EIC-1 ROD 3.1

In addressing the second task, the SWG 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[5] 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 like 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."*

Given the instruction to address revised technical guidelines from SWG Task 1 and against the programmatic background of the SWG Task 2 conclusions, the main features of ITER design activities since July 1998 has therefore been:

- the study of options for cost reductions against the new, reduced, technical objectives by reducing plasma performance and technical margins, using the advances in physics and technology understandings, and tools arising out of the ITER collaboration to date, and
- the studied convergence towards a specific single design, following newly adopted guidelines.

As a result, it is now possible to define the key elements of a device, referred to as ITER-FEAT. This report provides the results to date of the joint work in the form of an Outline Design Report on the ITER-FEAT design, which, subject to the views of ITER Council and of the Parties, will be the focus of further detailed design work and analysis in order to provide to the Parties a complete and fully integrated engineering design within the framework of the ITER EDA extension.

A companion paper[6] which documents the Technical Basis to this report was presented to the ITER Technical Advisory Committee for review at its meeting on 20-22 December 1999, in Naka.

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

[6] Technical Basis for the ITER-FEAT Outline Design, Draft for TAC review, 12 December 1999

Plasma Performance

The device should:

- *achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power of at least 10 for a range of operating scenarios and with a duration sufficient to achieve stationary conditions on the timescales characteristic of plasma processes.*
- *aim at demonstrating steady-state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5.*

In addition, the possibility of controlled ignition should not be precluded.

Engineering Performance and Testing

The device should:

- *demonstrate the availability and integration of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance);*
- *test components for a future reactor (such as systems to exhaust power and particles from the plasma);*
- *Test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high grade heat, and electricity production.*

<u>Design Requirements</u>

- *Engineering choices and design solutions should be adopted which implement the above performance requirements and make maximum appropriate use of existing R&D database (technology and physics) developed for ITER.*
- *The choice of machine parameters should be consistent with margins that give confidence in achieving the required plasma and engineering performance in accordance with physics design rules documented and agreed upon by the ITER Physics Expert Groups.*
- *The design should be capable of supporting advanced modes of plasma operation under investigation in existing experiments, and should permit a wide operating parameter space to allow for optimising plasma performance.*
- *The design should be confirmed by the scientific and technological database available at the end of the EDA.*
- *In order to satisfy the above plasma performance requirements an inductive flat-top capability during burn of 300 to 500 s, under nominal operating conditions, should be provided.*
- *In order to limit the fatigue of components, operation should be limited to a few 10s of thousands of pulses*
- *In view of the goal of demonstrating steady-state operation using non-inductive current drive in reactor-relevant regimes, the machine design should be able to support equilibria with high bootstrap current fraction and plasma heating dominated by alpha particles.*
- *To carry out nuclear and high heat flux component testing relevant to a future fusion reactor, the engineering requirements are*

  Average neutron flux $\geq 0.5\ MW/m^2$

  Average neutron fluence $\geq 0.3\ MWa/m^2$
- *The option for later installation of a tritium breeding blanket on the outboard of the device should not be precluded.*
- *The engineering design choices should be made with the objective of achieving the minimum cost device that meets all the stated requirements.*

<u>Operation Requirements</u>

The operation should address the issues of burning plasma, steady state operation and improved modes of confinement, and testing of blanket modules.

- *Burning plasma experiments will address confinement, stability, exhaust of helium ash, and impurity control in plasmas dominated by alpha particle heating.*
- *Steady state experiments will address issues of non-inductive current drive and other means for profile and burn control and for achieving improved modes of confinement and stability.*
- *Operating modes should be determined having sufficient reliability for nuclear testing. Provision should be made for low-fluence functional tests of blanket modules to be conducted early in the experimental programme. Higher fluence nuclear tests will be mainly dedicated to DEMO-relevant blanket modules in the above flux and fluence conditions.*
- *In order to execute this program, the device is anticipated to operate over an approximately 20 year period. Planning for operation must provide for an adequate tritium supply. It is assumed that there will be an adequate supply from external sources throughout the operational life.*

2.0 Revised Objectives

The revised performance specifications adopted by the ITER Council which are set out in full in the above Table, require, in summary:

- to achieve extended burn in inductive operation with $Q \geq 10$, not precluding ignition, with an inductive burn duration between 300 and 500 s, a 14 MeV average neutron wall load ≥ 0.5 MW/m^2, and a fluence ≥ 0.3 MWa/m^2;
- to aim at demonstrating steady-state operation using non-inductive current drive with $Q \geq 5$;
- to use, as far as possible, technical solutions and concepts developed and qualified during the EDA;
- to target about 50% of the direct capital cost of the 1998 ITER design with particular attention devoted to cash flow.

3.0 The approach to an Outline Design

System Studies

As a first approach to identifying designs that might meet the revised objectives, system codes were used which summarise in quantitative form the inter-relationships among the main plasma parameters, physics design constraints and engineering features, and can be combined with costing algorithms.

Such an analysis combines a detailed plasma power balance and boundaries for the plasma operating window, providing the required range of Q for the DT burn, with engineering concepts and limits. Four key parameters — aspect ratio, peak toroidal field, elongation, and burn flux — are intimately linked, allowing options in the systems analysis to be characterised principally by the aspect ratio (A), in addition to the device size, given by the major radius (R). Access to the plasma (e.g. for heating systems) and allowable elongation (simultaneously constrained by plasma vertical position and shape control and by the necessary neutron shield thickness), are functions of aspect ratio.

On this basis the 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 reduced requirements, with a shallow cost minimum across the range. The shallowness of the cost curve and the inevitable approximate nature of the system studies made it clear that no particular choice can be made on the optimal aspect ratio based on estimated costs alone. In addition, there are other important aspects whose cost or performance impact may not be easily factored into a systems optimization.

Study of representative options

In order to provide a basis for rigorous exploration and quantification of the issues and costings, representative options that span an appropriate range of aspect ratio and magnetic field were selected for further elaboration and more comprehensive consideration, as reported to the ITER meeting in Cadarache, March 1999[7].

[7] Study of Options for the Reduced Technical Objectives/Reduced Cost (RTO/RC) ITER, (ROM 1999-03-10 Attachment 8)

The development of specific representative options provided a more tangible appreciation of the key issues and a practical framework for the process of convergence was explored and clarified in a joint JCT/Home Team "Concept Improvement Task Force" constituted in April 1999, following the guiding principles:

- to preserve as far as possible physics performance and margins against the revised targets, and the scope for experimental flexibility, within the cost target and relevant engineering constraints;
- to exploit the recent advances in understanding of key physics and engineering issues to be drawn from the results of the ITER voluntary physics programme and the large technology R&D projects;
- to maintain the priority given to safety and environmental characteristics, using the principles, analyses and tools developed through ITER collaboration to date.

The Task Force recommendations, presented to the Programme Directors' Meeting in Grenoble (July 1999)[8], 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.

Intensive joint work through a JCT/Home Teams "Integration Task Force", has led to a single configuration for the ITER-FEAT design which represents an appropriate balance of the key technical factors and the cost target and the use of the conservative option for the energy confinement scaling.

[8] Study of options for the RTO/RC ITER, Director's Progress Report, ITER Meeting, Grenoble July 1999.

4.0 ITER-FEAT Parameters and Design Overview

The main parameters and overall dimensions of the ITER-FEAT plasma are summarised in Table 4.1 below. The figures show parameters and dimensions for nominal operation; figures in brackets represent maximum values obtaining in specific limiting conditions, including, in some cases, additional capital expenditures:

Table 4.1 Main Parameters and dimensions of ITER-FEAT plasma

Total fusion power	500 MW *(700 MW)*
Q — fusion power/auxiliary 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 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)*

The ITER-FEAT facility comprises the following systems:

- the tokamak itself, consisting of a vacuum vessel and its internal components, a blanket and divertor, and superconducting magnets and associated structure);
- a cryoplant and cryodistribution;
- a pulsed electrical power supply;
- a cryostat and its associated thermal shields;
- a fuelling and exhaust system including an exhaust tritium processing system;
- a cooling water system;
- a plasma measurement (diagnostic) system;
- a heating and current drive system and its electrical power supply;
- buildings and services;

The initial assembly of the tokamak and its remote maintenance are also important elements of the ITER-FEAT design.

A cross-section of the tokamak showing the vacuum vessel, its internal components and its ports, as well as some features of the magnet system and cryostat, is shown in Figure 4.1. Figure 4.2 shows an overall schematic of systems important for normal operation, and Figure 4.3 shows an indicative site layout for the entire ITER-FEAT facility.

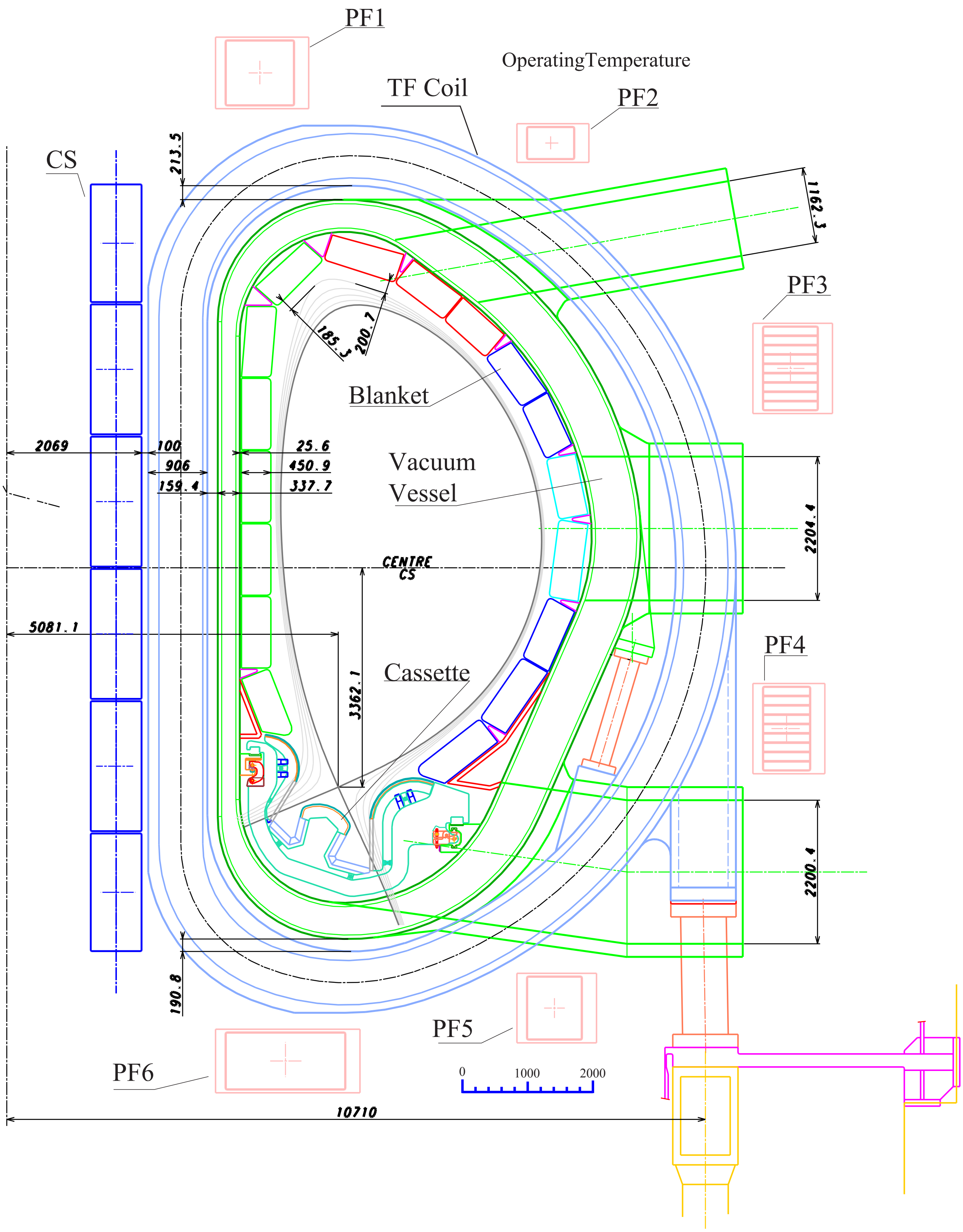


Figure 4.1 Cross-section of the ITER-FEAT tokamak

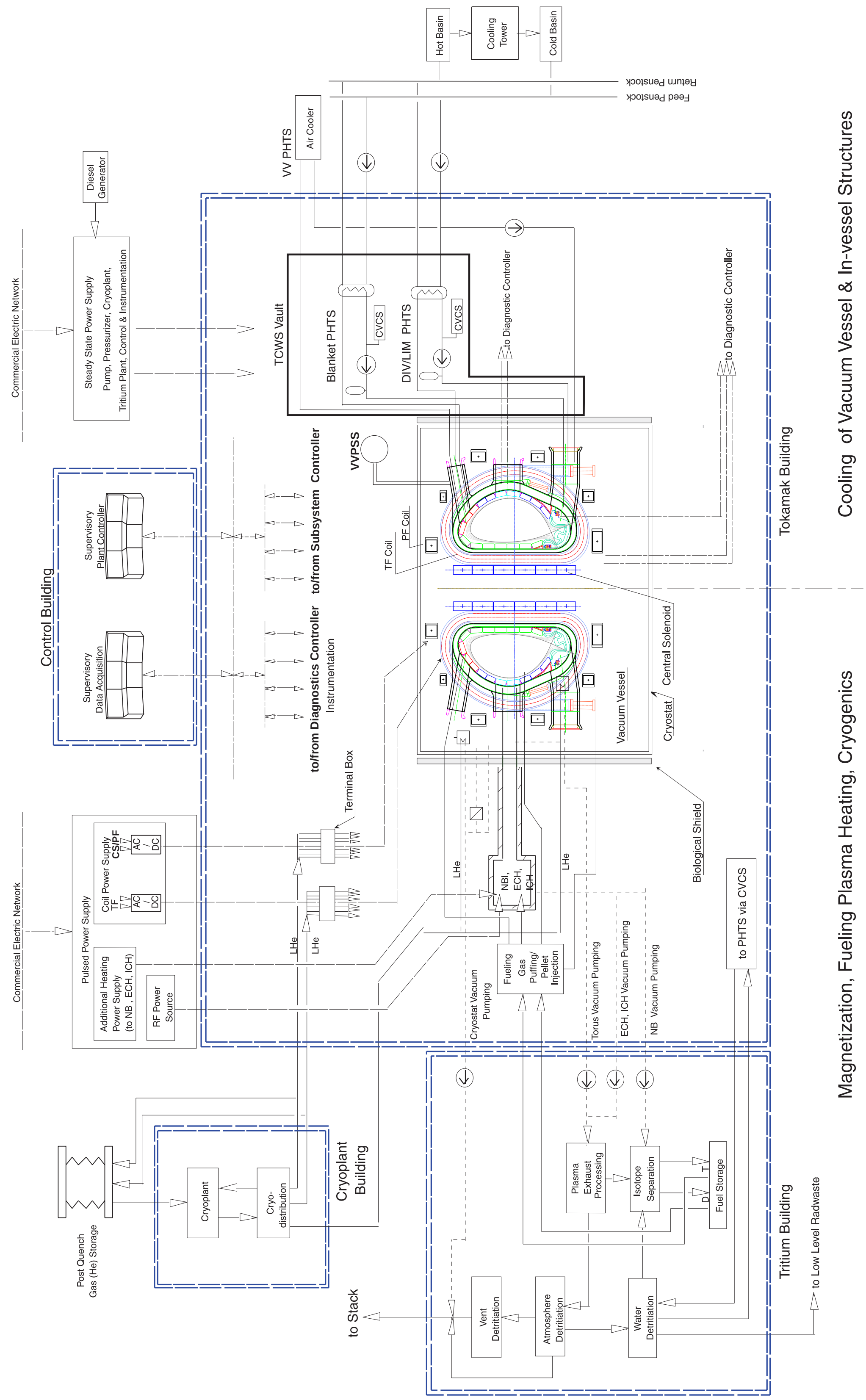


Figure 4.2 ITER-FEAT plant systems diagram

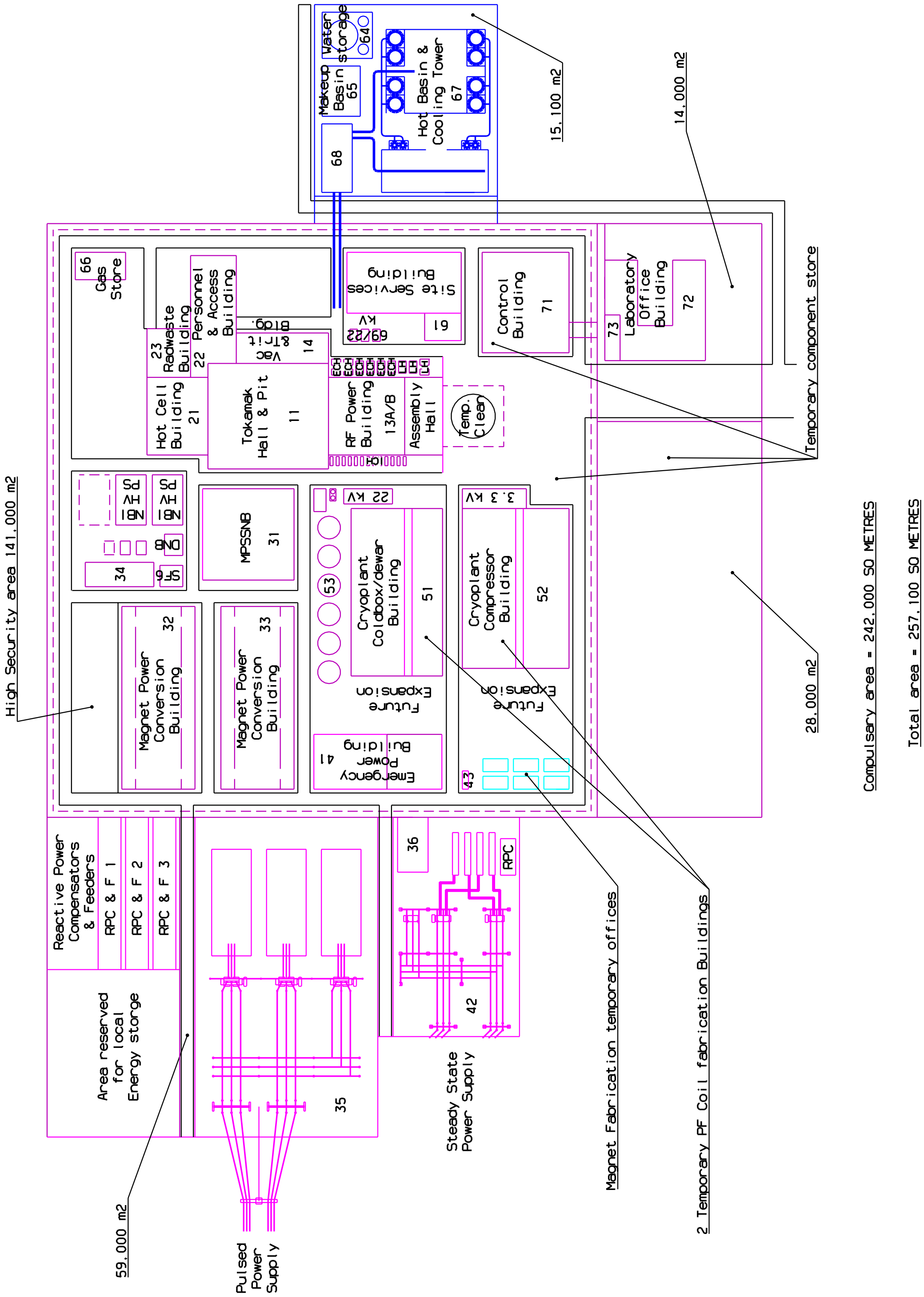


Figure 4.3 ITER-FEAT site and facilities layout

5.0 Physics basis and Plasma performance projections

Overview

The principal physics goals of ITER-FEAT are:

(i) to achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power (Q) of at least 10 for a range of operating scenarios and with a duration sufficient to achieve stationary conditions on the timescales characteristic of plasma processes;

(ii) to aim at demonstrating steady-state operation using non-inductive current drive with a ratio of fusion power to input power for current drive of at least 5.

In addition, the possibility of higher Q operation will be explored if favourable confinement conditions can be achieved.

The reference operating scenario for ITER-FEAT inductive operation is the ELMy H-mode and the rules and methodologies for projection of plasma performance to the ITER scale are those established in the ITER Physics Basis (IPB)[9], which has been developed from broadly-based experimental and modelling activities within the magnetic fusion programmes of the ITER Parties.

The key physics issues relating to plasma performance in the ELMy H-mode regime are:

- the maintenance of H-mode quality confinement at sufficiently high density, achieving adequate plasma β to produce the requisite fusion power, and hence Q value;
- the provision of satisfactory power and particle exhaust to ensure acceptable levels of helium and plasma impurities;
- the evolution of plasma confinement phenomena scaling with size;
- efficient transfer of α-particle power to the thermal plasma while limiting anomalous α-particle losses, via TF ripple or collective instabilities, to prevent damage to the plasma-facing components.

At the same time, global magnetohydrodynamic (mhd) stability and plasma control capability must be such that the thermal and electromagnetic loads, as well as runaway electron currents, arising from disruptions are within acceptable bounds.

H-mode operation at high plasma density is favoured by the choice of a high plasma triangularity and the exploitation of high-field-side (‘inside’) fuel pellet launch, while the overall choice of design parameters allows considerable headroom for Q = 10 operation well below the Greenwald density. Plasma performance predictions show that Q = 10 operation can be achieved at modest values of β_N (~1.5). However, in the event that the β threshold for the onset of neoclassical tearing modes (NTMs) scales unfavourably with size to ITER-FEAT, stabilization of the modes by localized Electron Cyclotron Current Drive (ECCD) is foreseen.

Extensive divertor model validation and analysis activities performed so far during the EDA give confidence that the proposed divertor design allows adequate power dissipation in

---

[9] ITER Physics Basis, ITER Physics Expert Groups et al, Nucl. Fusion, IAEA, Dec 1999

volume to be achieved, with peak time-averaged power loads below the acceptable level of 10 MWm^{-2}, and that the planned fuelling throughput of 200 Pam^3s^{-1} will limit the core helium concentration below 6%.

While the detailed evaluation of α-particle loss processes is still in progress, it is expected that the losses via TF ripple can be brought within acceptable limits by reducing the residual TF ripple level via ferromagnetic inserts. In many respects ITER-FEAT represents a key experimental step in the evaluation of α-particle losses due to collective effects at the reactor scale. Nevertheless, on the basis of studies carried out in support of the ITER FDR design, it appears unlikely that excitation of collective mhd instabilities, such as Alfvén eigenmodes, will limit plasma performance in ITER-FEAT inductive scenarios.

The development of plasma operation scenarios that exploit active profile control to access enhanced confinement regimes, which has occurred in the course of the EDA, has allowed greater emphasis to be placed on the use of such scenarios in ITER-FEAT. In particular, these regimes offer the prospect of establishing reactor-relevant steady-state operation in which a significant fraction of the plasma current is generated via the bootstrap effect. Flexibility in the ITER-FEAT design through plasma shaping, a mixture of heating and current drive systems, and mhd stability control techniques for NTMs and resistive wall modes (RWMs), favours the exploitation of plasma scenarios with either shallow monotonic or negative central shear. Although the precise conditions for the development of internal transport barriers (ITBs) are uncertain, the aim has been to provide ITER-FEAT with the necessary plasma control tools to facilitate access to such modes of operation. Moreover, sophisticated diagnostics of key profiles such as q, pressure, and rotation will be required to operate with a high level of reliability from the first phase of plasma experiments, and this has been acknowledged in assigning measurement priorities. The question of α-particle, losses via TF ripple losses or collective instabilities, is anticipated to be particularly acute in these regimes, and the design of the ferromagnetic inserts will reflect this consideration. Predictions of steady-state operation in ITER-FEAT, therefore, build upon these recent developments and reflect the expectation that considerable further progress can be achieved in the fusion programme in the future to resolve remaining uncertainties.

Physics Basis and Selection of Plasma Parameters

The reference plasma scenario for inductive Q = 10 operation, the ELMy H-mode, is a reproducible and robust mode of tokamak operation with a demonstrated long-pulse capability. The essential physics which enters into the prediction of plasma performance in ITER-FEAT derives from the two principal ELMy H-mode scalings, i.e. the H-mode power threshold scaling, which defines the lower boundary of the device operating window in terms of fusion power, and the energy confinement time scaling. The recommended form for the former scaling is,

$$P_{LH} = 2.84 M^{-1} B_T^{0.82} \bar{n}_e^{0.58} R^{1.00} a^{0.81} \qquad \text{(rms err. 0.268)}$$

in (MW, AMU, T, $10^{20}m^{-3}$, m), with M the effective isotopic mass of the plasma fuel. This scaling expression is based on the latest version of the threshold database (DB3) extended with results from recent dedicated H-mode threshold experiments in Alcator C-Mod and in JT-60U, the latter using the new 'W' shaped divertor. For ITER-like devices, this scaling yields an H-mode power threshold prediction which is approximately a factor of 2 lower than that predicted by an earlier version (IPB98(5)). There is, however, evidence from JET and

JT-60U that the heating power should be 1.3-1.5 times higher than the H-mode threshold to obtain a good H-mode confinement. Therefore, a boundary corresponding to $1.3P_{L\text{-}H}$ is also taken into account in the analysis of performance.

Thermal energy confinement in the type I ELMy H-mode is described by the IPB98(y,2) scaling,

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

where the units are (s, MA, T, MW, 10^{19}m^{-3}, AMU, m) and the elongation κ_a is defined as $\kappa_a = S_o/(\pi a^2)$ with S_o being the plasma cross-sectional area. A comparison of the H-mode thermal confinement times with the scaling for a subset of ELMy data in the ITER H-mode database is shown in Figure 5.1.

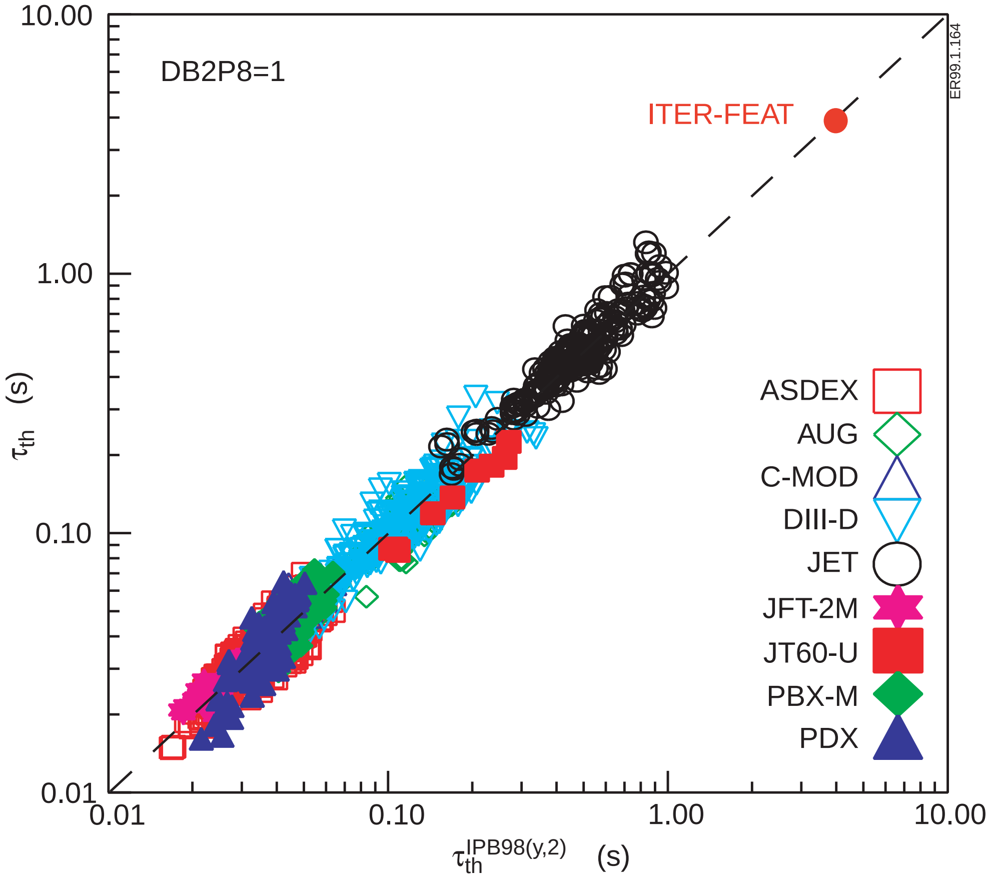


Figure 5.1 Comparison of ELMy H-mode thermal energy confinement times with the scaling expression IPB98(y,2) and scaling prediction for the energy confinement time in a nominal ITER-FEAT Q = 10 discharge.

IPB98(y,2) has been selected as a conservative option from among five empirical log-linear (power law) scaling expressions for the energy confinement time reviewed by the Physics Expert Group concerned. Other projections which include the (ohmic) H-mode data from small tokamaks predict ~ 20% higher confinement for an ITER-like machine.

• The principal mhd stability constraints which contribute to the definition of the device performance relate to the plasma current, elongation, plasma density, and plasma pressure.

1. There is now an extensive energy confinement database for plasmas with $q_{95} \sim 3$, and proven experience in operation with low disruption frequency. A quantitative analysis of disruption frequency on several tokamaks has shown that ITER's goal of achieving an initial disruption frequency of 10% has been attained in existing devices, with no specific problems due to proximity to $q_{95} = 3$. Although recent experiments found no significant degradation of confinement with decreasing q_{95} over the range $2.3 \le q_{95} \le 4$, selection of a lower q_{95} operating point would reduce performance margins (particularly for higher Q operation) and might also impair steady-state capability. Therefore, $q_{95} = 3$ has been retained as an acceptable compromise between good energy confinement and satisfactory mhd stability properties although flexibility to accommodate discharges with higher currents ($q_{95} \sim 2.7$) at reduced pulse length is under study.

2. Plasma shaping capability (elongation and triangularity) derives from a consideration of axisymmetric plasma stability and power required to maintain the plasma vertical position, equilibrium control including inner divertor leg length, limits to the acceptable vacuum vessel forces during a vertical displacement event, and the advantages in confinement which may accrue, both from a higher current capability and from direct dependencies of energy confinement on shaping parameters. The range of issues involved in determining the optimum shaping capability has motivated a reassessment of the shaping parameters for ITER-FEAT.

   An examination of the H-mode global confinement database confirms that the confinement times from JET and DIII-D are consistent with the IPB98(y,2) scaling up to the highest available values of κ_{95} (~ 1.8 at $q_{95} \le 3.5$). Nonetheless, in view of studies that show difficulties in maintaining vertical position control within an acceptable range of PF circuit power and coil voltage if only the passive stabilisation of the vacuum vessel and the active stabilisation action of external poloidal field coils are employed, an elongation of $\kappa_{95} = 1.7$ ($\kappa_X \approx 1.84$) has been selected for the reference parameter.

   Although there is no explicit dependence of energy confinement time on triangularity, the high triangularity of the ITER-FEAT design ($\delta_{95} \approx 0.33$ or $\delta_x \approx 0.49$) reflects several potential advantages:
   - the current-carrying capability of the device, and hence confinement capability, is linked to triangularity through $q_{95;}$
   - recent results from JET, demonstrate that operation at higher triangularity allows high confinement to be maintained at densities close to the Greenwald value, a result which has been confirmed in ASDEX Upgrade.;
   - in steady-state scenarios, where the pressure and current profiles are closely linked, it has been predicted that the β-limit should benefit from higher triangularity.

   One possible disadvantage is that the type I ELM frequency is known to decrease with increasing triangularity (increasing edge shear) and the resultant increase in the amplitude of heat pulses which may be produced by lower frequency ELMs is likely to lead to increased erosion of the divertor target.

3. The choice of an aspect ratio of 3:1 is a compromise between the benefits of lower aspect ratio such as lower magnetic field values and a larger margin relative to the H-mode power threshold and those associated with higher ratios such as higher plasma

densities. Practical considerations of accessibility and of maintaining acceptable margins for equilibrium and vertical stability control, also figure in the judgement.

4. The $\beta^2 B^4$ dependence of fusion power motivates operation at the highest attainable β. However, in recent years, neoclassical tearing modes (NTMs) have been shown to limit the achievable β_N (= $\beta(\%)/[I_p(MA)/a(m)B(T)]$) in ELMy H-mode plasmas to values below the ideal limit, $\beta_N \sim 3.5$, and this instability might occur in the ITER-FEAT target range of $\beta_N \sim 1.5$-2.5, leading to degradation of confinement (or disruptions). A stabilization technique for NTMs based on electron cyclotron current drive is, however, yielding promising results on present experiments and its application is foreseen in ITER-FEAT to allow control of such modes if necessary. Nevertheless, the assumption $\beta_N \leq 2.5$ has been taken as a pragmatic limit for calculations of the ITER-FEAT operating window.

5. Optimum use of the plasma pressure for fusion power production implies that densities in the vicinity of (and, in power plants, perhaps beyond) the Greenwald density ($\bar{n}_{GW}(10^{20}\mathrm{m}^{-3}) = I_p(\mathrm{MA})/\pi a^2(\mathrm{m})$) be attained. Although it has traditionally been difficult to maintain H-mode confinement at densities close to the Greenwald value, experiments at higher triangularity have obtained H-mode quality confinement at 80% of the Greenwald density. In addition, experiments with inside pellet launch and recent experiments with pumping at both the inboard and outboard divertor strike points have sustained H-mode level confinement at densities beyond the Greenwald value. On the basis of these results, the conservative assumption $\bar{n}_e \leq n_{GW}$ is used to limit the density range foreseen for the ITER-FEAT reference regime. In addition, as is below, ITER-FEAT can achieve its mission of Q = 10 at a normalized density of $n/n_{GW} \sim 0.6$, and inside pellet launch will be available to facilitate high-density operation.

- Several other physics considerations constrain the operating window of the chosen device. In particular, it has been decided to retain a single-null diverted equilibrium, since the scaling of the H-mode threshold power is more favourable in single null, as opposed to double null, plasmas. Moreover, the difficulty of maintaining a double null equilibrium which is fully up-down symmetric with respect to power handling is likely to impose unrealistic requirements on the accuracy of plasma vertical position control.

- Scrape-off layer and divertor behaviour influences plasma performance in several ways, but the principal issues for ITER-FEAT performance projections are the peak power to the divertor target, plasma helium fraction, and core plasma impurity content. There is substantial experimental evidence that helium exhaust rates are determined by the divertor throughput, rather than by helium transport rates in the bulk plasma, and that $\tau^*_{He} / \tau_E \sim 5$ can be achieved under relevant plasma conditions with the projected throughput of 200 $\mathrm{Pam^3s^{-1}}$. This would limit helium fractions in ITER-FEAT to acceptable levels, below 6%.

Domains of inductive operation

Based on the physics considerations and constraints outlined above, and with the major dimensionless geometrical parameters determined, it is possible to identify major radius and plasma current on the basis of the requirement that Q = 10 be achieved, that acceptable performance margins can be maintained, and that the projected cost of the device falls within

the required range. Smaller devices are more attractive from the cost point of view, but provide smaller margins for Q = 10, lesser likelihood of accessing Q > 10 and less flexibility to explore varying modes of operation. Increasing the size increases the operational domain and the margins but at an inevitable increase in cost. The reference parameter set, having a plasma major radius of 6.2 m and plasma current of 15 MA, was selected as it offers a satisfactory margin for Q ≥ 10 operation, has adequate flexibility and its cost satisfies the target.

Table 5.1. Nominal parameters of ITER-FEAT in inductive operation

Parameter	Units	Reference Q = 10	High Q, high P_{fus}
R/a	m/m	6.2 / 2.00	6.2 / 2.00
Volume	m^3	837	837
Surface	m^2	678	678
Sep.length	m	18.4	18.4
$S_{cross-sect.}$	m^2	21.9	21.9
B_T	T	5.3	5.3
I_p	MA	15.0	17.4
κ_x / δ_x		1.86 / 0.5	1.86 / 0.5
$\kappa_{95} / \delta_{95}$		1.7 / 0.35	1.7 / 0.35
$l_i(3)$		0.86	0.78
V_{loop}	mV	89	98
q_{95}		3.0	2.7
β_N		1.77	1.93
$<n_e>$	$10^{19}\,m^{-3}$	10.14	11.56
n/n_{GW}		0.85	0.84
$<T_i>$	keV	8.1	9.1
$<T_e>$	keV	8.9	9.9
$<\beta_T>$	%	2.5	3.2
β_p		0.67	0.62
P_α	MW	82	120

Parameter	Units	Reference Q = 10	High Q, high P_{fus}
P_{aux}	MW	40	23
P_{ohm}	MW	1.3	1.7
P_{tot}	MW	123	144
P_{brem}	MW	21	29
P_{syn}	MW	8	10
P_{line}	MW	19	20
P_{rad}	MW	48	59
P_{fus}	MW	410	600
P_{sep}/P_{LH}	MW/ MW	75/48	84/53
Q		10	24
τ_E, s		3.7	4.1
W_{th}	MJ	325	408
W_{fast}	MJ	25	33
$H_{H-IPB98(y,2)}$		1.0	1.0
τ_α^* / τ_E		5.0	5.0
Z_{eff}		1.65	1.69
$f_{He,axis}$	%	4.1	5.9
$f_{Be,axis}$	%	2.0	2.0
$f_{C,axis}$	%	0.0	0.0
$f_{Ar,axis}$	%	0.12	0.11

Performance calculations using the agreed physics guidelines yield a substantial operating window for Q ≥ 10 inductive operation for the selected parameter set.

Parameters of two representative plasmas in ITER-FEAT are listed in Table 5.1 The first column shows a reference Q = 10 discharge with the nominal plasma current of 15 MA and a fusion power of 400 MW, while the second column tabulates parameters for a regime with higher current, I_p = 17.4 MA ($q_{95} \sim 2.6$), that has the potential for a higher Q of ~ 25 and higher fusion power of ~ 600 MW, although with potentially higher risk of plasma disruption. In these simulations, the total power exhausted to the divertor target is held below 30 MW.

To illustrate the range of performance which can be achieved in ITER-FEAT, Figures 5.2 and 5.3 show values of P_{fus} and Q as a function of the auxiliary heating power for discharges with I_p = 13.1, 15.1 and 17.4 MA in which an operating point having $H_{H-IPB9(y,2)} = 1$ and $n/n_{GW} = 0.85$ is selected. The minimum fusion power at 15.1 and 13.1 MA is limited by the L-H back transition, taken as $1.3\times P_{LH}$. There is a strong increase in Q and P_{fus} with the plasma current and a strong increase in Q with reducing the auxiliary heating power. This emphasizes the

fact that the operation space is multidimensional and that plasma parameters can be adjusted to optimize the fusion performance according to whether high Q or high fusion power (e.g. to maximize the neutron wall loading) is required.

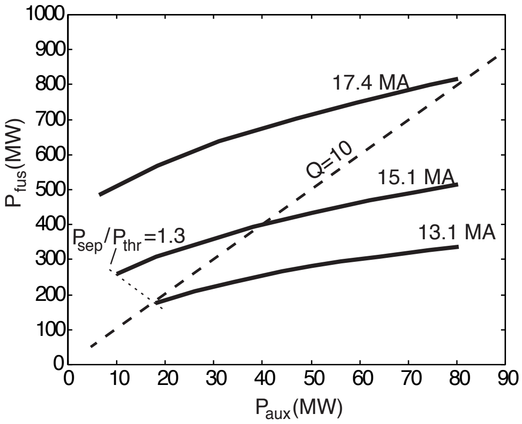


Figure 5.2 P_{fus} as a function of P_{aux} for I = 13.1, 15.1 and 17.4 MA at $H_{H-IPB9(y,2)}$ = 1 and n/n_{GW} = 0.85.

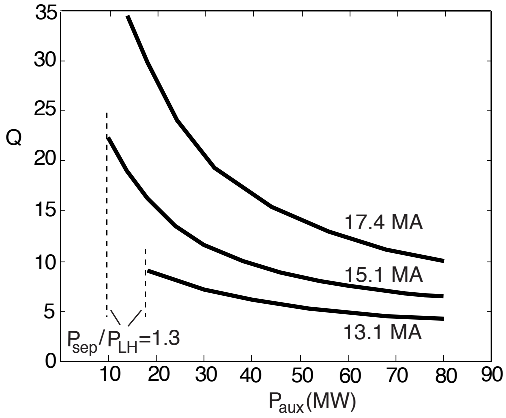


Figure 5.3 Q as a function of P_{aux} for I = 13.1, 15.1 and 17.4 MA at $H_{H-IPB9(y,2)}$ = 1 and n/n_{GW} = 0.85.

A more complete view of the range of plasma parameters at which Q = 10 operation is possible can be gained from an analysis of the operational domain in terms of fusion power and confinement enhancement factor, in which the various operational boundaries ($P_{loss} = 1.3P_{LH}$, $n = n_{GW}$, and $\beta_N = 2.5$) can also be traced, as shown in Figure 5.4 and Figure 5.5. Inside the indicated domain the Q = 10 is maintained, but the auxiliary power is adjusted together with the density.

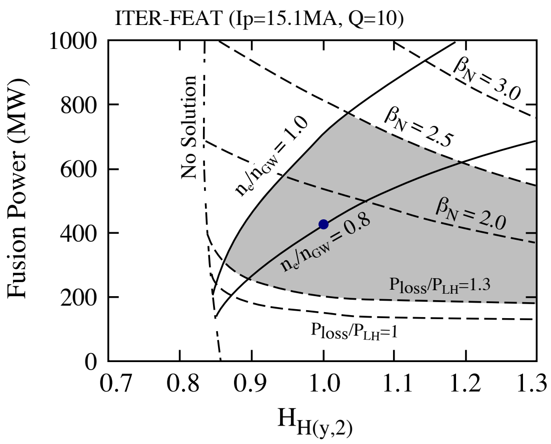


Figure 5.4. Q = 10 domain (shaded) for I_p = 15.1 MA (q_{95} = 3.0).

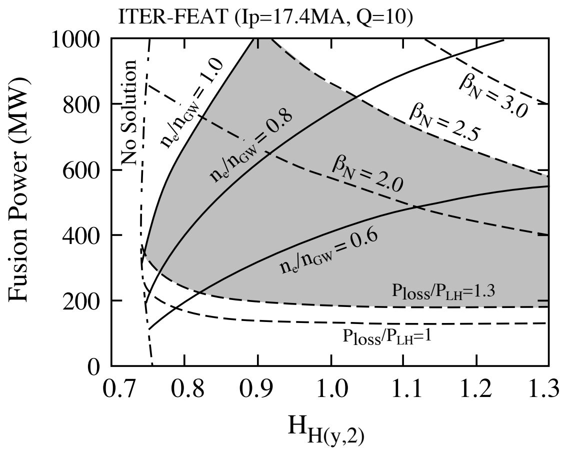


Figure 5.5 Q = 10 domain (shaded) for I_p = 17.4 MA (q_{95} = 2.6).

It is evident from the above, that:

- for operation at q_{95} = 3 the fusion output power from the ITER-FEAT design is in the region of 200-600 MW (at $H_{H(y,2)}$ = 1), corresponding to a mean separatrix neutron flux ('mean neutron wall loading') of 0.29-0.86 MWm^{-2}, so that the device retains a significant capability for technology studies, such as tests of tritium breeding blanket modules;
- the margin in H-mode threshold power (at $H_{H(y,2)}$ = 1) is significantly greater than the predicted uncertainty derived from the scaling;

- the device has a capability for Q = 10 operation at n/n_{GW} ~ 0.6 and β_N ~ 1.5 (when $H_{H(y,2)} = 1$).

The results also illustrate the flexibility of the design, its capacity for responding to factors which may degrade confinement while maintaining its goal of extended burn Q = 10 operation, and, by implication, its ability to explore higher Q operation as long as energy confinement times consistent with the confinement scaling are maintained.

For instance, Figures 5.6 (I_p = 15.1 MA) and 5.7 (I_p = 17.4 MA) illustrate the window for higher Q operation (Q = 50, representative of 'controlled ignition') in ITER-FEAT, showing that controlled ignition is not precluded: operation at a range of Q values is possible and values as high as 50 can be attained if $H_{98(y,2)}$ ~ 1.2 is achieved in a improved confinement mode, e.g. reversed shear or shallow shear mode with internal transport barrier, or high density operation can be extended beyond the Greenwald value, or operation at lower q_{95} (~ 2.6) can be sustained without confinement degradation.

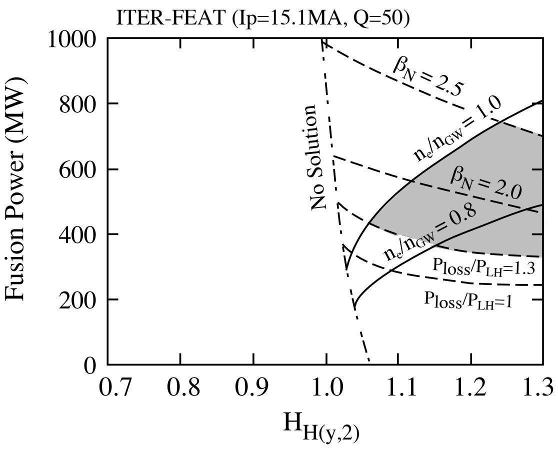


Figure 5.6. Q = 50 domain for I_p = 15.1 MA (q_{95} = 3.0).

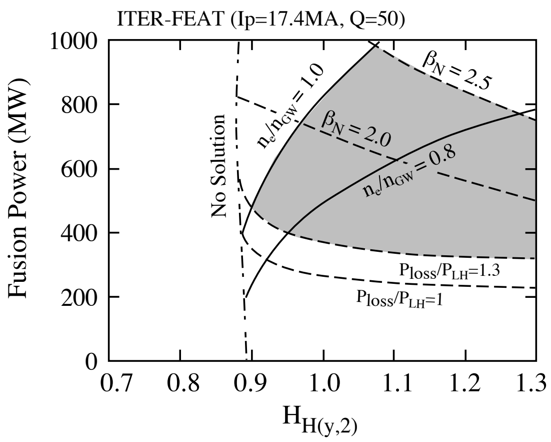


Figure 5.7. Q = 50 domain for I_p = 17.4 MA (q_{95} = 2.6).

Operating Flexibility

The design is capable not only of studying the standard operating regime, but also incorporates the flexibility and extended capability to achieve enhanced performance within the cost constraint. Several aspects of the design address the issue of pressing the boundaries of the operating domain and of accommodating uncertainties in physics predictions. For example, the inclusion of inside pellet launch opens the route towards operation at high density. Moreover, a variety of active feedback control techniques are provided for the stabilization of mhd instabilities. Active current profile control techniques could also provide an additional tool for the control of mhd activity. To extend the achievable range of Q values (and to counteract any unforeseen degradation of confinement), the possibility of operating the device with plasma currents up to ~ 17.4 MA (q_{95} ~ 2.6) is being explored, albeit at reduced pulse length (>100 s). Finally, the capability of operation at fusion powers up to 40% higher than the reference value (though under the assumption of no increase in total neutron fluence) is included in the design to enhance the possibility of ignited operation and to accommodate the possibility that higher β values than assumed are achieved.

Steady-state and hybrid operation

A complete scenario for steady-state operation with Q = 5 which treats energy confinement plasma profiles, current drive requirements, divertor performance and plasma equilibrium self-consistently, and which satisfies all relevant constraints is yet to be developed. In ITER it is likely that a variety of candidate steady-state modes of operation will be investigated and it is therefore essential that the requisite tools for the control of plasma geometry and profiles are available for on-axis and off-axis current drive capabilities to enable plasmas with shallow or reversed shear configurations to be sustained, in the latter regime simultaneously maintaining the central safety factor well above unity, while the minimum safety factor is held above two; a poloidal field system capable of controlling the more highly shaped plasmas characteristic of high-β_p operation; and methods to allow reliable long pulse operation at high-β, including techniques for the stabilization of neoclassical tearing modes and resistive wall modes.

The capability of the ITER-FEAT designs for steady-state operation with Q = 5 are being studied numerically using 0-dimensional analysis within the limitations of current assumptions. Two operational scenarios are under consideration for steady-state operation: high current (12 MA) with monotonic q or shallow shear, and modest current (8 MA) with negative shear. The high current steady-state operation requires all the current drive power (100 MW) available for ITER-FEAT, but the requirements on confinement ($H_H \sim 1.2$) and beta ($\beta_N \sim 3$) are modest. On the other hand, the low current steady-state operation requires more challenging values of confinement improvement $H_H \sim 1.5$ and beta ($\beta_N \sim 3.2$-3.5). Performance predictions for this mode of operation are much less certain than for inductive operation and high current steady-state operation. In particular, the operating space is sensitive to assumptions about current drive efficiency and plasma profiles.

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

An operation space, in terms of fusion power versus confinement enhancement factor, and showing the transition from hybrid to true steady-state operation is illustrated in Figure 5.8 for I_p = 12 MA and P_{CD} = 100 MW. Contours of constant n/n_{GW} and β_N are indicated, as is the threshold for Q = 5 operation. It is assumed that the plasma minor radius is reduced by shifting the magnetic axis outward. For a given value of fusion power (and hence Q), as the confinement enhancement factor, $H_{IPB98(y,2)}$, increases (simultaneously decreasing plasma density and increasing β_N), the plasma loop voltage falls towards zero. For example, operation with V_{loop} = 0.02 V and I_p = 12 MA, which corresponds to a flat-top length of 2,500 s, is expected at $H_{IPB98(y,2)} = 1$, Q = 5, $n_e/n_{GW} = 0.7$, and $\beta_N = 2.5$. True steady-state operation at Q = 5 can be achieved with $H_{IPB98(y,2)} = 1.2$ and $\beta_N = 2.8$. This analysis indicates that a long pulse mode of operation is accessible in ITER-FEAT.

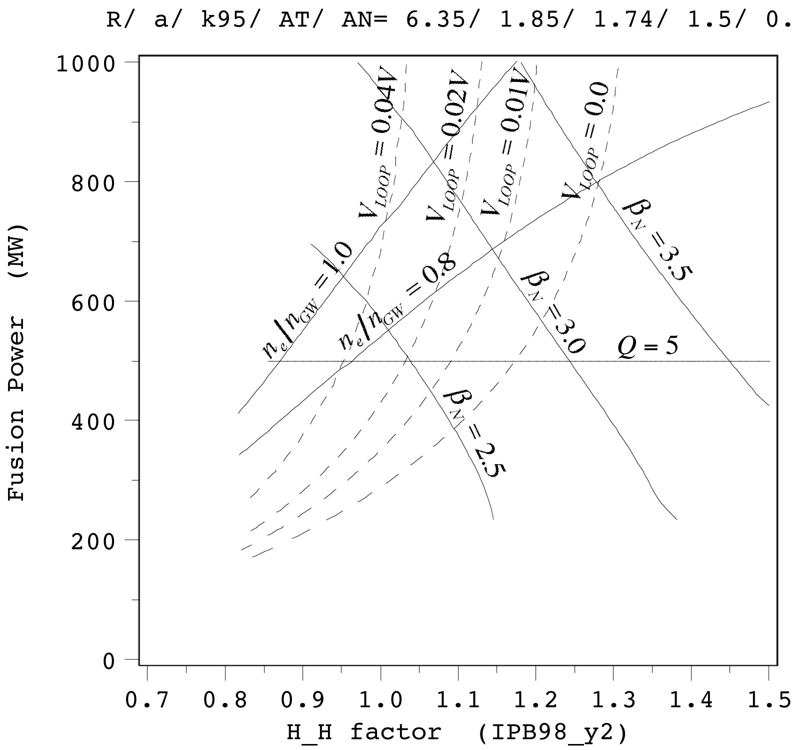


Figure 5.8 Operation space for ITER-FEAT for hybrid (long pulse) and steady-state operation. Here, I_p = 12 MA and P_{CD} = 100 MW.

Probabilistic Performance Analysis

As with any extrapolation from current experience to unprecedented domains, there are unavoidable ranges of uncertainty on either side of the performance projections for ITER.

To provide an evaluation of the probability of achieving a fusion gain of Q ≥ 10 in ITER-FEAT, an analysis based only on the estimated uncertainty in the form of the confinement scaling has been developed. This approach does not provide information on, for example, the probability of achieving a specified fluence, since many other factors influencing the average duty cycle and total operational time of the device must be considered.

It is assumed that the energy confinement time (or, in practice, the $H_{H(y,2)}$ factor) for a given set of plasma parameters can be described by a Gaussian distribution having a standard deviation of either 10% or 20% about the mean value. However, for similar discharge conditions, the distribution of $H_{H(y,2)}$ in the database has a smaller spread: for example, with $n_e/n_{GW} \geq 0.65$, $q_{95} \leq 3.5$, $P_{RAD}/P_{HEAT} \leq 0.5$ and $\kappa \geq 1.5$, the spread of $H_{H(y,2)}$ values in the database is only 5%. This illustrates the important point that only a fraction of the scatter in the experimental data is associated with irreproducibility in discharge conditions.

Using conservative operating conditions, it is possible to delineate operation domains for various values of auxiliary heating, which when integrated with the assumed probability density function for H_H, allows to quantify the probability that Q exceeds a given Q_0 value. Combining the analysis for the range of values generates a family of probability curves values that can be used to summarise the overall probability to achieve a target Q value. This analysis is illustrated in Figure 5.9 for the case of σ = 20%, and is summarised in Figures 5.10 and 5.11 for σ = 10% and 20%, and for values of plasma current at 15.1 MA and 17.4 MA respectively.

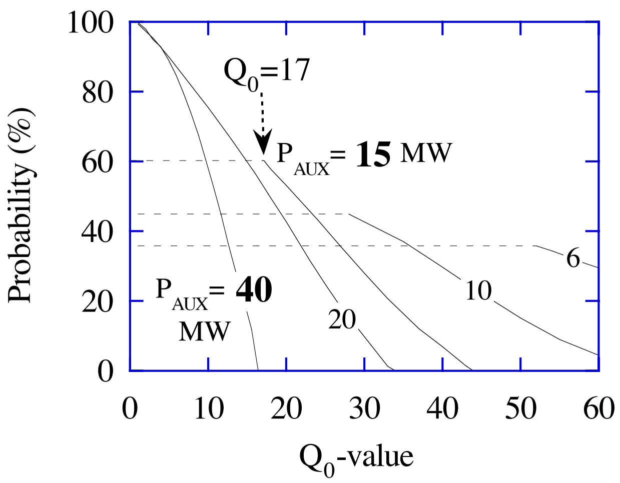


Figure 5.9 Probability of achieving $Q \geq Q_0$ in ELMy H-mode for a range of fixed heating powers, P_{AUX}, when σ = 20%. Here $n_e/n_{GW} \leq 0.85$ and $\beta_N \leq 2.5$. The flat part of each curve corresponds to $P_{loss} \leq P_{LH}$ (at P_{AUX} = 6, 10, 15 MW). In these cases the probability of $Q \geq Q_0$ is equal to that of $Q \geq Q_{MAX}$, where Q_{MAX} is the value which gives the maximum probability.

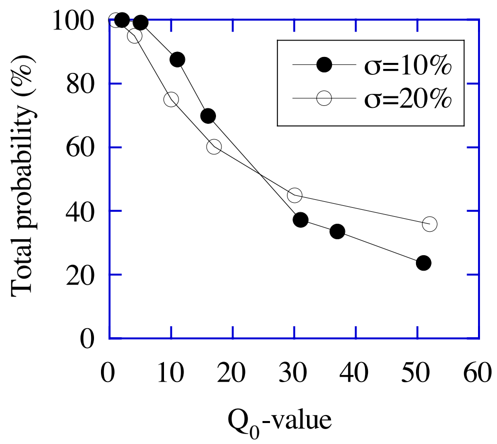


Figure 5.10. Probability of achieving $Q \geq Q_0$ in ELMy H-mode for σ = 10% and 20% with I_p = 15.1 MA, $n_e/n_{GW} \leq 0.85$ and $P_{loss} \geq P_{LH}$.

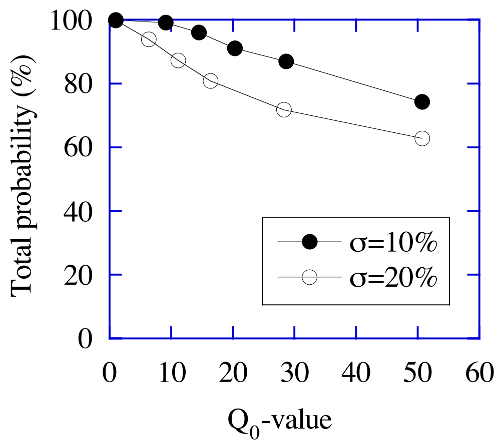


Figure 5.11 Probability of achieving $Q \geq Q_0$ in ELMy H-mode for σ = 10% and 20% with I_p = 17.4 MA, $n_e/n_{GW} \leq 0.85$ and $P_{loss} \geq P_{LH}$.

On this basis, the probability of achieving Q ≥ 10 in the ELMy H-mode regime is high. However, if for unexpected reasons Q ≥ 10 were not achieved under nominal operating conditions, there are, as noted previously, various options for increasing the probability of achieving the required Q. For example, raising the plasma current to 17.4 MA increases the probability of achieving Q ≥ 10 to ~ 90% and 100% if σ= 20% and 10% respectively. Another option is to increase the fuel throughput in the divertor beyond the reference value of 200 Pam^3s^{-1} to, say, 400 Pam^3s^{-1} (which can be maintained for 200 s), allowing the helium concentration to be reduced by 2% (incremental), which, in fusion performance terms, is

equivalent to a 1 MA increase in plasma current. Furthermore, regimes with active profile control could allow enhanced confinement to be accessed in inductive operation.

The probability calculation outlined above is essentially a ‘model’ calculation, i.e. it represents a numerical result based on simple, well-defined assumptions. It does not, however, amount to a complete evaluation of the true probability of achieving $Q \geq 10$. In addition, it is a model calculation carried out in only one dimension of the multi-dimensional operating space which describes a burning plasma and it neither fully reflects the complexity of the behaviour close to operating limits, nor the degree to which experimental optimization of plasma parameters can improve plasma performance. In summary, the optimum operating point of a tokamak plasma consists neither of a random selection of parameters, nor a random response to the operating conditions selected, but corresponds, rather, to a well-defined and reproducible plasma state resulting from extensive experimental development.

Deterministic assessment of performance

As observed above, the uncertainty in extrapolation of the energy confinement time along the scaling law — expressed in a probability distribution of the scalar H_H around a mean value — includes, from a statistical analysis, the consequences on confinement performance from varying machine operating conditions, particularly when their non-dimensional plasma parameters approach their respective limits. As examples, the effect of magnetic shear (from δ,q,κ,A) on confinement in high density discharges, or the effect of saw teeth on low-edge-safety-factor discharges at higher elongation and triangularity are not obviously reflected in the scaling law.

To overcome some of these difficulties, a deterministic procedure has been established so as to define a more focussed experiment, similar in all non-dimensional parameters to a fully documented experiment available in the database from present machines.

- Each discharge from the database is used to size, by means of a system code (in accordance with ITER engineering criteria) a Q = 10 machine with the same geometry and parameter values for κ,δ,A,q_{95} and n/n_{GW} as in the experimental case.
- As a first step the extrapolation in the energy confinement time is performed using a ratio to the experimental value coming from the relative values of the parameters not kept constant as follows:

$$\frac{\tau_{E,Q10}}{\tau_{E,Ex}} = \left(\frac{P_{Q10}}{P_{Ex}}\right)^{\gamma_P} \cdot \left(\frac{B_{Q10}}{B_{Ex}}\right)^{\gamma_B} \cdot \left(\frac{R_{Q10}}{R_{Ex}}\right)^{\gamma_R} \cdot \left(\frac{M_{Q10}}{M_{Ex}}\right)^{\gamma_M}$$

From the dimensionally correct IPB98(y,2) scaling law,

$\gamma_P = -0.69$, $\gamma_B = 1.49$, $\gamma_R = 2.49$, $\gamma_M = 0.19$

From the more than 1000 discharges in the ELMy H-mode database, only half of them turn out to extrapolate to a Q = 10 machine of major radius smaller than 8 m, 70 to a radius smaller than 6.2 m and only a few to radius smaller than 6 m, the smallest value being 5.6 m with $q_{95} = 3.0$.
- As a second step, a more general case is worth considering, which avoids completely the use of empirical scaling formula, the extrapolated device being sized to obtain a required fusion power (and not a Q value) from experimental discharges with the same parameter package as before and, in addition, a fixed value for β_N.

  In this case, Figure 5.12 shows the major radius versus the safety factor q_{95} for all machines extrapolated from experimental discharges and capable of 500 MW of fusion

power. A good number of discharges can be extrapolated to 500 MW devices of radius between 6 and 6.2 m (although the relevant values of Q cannot be determined in this case).

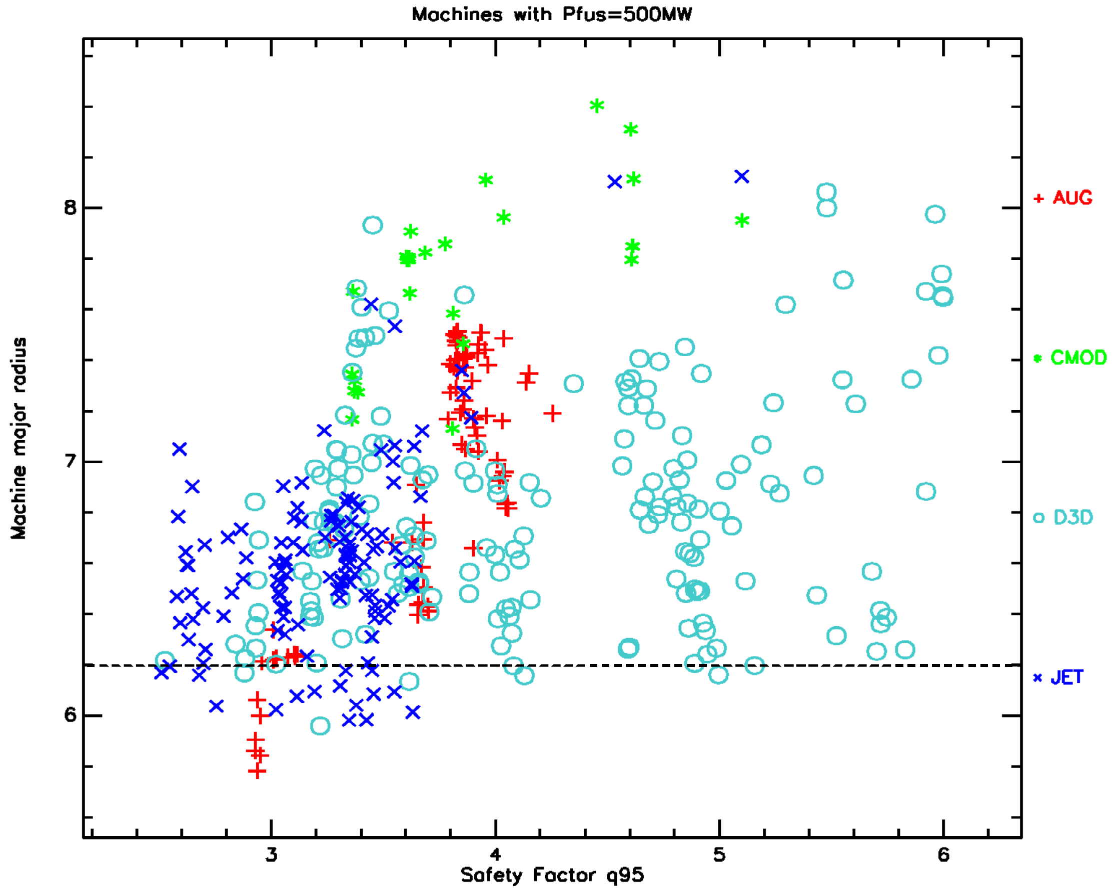


Figure 5.12 Major radius of 500 MW fusion power machines versus safety factor in the database under the assumption of constant beta

In summary, according to this deterministic approach to extrapolation using similar non-dimensional parameters from the documented experimental discharges in present machines (assuming continuity of variation with ρ* of the physical behaviour of plasma and of radial parameter profiles), the ITER-FEAT design parameters appear adequately chosen to achieve the performance goals, on the basis of extrapolation from many high performance ELMy H-mode discharges from JET, DIII-D and Asdex-U.

6.0 Design Features and Assessments

Summary of design features

Following the revised technical guidelines, the design of ITER-FEAT aims to use as far a possible technical solutions and concepts developed and qualified during the EDA so far. Nonetheless, changes in overall scale and in some physics requirements (e.g. more plasma shaping); the pressure to preserve the plasma performance capacity and flexibility, whilst approaching the 50% cost target ITER-FEAT have induced some significant changes in the design features from the 1998 ITER Design; in addition, the continuing flow of new data from the technology R&D projects have enabled changes in design criteria associated with a better knowledge of the available margins.

The main engineering features and materials in the design are summarised in Table 6.1 At this stage of the development of the design, not all components/subsystems are “frozen”. Nevertheless, all systems which interfere with the global layout of the project have their parameters frozen. Inside some systems with no external influence more than one option is maintained pending further analysis, and, in general, detailed design work and optimisation will lead to limited modifications. As noted above, the proposed engineering design relies mainly on technical solutions which have been or are being qualified by the on-going R&D in the Parties’ laboratories and industries. Most of the remaining issues are related to the choice of options which will provide the largest cost saving through improved and more efficient manufacturing processes.

Because of the unwillingness to compromise with physics extrapolation so as to provide enough margins in the physical parameters and physics-related systems e.g., plasma size, fuelling, and heating and current drive, a major focus of effort will be to press on the manufacturing processes (with their feedback on design) to approach as closely as possible the target of 50% saving in direct capital cost from the 1998 ITER design.

The following paragraphs summarise and assess the key features of the main the ITER-FEAT components and subsystems, and the overall plant systems integration and illustrate some of the issues and options remaining to be decided.

Magnets and structures

The superconducting magnet system which confines, shapes and controls the plasma inside a toroidal vacuum vessel comprises three main systems and their power supplies:

- 18 Toroidal Field (TF) coils which produce the confining/stabilizing toroidal field;
- 6 Poloidal Field (PF) coils which contribute to the plasma positioning and shaping; and
- a Central Solenoid (CS) coil which provides the main contribution to the induction of poloidal field current in the plasma.

Correction coils (including three sets located above, outboard of and below the TF coils) are also required to correct error fields that arise due to imperfections in the actual PF and TF coil configuration and to stabilize the plasma against resistive wall mode instabilities.

The magnet system weighs, in total, about 8,700 t — about one third of the weight of the 1998 design.

Table 6.1 Main engineering features of the ITER-FEAT systems

Superconducting toroidal field coils (18 coils)	
Superconductor	Nb_3Sn in circular stainless steel (SS) jacket in grooved radial plates, or in square SS conductor
Structure	Pancake wound, in welded SS case wind, react and transfer technology
Superconducting Central Solenoid (CS)	
Superconductor	Nb_3Sn in square Incoloy jacket, or in circular Ti/SS jacket inside SS U-channels
Structure	Pancake wound, 3 double or 1 hexa-pancake wind react and transfer technology
Superconducting poloidal field coils (PF1-6)	
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 (429 modules)	(Initial DT Phase)
Structure	Single curvature faceted separate FW attached to shielding block which is fixed to vessel
Materials	Be armour, Cu-alloy heat sink, SS 316 LN structure
Divertor (54 cassettes)	
Configuration	single null, cast or welded plates, cassettes
Materials	W alloy and C plasma facing components Copper alloy heat sink, SS 316 LN structure
Cryostat	
Structure	Ribbed cylinder with flat ends
Maximum inner dimensions	28 m diameter, 24 m height
Material	SS 304L
Heat Transfer Systems (water-cooled)	
Heat released in the tokamak 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 80K	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

The CS and TF coils use a conductor with a large number of Nb_3Sn strands (~ 1,000), whereas the remaining PF and correction coils use a similar conductor with NbTi strands. All coils are cooled by supercritical helium at ~ 4.5K. The TF coil case is the main structural component of the magnet system and the machine core. The PF coils and vacuum vessel are linked to the TF coils such that all interaction forces are resisted internally in the system thus eliminating the need for large external load transferring structures and the mechanical moments associated with such structures. The TF coil inboard legs are wedged all along their side walls in operation and they are all linked at their two ends to two strong coaxial rings which resist the local de-wedging of those legs under plane loads, a detrimental effect to resist against out of plane loads where these are at their maximum. At the outboard leg, the out-of-plane support is provided by intercoil structures integrated with the TF coil cases. Views of the TF coil case are shown in Figure 6.1.

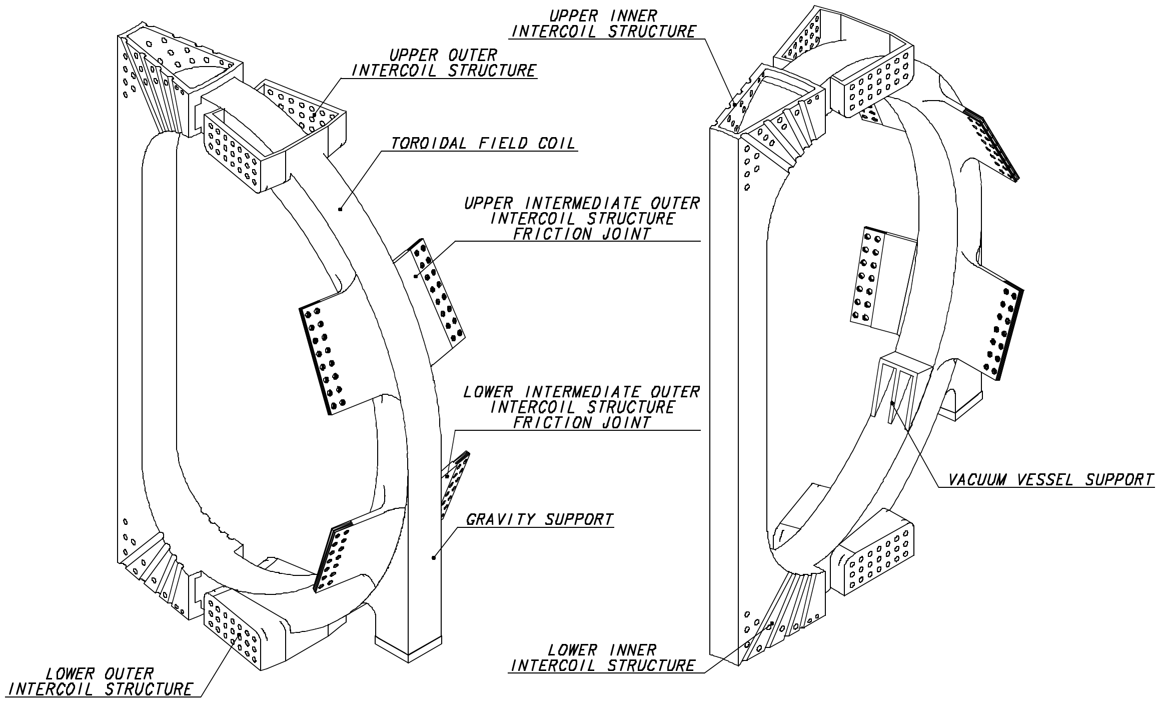


Figure 6.1 3-D views of the TF coil case

A power supply provides the DC coil currents to the different coils from the AC high voltage grid which supplies the ITER-FEAT site. The various coil electrical loads have different characteristics in terms of the currents, power, length of pulse, and so the coil power supply is made up of several subsystems. In addition, protective circuitry is provided to discharge the magnetic energy to external resistors in the possible event of superconducting coils quenching (rapid loss of superconductivity) under certain conditions.

Two options are still under investigation for the TF coil winding: one with a circular conductor embedded in radial plates and the other with a square conductor. The radial plate option has advantages in terms of the insulation reliability and fault detection capability, but suffers from cost and radial build penalties.

For the CS winding, there are two options to provide the structural material which is subject to fatigue due to the large number of pulses. The first one uses an Incoloy square jacket with a co-wound strip and the second one uses two, stainless steel, U-channels welded around a thin circular, jacket made of Incoloy or titanium. The selection of the option has some limited

impact on the CS flux capability but the choice can be postponed until more R&D results are available.

External conditions (static and variable magnetic field, stress and strain levels, cooling etc.) to be met in operation by the TF and CS coil conductors will be simulated in the testing facility built for the CS model coil. Insert coils made of TF and CS conductors will be tested. Results will provide a measure of the margins available around the reference operating point. The model coil programme has also addressed and resolved a number of key manufacturing issues. Production of the CS and TF conductors at industrial scale has been achieved and the "wind react and transfer" process for the conductor has been qualified.

The TF coil case manufacturing issues are being addressed in an R&D programme which includes welding development and the production of large forged pieces and castings as required in the full size coil cases and outer intercoil structures. This development is expected to facilitate manufacture and reduce cost.

In total, the large effort in R&D provides confidence that the remaining issues for the magnet design are not ones of feasibility, but rather, issues which relate to options to reduce the capital cost and to fulfil new requirements for plasma operation (e.g., the segmented CS and wedged TF coils).

For the PF coils, R&D has been initiated to verify the performance of the NbTi conductors for the PF coils; this work has to include the manufacture of a coil with a full size conductor and the testing of this coil in pulsed conditions.

For the main components and subsystems for the magnet power supplies, including AC/DC conversion system, the reference designs are based on existing technology and products available in the world market, or on the progress that is expected to be achieved in the near future.

Cryoplant and Cryodistribution System

Liquid helium from a cryoplant is distributed by a cryodistribution system to auxiliary cold boxes feeding the magnet and other loads as well (e.g. cryopumps for the pumping of the vacuum vessel). Circulating pumps force the flow of supercritical helium through the load in each separate circuit, which exchanges heat with a helium bath, whose pressure (and thus temperature) are controlled by a cold compressor in the return path towards the cryoplant. The plant design has to reconcile the pulsed character of the heat deposited in the magnet coils and the cryopumps, with the steady operation of the cryorefrigerator, which handles only the average heat load.

Although the envisaged cryoplant is a very large and complex facility, the confidence of building such a plant with the required performance is very high since the cryorefrigerator and cryodistribution systems for large particle accelerators provide good bases that can be directly applied to the ITER-FEAT system design.

Cryostat and Thermal Shields

The whole tokamak (vacuum vessel, magnet and associated structures) is located within a single-walled cryostat and within the cryostat there are thermal shields at 80K to prevent the cold portions (~ 4K) from receiving heat from the "hotter" parts. Rectangular bellows made

from elastomer are used to connect the interspace duct wall extensions of the VV ports with the cryostat port to compensate for differential movements.

The design and size of the cryostat are within industrial experience. Provided R&D results confirm the suitability of the intended use of elastomer bellows and Ag-plating to large panels, there is no reason to doubt that the cryostat and thermal shields can be procured and assembled as intended. Should the R&D results be negative, alternative, back-up options are available.

Vacuum Vessel

The double-walled vacuum vessel is lined by modular removable components, including blanket modules composed of a separate first wall mounted on a shield block, divertor cassettes, and diagnostics sensors, as well as port plugs such as the limiter, heating antennae, and test blanket modules. All these removable components are mechanically attached to the VV. These vessel and internal components absorb most of the radiated heat from the plasma and protect the magnet coils from excessive nuclear radiation. This shielding is accomplished by a combination of steel and water, the latter providing the necessary removal of heat from absorbed neutrons. A tight fitting configuration of the VV aids the passive plasma vertical stability and ferromagnetic material in the VV localised under the TF coils reduces the TF ripple. The overall arrangement of one of the 9 vacuum vessel sectors is shown in Figures 6.2 and 6.3.

Integrated functionally with the VV is the vacuum vessel pressure suppression system (VVPSS). This system minimizes the peak pressure inside the vacuum vessel during an in-vessel LOCA by relieving the pressure, caused by the ingress of a water steam mixture from damaged water-cooled, in-vessel components, through rupture discs via pipework into a steam condenser tank.

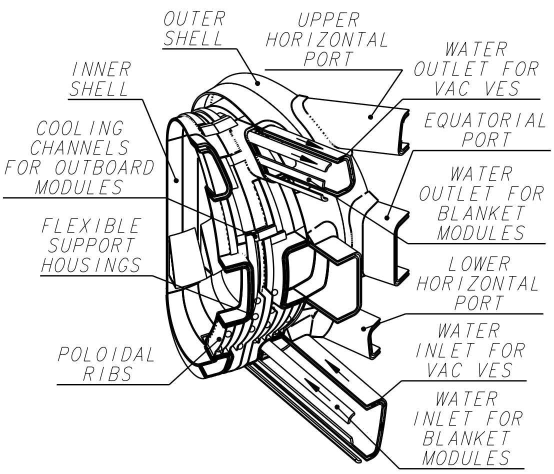


Figure 6.2
Vacuum Vessel overall arrangement

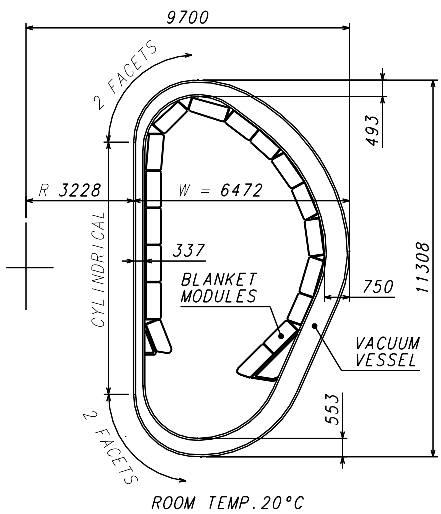


Figure 6.3
Vacuum Vessel cross-section

The manufacture of a full-scale sector of the 1998 ITER design gives a sound basis for the design of the present vessel. To reduce the VV fabrication cost, forging, powder HIPing and/or casting is being investigated for the large number of the housings in the VV for the blanket module support that have a relatively small and simple structure. The preliminary comparison of their fabrication costs with welded structures shows a cost benefit.

Blanket

The blanket system is made up of 429 modules, including those around the Neutral Beam injection lines. The initial blanket acts solely as a neutron shield and tritium breeding experiments are confined to the test blanket modules which can be inserted and withdrawn at radial equatorial ports. The blanket module design consists of a separate faceted first wall (FW) built with a Be armour and a water-cooled copper heat sink attached to a SS shielding block; this minimises radioactive waste and simplifies manufacture.

Two methods are being considered for FW attachment to the shield: a central mechanical attachment, which is bolted to a shield block at its rear side, or a system of bolts (accessed from the first wall) and small shear ribs, to support electro-mechanical loads and to prevent sliding due to thermal expansion.

Two options are being considered for blanket cooling: one with cooling channels integrated inside the vessel structure between the two walls, the other with channels mounted on the vessel in vacuum.

Overall, the manufacture and testing of blanket and FW mockups of the 1998 ITER design gives a sound basis for the present blanket design.

Divertor

The divertor exhausts the helium reaction product of the DT fusion reactions and limits the concentration of impurities (non-hydrogen isotopes) in the plasma by providing a region in which the magnetic field lines just outside the plasma boundary are "diverted" to meet a target plate at a small angle of incidence. Charged particles escaping from the confined plasma will flow to the target, but on the way will lose a large fraction of their energy by radiation and charge exchange with neutrals, thus limiting the power density on the target plate.

The divertor itself is made up of 54 cassettes. Figure 6.4 is a sketch in a poloidal cross-section of the diverted magnetic field and the divertor showing some features of the construction of a cassette, in particular the targets which are the surfaces subjected to the heat load from the diverted particles (peak heat fluxes are less than 20 MW/m^2).

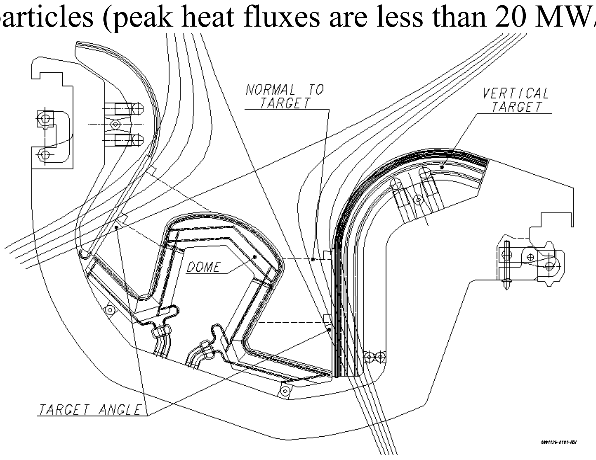


Figure 6.4 Divertor Plasma Facing Components Arrangement

The current design uses carbon at the vertical target strike points. Tungsten is being considered as a backup, and both materials have their advantages and disadvantages. The two options need continuous development so that the best judgement of the relative merits

can be made by the time of procurement. Carbon has the best behaviour to withstand large power density pulses (ELMs, disruptions), but gives rise to tritiated dust. Procedures for the removal of tritium codeposited with carbon and tritiated dust from various components by a number of schemes are under consideration and need further development.

The development of carbon and tungsten armoured plasma facing components has advanced to a level capable of meeting the demanding requirements of the ITER-FEAT divertor for the average target heat load. The armour behaviour against large power density pulses could be the limiting factor. A successful R&D campaign has demonstrated that armoured components can routinely operate with heat loads of up to 20 MW/m^2 for carbon and > 10 MW/m^2 for tungsten, with a promise of also reaching 20 MW/m^2. A prototypical armoured vertical target, which is compatible with the ITER-FEAT divertor requirements, has been built and fully tested. Furthermore, successful operation in tokamaks, with the SOL partially attached to the divertor targets, has demonstrated that the average heat flux to the divertor can be reduced to a value where the armour life-time is adequate. This is the basis for confidence in the design.

Water Cooling system

The heat deposited in the vessel-internal components and the vessel is rejected to the environment via the tokamak cooling water system, 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.

In the worst situation, where all active cooling to in-vessel components is lost because of pipe breaks or power failure, natural convection in the vessel is able to exhaust their decay heat and keep components well below the temperature at which there is no significant chemical reaction between steam (air) and Be-dust.

The normal operation of active components of the water cooling system such as the main pumps, small pumps and motor-operated valves under the operational magnetic field must be guaranteed. The allowable strength of the magnetic field and the required shielding for each component is under study now.

The option of using sea/fresh water instead of forced flow cooling towers as the ultimate heat sink is being considered for a site-specific design. It may be that, in this case, an intermediate cooling water system may be required.

Whilst details of the different elements of the system have yet to be finalised, the general capacity of the main components in the water cooling system is within the industrial experiences (or industrial proven technology), therefore no problematic issues on the component design and manufacturing are expected

Fuel Cycle

The fuelling and pumping system also provides plasma density control. The tokamak fuelling system is capable of gas puffing, and pellet injection from the high field side, into the plasma. These gases are subsequently removed from the plasma together with the helium ash using the torus cryopumps, which exhaust to the tritium plant where impurities are removed from the hydrogen stream and the various isotopes of hydrogen are separated and stored. Tritiated

impurities are processed to lower their tritium content enough to allow their release. The tritium plant also detritiates water, ventilation air and process fluids and solids.

Pellet launch is from the high field side of the tokamak to maximise pellet penetration for a given pellet speed, and fuelling efficiency. However, the pellet speeds required are somewhat beyond those currently achieved without pellet disintegration inside the curved guiding flight tube. Thus R&D is needed to improve the design and geometry of the flight tube.

Regarding the tritium plant, nearly all the separation systems have to be present by the start of DD operation since tritium will be generated during this phase of operation. However systems for water detritiation can be deferred to some extent until full DT operation; for how long needs further quantification.

Many subsystems in the ITER tritium plant are based on proven, industrial processes at relevant scale. In some instances the dynamic nature of ITER operation requires additional confirmation and this is targeted by R&D, e.g., on the isotope separation system and hydrogen storage beds.

Overall there is confidence, that, given the expected outcome of the R&D, the necessary subsystems can be designed, procured and operated as required.

Heating and Current Drive

The plasma heating systems must also have the ability to drive current in the plasma (current drive) to extend the tokamak plasma duration beyond the limitations imposed by the inductive current drive provided by the central solenoid. This lengthening of the tokamak pulse is an attempt to reach “steady-state” conditions where the current drive would be entirely non-inductive. The H&CD systems under consideration for ITER-FEAT are shown in Table 6.2 below:

Table 6.2 Heating and Current Drive Systems

	NB	EC (170 GHz)	ICRF (~ 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

1) Each standard equatorial port can provide 20 MW of RF (EC or IC or LH)
2) 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.

Whilst the designs draw, in general, on existing operational systems, all the options require further R&D to validate the designs and to ensure the performance targets, in the conditions foreseen for ITER. If reasonable R&D programmes are maintained to address the various issues, there is confidence that a range of heating and current drive capacity capable of providing all the requested services can be made available.

Diagnostics

In order to understand the behaviour of the plasma in ITER-FEAT, a large number of special devices (diagnostics) will be applied to the tokamak to measure various properties of the confined plasma, the confining magnetic field and the fusion reaction products. Some of these diagnostics are not only required to evaluate the experiments but are required for machine protection (e.g. to avoid excessive heat loads on vessel-internal surfaces and the consequent damage), and for plasma control (e.g. magnetic field measurements which are required for the control of the plasma shape and position by the PF coils).

For magnetic diagnostics, the lifetime of the in-vessel coils and loops are the important issues. The results of the supporting radiation effects R&D indicate that the necessary lifetime can be achieved.

The ability of the neutron cameras to provide the total fusion power and the alpha particle source profile is directly linked to the available access. A wide angle of view is desirable in both the radial and vertical directions. A view through the intercoil structure for the vertical camera is being considered but the feasibility has yet to be established.

The optical/infrared systems view the plasma with a mirror, and a critical issue is the lifetime of this component. A solution for the mirrors is believed to exist for those systems which operate in the visible and infrared regions. Further work is required for diagnostics which require a relatively large solid angle of observation, for example, active charge exchange recombination spectroscopy and motional Stark effect.

Most of the measurements required for the machine protection and basic plasma control can be made using established techniques. In a few cases, however, novel approaches are required to take account of the expected operating conditions such as intense gamma background.

For sustained operation in high confinement modes, for example reverse shear, it is expected that the profiles of many parameters will have to be brought under active control. Measurements of most of the required profiles can be made but further work is required to confirm that the accuracies and resolutions that can be achieved will be sufficient.

Buildings and Services

The above systems are housed within buildings and structures along with plant services. Table 6.3 lists the main buildings and their footprints and other structures and areas which are required. Considerable effort has been made to make the best use of building space while providing an optimised layout for the required performance of the plant at a minimum cost. The tokamak and its closely associated systems are located mainly in the lower areas of the buildings as illustrated in Figure 6.5 which shows a section through the Tokamak building.

Table 6.3 Site Buildings, Structures and other areas

	Foot Print m^2
Tokamak Hall	5,482
Assembly Hall RF Heating Area in Assembly Hall (2,550 m^2)	3,825
Tritium, Hot Cell and Radwaste Buildings, Personnel and Access Control Structure	6,550
Power Supply Buildings	15,264
Cryoplant Buildings	13,950
Site Services,Control and Laboratory Office Buildings, etc	10,861
Building Totals	**55,932**
Power Supply Areas	59,282
Diesel Fuel, Cryo-Gas, Water and Gas storage, Makeup Basin	3,517
Hot Basin & Cooling Tower, Pumping Yard	9,674
Sub-Total Other Structures and Areas	**72,473**
Outdoor Storage /Expansion Areas	25,050
Parking Areas	31,410
Roadways	34,684
Sub-total Other outdoor areas	**91,144**
Grand Total	**219,549**

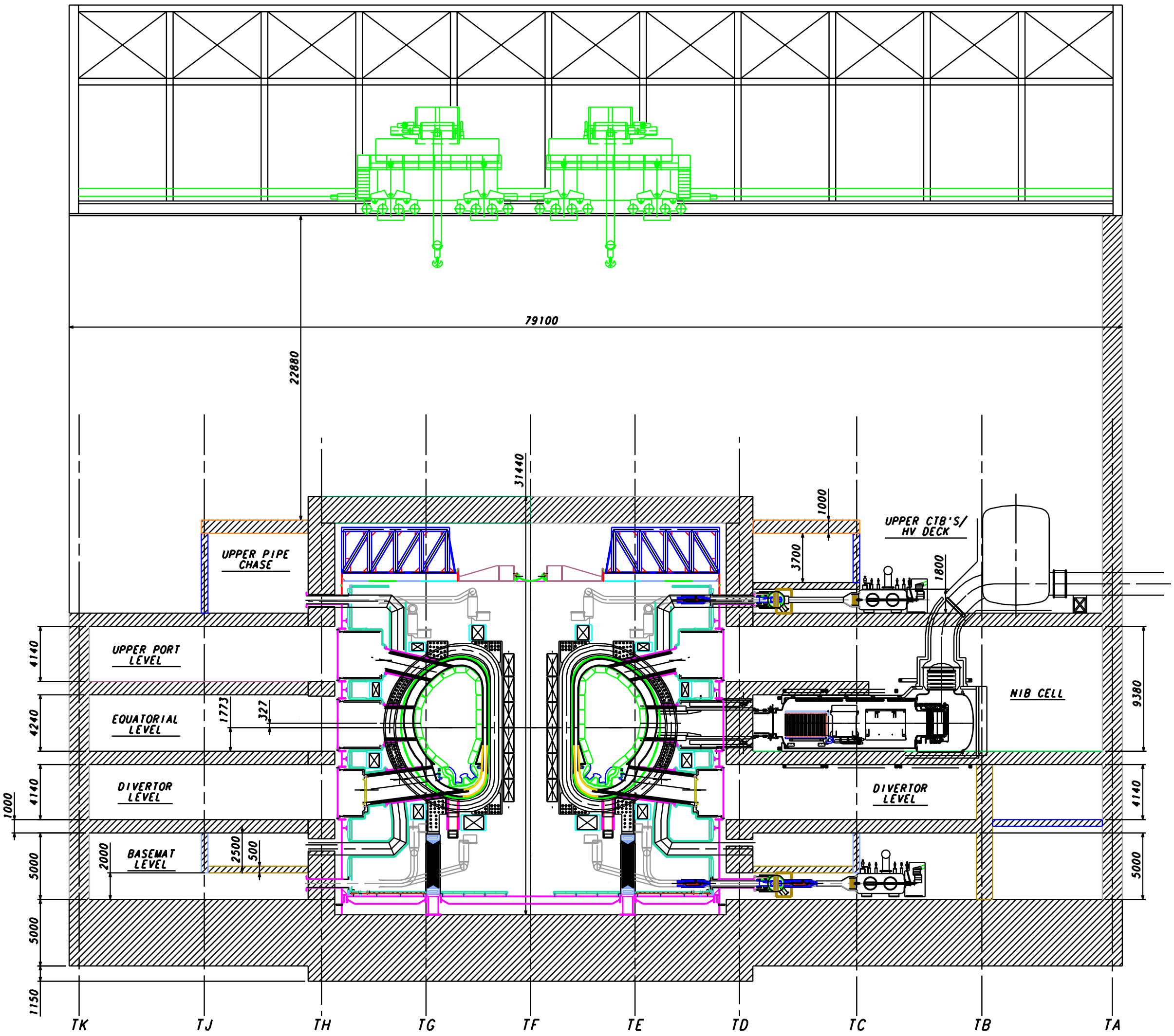


Figure 6.5 Tokamak building north-south section

Although the scale of the buildings and the components is towards the upper end of conventional building and construction experience, there is nothing about the buildings or structures that is outside the realm of current engineering and industrial practice. There are bigger structures, heavier equipment placed, and tighter tolerances used., but not so many all in a single project. Hence, it is an engineering challenge, but well within present engineering and construction capabilities.

Tokamak Maintenance

Because of the production of neutrons in plasmas of deuterium, and of deuterium and tritium, systems near the plasma will become radioactive and will require remote maintenance, with special remote handling equipment. The equipment involves an in-vessel transporter system for the removal and reinstallation of blanket modules, multifunction manipulators for divertor cassette removal, and specialised manipulators to handle vacuum vessel port plugs. Special casks, which dock horizontally to the access ports of the vacuum vessel, are designed to house such equipment and to transport radioactive items from the tokamak to the hot-cell where refurbishment or waste disposal operations can be carried out. Docking of these casks to the vessel and the hot cell flanges is tight, to avoid spreading of contamination. Hands-on assisted maintenance is used wherever justifiable.

A remote handling strategy for ITER has been confirmed by a comprehensive design and R&D programme which has successfully demonstrated that key maintenance operations like blanket and divertor replacement can be achieved using common remote handling technology. Several crucial issues like vacuum vessel remote cutting and re-welding, viewing, materials and components radiation hardness have been addressed and demonstrated. The above strategy is directly applicable to ITER-FEAT.

Some maintenance-related items still need to be addressed. In particular, the possibility of adopting a compact hot cell design based on the possibility to refurbish the divertor cassettes during the maintenance period is being assessed.

Overall, the development programme results so far obtained indicate that the remote maintenance strategy for ITER-FEAT is sound and sufficiently mature to support the ITER programme.

Tokamak Assembly

An outline procedure has been developed for the tokamak assembly, as the basis for determining the assembly schedule, manpower and tooling requirements and the associated cost.

The overall sequence is divided into the following six main sub-sections:

- lower cryostat activities, which cover activities from the initial assembly in this area up to the placement of the first TF/VV/VVTS sector;
- TF/VV/VVTS sub-assembly: Each sector includes a pair of TF coils, a 40˚ segment of the VV and three VVTS parts, an inboard 40° sector and two outboard, opposite hand 20° sectors.
- integrated TF/VV/VVTS assembly: The sequencing of the TF/VV/VVTS assembly in the cryostat;
- establish magnetic axis: survey procedures to establish the tokamak magnetic datum;
- ex-vessel activities: these activities occur in parallel with the in-vessel assembly procedures;
- in-vessel activities: further activities up to the preparation for commissioning.

A high level assembly plan has been established; details of many assembly activities, and related design of the assembly tooling now remain to be established. The need for a very accurate fit of the mating surfaces between adjacent TF coils may necessitate lengthy and precise matching operations, such as shimming, with a possible significant impact on the assembly schedule if the operations are to be carried out on the ITER site. Concepts and procedures for in-situ surveying and shimming have to be developed.

Plant control system

The integrated control and protection of the entire ITER plant will be achieved by the plant and plasma control system, and an independent interlock system.

The operation of the ITER plant is characterised by five major plant states as outlined below, in which many of the plant subsystems wait for commands before changing to another state, or some subsystems are undergoing maintenance or testing, or are in normal operation. The plant control system controls these states and the transitions between them, which occur through a sequence of steps.

The five defined plant operation states are:

- **Construction and Long-Term Maintenance State (LTM),** during which most of the tokamak subsystems which require maintenance will be shut down. Typical activities are large in-vessel and ex-vessel component replacement and maintenance.
- **Short-Term Maintenance State (STM),** for maintenance activities which typically last for 1 to 30 days. In this state, component maintenance and replacement will be carried out mainly outside of the vessel which remains under high vacuum conditions.
- **Test and Conditioning State (TCS)**, during which the tokamak systems are conditioned and heating and other ancillary systems might be tested; no in-vessel or major ex-vessel maintenance may be initiated.
- **Short-Term Standby State (STS),** which implies that the final preparation of each subsystem is completed and that the plant is ready for plasma operation.
- **Plasma Operation State (POS)**

The control system consists of a centrally-positioned supervisory control system (SCS) and subcontrol systems dedicated to each plant subsystem under the supervision of the SCS. Individual plant and diagnostic subsystems are directly controlled and monitored by their own dedicated intelligent control system. All systems use the same control method of conditional transitions between well-defined sequences of steps to be followed (i.e. SFC - Sequential Functional Control). The SCS controls the transition of the entire ITER plant from one operation state to another, and provides high level commands to plant subsystems, in order to achieve integrated control of the entire plant. The SCS also monitors the operation state of each plant subsystem to ensure it is operating within its proper operational envelope.

The interlock system monitors operational events of the plant, and performs preventive 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 control system for ITER-FEAT follows well established principles of system control. Accordingly, no major problems are expected in implementing the design and it has been possible to draw significant conclusions for the safety of the plant, as summarised below.

7.0 Safety and environmental characteristics

Safety Objectives and Design Principles

A main goal of ITER is to demonstrate from the viewpoint of safety the attractiveness of fusion and thereby provide a good precedent for the safety of future fusion power reactors. However, it is necessary to account for the experimental nature of the ITER facility, the related design and material choices, and the fact that not all of them are suited for future fusion power reactors. To accomplish this, ITER safety needs to address the full range of hazards and minimise exposure to these, and to permit siting by any Party.

Detailed safety related principles and environmental criteria have been adopted based conservatively on internationally recognised safety criteria and radiological limits following ICRP and IAEA recommendations and, in particular the ALARA principle.

The following safety objectives are taken into account in setting the requirements that guide the design for ITER-FEAT:

- **General safety**: To protect individuals, society and the environment. To ensure in normal operation that exposure to hazards within the premises and exposure due to any release of hazardous material from the premises are controlled and kept below prescribed limits. To prevent accidents with high confidence, to ensure that the consequences of more frequent events, if any, are minor, and to ensure that the consequences of accidents are bounded and their likelihood is small.
- **No evacuation**: To demonstrate that the favourable safety characteristics of fusion and appropriate safety approaches limit the hazards from internal accidents such that there is, for some countries, technical justification for not needing evacuation of the public.
- **Waste reduction**: To reduce radioactive waste hazards and volumes.

The general principles outlined below both provide direction to guide the design, and provide for on-going, independent review and assessment to ensure the design will meet the safety objectives.

1) Deployment of fusion's safety characteristics

The safety approach is driven by a deployment of fusion's favourable safety characteristics to the maximum extent feasible. Relevant characteristics are:

- the fuel inventory in the plasma is always below 1 g
- plasma burn is terminated inherently when fuelling is stopped due to the limited confinement of the plasma energy and particles
- plasma burn is self-limiting with regard to power excursions
- plasma burn is passively terminated by the ingress of impurities under abnormal conditions (e.g. by evaporation or gas release or by coolant leakage)
- the energy and power densities are low
- the energy inventories are relatively low
- large heat transfer surfaces and big masses exist and are available as heat sinks
- confinement barriers exist and must be leak-tight for operational reasons.

2) Exploitation of the passive safety features

Passive safety, based on natural laws, properties of materials, and internally stored energy are used to help assure ultimate safety margins.

3) Incorporation of defence-in-depth

The ITER safety approach incorporates ‘defence-in-depth’, the recognised basis for safety technology. All activities are subject to overlapping levels of safety provisions so that a failure at one level would be compensated for by other provisions.

There are three sequential defence levels - ‘prevention’, ‘protection’, and ‘mitigation’. Defence-in-depth, features at each of the three fundamental levels. All elements of defence in depth have to be available at all times during normal power operation and appropriate elements must be available when power is off (shutdown, maintenance, repair, decommissioning).

4) Provision for the experimental nature of ITER-FEAT

A robust safety envelope is provided to enable flexible experimental usage. Since ITER is the first experimental fusion device on a reactor scale, it will be equipped with a number of ‘experimental components’, in particular inside the vacuum vessel. In view of uncertain plasma physics and lack of operational experience, the experimental components are designed to allow for the expected loads from plasma transients so as to reduce the demands on systems which are required for safety. In particular, no safety function is assigned to experimental components.

Nevertheless, faults in experimental components that can affect safety are subject to safety assessments. On this basis, related measures will be incorporated in the design as appropriate.

The experimental programmes will be developed in such a way that design modifications will take account of experience from preceding operations and will stay within the safety envelope of the design.

Safety — Review and Assessment

Safety assessments covering normal operation, all categories of accidents, and waste management and disposal are an integral part of the design process, the results of which will be available to assist in the preparation of safety documentation for regulatory approval. The preliminary assessments of the ITER-FEAT design build on and develop further the detailed safety assessment of the 1998 ITER design.

Normal Operation

Effluents

Operational effluents are expected to be at a level which would cause public doses to the most exposed individual below 1% of the natural radiation background (postulating a typical 'generic' site).

Most releases are expected during maintenance operations. Presently available assessments suggest that the total tritium releases from the plant are about 0.25 g per year. In terms of doses, the releases of activation products are comparable to those of tritium.

Occupational Safety

Design criteria for personnel access have been established to ensure an acceptable level of occupational safety. Radiological hazards are being estimated by neutron activation analysis of components and structures, associated gamma transport calculations, and activated corrosion product build-up analysis, to assess against the design criteria. Non-radiological

hazards (EM fields, beryllium, etc) are also being estimated. In addition to meeting the criteria for access, an iterative assessment process will be applied to operational and maintenance activities to reduce radiation exposure based on the ALARA principle.

Radioactive Waste
Activated and contaminated materials arise during the operational phases and remain after final shutdown. Not all these materials would need to go into a waste repository, rather after some decay time a significant fraction can be 'cleared', i.e. declared to be no longer radioactive waste. The related processes (e.g. as recommended by IAEA) range from 'unconditional' clearance to clearance 'for recycling'.

A provisional waste characterisation assessment for the ITER-FEAT has been performed although a detailed study cannot be done until the design is finalised (or nearly so). A scaling approach based on the detailed assessment of the 1998 ITER design indicates that the amounts of radioactive material falls by about a half.

Possible accident conditions
In case of accident, ITER-FEAT protects personnel and the public using radioactivity confinement. Sources of tritium or activated materials that occur within the vacuum vessel, in the tokamak cooling water system, in the fuel cycle and within the hot cell, are housed behind multiple physical and functional barriers, which protect against the spread and release of hazardous materials. The primary confinement barrier is designed to have high reliability to prevent releases. A secondary barrier is provided close to the primary one to limit the spread of contamination and protect personnel from leaks. Exhaust from rooms that can be contaminated is treated by filters and/or detritiation systems, and monitored.

Possible Loss of Coolant Accidents (LOCAs) are accommodated in the design by means of the vacuum vessel pressure suppression system, the various confinement structures, and the detriation and filtering systems, with the result that the assessed consequences of possible LOCAs are commensurate with the conservative ITER release guidance.

Decay heat densities in ITER-FEAT are so small that no emergency cooling of the in-vessel components is needed. The vacuum vessel cooling system has the capability to passively remove all decay heat via natural circulation. Maximum temperatures of the in-vessel components during accidents are below 330°C with vacuum vessel cooling only. These temperatures are sufficient to radiate the power from the in-vessel components to the vacuum vessel which transports the power to the ultimate heat sink. No significant chemical reactions occur between steam/air and Be-dust at these temperatures. Venting of the cryostat and air convection at the outer cryostat surface limit the maximum temperatures of the in-vessel components to about 350°C without any cooling of the vacuum vessel.

Safety – Conclusions
In recognition of the central importance of the safety and environmental aspects of ITER-FEAT, a rigorous approach is being pursued, by establishing firm and widely recognised safety principles and criteria for the design process against which then to assess the ongoing design work. Building on the comprehensive and detailed safety and environmental analysis undertaken for the 1998 design, the preliminary assessments of ITER-FEAT tends to confirm that the design will meet the project safety objectives and will have, in many respects a reduced overall safety and environmental impact.

8.0 Costs and Schedule

Indicative Cost Estimates

A valid cost estimate of ITER-FEAT will be obtained only after the engineering details have been worked out to provide specifications for an industrial cost analysis to be undertaken by firms of the Parties in the second half of 2000. Pending such analysis, only a rescaling from the costs of the 1998 ITER design can be done as outlined below. However, this simple scaling cannot take into account the improvements in the design and in the industrial fabrication process expected as a result of current design work and supporting analysis and R&D.

For this initial indicative exercise, the 1998 ITER design cost basis was used as fully as possible, retaining the detailed system cost structures developed for that design, with cost scaling being done, as far as reasonable, at the component levels.

All costs are again expressed in the ITER Unit of Account (IUA) defined as $1000 US in January 1989. The relationship between the IUA and the ITER Parties' currencies in January 1989, and the internal escalation factors to early 1999 are shown in Table 8.1.

Table 8.1 Currency parities in January 1989 and escalation factors to Jan 1999

	IUA	US $	ECU (Euro)	¥
Jan 1989 exchange rates	1	1000	875.8	127,510
Internal escalation factors	1	1.35	1.4	1.14

In the many cases where the ITER-FEAT systems have retained their basic design features from the 1998 ITER design, cost can be simply scaled down within an unchanged cost structure. For each system the major cost drivers are identified to recalculate the component materials costs, the tooling, the fabrication, assembly, testing and shipping costs.

The amount of materials is typically associated with the number of components and characteristic size or weight. The tooling cost drivers are selected depending on the specific technological procedures used for each system; often these drivers are used with power scaling factors less than 1, typically 0.7. A similar approach is used for recalculating the labour costs associated with fabrication, assembly, testing and shipping.

Some new design options require the adjustment of the previous cost structure and identification of additional cost drivers. Such changes have to be applied, e.g., to the multi-sectional central solenoid, and the vacuum vessel with added back plate functions following elimination of the backplate etc.

The results of this initial scoping for direct capital costs are summarised in Table 8.2 below. The figures show estimates for total system costs; the impact of deferring certain of the costs have yet to be quantified.

Table 8.2 Indicative cost breakdown for ITER-FEAT

Components/systems	**Indicative Cost (kIUA)**	**% of Total**
Magnet Systems	880	27
Vacuum Vessel, Blanket & Divertor	507	16
Power Supplies	224	7
Diagnostics	215	6
Other Main Tokamak Systems	664	21
Heating Systems (73 MW total)	229	7
Buildings, Site Facilities and Balance of Plant	503	16
Total Direct Capital Costs	**3,222**	**100**

Compared to the cost estimate of the 1998 design, the largest cost savings occur for the Magnet systems and tokamak buildings, where reductions of more than one half are indicated. Cost savings approaching 50% can also be expected for other size-dominated systems such as the Vacuum vessel/Blanket, Divertor, Pulsed Power supply, etc. Lesser savings are indicated for function-dominated systems such as balance of plant and even less for auxiliaries such as fuelling, pumping, tritium plant, cryoplant, remote handling and assembly. No savings are indicated for the diagnostics and CODAC systems.

The net result indicates an overall reduction to about 56% of the estimated direct capital costs of the 1998 design. The scope to approach closer to 50% will be better understood only after the further detailed design and analysis needed to optimise choices and after the Parties' industries will have had the opportunity to study and estimate procurement packages which incorporate expected improvements in design and fabrication process. These are now the most important areas of activity for reducing capital costs further towards the target.

Operating costs depend highly on the cost of electricity (assumed at an average cost of 0.05 IUA/MWh), the salaries of the 200 professionals and 400 support personnel, the cost of the divertor high heat flux component replacements and general maintenance expenses, most of which may vary quite substantially amongst the potential host sites for ITER. Simple scalings from the operating cost estimates for the 1998 ITER design suggest an indicative annual figure of about 180 kIUA over the first ten years of ITER operation — a saving of almost 50%.

The main driver for decommissioning costs included in this estimate is the amount of work necessary to de-activate the machine at the end of the plant operation, remove all in-vessel components and then, after activity decay, finally remove the ex-vessel components and dismantle the vacuum vessel. The required manpower for these operations is scaled according to the size and number of sections of the vacuum vessel, assuming a constant cost for additional equipment envisaged in the 1998 ITER design. The costs of transportation and long term storage of the activated material is not taken into account. On this basis a cost of about 170 kIUA for the assumed decommissioning is indicated - a saving of about 45%.

Schedules

The overall project plan is composed of an eight years construction phase including the commissioning necessary for the first hydrogen plasma discharge, followed by approximately

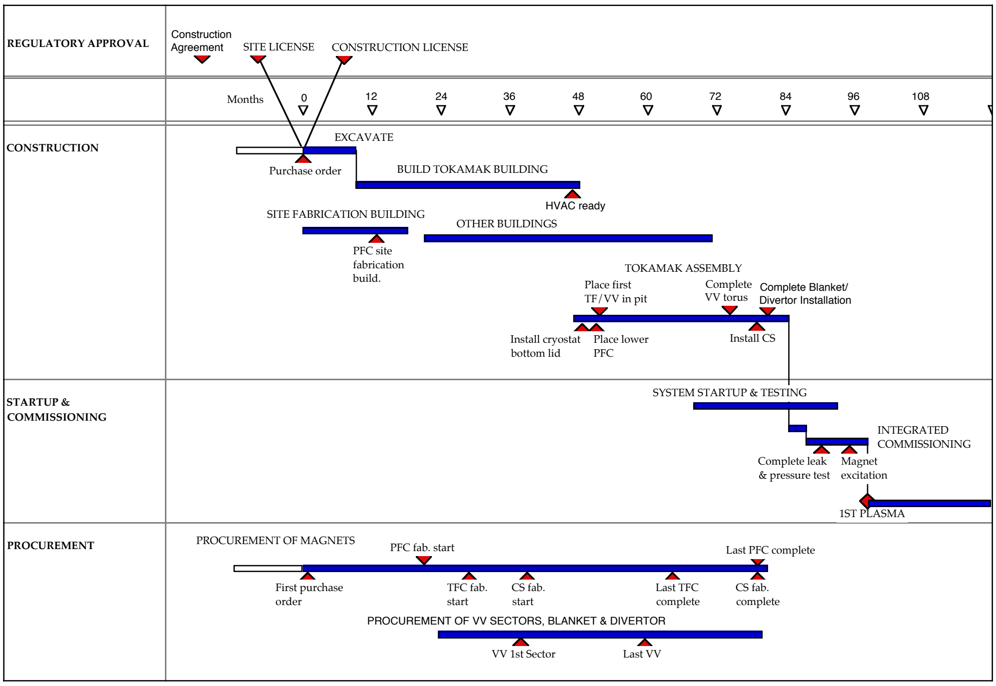
20 years of machine operation. For illustrative purposes this is divided into four phases: two and half years of hydrogen plasma operation, one and a half years of D operation, 3 years of DT operation to low fluence, and for the remaining time a higher fluence DT operation phase. There may be a one to two year machine modification phase before starting the second DT phase in which the outboard shield blanket can be replaced with a breeding blanket. A three year de-activation phase follows after twenty years of operation. The ITER organization has responsibility up to the end of this phase for the ITER facility, which is then handed over to an organization inside the Host Party for dismantling and disposal processes. Figure 8.1 shows the construction planning and figure 8.2 shows the planning for the first ten years of operation.

Construction Schedule

On the assumption that an appropriate amount of technical work has been completed and that an appropriate ITER organisation comes into operation when an agreement to construct ITER is signed, the start of the actual construction on the site depends upon when a site license or construction license is issued by the regulatory authority of the Host Party. Therefore, the dates in the construction schedule are measured in months from a start date (“T = 0”) defined as the date at which the actual construction work of excavation for the tokamak building is started, immediately after the site license or construction license is issued. Documents required for the formal regulatory process are prepared by the Host Party to allow the regulatory process to start immediately after the signing of the Construction Agreement and to provide a licence after 12-24 months.

As previously, the construction plan is based on “just in time” delivery; the construction planning therefore depends strongly on timely preparations and efficient processing of contracts especially for the long lead-time items and for the critical buildings. A prompt start to the Tokamak Building and to the PF fabrication building (which will later be converted to the two cryoplant buildings) are the first steps to the critical path. This presumes that any necessary site preparations will have been completed by the Host before T = 0. The procurement schedules for the superconductor strand, for the TF coils are also critical early actions and it is important to ensure completion of the necessary preparations to launch these procurements.

Fig 8.1 Construction schedule for ITER-FEAT



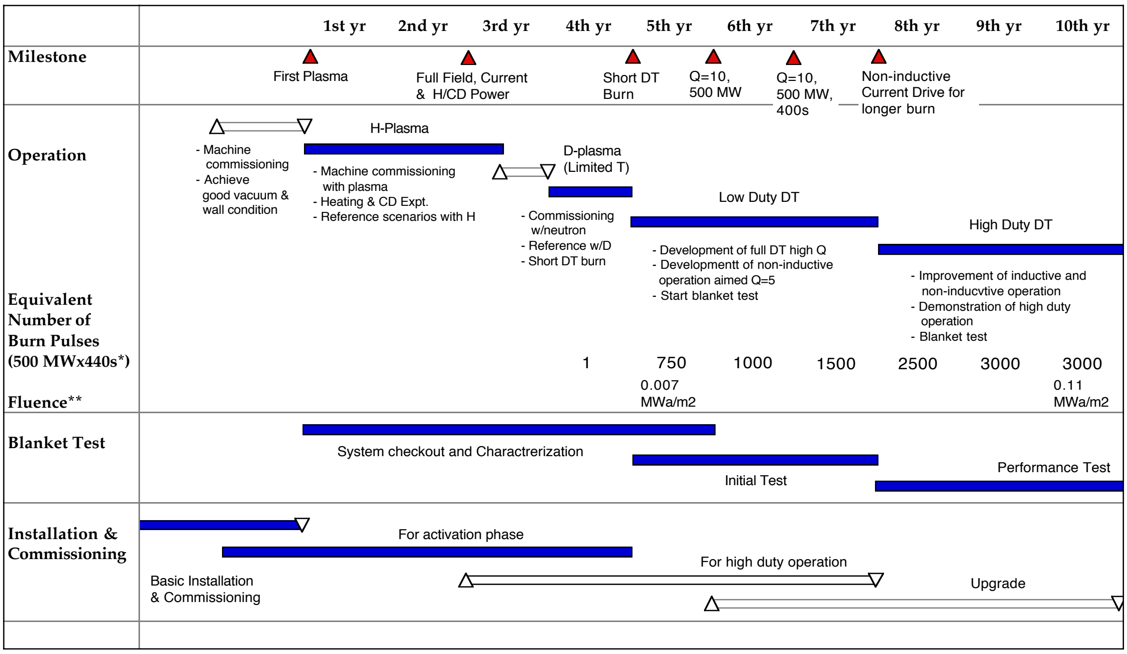
Operation Plans

Operation starts from the first plasma with hydrogen. The ITER machine will be fully commissioned and operated at full plasma current and the full heating power with H plasma discharges. At this time, operating with D plasma discharges with limited tritium, will allow all components and processes to be commissioned ready to work with tritium and with neutron irradiation, before the full deuterium-tritium operation starts to develop high-Q inductive, non-inductive and highly reliable operations suitable for blanket testing.

Hydrogen Phase (H-Phase)

In this first phase of operation (H-phase), no fusion reactions occur, and ITER in-vessel components are not activated and are not contaminated by tritium. ITER will be commissioned with tokamak discharges with the same electromagnetic characteristics as during active operation. By the end of this phase, the nominal plasma current will have been achieved at the maximum toroidal magnetic field and about 70 MW of external heating power with a flat-top duration of about one hundred seconds. The plasma scenario and its control in normal and off-normal conditions will have been established. The heat flux on the limiter and the peak heat flux on the divertor target will be in the same range of average values as for the reference operation for the DT-phase. Depending on plasma confinement characteristics with hydrogen (achievement of good H-mode at large enough densities), many features of the future operation of DT can be explored. Therefore the duration of this period may be lengthened if optimistic results are achieved.

Fig 8.2 Schedule for the first ten years of ITER-FEAT operation



* The burn time of 400 sec includes 400 sec flat top and equivalent time which additional flux is counted during ramp-up and ramp-down.
** Fluence at outboard midplane (Neutron wall load is 0.57 MW/m2 in average, 0.65MW/m2 at outboard and 0.41 MW/m2 at in board.)

Deuterium Phase with Limited Tritium Use (D-Phase)

The main purpose of this phase is to assess the mass scalings of performance, by comparison with H operation, and more accurately predict performance with DT, taking any necessary steps to correct or improve plasma control in preparation for full DT operation. By using limited amounts of tritium in a deuterium plasma, the integrated ITER system can be commissioned, especially with regard to shielding performance, including:

- "nuclear commissioning" of the machine with D/(T) plasma, including check and calibration of nuclear diagnostics, shielding test and radiation monitoring;
- research confirming operation with D/(T) plasma, albeit for short pulses;

Characteristics of D plasma behaviour are expected to be very similar to that of DT even if the alpha heating power is much less than the external heating power. Therefore, the reference plasma operational scenario including L-mode to H-mode transition, very short burn, demonstration of ELMy H-mode for a long period and plasma termination may be confirmed in this phase. The tritium balance can also be studied, and no vacuum vent is planned.

Deuterium-tritium plasma phases (DT-Phases)

Initially, physics studies will be done gradually by increasing and optimising the plasma operation space especially by developing reference scenarios for inductive and non-inductive operations. After developing reliable operation scenarios, series of pulses repeated continuously for a few days are planned mainly for engineering tests particularly relevant to breeding blanket test modules. The fluence at the end of this ~ 6 year phase will be typically ~ 0.1 $MWam^{-2}$.

A detailed operational plan for a second DT phase beyond the first ten years of operation has not been developed because it will depend on the plasma performance and operating experience obtained thus far. However, it is foreseen that there will be more emphasis on optimization of performances and reliable operation to produce higher neutron fluxes and fluence, using the most promising operational modes developed in the previous phases. The average neutron fluence on the first wall is planned to reach at least 0.3 $MWam^{-2}$ at the end of the 20-year operation program.

Tritium Supply

During the first ten years of ITER operation, the equivalent total burn duration at 500 MW is planned to be about 0.15 years or the total equivalent number of pulses is 11,800 at 500 MW. The net consumption of tritium with 500 MW and 400 s burn is about 0.4 g. including heating-up and cool-down phases, and the total consumption during the first ten years are about 5 kg. To achieve the reference average neutron fluence on the first wall of 0.3 MWa/m^2, a total net burn duration of 0.53 years at 500 MW of fusion power is needed, and about an additional 10 kg will be consumed. This tritium can be supplied by external sources.

Tritium Breeding Blanket Test Programme

ITER should "test tritium breeding blanket concepts that would lead in a future fusion reactor to tritium self-sufficiency, the extraction of high-grade heat, and electricity generation." To achieve this testing the ITER Parties will provide specific modules of their own design to be introduced in a few ITER equatorial ports. A common test blanket programme will be established. with the following main objectives:

1) to demonstrate tritium breeding performance and verify on-line tritium recovery and control systems;
2) to demonstrate high-grade heat extraction suitable for electricity generation;
3) to validate and calibrate the design tools and the database used in the blanket design process including neutronics, electromagnetic, heat transfer, and hydraulics;
4) to demonstrate the integral performance of blanket systems under different loading conditions;
5) to observe possible irradiation effects on the performance of the blanket modules.

Decommissioning Plan

It is assumed that the ITER organisation at the end of operation will be responsible for starting the machine decommissioning through a deactivation period after which the facility will be handed over to an organisation inside the ITER Host Country. The decommissioning plan is based on a logic of resources and equipment usage optimisation and takes into account the statutory Occupational Radiological Exposure (ORE) limits. The plan provides a framework within which the organisation that takes over responsibility for decommissioning can decide when and how to implement the ITER facility dismantling, depending on the financial, schedule, resources and/or any other priorities applicable at the time. Flexibility is provided by the use of two separate phases. Each phase duration and activity can be modified (to a certain extent) to accommodate the organisation requirements and constraints.

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 (assuming that no components cooling will be further required) and processed to remove the 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. 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 spread of contamination or present unacceptable hazards to the public or workers. These activities, part of phase 1 of the decommissioning schedule, will be carried out by the ITER organization using the remote handling facilities and staff existing at the end of the project. 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 of some decades for radioactive decay after which final dismantling and disposal could proceed.

Conclusions

1 The ITER-FEAT outline design meets the requirements set at the ITER Meeting in Cadarache, March 1999. It facilitates the exploration of a domain of inductive operation around $Q \geq 10$, in which isotropic α-particles are the dominant source of plasma heating and the main determinant of plasma behaviour. It could be used, with appropriate enhancements to the heating and current drive systems, to approach steady-state operation with $Q \geq 5$.

2 The design point for ITER-FEAT results from systems analysis and intensive joint assessments by Task Forces involving all the Parties. It represents an agreed appropriate compromise among the many interacting scientific, technical and cost constraints and objectives, recognising the importance of providing robustness to the unavoidable uncertainties of the plasma performance projections and of the need to offer capacity to exploit new physics results and understandings. The performance projections are based on a conservative choice of scaling extrapolation. Tools have been identified to allow the boundaries of the operating domain to be expanded and to mitigate possible problems should actual performance prove to lie in the adverse regions of the uncertainty ranges.

3 The parameter set and size appear to be sound even though the margins against uncertainty of the confinement extrapolation are not so large as first envisaged. It would be imprudent to compress or constrain the design further in the hope of achieving further cost savings.

4 Thus, in meeting the ITER mission to demonstrate the scientific feasibility of fusion energy, the conditions of plasma operation in ITER-FEAT will make technological demands that necessarily integrate and demonstrate key technologies for fusion energy - notably superconducting magnet coils, fusion fuel cycle operation, accommodation of heat flux and neutron flux and fluences and the application of remote handling and maintenance technologies. ITER-FEAT will have the capacity to test principles and concepts for other fusion technologies such as breeding blanket modules.

5 The engineering features of the design rest on the approaches and solutions developed and qualified for the 1998 ITER design. The continuing flow of results from the large technology R&D projects provide a basis for confidence in the feasibility, performance and operating margins for the various systems.

6 The initial cost scoping exercise, based mainly on simple scaling from the industrial cost studies of the 1998 ITER Final Design Report, indicates a total capital cost estimated at about 56% that of the 1998 design and provides mainly the relative cost of the different systems as a percentage of the total. Indicative estimates of operating costs are about 51% those estimated for the 1998 design. A proper cost estimate of ITER-FEAT will be obtainable only after a new round of industrial estimates of procurement packages based on detailed design of components and systems. Such studies will be able to quantify the opportunities for re-optimising the manufacturing around the new design and for taking full benefit of the results emerging from the large technology R&D programmes. Thus, there appears to be scope for further reductions in total capital cost.

7 Analysis to date indicates that the favourable overall assessment of the safety and environmental characteristics of the 1998 ITER design applies also to the ITER-FEAT outline design; the analysis is being further refined around the specifics of the design and planned operations. The safety principles and criteria for ITER safety are being further developed with the aim of developing a consensus among the Parties in this area with their regulatory authorities before the case will be presented to the Host Authority.

8 Subject to the views of the ITER Council and of the Parties on the content of the present report and its supporting technical information, the project is in a position now to proceed to detailed design of the ITER-FEAT systems, to resolve the remaining open design issues, to prepare the inputs for industrial cost estimates, and to extend the safety and environmental assessments with the view to providing by the end of the ITER EDA extension, a sufficient technical basis for a possible decision by the Parties to proceed to joint construction and operation of ITER-FEAT.

TAC
15 June 2000

ITER
TECHNICAL ADVISORY
COMMITTEE MEETING

25-27 June 2000
St Petersburg

ITER Technology R&D Progress Report

Report by the Director

ITER Technology R&D Progress Report

Table of Content

1. Introduction and Overview of R&D status

The major technical challenges in ITER are:

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

The overall philosophy for the ITER design has been to use established approaches through detailed analysis and to validate their application to ITER through technology R&D. including fabrication of full scale or scalable models of key components. All this R&D work has been done for ITER under collaboration among the Home Teams, with a total resource of about 660 KIUA.

R&D issues for ITER-FEAT are almost the same as for the 1998 ITER design. Major developments and fabrication have been completed and tests have significantly progressed. The technical output from the R&D validates the technologies and confirms the manufacturing techniques and quality assurance incorporated in the ITER design, and supports the manufacturing cost estimates for important key cost drivers.

The testing of models is continuing to demonstrate their performance margin and/or to optimize their operational use.

Their realisation offers insights useful for a possible future collaborative construction activity. Valuable and relevant experience has already been gained in the management of industrial scale, cross-party ventures.

The successful progress of these projects increases confidence in the possibility of jointly constructing ITER in an international project framework. The R&D present status is summarized in the following: details are given in Chap. 2 and Chap. 3.

Significant efforts and resources have been devoted to the Seven Large R&D Projects which cover all the major key components of the basic machine of ITER and their maintenance tools.

Central Solenoid (CS) and Toroidal Field (TF) Model Coils Projects (Ll and L2))

These two projects are working towards developing the superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. The Model Coil Projects are intended to drive the development of the ITER full-scale conductor, including the manufacturing of strand, cable, conduit and terminations, and the conductor R&D in relation to AC losses, stability and joint performance. These Model Coil Projects also integrate the supporting R&D programmes on coil manufacturing technologies,

including electrical insulation, winding processes (wind, react, and transfer) and quality assurance. 29 t of Nb_3Sn strand, from seven different suppliers throughout the four Parties, has been produced and qualified. This reliable production expanded and demonstrated the industrial manufacturing capability for the production of the 480 t of high performance Nb_3Sn strand as required for ITER FEAT.

For the CS model coil, the cabling and jacketing technologies and winding techniques have been established and all these activities have been completed. The next critical step, the heat treatment to react the superconducting alloy without degrading the mechanical properties of the Incoloy jacket, has been successfully completed. By using approximately 25 t of the strand, the inner module (US), the outer module (JA), and the insert coil (JA) were fabricated and assembled. In April 2000, the maximum field of 13 T with a cable current of 46 kA has been successfully achieved in the ITER dedicated test facility at JAERI. The stored energy of 640 MJ at 13 T has been safely dumped with a time constant as short as 6 s. By comparison, the energy discharge time in the full size CS is 11 s. The insert coil has been also tested at 13 T. The size of the CS model coil (3.6 m in diameter and 2 m in height) is almost the same as a module (4 m in diameter and 2 m in height) of the ITER-FEAT Central Solenoid and the maximum field and the coil current are the same.

For the TF model coil, forging and machining of the radial plates have been completed. Cabling, jacketing, winding, reaction treatment and transfer of the reacted conductor in the radial plates have also been successfully demonstrated. The coil is fully assembled except for the final impregnation of the winding pack in the coil case, which is underway. All the work has been performed in EU. The coil is expected to be delivered to FZK, Karlsruhe in the summer of 2000. The Model Coil uses a cable similar to the full-size TF coil cable and the cross section of the TF model coil is smaller but comparable in size to that of the ITER-FEAT TF coil. The model coil will be tested first on its own and later in conjunction with the LCT coil in the TOSKA facility. With the LCT coil, a field of 9.7 T at 80 kA will be achieved. By comparison, the peak field and the operating current are 11.8 T and 68 kA in ITER-FEAT.

In addition, a TF insert coil with a single layer will be tested inside the bore of the CS model coil test facility at JAERI at a field up to 13T. This insert coil will be completed in the RF this year.

A 1 km jacketing test, which exceeds the ITER-FEAT requirements, has been separately demonstrated in the RF.

For the development of the manufacture of the TF coil case, large forged and cast pieces (about 30 t and 20 t respectively) have been produced in the EU. Investigation of the properties of the forging has revealed values exceeding the requirements of 1000 MPa yield stress and 200 $MPam^{1/2}$ fracture toughness, with low fatigue crack growth rates. The casting also shows properties adequate for the low stress regions of the case (yield stress about 750 MPa). Welding trials have demonstrated successful welding of the cast to forged sections.

For the case assembly welds, electron beam (EB) welding is planned for the first pass followed by submerged arc welding for the remainder, to minimise distortion. The welding processes have been qualified, and preparations for the final welding demonstration are underway.

Vacuum Vessel Sector Project (L3)

In the Vacuum Vessel Sector Project, the main objectives are to produce a full-scale sector of the ITER vacuum vessel including the equatorial port, to establish the tolerances, and to undertake initial testing of mechanical and hydraulic performance. The key technologies have been established and, in relation to manufacturing techniques, two full-scale vacuum vessel segments (half-sectors) have been completed in JA industry, using a range of welding techniques, within the required tolerances. At JAERI, they were welded to each other and the equatorial port fabricated in the RF was attached to simulate the field joint planned to be done at the ITER site during assembly of the machine. Remotised welding and cutting systems prepared by the US were also tested and applied.

Blanket Module Project (L4)

The Blanket Module Project aims at producing and testing full-scale modules of the first wall elements and full-scale, partial prototypes of mechanical and hydraulic attachments, as well as demonstrating prototype integration in a model sector. (Originally, R&D on the back plate was planned but cancelled.) The key technology has been successfully developed, tested and qualified.

- A range of crucial material interfaces such as Be-Cu and Cu-stainless steel have been successfully bonded by using hot isostatic pressing (HIP) and other advanced techniques inside each of the four Parties.

- A full-scale model, without the attachments, has been completed in JA.

- The module attachments have been developed and tested in the RF.

- A full-scale module with attachments is under fabrication in the EU. The full-size shield block has been completed by using powder HIP. After the first wall is attached to this block, the module will be tested to confirm that it meets the requirements for anticipated loads, electrical insulation and remote handling together with the necessary accuracy of positioning.

- A port limiter mock-up with Be tiles has been fabricated by using HIP in the RF.

- In parallel with these fabrications, heat cycle and irradiation tests have been performed for the base materials and the bonded structures and have demonstrated that the performance is well within the acceptable level.

Divertor Cassette Project (L5)

The Divertor Cassette Project aims at demonstrating that a divertor can be built within tolerances and withstand the high thermal and mechanical loads.

- A full-scale prototype of a half cassette has been built by the four Parties. Plasma facing components shipped from JA and the RF were installed in the divertor cassette body

fabricated in the US, and hydraulic flux and mechanical tests were performed at Sandia National Laboratory.

- Various components for high heat flux were fabricated and tested in the four Parties. High heat cycle tests show that CfC monoblock survives 20 MW/m^2 x 2000 cycles (EU) and W armours survive 15 MW/m^2 x 1000 s (EU / RF). A large divertor target mock-up with CfC attached to DSCu through OFCu has been successfully tested with 20 MW/m^2 x 1000 cycles from a large hydrogen ion beam with a diameter of 40 cm.

- Irradiation tests have been also performed. For example, CfC brazed on Cu survived 20 MW/m^2 x 1000 cycles after 0.3 dpa irradiation at 320°C. Tests with pulse heat deposition simulating the thermal load due to disruptions have demonstrated erosion but no disruptive failure of CfC armours even with 0.4 dpa irradiation. (The average neutron fluence of 0.3 MWa/m^2 at the first wall gives 0.38 – 0.59 dpa on the CfC divertor target.)

Blanket Remote Handling Project (L6) and Divertor Remote Handling Project (L7)

The last two of the Large Projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions on reasonable timescales. These technologies should provide the flexibility needed for ITER to pursue its scientific and technical goals whilst satisfying stringent safety and environmental requirements. In this area, full-scale tools and facilities have been developed. Their testing will be extended over a long period of time including the ITER operation phase. This is necessary not only for developing the right procedures but also for optimizing their use in detail and minimizing the intervention time. Rescue procedures and equipment to recover equipment and components are also being developed. The facilities will also allow training of operators.

The Blanket Module Remote Handling Project aims at demonstrating that the ITER blanket modules can be replaced remotely. This involves proof-of-principle and related tests of remote handling transport scenarios, including opening and closing of the vacuum vessel, and of the use of a transport vehicle on a monorail inside the vacuum vessel for the installation and removal of blanket modules. At first, the procedures were demonstrated at about one fourth scale so as to reduce the risk and the cost compared with the development of full-scale equipment. Work is now in progress on a full-scale demonstration. The fabrication of the full-scale equipment and tools, such as a 180° rail, a vehicle with telescopic type manipulator, and a welding / cutting / inspection tool, have been completed in JA. The simulation of installation and removal of a simplified, dummy shield blanket module of 4 tons has been successfully performed by using a teach and play-back procedure. The dummy module was installed with only 0.25 mm of clearance between dummy keys and keyways using the intrinsic compliance of the manipulator. Integrated tests in a blanket test platform which simulates the full-scale structure of a 180° ITER in-vessel region are providing comprehensive validation of the remote handling system so as to allow completion of the detailed design of the components and the remote handling equipment. The real in-vessel operation will be done in a gamma field of 10^4 Gy / h. Key elements such as motor, position sensor, wire/cable, glass lens, electrical insulator, periscope and strain gauge have shown to survive tests at 10^6 –10^7 Gy.

In the Divertor Remote Handling Project, the main objective is to demonstrate that the ITER divertor cassette can be installed and removed remotely from the vacuum vessel and remotely refurbished in a hot cell. This involves the design and manufacture of full-scale prototype remote handling equipment and tools, and their testing in a divertor test platform (to simulate a portion of the divertor area of the tokamak) and a divertor refurbishment platform (to simulate the refurbishment facility). Construction of the necessary equipment and facilities has been completed and successful tests carried out with the remote handling transporters and tools procured in the EU, including a central cassette carrier from JA and a transporter from Canada. The system is based on a toroidal transporter that moves on the same rails to which the individual divertor cassettes are attached. This can move a cassette in front of a remote handling port from where the cassette is extracted with a radial transporter that is deployed from a transfer cask. Redesign of equipment is underway commensurate with design changes to the divertor cassette that were necessary for ITER-FEAT.

Other R&D

In addition to the Seven Large R & D Projects, development of key components for fuelling, pumping, tritium processing, heating / current drive, power supply, diagnostic as well as safety-related R &D have significantly progressed.

- For example, a tritium pellet injector has been tested with a total amount of 36 g T_2 and 28 g DT and ejection of a large pellet (10 mm) from a 80 cm radius curved guide tube has been successfully achieved with 285 m / s in the US. Further tritium pellet injector development is continued in the RF.

- A full-scale cryogenic pump for DT, He and impurities has been completed and is under testing in the EU.

- The tritium processing system with 180 g T was successfully operated for 12 weeks in the US.

- Gyrotrons at 170 GHz have been developed and successfully operated at 0.5 MW x 8 s in JA and at 1MW x 1s in the RF.

- Key components for the ICRF antenna and the transmission line have been developed and tested at a higher voltage than the expected operational voltage.

- Almost full-size negative ion sources and high voltage technology (1 MeV) have been developed for NB injection in JA and the EU.

- Mechanical bypass switches and fast-make switches have been developed and successfully tested at 66 kA and explosively actuated circuit breakers at 66 kA and 170 kA at Efremov Institute (170 kA is no longer required for ITER-FEAT).

- Irradiation tests of key components of diagnostics have provided values required for shielding of components and replacement. The radiation-induced emf (RIEMF) effect especially on measurement by magnetic probes is an important issue and is under study. Lifetime of mirrors set near the plasma will be limited by deposition / sputtering and are under investigation.

- Safety-related R&D, such as characterisation of dust in tokamaks, Tritium co-deposited with carbon, and experiments on steam-material reactions, has provided inputs for the key phenomena and data for ITER safety assessments, and the current R&D emphasis is now on verification and validation of data, models and computer codes. Measurement and removal of radioactive and tritiated dust in the vacuum vessel are under investigation.

- Neutron shielding tests by using a 14 MeV neutron source in JA and the EU demonstrates that the accuracy of shielding calculation is within 10 %.

2. R&D Resources Summary and New R&D for ITER-FEAT

1 Introduction

Total resources of 550kIUA (IUA = 10^3 89US$) have been committed by the Home Teams in 680 Technology R&D Task Agreements during six years from July 1992 to July 1998. To date, about 660 Task Agreements have been completed with the appropriate task final reports submitted by the Home Teams and accepted by the JCT. To maintain continuous progress of the project towards its overall objectives, another 140 R&D tasks have been planned and are expected to be completed during the three years extension period for an additional 110kIUA in total.

These additional R&D are identified mainly in two categories (1) continuation and extending of prototype testing to provide further data on operation margins, and (2) R&D associated with cost reduction for the ITER-FEAT design. This last category is briefly described in 2.3.

2 R&D Resources Allocation

Table 2.1 shows the allocation of the R&D resources to the fifteen R&D areas including their sharing among the Parties. The US contribution up to July 1999 are included in italic in the table. The magnets, VV, in-vessel components and remote handling have been identified as the most critical R&D areas and therefore are the focus of the "Seven Large R&D Projects". Table 2.2 shows the distribution of the R&D resources among the Seven Large R&D Projects and the sharing of the resources among the Parties. Table 2.3 shows a summary of the R&D resources distribution in percentage among the different areas.

Table 2.1 Resource Allocation Summary for the R&D areas per Party

Unit : kIUA

R&D Area	EU	JA	RF	*US**	Total
Magnets	66.6	69.8	11.8	*32*	180.2
Vacuum Vessel	7	17.4	5.4	*4.3*	34.1
Blanket and First Wall	36.7	26.4	23.2	*19*	105.3
Divertor & PFC	27.7	22.2	21.8	*25.4*	97.1
In-vessel Remote Handling	36.7	34.1	0	*1.8*	72.6
Fuelling & Pumping	5.5	0.9	2.5	*3.4*	12.3
Tritium System	8.7	5.1	2.2	*5.9*	21.9
Power Supply	3.2	1	7.1	*0*	11.3
IC H&CD	3.9	0.6	0	*2.7*	7.2
EC H&CD	4.8	9.8	6.8	*2.6*	24
NB H&CD	4.4	13.5	2.1	*0*	20
Diagnostics	2.8	5.4	5.2	*2.5*	15.9
Safety Related R&D	5.2	5.3	4.6	*6.9*	22
Miscellaneous	9.2	8	2.4	*1.6*	21.2
Total	222.4	219.5	95.1	*108.1*	645.1

* US contributed until July 1999.

Table 2.2 Resource Allocation Summary for the Seven Large R&D Projects

Unit : kIUA

7L Projects	EU	JA	RF	*US**	Total
L1 CS Model Coil	10	61	4	*22*	97
L2 TF Model Coil	40	0	0	*1*	41
L3 VV Sector	4	19	4	*2*	29
L4 Blanket Module	29	14	12	*9*	64
L5 Divertor Cassette	13	12	9	*21*	55
L6 Blanket RH	3	18	0	*0*	21
L7 Divertor RH	26	3	0	*0*	29
Total	125	127	29	*55*	336

Table 2.3 Percentage of Resources Devoted to the Different R&D Areas

R&D Area	%
Magnets (incl. L-1 & L-2 Projects)	27.9
Vacuum Vessel (incl. L-3 Project)	5.3
Blanket and First Wall including Materials (Incl. L-4 Project)	16.3
Divertor & PFC including Materials (incl. L-5 Projects)	15.1
In-vessel Remote Handling (incl. L-6 & L-7 Project)	11.3
Subtotal	75.8%
Fuelling & Pumping	1.9
Tritium System	3.4
Power Supply	1.8
IC H&CD	1.1
EC H&CD	3.7
NB H&CD	3.1
Diagnostics	2.5
Safety Related R&D	3.4
Miscellaneous (incl. Standard Component Development)	3.3
Total	100.0

3 R&D Associated with the Design Changes or the Cost Reduction for ITER-FEAT

New R&D tasks associated with the ITER-FEAT design and the reduced-cost options have been identified and detailed task specifications have been developed. Their objectives and brief work descriptions are summarized below.

3.1 Magnets (about 5 kIUA)

For the magnet system, two main R&D subjects have been identified to improve the database and allow cost reductions. They relate to the materials and manufacturing technology database for the Central Solenoid and the TF coil winding pack.

<u>R&D for the CS Conductor</u>

The CS conductor is required to operate under high cyclic stresses, up to 400 MPa of tensile stress, for at least 30000 pulses in the ITER-FEAT design. In the CS Model Coil programme, Incoloy is used as the conductor jacket material. There are some outstanding issues with this material, in particular the weldability, for which R&D is underway in the EU. In addition to the Incoloy jacketed conductor, two other conductor designs are under consideration:

(1) a conductor with a thin circular titanium jacket reinforced by steel U-channels;

(2) a conductor with a seamless circle-in square steel tube jacket.

These alternative designs could potentially satisfy the design requirements and thus need R&D. The R&D for the first option should cover the U-channel material characterisation, the Ti tube characterisation and joining, the demonstration of U-channel production and assembly. The R&D for the second option should demonstrate the industrial production of steel extrusions with an optimized metallurgical process to improve fatigue properties.

<u>R&D of TF Coil Winding Pack</u>

The use of radial plates in the TF winding pack employed in the 1998 ITER design is expected to result in a more reliable conductor insulation. However, the fabrication cost of the winding pack with radial plates is expected to be significantly higher than the cost of a more conventional winding pack with square conductors. The R&D work should involve the fabrication of a curved section of a radial plate with investigation of techniques leading to a cost reduction.

<u>Other R&D topics for ITER FEAT</u>

i) R&D of CS Conductor Helium Inlets

In ITER FEAT, the Central Solenoid consists of a stack of pancakes. The helium inlets are placed at the high field inner bore at the point where the conductor transitions between double pancakes. The inlet hole must be the minimum size compatible with a proper distribution of the helium flow into the conductor. Measurements of the hydraulic properties of the flow through the hole are required using various geometries to define the smallest acceptable penetration.

ii) R&D of TF Coil Out-of-plane Support Concept

In ITER FEAT, the TF coil out-of-plane support requires the use of pre-compression rings at each end of the inboard straight legs. These rings provide additional compression between the coils in the region of the shear keys. This arrangement reduces also the toroidal tensile load on the Outer Intercoil Structures. The most promising concept for the precompression ring is a bonded unidirectional glass fibre reinforced composite. Data on the allowable stress and creep behaviour of such composites is required and should be obtained with the fabrication and test of a reduced scale model.

iii) Test of Central Solenoid (CS) joints in pulsed field

The CS joints are placed on the outside of the coil. With a lap joint configuration, the most convenient orientation for the pancake to pancake joints is with the joint axis along the circumferential direction and the contact surface normal to the vertical axis. This exposes the joints at the top and bottom ends of the CS to transverse field that can drive eddy currents across the joint surface. There is little data on the performance of ITER lap joint in this configuration as it did not occur in the 1998 ITER design. The testing could be done in the Pulse Test Facility (PTF) at MIT (USA) since it was built for the purpose of pulse testing of joints. Several types of joint (lap and possibly butt) could be tested to allow a final selection of the joint design for the CS.

A new R&D programme on NbTi conductors is not specific to ITER-FEAT. However, this type of R&D was always considered as necessary but could not be included in the EDA (1992-1998) due to resource limitations. The TAC-14 recommendations acknowledged this situation and requested NbTi R&D. Therefore this useful R&D programme is also described here.

R&D on NbTi conductors

The R&D programme includes the following items.

i) Investigation of strand coatings for AC loss control.
ii) Cabling trials and AC loss/ Transverse resistance measurements.
iii) Joint trials and resistance distribution measurements.
iv) Fabrication and test of a NbTi conductor/joint sample in SULTAN.
v) Production of strand and cabling for the NbTi Insert Coil.
vi) Jacketing of the 45kA NbTi conductor.
vii) Manufacture of insert coil.
viii) Test of NbTi coil.

Work has been started in the EU and RF and the results already available on items i) and ii) above have allowed the selection of the preferred strand coating. Strand production is underway for conductor and joint sample preparation. At a meeting held in March 2000, agreement was reached with all Home Teams on the work programme and sharing of activities for the NbTi Insert coil. The cable is to be produced in the RF and the jacketing and coil manufacture will be carried out in the EU. The Insert Coil manufacture could be completed in one year, and testing could then take place after July 2001. The preferred test facility is the CS Model Coil facility at Naka but in view of the uncertainties regarding the availability of this facility after July 2001, it was agreed to design the Insert Coil such that it will fit in either the Naka or the TOSKA FzK facilities.

3.2 Divertor (about 5 kIUA)

R&D on fabrication of the PFCs to reduce cost for ITER-FEAT:

During the EDA there was considerable success in constructing and testing full-scale vertical target mock-ups. These were armoured with CfC in the region of the strike point of the SOL and elsewhere with tungsten. However, driven by the requirements of ITER-FEAT to reduce costs, this R&D aims to investigate the possibility of developing cost-effective alternatives to the reference design. These include a simplified vertical target that uses a heat sink employing annular coolant flow, and armour – Cu joining techniques suitable for use with CuCrZr that overcome the need for HIPing. Furthermore, with the success of the tungsten

brush armour in accommodating the differential thermal expansion coefficients of tungsten and copper, it has become apparent that it should be possible to manufacture a tungsten armoured target with 20 MWm^{-2} heat flux handling capability for the strike point of the scrape-off-layer (SOL). This would reduce the tritium inventory in the vessel.

R&D on Water Radiolysis at Low Temperature:
The objective of this R&D is to assess the radiolytic decomposition of the water under ITER-FEAT operation conditions, taking particularly into account the decrease of the water coolant temperature to 100 °C.

R&D on modification of the Divertor Remote Handling to ITER-FEAT:
The divertor handling system of ITER-FEAT is conceptually similar to that proposed for the 1998 ITER design (involving radial motions to and from the RH ports together with toroidal motions of the cassettes inside the vessel). The fact that the cassette is now handled in a cantilevered way and that cassette to vessel supports has changed will not be studied by R&D in the EDA. However, a new design for the RH pipe tools suitable for use with curved coolant pipes instead of straight pipes is being studied, as are a laser viewing and metrology system and a first wall inspection robot manipulator, both suitable for deployment through the first wall at the divertor level.

3.3 Cryostat (0.5 kIUA)

R&D on Elastmer Bellows
In the cryostat design of the 1998 ITER design, circular metallic bellows have been employed to connect the interspace duct wall, which is attached to the vacuum vessel port, with the cryostat port. These bellows are required to compensate the differential movements between VV and cryostat. In ITER-FEAT, the same design cannot be used as there would then be insufficient space left for accessing, for repair operations, the region between the equatorial and divertor ports inside the cryostat. Two alternative designs have been proposed involving either metallic, circular bellows that are attached outside the interspace, or rectangular bellows made of reinforced elastomer material. The latter leave maximum space for interventions inside the cryostat near the equatorial and divertor port regions. Hence the incorporation of elastomer bellows is at present considered in the reference layout but their applicability needs to be confirmed by R&D. Rectangular, metallic bellows are not considered due to excessive space requirements.

3.1 CS Model Coil Project (L-1) Progress Summary

1. Objectives

1.1 Overall Objectives

The overall objective is to develop magnet technology to a level that will allow the ITER magnets to be built with confidence. The model coils will provide for the validation of design and analysis, the demonstration of industrial manufacturing methods and of the performance of each component integrated in the magnet, and the demonstration of reliable operation.

The model coils are also intended to drive the development of the ITER full-scale conductor including strand, cable, conduit and terminations. In addition, the model coils serve to integrate the supporting R&D programmes on insulators, joints, material characterization, ac losses, and stability/ramp rate effects. These are to be realized by the implementation of the test programme for the CS model coil and inserts.

1.2 Component Objectives

Stage	Production Objectives	Testing Objectives
Strand	• establish production rates • find jc, hysteresis, unit length compatible with large production	• develop reliable QA procedures (standardise jc, hysteresis tests)
Cable	• gain production experience with long unit lengths, full cross-section	• develop QA procedures (sub-wrap, center channel)
Conductor	• establish jacketing method, welding, pull through and critical parameters (gap, force, cable protection, compaction) • establish production rate • establish jacket material production routes and achievable tolerances	• develop QA procedures for jacket assembly (NDT, leaks etc) • demonstrate short length conductor performance (jc, AC losses)
Coil fabrication	• establish winding method for large current conductors • establish reaction heat treatment conditions for conductor (inside & out) • establish insulation application procedure • demonstrate handling of reacted conductors for assembly of coil • demonstrate joint fabrication under industrial conditions	• develop QA procedures for conductor reaction • develop QA procedures for insulation testing • develop QA procedures for conductor handling • develop acceptance test QA
Overall	• develop quality control procedures for interfaces • clarify shipping and customs procedures	
Supporting R&D	• produce representative components for the ITER CS	• demonstrate joint performance using full size samples • demonstrate insulation materials performance

		• characterize jacket material

1.3 Test Programme Objectives

The overall test programme goal is to validate and solidify the design principles, design procedures, design criteria, operating margins, analysis methods and manufacturing processes, including QA, which are employed in the CS model coil and inserts and are, in essence, capable of application to the ITER magnets. The objectives of the test programme are:

- Perform model coil demonstration tests under ITER operational conditions:
  - DC operation up to 13 T.
  - Pulsed operation to simulate a scenario for the ITER Central Solenoid.
  - Fast discharge
- Characterization of the performance of conductors, joints and terminals.
- Tests on AC losses, ramp-rate limitation, current sharing temperature and quench properties.
- Characterization of coil mechanical, thermal and hydraulic behaviors.
- Tests of all inserts using different types of conductors.
- A limited lifetime simulation with more than 10,000 cycles for the inserts.

To determine the actual margins which have been achieved in the design and manufacture of the coils in selected areas, the CS model coil and inserts will be operated beyond the design conditions in terms of current, temperature, magnetic field and variation of magnetic field.

2. Achievement up to May 2000

2.1 Present Status

Inner and Outer Modules, CS Insert

The manufacture of the CS model coil inner and outer modules and CS insert was completed. These were assembled together with the structures and helium plumbing in the facility at Naka in 1999, as shown in Figures 2.1-1 and 2.1-2. Testing of the CS model coil and CS insert started in April 2000, after the cooldown of the assembly.

TF Insert

The design of the TF insert was changed in October 1999. The conductor will use a titanium conduit instead of Incoloy due to defects found in the longitudinal welds of the Incoloy tubes. It should be noted that titanium is a candidate material for one of the CS conductor jacket options. The development of a titanium jacket, therefore, is part of the new R&D programme that addresses specific ITER-FEAT issues.

A new design of the TF insert, as well as manufacturing tools, are in progress in the RFHT. The RFHT are aiming to complete the manufacture of the TF insert by the end of September 2000.

Nb_3Al Insert

Nb_3Al conductor was fabricated and a trial winding using dummy conductor was also completed. Trial heat treatment of Nb_3Al conductor with termination revealed a major damage at the termination. The difference in the thermal expansion coefficients (from 750°C

to RT) is a possible reason for it. A design modification is being performed with a demonstration using small samples. The Nb_3Al insert is now expected to be delivered to JAERI in December 2000.

2.2 Major Achievements of the L1 Project since June 1997

The following describes the major achievements of the L1 Project since June 1997.

- The last conductors were shipped from Ansaldo in August 1997 for the inner module and in June 1997 for the outer module.
- The first heat treatment for Nb_3Al formation was performed in May 1997 by the USHT, and the last one was in March 1998.
- The outer module was completed at Toshiba and delivered to JAERI Naka on November 11, 1998.
- The inner module was completed at Lockheed Martin and left San Diego on Feb. 8, 1999. It arrived at JAERI Naka on May 10, 1999.
- Installation work at the facility in Naka started on May 10, 1999.
- The CS insert arrived at Naka on May 21, 1999.
- Installation of the CS model coil and CS insert was completed on October 14, 1999.
- The first cooldown of started on November 29, 1999. However, due to a leak found in the helium pipe of structure cooling, the coils were warmed up for the repair.
- The second cooldown of the CS model coil and insert started on March 13, 2000, and was successfully completed on April 6.
- Testing of the CS model coil started on April 11.
- The first test campaign of the DC and AC operations for the CS model coil was completed on May 26, 2000.
- Testing of the CS insert started on May 30, 2000.

2.3 Details of the Achievements in the Test Programme

The test programme has been initiated. So far, a lot of significant achievements have been obtained as follows:

<u>**Cooldown**</u>

- The cooldown took about 24 days, from March 13, 2000 to April 6, including an interruption of two days due to the annual maintenance work of the power distribution system in JAERI. Total cooldown mass is 176 tonnes.
- Detailed cooldown analysis is being performed to compare the results with simulations.

<u>**DC Operation of the CS Model Coil**</u>

- Ramp-up to the nominal current of 46 kA and generation of a maximum field of 13 T (stored energy 640 MJ) with an inlet temperature of 4.5 K. No coil quench occurred, even in the virgin run.
- Ramp-up to 13 T with elevated inlet temperatures of 5.3 K and 6.3 K to the innermost turn (highest field layer) without quench.
- Fast discharge from 13 T with a time constant of 8.5 s, corresponding to a field change of 1.5 T/s, to simulate the operation of ITER CS at plasma breakdown phase.
- Fast discharge from 13 T with a time constant of 5.3 s (shortest), corresponding a maximum filed change 2.5 T/s. The peak voltage of the coil terminals was around 4800 kV.

- Current sharing temperature (Tcs) measurements up to 46 kA for the innermost layers (layer 1 and 11) of the inner and outer modules. A detailed comparison between the coil layer performances and strand characteristics is being performed.
- All 37 conductor joints (35 interlayer joints + 2 terminal joints) worked as designed, generating around 4 W per joint at a current of 46 kA.
- Preliminary estimates on the AC loss performance from the manual dump tests.

AC Operation of the CS Model Coil

The following pulsed operation was performed to simulate the ITER CS operation as closely as possible within the facility capabilities:

- Ramp-up to 46 kA, 13 T, with an inlet temperature of 4.5 K at a ramp rate of 0.4 T/s, flattop of 5 s, followed by ramp-down to zero at 0.7 T/s (Figure 2.3-1).
- Ramp-up to 46 kA, 13 T, with an inlet temperature of 4.5 K at a ramp rate of 0.4 T/s, flattop of 5 s, followed by a fast discharge with a time constant of 8.5 s (1.5 T/s)
- Ramp-up to 46 kA, 13 T, in 26 s (0.5 T/s), flattop of 5 s, ramp-down to 41 kA in 2 s (0.7 T/s), flattop of 5 s, followed by ramp-down to zero in 18 s (0.6 T/s) (Figure 2.3-2).
- Bipolar operation of zero to -11 kA in 6 s, 2 s flattop, -11 kA to +35 kA in 40 s, 4 s flattop, and ramp-down in 19 s (Figure 2.3-3).

All of these operations were performed successfully without a coil quenches. The following pulsed operation was performed to explore the operation margins:

- Ramp-up to 46 kA, 13 T, with inlet temperatures of 6.0 K and 6.5 K at 0.4 T/s, flattop of 5 s, followed by ramp-down to zero at 0.4 T/s. A quench occurred at around 45 kA during the ramp-up in the 6.5 K operation. The others were successful.
- Ramp-up to 46 kA, 13 T, with an inlet temperature of 6.5 K at 0.6 T/s, flattop of 10 s, followed by ramp-down to zero at 0.6 T/s. A quench occurred at around 41 kA during the ramp-up.

Preliminary assessment indicated that these operational limits are relatively in good agreement with the Tcs measurements of layer 1, which may mean that there is no obvious "ramp-rate limitation" in the coil. Further assessment is required to draw final conclusions on this subject.

2.4 Comparison between the Goals and Achievements

Component Development and Coil Manufacture

Table 2.4-1 summarizes the achievements in reference to the component objectives described in section 1.2. All of these achievements, such as the component and tooling designs, manufacturing procedures, and QA programs, are basically applicable to ITER-FEAT, and provide solid technical basis for ITER magnet fabrication.

Table 2.4-1 Achievements in component development and coil manufacture.

Goal	Notes
Conductor Fabrication	
Strand production for CSMC and CS insert	25 tonnes of strand for CSMC produced by 4 Parties using 7 companies, according to the specification

Cabling for CSMC and CS insert	Cabling done in 4 Parties for two grades of conductor
CS jacket material production	6000 m of jacket material supplied by USHT in two types for 2 grades
Jacketing of CSMC and CS insert conductor	Jacket welding, cable insertion & compaction by EUHT
TF jacket material production	2000 m of TF jacket material supplied by USHT to RFHT & EUHT (turned out to be defective).
Demonstration of 960 m pull through	RFHT jacket welding, cable insertion & compaction of TF dummy for full scale demonstration
Coil Fabrication	
Reaction heat treatment specification for CS	Process specification development & demonstration for Nb3Sn formation, consistent with SAGBO control
Winding of all layers for CSMC	Demonstration of 2 types of tooling capability
Terminations, heat treatment, insulation of all layers for CSMC	Manufacturing demonstration of two types of terminations & demonstration of Nb_3Sn formation, consistent with SAGBO control
Factory acceptance of inner & outer modules for CSMC	Confirmation of manufacturing processes for geometry, insulation, helium leak tightness and pressure proof by two fabricators (JAHT & USHT)
Test facility Preparation	Demonstration of reliable operation of 500 g/s Helium circulation system and 50 kA power supply with protection system
Structures & components for assembly	Provide for axial pre loading in the CSMC as well as mechanical integrity of the assembly
CS insert fabrication	Demonstration of winding, factory acceptance tests & QA by third fabricator (JAHT)
Overall	
Interfaces	Demonstration of interface control among four Parties, including QA and licensing (e.g. JA High Pressure Gas Safety Law)

Performances of the Coils

A lot of significant achievements are being made in the test program and data processing is performed in parallel with the testing, as is mentioned in section 3 below. Table 2.4-2 shows a summary of the achievements in comparison with R&D targets and relevance to ITER-FEAT. These results obtained so far are very encouraging and it is confidently expected that the coil performances will meet all of the requirements of ITER CS.

The tests of the CS insert tests, as well as the additional tests of the CS model coil, will allow more detailed and precise assessments.

Table 2.4-2 Achievements of the CS model coil (as of May 31).

Items	R&D target	Achievement	ITER-FEAT Design Values
Operation Current (kA)	46	46	42 / 45
Maximum Field (T)	13	13	13.5 / 12.8

Current Ramp-up Rate	0 => 13 T at 0.027 T/s	0 => 13 T at 0.4 T/s	0 => 8 T[*1] at 0.1 ~ 0.2 T/s 0 => 13.5 T[*2] at 0.045 T/s
Current Ramp-down Rate	-1.2 T/s at 13 T	-1.5 T/s at 13 T	-1.2 T/s at 13 T[*3]
Fast Discharge - delay - decay time constant	 5 s 20 s	To be performed with an initial normal zone	 2 s 7.5 s
Quench Properties (hot spot temperature at cable space)	150 K	Same as above	150 K
AC loss coupling time constant (nτ)	25 ~ 100 ms	~ 100 ms (**tentative value**)	50 ms
Joint Resistance	TF < 1.5 nΩ CS < 6.5 nΩ	1 ~ 2 nΩ (**tentative value**)	TF < 1.5 nΩ CS < 4.5 nΩ
Operation Margin	Stable operation with 2 K Tcs margin	Stable operation with Tcs margin of ≤ 1 K (**tentative value**)	Stable operation with 1 K Tcs margin

[*1] from X-point formation to start of flattop
[*2] for initial magnetization
[*3] at plasma breakdown

2.5 Schedule Delays and Technical Issues Overcame

In January 1995, the project schedule was reviewed and restructured in order to accommodate a delay in the conductor fabrication and some difficulties happened in the RFHT. Even so, the project still had a significant delay due to various technical reasons.

A delay of one year in the conductor fabrication is a result of the accumulation of the delays in strand, cable and jacket material productions and jacketing work. This includes the reallocation of strand production from the USHT to the JAHT in 1996 for layers 9 and 10 of the CS model coil due to budget and schedule constraints in the USHT.

The coil manufacturers encountered a number of major and minor technical difficulties and delays were due to the time required to overcome them. Details of the fabrication processes of the CS model coil modules are summarized in the document attached to FDR in 1998, Seven Large R&D Projects, "L-1 CS Model Coil Project - Achievements -," July 9, 1998.

The main issues are as follows:

<u>Conductor Jacketing</u>

- Unit length of Incoloy jacket sections supplied to the jacketing company was shorter (5~6 m) than the specification (10 m). This was because part of the section had to be cut at both ends to meet dimensional tolerances on the eccentricity. This substantially increased the number of welds in the jacketing line.
- Some butt welds of jacket sections showed small flaws after the compaction and bending, and required repair. The welding technique was improved but the main problem was

identified as incorrect weld wire supplied by the USHT. Recent trials by the EUHT with new wire have identified that the problem is solved.

<u>Winding Conductor and Lead Forming</u>

- Fabrication of the first layer in two-in-hand configuration, including lead forming, took much longer time than expected.
- The conductor winding was an unprecedented operation that needed to handle a heavy-walled square jacket conductor. Significant modifications of the procedures and tooling were performed to achieve the required tolerances and quality of the layer during the initial manufacturing stages.
- In the USHT, each hand was wound in a coil separately and two coils were then assembled by corkscrewing. For some layers the winding operation had to be repeated several times until the final diameter was obtained. The first layer took more than seven months for these operations, which should be compared with 1.5 months in the last (10th) layer.
- In the JAHT, one hand was wound on the mandrel and second hand was wound between the turn of the first hand. Because they used a winding mandrel, it was difficult to know the actual winding diameter in the freestanding condition. The first layer took four months and the last (8th) layer took only a month.
- Lead forming operation required bending of the conductor in 3 dimensions and correction of conductor twist in order to provide a proper interface surface matching the lead of next layer. The JAHT and USHT cooperated together to develop proper processes and tooling for this operation.

<u>Heat Treatment</u>

- Incoloy was well known as a material, which causes SAGBO. Since the beginning of the project, a lot of efforts were made by the JAHT and USHT to establish heat treatment procedures, which should be consistent with SAGBO avoidance. These efforts included trial heat treatments using short samples as well as full-scale dummy coils.
- Results had been successful until the heat treatment of one of the dummy coils performed in March 1997 indicated serious damages due to SAGBO. The JAHT made extensive investigation and to establish reliable precaution measures.
- Another trial heat treatment using dummy conductor identified the needs of an additional heating step in the heat treatment in order to remove contamination on the surface of strands. The detailed procedure was developed by the USHT.
- These problems caused only a few months of delay to start heat treatment since most of the delay was hidden by the delay in winding operation. In addition, many trial procedures minimized the delay in the actual production: there was almost no difference between the 1st and last layers.

<u>Turn Insulation</u>

- Turn insulation operation required stretching of the turns to allow the taping machine to rotate around the conductor without imposing a strain of more than 0.2% on the conductor. After the insulation, rubber pads were installed to provide pressure for curing resin.
- Since these were complicated operations, experiences, as well as optimization of the procedures and tooling, were required to provide sufficient quality and efficient work. Both JAHT and USHT took about three months for the first layer and only one-month for the last layer.

Transfer and Assembly of Layers

- After the insulation, the layers were transferred onto the previous layer to form a module. This operation consisted of stretching of the layer in diameter for insertion of the layer and reduction to a final diameter on the assembly.
- These required large forces to stretch and to reduce diameters, resulting in a serious damage of the protection wrap made of dry glass tape around the turn insulation.
- Shape of the clamps and pushing rollers were modified. Both JAHT and USHT took about three months for the first layer and less than one month for the last layer.

During the testing some technical issues also occurred as follows:

Cold Leak

- A leak appeared in one of the cooling lines to the structure when the cooldown was almost completed. There was no leak found in the inspections performed at factory and at site. However, QA procedures at the manufacturer were identified to be not appropriate. The leak caused a delay of three and half months in the test program.

Pre-loading

- The level of pre-loading of the CS model coil modules did not reach target values. This was mainly due to improper estimate of the Young's moduli of the modules during the design phase.
- The JAHT and US showed that this has no effect on the operation of the CS model coil. Detailed thermal and mechanical analyses will be performed to explain this behavior and to incorporate in the ITER CS design, which also requires pre-loading.

3. Additional Achievement Expected by July 2001

The following will be completed:

- DC and AC operation of the CS insert to allow detailed characterization of the CS model coil conductor.
- Additional DC and AC operation of the CS model coil to perform a fuller assessment of the limiting performance.
- A limited lifetime simulation with more than 10,000 cycles for the CS insert.

With regard to the TF conductor and Nb_3Al inserts,

- Completion of the fabrication
- Testing in the CS model coil assembly

To complete the testing and close the project, it should be ensured that all necessary data at 4 K are taken, and after that, the final reports to cover whole project have to be prepared. These consist of:

1. Quality assurance records of the conductor
   QA records of the conductors, including strands, cables, Incoloy jacket material, and jacketing, are already available.

2. Manufacturing report

The JAHT and the US have prepared the manufacturing reports of the inner and outer modules, including Quality Assurance records. Most of these documents have been delivered to the JCT.

3. Test results, analysis and implication for the ITER magnet design

A testing group, members of which are from the EUHT, JAHT RFHT, JCT and the US, has been organized. The group is responsible for making a test plan, developing detailed test procedures and conducting tests at site, as well as for an analysis of the results and preparation of reports. The members include those on-site and off-site who can directly access to the JAERI computer system remotely. The group is monitoring the progress of the testing and making assessment of the results on daily basis. The reports are being written as the tests proceed, and implications for the ITER magnet design are discussed as appropriate.

Figure 2.1-1 The outer module is being placed outside the inner module which has already been installed in the vacuum chamber.

Figure 2.1-2 CS model coil and CS insert installed in the vacuum chamber at the test facility in JAERI Naka. The preload structure (upper beams and tension rods), helium pipes and top of the coils are shown. Behind is the vacuum chamber lid.

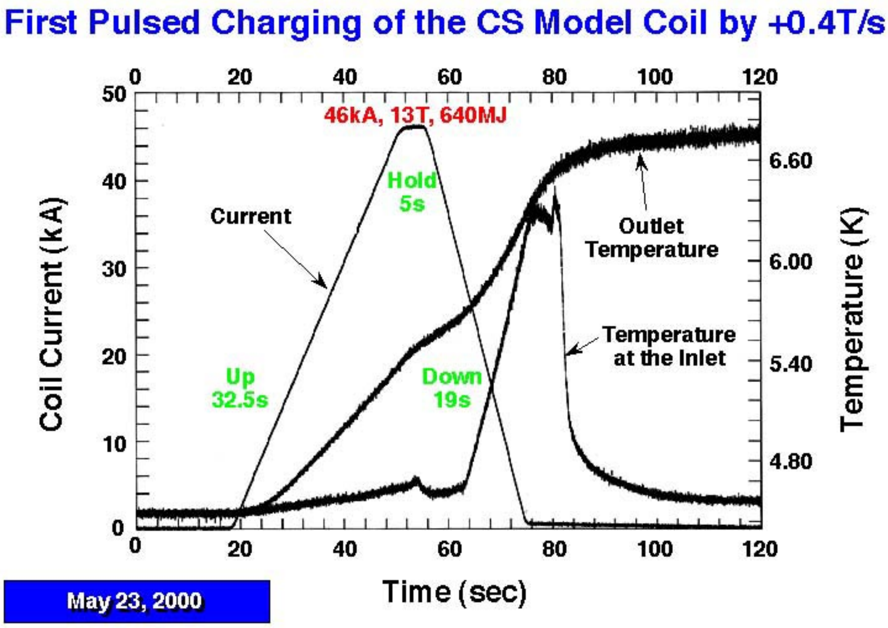


Figure 2.3-1 Ramp-up to 46 kA, 13 T, with an inlet temperature of 4.5 K at a ramp rate of 0.4 T/s, flattop of 5 s, followed by ramp-down to zero at 0.7 T/s. The outlet temperature increased to 6.8 K after the pulse.

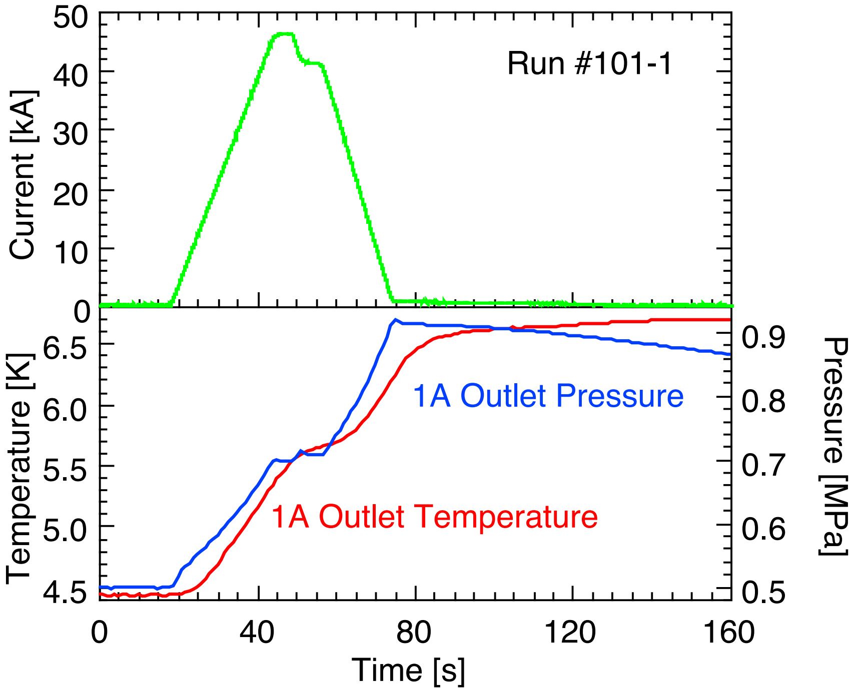


Figure 2.3-3 Ramp-up to 46 kA, 13 T, in 26 s (ramp rate of 0.5 T/s), flattop of 5 s, ramp-down to 41 kA in 2 s (0.7 T/s), flattop of 5 s, followed by ramp-down to zero in 18 s (0.6 T/s). This run was performed to resemble the operation of ITER CS during the plasma breakdown phase.

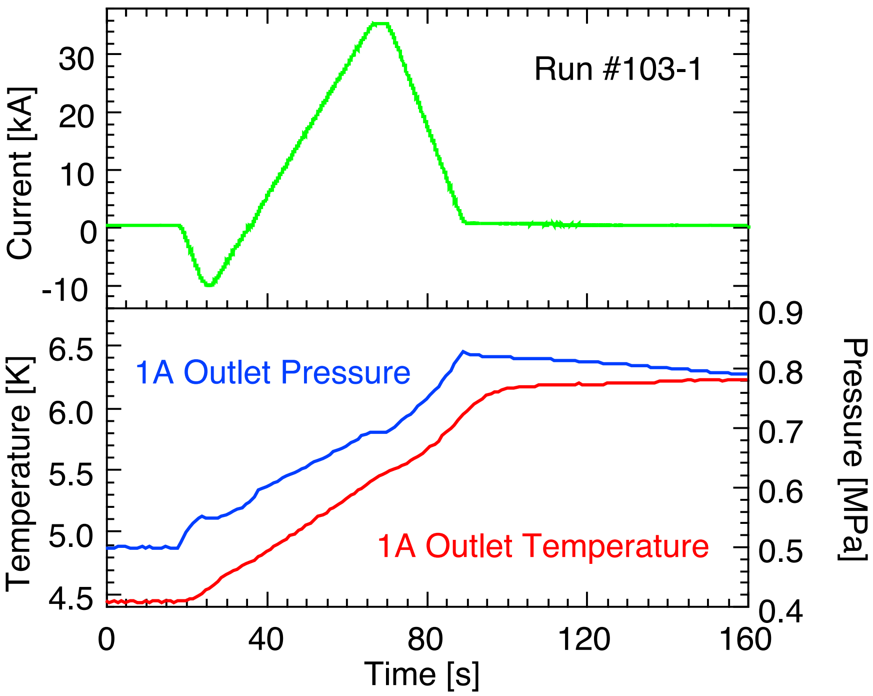


Figure 2.3-3 Bipolar operation of the CS model coil: zero to -11 kA in 6 s, 2 s flattop, -11 kA to +35 kA in 40 s, 4 s flattop, and ramp-down to zero in 19 s. The run was performed to simulate part of the CS operating scenario and to assess the effect of hysteresis loss in the superconducting conductor, which become the largest when the coil current crosses zero.

TF Model Coil Project (L-2) Progress Summary

1. Introduction

The L2 task consists of two main parts:

1) The fabrication of a racetrack shaped TFMC and its test in the TOSKA facility at FzK. The coil has external dimensions about 3.6x2.7m, a stored magnetic energy of about 80MJ and a weight of about 30t (winding pack and case). The winding pack is formed from 5 radial plates each containing a double pancake of the conductor, similar in concept to those used in the TF coils. The conductor current of up to 80kA is greater than the 68kA proposed for the TF coils and coil voltage levels are comparable to those in the full size coil

2) The fabrication of two full size sections of the TF coil case and the qualification of the welding procedures necessary to fabricate these sections into the full size case. Each case section weighs about 35t after completion

The work is almost entirely within the EU HT and is administered by EFDA. The main participants in the work are as follows:

Europa Metalli LMI: Conductor fabrication
AGAN Consortium (Ansaldo, Noell, Alstom): Coil fabrication, busbars and installation in TOSKA. Within the consortium the work division is broadly:

Accel: programme management, analysis and QA
Noell: radial plates, support structures and installation
Ansaldo: winding, joint forming, conductor heat treatment, conductor insulation and installation on the radial plates, radial plate insulation, internal joints
Alstom: assembly of plates into winding pack, joints between plates, winding pack insulation, case fabrication, assembly of winding pack in case, instrumentation, busbars and helium pipework

CEA Cadarache: Joint specification
FzK Karlsruhe: Test facility, current leads and testing in TOSKA
Belleli SpA: Case section fabrication. Kind (Germany) and Creusot-Loire (France) have been major subcontractors in the work.
Outside EU
RFHT and VNIIKP (Moscow): Long length conductor jacketing demonstration

2. TF Model Coil

The main steps of the TFMC that have been completed are as follows:

- fabrication of 4t of Nb3Sn strand with parameters suitable for the ITER TF coils
- extrusion of 1.5km of austenitic stainless steel seamless tubing for the conductor jacket
- fabrication of 1km of Nb3Sn conductor using a pull through technique
- fabrication of 800m of dummy (copper) conductor in a single length using a pull-through technique

- fabrication of 5 radial plates, by forging as a racetrack shaped hollow disc and then machining the grooves, with stress relaxation steps to avoid accumulation of out of plane distortion
- winding the conductor into a mold for the heat treatment, with some undersize to allow for distortion during the superconductor heat treatment
- wrapping the conductor with dry kapton-glass insulation, insertion in the grooves on each side of the radial plate, closure of the groove with a cover plate retained by laser welding.
- soft soldering of the inner joint surfaces (between the two pancakes in the radial plate)
- wrapping the radial plate with an outer layer of dry glass-kapton insulation, impregnation of this layer and the conductor within the groove by vacuum impregnation
- assembly of the five insulated radial plates into a winding pack, impregnation to bond them together
- EB welding of the joint surfaces between the radial plates (soldering is not acceptable due to high thermal stresses created in the TFMC geometry: this issue is not present in the full size coil)
- Wrapping with ground insulation and impregnation
- Fabrication of the coil case from austenitic stainless steel by welding of curved plate sections. The case shape is that of a 'U' with a flat lid (different from that proposed for the TF coil itself)
- placing the coil in the case, filling the gap with glass grains and dry glass fibre (this process may not be appropriate for the full size coil), closure of the case
- fabrication of a support structure for the TFMC in the TOSKA facility (to support the coil on its own and to support the coil when tested next to the EU-LCT coil)
- preparation of the TOSKA facility including supply of 80kA current leads, power supplies, cryogenic supplies and the operating and measurement systems for the testing

The following items remain to be done:

- completion of the impregnation of the case-winding pack gap with epoxy resin
- final machining of the outside of the coil case (including cooling channel grooves)
- installation of the headers for the helium cooling system
- leak testing
- installation of instrumentation
- delivery of the coil to FzK and installation in TOSKA

Delivery of the coil to FzK is officially scheduled for August 2000 but is unlikely to achieve this date. The time schedule has been substantially delayed in the 4 years since the fabrication started. The delay to the start of testing amounts to about 2 years (i.e. fabrication and installation was originally foreseen to last 3 years but is now expected to take 5). Cooldown of the coil is now foreseen for May 2001. The main causes of the delay have been

- several failures of the heat treatment oven
- longer than expected machining time for the radial plates
- unexpected distortion of the joints during heat treatment
- inadequate number of molds for impregnation of each of the radial plates, and improper mold design that added extra tolerances to the plates in the out of plane direction
- delay due to vacuum chamber availability in EB welding of plate to plate joints
- MAG welding problems during closure of the coil case
- inadequate impregnation of the gap between winding pack and case

Much of the data from the TFMC comes from the manufacturing experience gained during the fabrication, rather than the testing. TOSKA testing is limited to about 8T and steady state:

only the high conductor current (up to 80kA) will provide new data on the conductor performance. The manufacturing experience so far has confirmed:

- the feasibility of the pull-through method for conductor jacketing on lengths up to at least 800m
- the level of tolerances required for the conductor, radial plate grooves and radial plate flatness
- the feasibility of laser welding to seal the cover plates over the conductor with acceptable distortion of the plate flatness
- the feasibility of the transfer process of the reacted conductor from the heat treatment mold into the grooves of the radial plates
- the dominant cost items in the fabrication of the radial plates and possible routes to reduce these.

Fig. 2.1 Radial Plates, Blanks and during Machining of Grooves

Fig. 2.2 Winding Conductor into Mold

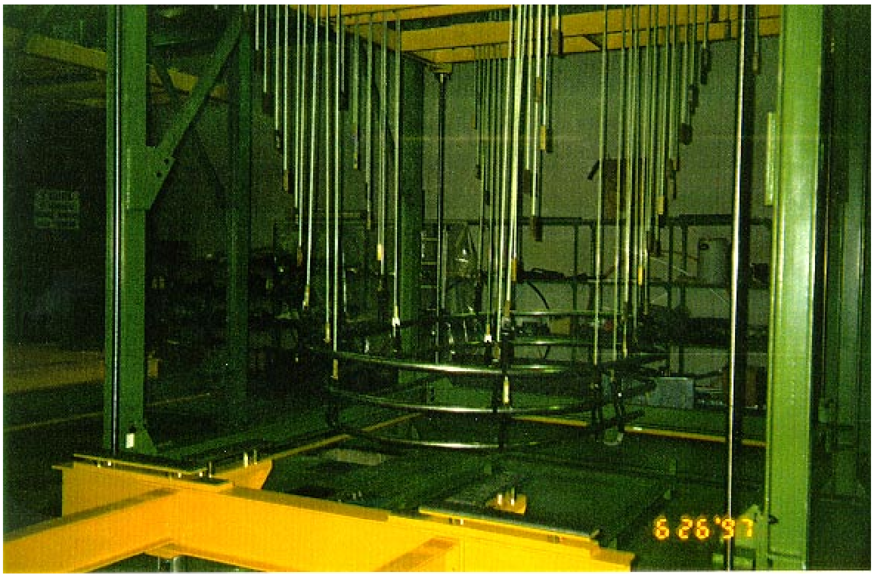


Fig. 2.3 Tool for Transferring Reacted Conductor from Mold to Radial Plate

Fig. 2.44 Putting the Case over the Winding Pack

3. Case Sections

For the case sections, it was decided to make one forged case section to represent the curved region of the TF coil inner leg (where the peak stresses occur) and one cast section to represent the outer leg and outer intercoil structure, where the stresses are low. 316LN was selected as the basic material but was modified slightly outside the ANSI standard for both forging and casting, to improve the cryogenic strength and workability for each process

The achievements at present are as follows:

- two 50t ingots were procured, one from Thysen for the forging and one from Creusot-Loire for the casting
- the inner curved section has been forged with a novel procedure, where the ingot was formed into a hollow square tube which was then curved to the required radius. The curved square tube was then cut lengthways to form the two U sections of the case, ready

for machining. Both sections have been completed, including final machining and NDT inspection

- the outer section (representing one U section of the case and two of the OIS flanges) has been cast. The casting process was not fully successful due to incorrect mold design (the single riser that was provided was inadequate and a cavity developed). However, enough material is available for welding trials and it was not found necessary to repeat the casting with a corrected mold.
- extensive welding trials on cast, forged and cast-forged joins have been performed and appropriate fillers selected and qualified. For both the butt welding of the sections around the perimeter of the coil and the final closure weld, EB will be used for the first 50mm, followed by submerged arc for the remainder. NDT procedures for the welding are being defined
- characterisation of the forging shows very good static and fatigue properties, close or equivalent to those achieved with the specially developed cryogenic steels from Japan. The casting shows reduced properties, as expected, but these appear fully adequate for the outer part of the TF coil case.

The following work remains to be done:

- the forged section will be cut and rewelded to demonstrate the butt welding procedure.
- the two sections of the forging will be welded to demonstrate the case closure procedure
- section of full thickness casting will be welding to each other and to forged section
- a more extensive material characterisation (including the final welds) will be performed
- the NDT procedures will be demonstrated on the welds.

The work has been delayed by about 1 year behind the original schedule due to technical problems during the forging (hot cracking required more frequent reheating and extra machining) and the final machining (the material was harder than anticipated by the machining company). It will be completed before July 2001.

The case section manufacture has provided extensive information for the manufacture of the full size cases. For the first time, a material has been properly qualified both for its final properties and also its welding and forming properties, on full size case sections. The casting offers a cost saving alternative for the manufacture of part of the case, avoiding extensive (and expensive) welding. The forging of tubes for the inner coil is also more efficient than the fabrication of separate sections. The qualification of the NDT procedures for the sections and associated destructive examinations are providing data to support procedures for assessing the operational lifetime of the case.

Fig. 3.1 Forging of Inner Leg Curved Section as Hollow Tube

3.3 VV Sector Project (L-3) Progress Summary

1. Objectives

The issues associated with the fabrication of a Vacuum Vessel (VV) sector are being addressed by the L-3 Project. The primary purpose of this project is to resolve critical VV issues and provide the inputs required to complete the design. The biggest issues that could not be addressed with small scale models were related to fabrication technology. The most critical of these was the determination of the magnitude of welding distortions, dimensional accuracy and achievable tolerances. The achievable tolerances for a sector will impact the positional accuracy of in-vessel components, the required clearances to components on both the inside and outside of the VV, and the design of the field joint used to connect adjacent sectors. Other important objectives of the project are to verify the mechanical and hydraulic characteristics of the double wall design, to demonstrate the assembly and field joint welding of the VV field joints located at the center-line of the ports, and to demonstrate the integration testing of the port extension with the sector.

The main part of this project is the fabrication of the full-scale sector model that has been made by the JAHT. The other parts of the project include:
- The fabrication of the full-scale equatorial port extension which has been made by the RFHT;
- The assembly and field joint testing by the JAHT;
- The integration testing of the port extension with the full-scale sector model by the JAHT and RFHT;
- The development of the advanced cutting, welding and inspection methods by the EUHT.

2. Achievement up to May 2000

2.1 Fabrication of Full-scale Sector Model

The full-scale sector model consists of two 9° half sectors which are designated sectors A and B (see Fig. 2.1). Overall dimensions of the half sectors are 15 m in height and 9 m in width. The material and geometry of the model is very close to that of the 1998 ITER VV. The basic structure is a double wall design with the inner and outer shells made of welded plates, 40 to 60 mm in thickness and separated by ribs which space the shells 0.45 - 0.83 m apart. The design of these two half sectors is different to allow several fabrication methods to be investigated. For example, half sector A was poloidaly segmented into 9 parts during fabrication while half sector B was only segmented into 4 parts. The two segmentation schemes will help assess the impact of poloidal segmentation on sector tolerances. The weld schemes for the two half sectors were also different. The technique selected for half sector A is tungsten inert gas (TIG) and electron beam (EB) welding while TIG and metal active gas (MAG) welding were selected for half sector B. Both half sectors have been successfully fabricated. The total duration of the fabrication took approximately 22 months. Both welding techniques and poloidal segmentation schemes were shown to be feasible. The fabrication technologies required to produce a high quality structure are conventional and available to multiple qualified VV manufacturers. The magnitude of welding distortions and achievable tolerances were determined. Overall achieved tolerances for

the half sectors were less than ± 5 mm for the half sector height and ± 6 mm for half sector width. Non-destructive testing (NDT) including PT, UT and RT was performed for about 10% of the weldment and no defect was found. In addition, the pressure and leak tests were performed and no pressure drop and leakage was detected. Mechanical testing of the sector model under gravity loading resulted in deformations that were less than predicted with a FE model. Testing of a 1/6 scale flow model indicates the variation in the mass flow distribution from the slowest to the fastest channel is about a factor 4, which was consistent with analytical results.

Compared to the ITER-FEAT VV, differences from the sector model are mainly the size and the inclusion of the support structure of the blanket modules into the VV structure. However, since the basic design of the ITER-FEAT VV is the same as the fabricated sector model (i.e., the material, the basic torus shape, and the double wall structure with shielding and cooling water between the shells), this R&D also validated the fundamental feasibility of the ITER-FEAT double wall design. Additional R&D, such as the fabrication of a partial VV sector model including the support structure of the blanket modules, may be required to confirm the fabrication technology and associated tolerances for the ITER-FEAT VV.

2.2 Fabrication of Equatorial Port Extension

The equatorial port extension is a full-scale double wall structure 2.2 m wide by 3.4 m tall by 1.5 m long (see Fig. 2.2). The material is SS 304 and was selected to minimize costs. The welding technique used is a combination of TIG and MAG. This model, like the sector model, was used to develop fabrication technologies for the port extensions. The extension achieved tolerances of ~ ± 4 mm for height and width and ± 1 mm for the wall thickness. No flaw and leakage was detected by the NDT and by the pressure and leak tests, respectively. The duration of manufacturing activities for the port extension was 3 months.

Even though the extension for each port of the ITER-FEAT is a different size and shape, the basic design and fabrication procedure should be common so the technologies developed for the model should be applicable to all extensions for the ITER-FEAT.

2.3 Assembly and Field Joint Testing

Both half sectors were shipped to the Tokai Establishment in JAERI for the assembly and field joint welding test. Alignment of the outer shell butt weld joint was accomplished by adjusting the sector positions and locally jacking the shells into position. Alignment of the root gap and root offset of this joint was achieved to < 0.8 mm. Automatic TIG welding machines mounted to guide rails were used to make the weld joint (see Fig. 2.3). Two welding systems, which were mounted on opposite sides of the vessel (in the poloidal plane), were operated simultaneously in a symmetrical manner to minimize the deformation of the D-shape. A total of 11 to 12 weld passes were required to fill the joint in the 40 mm thick plate of the main VV. A double shielded and twin wire system has been developed for this application. The inner shell field joint includes 8 splice plates for the main shell. Before welding, the individual splice plates were adjusted to fit the joint, installed and tack welded in place. Automatic TIG welding machines mounted to guide rails were again used to make the weld joint on each side of the splice plate but a different

welding head was used. The machines were single wire systems which were developed based on the hot wire switching TIG process. Three systems were mounted in symmetric positions around the VV and operated simultaneously. A total of 11 to 12 weld passes were required to fill the joint along each side of the splice plates. Both the butt weld joint for the VV outer shell and the splice plate joint for the VV inner shell were completed and inspected. The total time required to make the outer shell joint and the inner shell joint were ~320 hours and ~280 hours using the two machines and three machines, respectively. No unacceptable weld flaws were detected. High deposition rates (> 30 g/min) and arc times (~ 20%) were achieved. Average deposition rates of ~ 15-20 g/min and average arc times of ~15-16% were achieved for the field joint welding of the outer shell and inner shell, respectively. A poloidal distribution from average of the weld deformation for the joint was controlled to be uniform to within 1 mm (total values were ~ 4-5 mm) for both butt welding of the outer shell and splice plate welding of the inner shell.

The welding deformation of the VV sector due to the field joint welding has been analyzed using an FE method to establish an analysis method which estimates welding deformations. The comparison of analyzed result with the measured welding deformation showed that the deference is up to 30% in shrinkage and up to 40 % in sectional deformation, respectively.

The assembly and field joint welding required for the sectors is common for the 1998 ITER and the ITER-FEAT VV. The technologies developed for the assembly and field joint welding should be applicable to the ITER-FEAT VV. In addition, the analysis method to estimate welding deformation will be used for the ITER-FEAT VV.

2.4 Integration of Port Extension with VV Full-scale Sector Model

The current L-3 project under the T412 (from 1999) involves the demonstration of remotized welding, cutting and NDT systems on port extension field joints. The technologies under development are fully applicable to the ITER-FEAT VV.

The remotized welding and cutting systems developed by the USHT under the T301 were shipped to the JAHT. To transfer the technology of the systems from the USHT to the JAHT, performance test of the welding system using the VV/port mock-up was conducted for three months with the US's personal participation. Further performance test is planned for the cutting system and the integration of the RFHT's NDT system. This allows the JAHT to fully replace the USHT in this area.

The shape of the field joint between the sector model and the port extension is a rectangle 3.4 m high and 2.2 m wide. A butt weld joint was employed for the outer shell field joint, while splice plates were used for the inner shell field joint. For the outer shell connection, the welding system developed by the JAHT and applied for the field joint welding between the half sectors was employed. The US system was used for the inner shell connection. Prior to the integration testing, the existing test stand of the full-scale sector model was modified to support the port extension (see Fig. 2.4). Detailed dimensions of the sector model and port extension were measured and the joint edge of the port extension was machined to achieve a better alignment of within +1.5/-1.0 mm for surface-to-surface mismatch and within 0.5 mm for the gap. Automatic TIG welding machines mounted on guide rails along the port perimeter were used to make the

weld joint. A total of up to 15 weld layers were required to fill the joint in the 40 mm thick plate. The total time required for the two welding machines to make the joint was about 8 weeks. An assessment of weld shrinkage in the port axis direction indicates that the shrinkage was about 5.0 mm. NDT was performed for about 10% of the weldment and no defect was found.

Preparations for welding of the inner shell field joint (which includes splice plates) were completed including detailed measuring of the joints, individually adjusting each splice plate to fit the joint and installing and aligning the US welding system. The splice plate joints were aligned within ± 2 mm toroidally and radially and tack welded in place. Welding of the joints on each side of the splice plates took four months to complete. Two automatic TIG welding machines mounted on guide rails were used (see Fig. 2.5). The inner shell field joint includes 12 splice plates for the main shell. The 40 mm thick splice plate joints required up to 15 weld layers. Dimensional change observed for the port structure during the welding was up to ±1 mm. NDT of the weld joints using the RFHT's system will be started in July, 2000.

NDT systems to be used for inspection of the weld joints were designed and fabricated by the RFHT. The systems will be adapted to the US remote handling system in Japan and will be used to inspect the field joints welded by the JAHT. The ultra-sonic testing (UT) system has been selected for the purpose. The main feature of this UT system is high resolved defect visualization that allows to determine defect sizes. It is expected that artificial defects as small as 2 mm x 5 mm can be detected for the 60 mm thick plates. Two UT systems for straight flat inspection regions (see Fig. 2.6) and for corner curved regions are to be used for the field joint. Performance test of the UT systems using test samples is being carried out and the systems will be shipped to Japan.

In parallel with the above activities, the advanced methods of cutting, welding and inspection for the VV have been under development by the EUHT. The main objective of the part is to further develop the methods of the cutting, welding/rewelding and NDT, which have the potential of improved cost and technical performance, such as maintaining the stringent tolerances, tight space constraints and reweldable surface conditions, applicable for the ITER-FEAT VV assembly and disassembly. The investigated methods include NdYAG laser cutting and welding, reduced pressure electron beam welding, and UT NDT system.

2.5 Overall Summary

The full-scale sector and port extension models have been fabricated, and the assembly and field joint welding between the half sectors has been demonstrated. The fabrication and assembly technologies required for the VV have been developed. The magnitude of sector fabrication tolerances and welding distortions due to field joint welding have been determined. Computational mechanical analysis provides a reasonable representation of the welding deformation, while thermo-hydraulic analysis provides a reasonable estimate of the flow characteristics for the VV. The feasibility of the field joint welding at the center-line of the ports has also been shown.

Under the current phase of the L-3 project, shipment of the US remotized welding and cutting systems to the JAHT and performance test of the welding were competed. The port extension

has been aligned with the full-scale sector model and the field joint welded using the JAHT's system for the outer shell connection and the USHT's system for the inner shell connection, respectively. UT inspection systems have been designed and fabricated. A performance test of the UT systems is being carried out and the systems will be shipped to Japan to be used for the inspection of the field joints. In addition, the advanced cutting, welding and inspection methods have been under investigation.

The L-3 Project, the Vacuum Vessel Sector Project, is the most important VV R&D during the EDA, which is associated with the fabrication of a sector and demonstration of remotized welding, cutting and NDT systems. The full-scale sector model, fabricated and tested, has been providing critical information related to fabrication and assembly technologies required to produce a high quality sector, and the magnitude of welding distortions, dimensional accuracy and achievable tolerances. Since the basic design of the ITER-FEAT VV is the same as the fabricated sector model, this R&D also validates the fundamental feasibility of the ITER-FEAT VV design. Additional R&D, such as the fabrication of a partial VV sector model including the support structure of the blanket modules, may be required to confirm the fabrication technology and associated tolerances.

3. Additional achievement expected by July 2001

The current phase of the L-3 Project continues until the end of the EDA. Remaining activities from June, 2000 include the NDT demonstration of the weld joints between the port and the port extension using the RFHT's system, cutting sections of the field joint, and cutting and rewelding a test piece. Deformation measurements will be made before and after cutting the port extension. Mechanical test of the rewelded test piece will be performed. In addition, the NDT system will be updated to be fully remotized for the VV application.

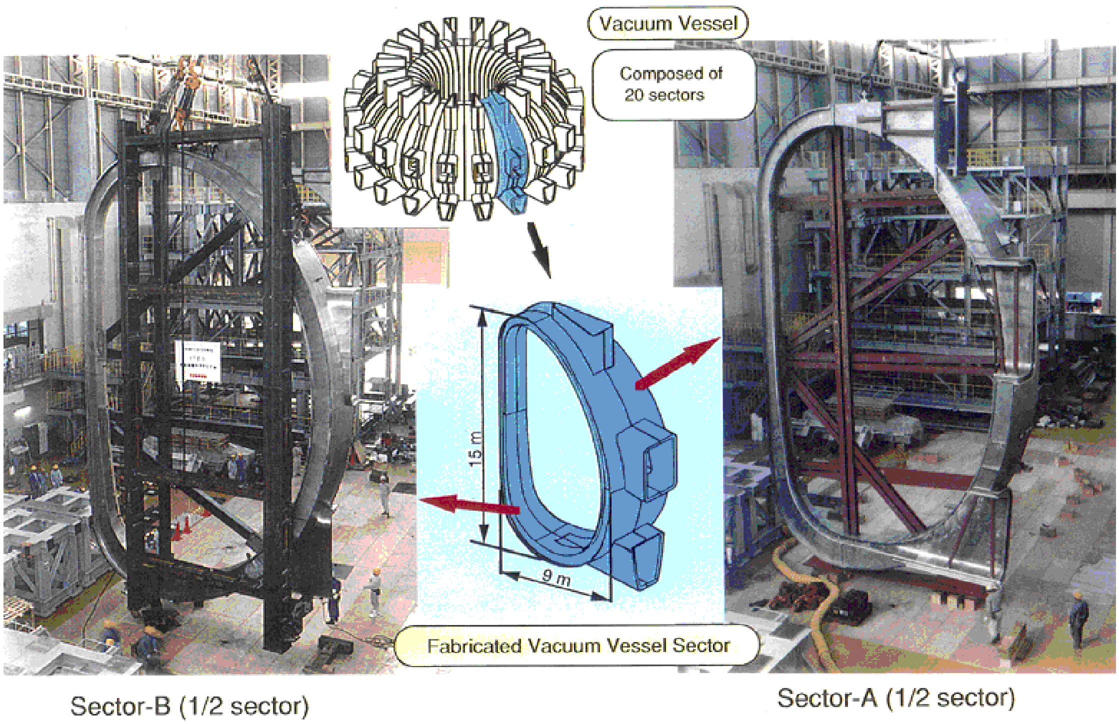


Fig. 2.1 Full-scale Sector Model

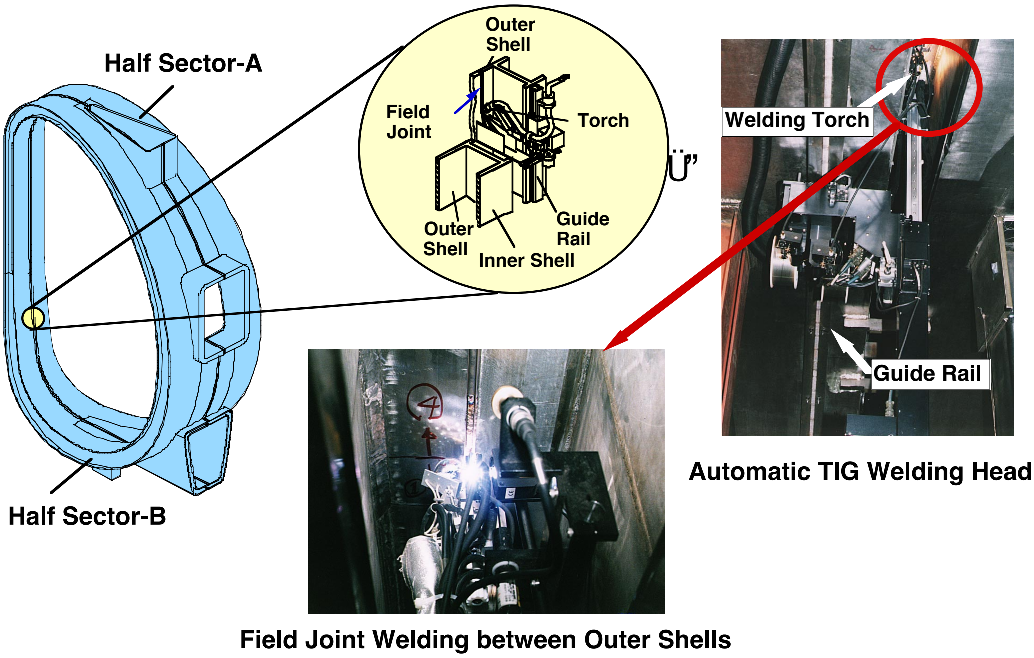


Fig. 2.3 Field Joint Welding Test of VV Sector Model

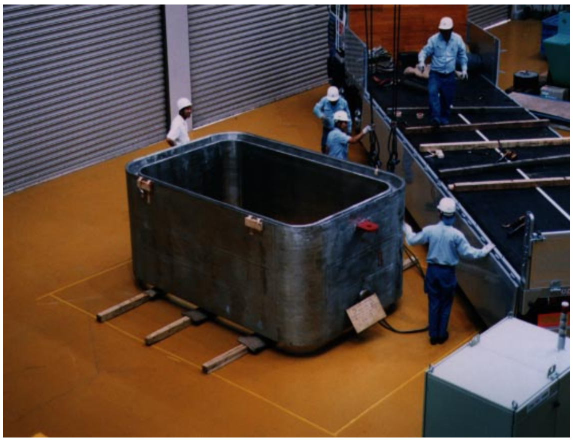


Fig. 2.2 Full-scale Port Extension Model

Fig. 2.4 Port Extension Assembled to VV Sector Model

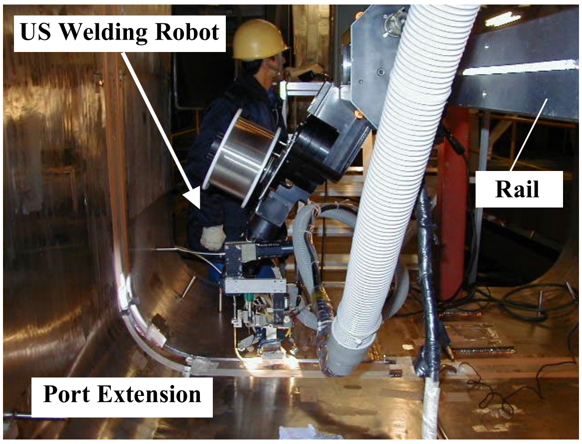


Fig. 2.5 Inner Shell Welding for Integration of Port Extension

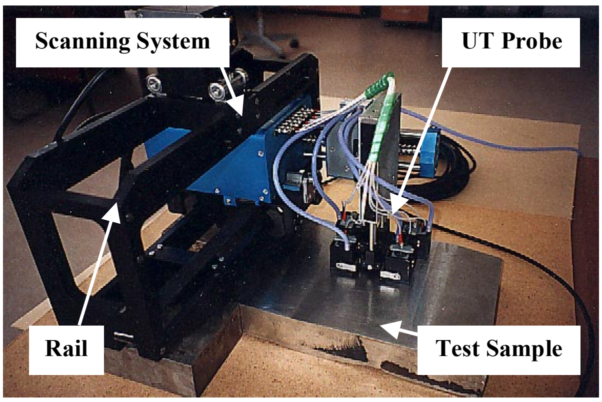


Fig. 2.6 UT Inspection System for Straight Flat Inspection Region

3.4 Blanket Module Project (L-4) Progress Summary

1. Objectives

The L-4 main objectives are:

(1) The development of technologies required for manufacturing the blanket system.
(2) The assessment of the manufacturing feasibility, the performances and the integration of the most important components of the blanket system.

From July 1998 the L-4 Project has been extended to include:

(3) The development and application of lower cost materials and technologies.
(4) The feasibility assessment of ITER-FEAT improved component design by manufacturing new prototypes.

The L-4 objectives have also included the selection, development and characterization of all the blanket materials and the experimental validation of the neutronics calculation tools, upon which depend the shielding sizing of the reactor (reported in other presentations to the TAC).

2. Achievements up to May 2000

The materials and the manufacturing technologies, which have given the best R&D results in terms of reliability and performance, have been selected as reference and used for manufacturing prototypes and mock-ups of the most important blanket components.

2.1 Primary Module and Baffle Region Modules

2.1.1 Primary FW Be Armor to Cu-alloy Heat Sink Joining Technology

<u>Hot Isostatic Pressing with Ti interlayer</u> (800-850°C, 2h, 120 MPa, large Be tiles) is the reference technology for low and medium heat flux and large surfaces. Main results:

- A first EUHT primary FW Be/GlidcopAl-25IG/SS mock-up, (dim.96lx86wx59.4t mm; 1 large tile 9.4 mm thick) has survived 1000 cycles at 2.5 MW/m^2.
- Two new EUHT prototypical primary FW Be/GlidcopAl-25IG/SS mock-ups (dim.250lx66wx80t mm, Fig. 2.1) with several tiles dimensions (ranging from ~40 to ~120 mm in length, 10 and 20 mm in thickness), withstood without damage 13,000 cycles at a surface heat flux of 0.7 MW/m^2 plus a 0.1 MW/m^2 flux in channels inside the SS back-plate to simulate the nuclear heating.
- A HIP technology is also being developed by the JAHT (Fig. 2.2). Strong bonding have been obtained (e.g., up to ~260 Mpa in 4 point bending tests so far using two methods:
  - Coating the DS Cu base material with Cu (PVD, 10μm) and then HIP Be tiles at 620 ^{o}C, 140 Mpa for 2 h;
  - Vacuum plasma spray Aluminum (0.7 mm) on Be, plus coating layers of Al+Ti+Cu on the DS Cu base material (PVD, 10 μm), and finally HIP using an additional intermediate Al layer (0.1 mm thick at 555 ^{o}C 140 Mpa for 2 h).
- Three EUHT small scale mock-ups, with smaller tile dimensions 10x10 up to 20x20 mm, for baffle region modules, were submitted to screening tests to assess the maximum heat flux performance. The mock-ups resisted up to 4.5 MW/m^2 when first tiles started to detach. Some tiles withstood 7 MW/m^2.
- Four EUHT small scale qualification mock-ups for baffle region modules are being manufactured in preparation of a large scale component. They include two straight mock-

ups (one flat tile and one monoblock type) and two bent mock-ups (one flat tile and one monoblock). For these mock-ups HIPing conditions with lower temperature (580 °C) are being alternatively developed and tested. Diffusion bonding trials showed a shear strength of 139 MPa at RT. The equivalent tensile strength is close to the UTS of Be (338 MPa at RT).

2.1.2 Manufacturing Technology for the Primary FW Structure

Solid to solid HIP is the reference (main) manufacturing technology for the primary FW. It has given the best results in terms of quality (e.g. tolerances) and performance either for joining the Cu-alloy heat sink to SS tubes and structure or for joining SS to SS parts. Thermo-mechanical tests including fatigue have shown the robustness of the joints far beyond the specified values. Most of the FW mock-ups manufactured during EDA have been provided with a relatively thick SS back-plate, and are therefore directly relevant for the improved ITER-FEAT separable FW design. Main results:

- A JAHT mock-up of the structural part of the FW, consisting of a DS-Cu heat sink with HIPed SS coolant tubes and a 7 mm SS back-plate (HIP conditions 1050 ^{o}C, 150 Mpa, 2h) withstood up to 5 to 7 MW/m^2 for a total of 2500 cycles.
- Similar results have been obtained by testing four EUHT FW mock-ups; two were with DS-Cu heat sink, one with a CuCrZr and one with a powder HIP CuCrZr heat sink. All were provided with a thicker SS back-plate (60 mm). For these mock-ups two cycles HIP were used, both at 940 ^{o}C, 140 Mpa, 2h.
- A medium scale mock-up (~580 mm h x 100 w x 85 t, Fig. 2.3) of the ITER-FEAT FW (option B) has been already completed by the JAHT. The mock-up is manufactured by HIPing the DSCu to the SS and is without Be armour, but includes the slot penetrating through the 85 mm thick DSCu/SS made by water jet cutting (in most places except at the outlet manifold region where the slot is only 35 mm deep inside the DSCu).
- Three additional mock-ups, manufactured by the EUHT (2 with DSCu and 1 with CuCrZr heat sink) have withstood without any damage long thermal fatigue tests (30000 cycles at 0.75 MW/m^2), Fig. 2.5.

2.1.3 Manufacturing Technology for the Shield Block and Module Manufacturing Feasibility Assessment

Two main technologies are pursued for the shield block manufacturing, Forge & Drill and Powder HIP, developed by the JA HT and EU HT respectively. The first is more suitable for simpler geometry (i.e. the straight modules of the ITER 1998 design and ITER-FEAT), the second can be more convenient for relatively complex module shapes and/or internal cooling channel lay-out, when precise tolerances are not required. The EU and the JA HTs have systematically developed these technologies by firstly manufacturing samples and small scale mock-ups and then manufacturing several medium and large scale mock-ups. The final assessment of the manufacturing feasibility of the blanket module is performed by manufacturing prototypes. Main results:

- The JAHT has manufactured a full size primary wall prototype module (#8 ITER 1998 design, excluding the Be armour, Fig. 2.4 with dimensions ~930mm high x ~1600mm wide x ~350mm thick. The shield block was manufactured by forge & drilling plus cold bending (to shape). The FW was manufactured and attached to the shield block using one step HIP (1050 ^{o}C, 150 Mpa, 2h). No decrease of pressure was observed during the pressure tests (6 MPa, 30 min), and the leak rate was less then 10^{-9} Torr l/sec during the leak tests. The tolerances of the cooling channels were ±0.3 mm and ± 0.2 mm for the

channels with 40 and 24 mm diameter respectively. The maximum tolerances on the pitch were ±0.5 mm. The maximum deformations on the outer diameter of coolant tube are ~ ± 1.0 mm. No defects in the joints have been detected. Additional destructive examinations are in progress to verify these results

- A shield block prototype of the most complex (in shape and pipe gallery layout) shield module (#11 ITER 1998 design) has been manufactured by the EUHT (Fig. 2.6). HIP parameters used 1100 ^{o}C, 100 Mpa, 4h. After completion NDE was performed. Results show that the deviations of the tubes in the first row are less than ± 2 mm for 65% of the tubes. The remaining 35% have a maximum deviation of up to ± 5 mm. The main reasons of these deviations are:
  - A cold bending was necessary after HIP to correct a module manufacturing bowing, and the final radius of the block was smaller than required. The subsequent machining of the front surface to the required shape led then to slightly different tube locations.
  - Local deformations from the straightening process.
  - Different shrinkage of local volumes involving different proportion of powder and tubes due to the fact that the module design was not optimised for powder HIP.

  It can be preliminary concluded that the use of the powder HIPing technology for the shield block is a viable solution, but it requires that the design takes into account the process requirements in order to meet the manufacture tolerances.

2.1.4 Plasma Spray

Be plasma spray coating has also been developed mainly as an alternatively bulk FW protection material and possible "in hot cell" repair method of low heat flux, locally damaged FW areas. Low Pressure Plasma Spray (LPPS) is the reference technique:

- LPPS results in high density coatings, approaching the thermal conductivity and the theoretical density of bulk beryllium (~93%) from room temperature (RT) to 600°C. Coating thickness of the order of 10 mm can be obtained.
- Mock-ups with and without special preparation of the Cu alloy substrate have been manufactured by SANDIA/Los Alamos (US) and tested. They survived 3000 cycles at 1 MW/m^2, with failure limit ~ 2.5 MW/m^2.
- In heat shock experiments instead, cracks parallel to the coating layers occurred in samples, particularly after irradiation. The failure is ascribed to the fabrication method which caused the presence of non melted Be powder between each plasma sprayed layer. An improved method is being developed during EDA extension

2.2 Limiter

2.2.1 Limiter Be Armor to Cu-alloy Heat Sink Joining Technology

Fast amorphous brazing of small tiles is the joining technology that has given the best performance for high heat flux components such as the limiter. The RFHT has developed a fast brazing technique (~ 800°C, for max. a few minutes) with amorphous CuInSnNi braze and using small tiles, which results in a high performance, reliable and robust joint. The fast braze cycles reduce the formation of intermetallics to a minimum. Additionally if a CuCrZr alloy is used as heat sink there is no risk of material properties degradation. The fast heating methods used are direct electrical heating and electron beam. Several mock-ups have been tested. A mock-up with tiles 5x5x5 mm did not fail after 4500 cycles at 12 MW/m^2. For this technology the effect of the tile size on the durability of the joint has also been determined,

Fig. 2.7. Tiles with size 20x20x5 mm detached at 12 MW/m^2, while tiles with size 5x5x5, 7x7x5 and 10x10x5 mm did not detach in this condition. Curved mock-ups have also been tested: no damage has been observed after 1000 cycles at 11 MW/m^2.

2.2.2. Limiter Module Manufacturing Techniques and Limiter Manufacturing Feasibility Assessment

The relevant technologies necessary for the manufacturing the ITER limiter module have been performed by the RFHT. The main manufacturing technologies are HIP to form the plates and e-beam welding for joining the plates together on their rear side:

- All the mechanical parts for a total of 12 limiter plates have been manufactured by the RFHT and used for specific R&D (manufacturing trials).
- Three qualification limiter plates have been HIPed (150 MPa, 1030+5 ^{o}C, 2.5 h):
  - Two have been cut and used for metallurgical examination.
  - One has been used for the optimization of the Be brazing process.
- The final two mock-ups (250lx250wx~45t mm), have also been HIPed with the same parameters. One mock-up with DShG-200 Be 'fast' brazed tiles is completed, Fig. 2.9. The second mock-up which will have the Be armour attached with a slower heating technique (Bochwar) will be completed soon. The two plates will be joined together at their rear side by e.b.welding and will be tested at TSEFEY facility, RF.

2.3 Blanket Attachment System and Assembly

2.3.1 Flexible Cartridge Manufacturing

The reference manufacturing technique developed by the RFHT for the Ti alloy cartridge is machining from a plain rod to form the cartridge hollow cylinder and then milling to produce the flexible spokes. Main results:

- Several flexible cartridges have been manufactured and tested by the RFHT (Fig. 2.8). The thermal tests show that the maximum temperature in the spokes is 220 ^{o}C, (at volumetric heating of 0,6 W/m3) when the supporting surfaces are at 100 ^{o}C. The calculations give a different result (260 ^{o}C). The RFHT considers this difference to be due to enhanced heat transfer because of the low vacuum used (1-0.1 Pa).
- Buckling tests, Fig. 2.10, show that the loss of stability of the flexibles occur at a compression force ~2.6 times higher than the expected disruption load (tests were performed at temperature and with pre-defined spoke displacements).
- The cyclic mechanical fatigue tests showed that the flexibles withstand 10,000 cycles at the required 600KN loads. However during the tests a systematic loss of bolt pre-load occurred, causing fatigue rupture. Further investigations are necessary because these results may indicate the need for unscrewing devices in the attachment.

2.3.2 Assembly Testing

The EU HT has completed preliminary assembly test of the blanket module on the attachment system of the ITER 1998 blanket design. Main results:

- Assembly of the central post alone (without keys) could be made with half of the required tolerances (total gap in average 0.25 mm). The insertion path was linear but skewed so that it required the contemporaneous movement of two translating devices.

- After metrology the big key-pads were machined to fit within tolerances and the assembly of the central post together with the big keys could be performed with the required gaps (0.5 mm).
- Finally the assembly tests with central post, big keys, (dummy) flexibles cartridges and branch pipes were successfully performed. The branch pipe guiding pins tend to jam during disassembly. It is expected that a slight modification of the mating surfaces can solve the problem.
- Rather low pushing and pulling forces (~ 50 N for the insertion and 500 N for the extraction) are needed for the module assembly in the tested conditions (ideal translation movement with minor interference between the mating parts). When the module comes in contact with the branch pipes a much higher force is required (6kN) to complete the assembly. This hydraulic connection design may be a problem for the RH arm.

2.3.3 Bolting Tool

- The EUHT has completed the manufacturing of the remotized hydraulic bolting tool (Fig.2.11) and its instrumentation. It has also started to perform preliminary bolt tightening tests.
- It was not possible to self engage the bolt tool lance in the bolt head. At present this operation is performed manually (the lance must be rotated until it can enter inside the bolt head recess). The design of the bolt head should be modified so that the lance self-engages in the bolt head.

2.3.4 Laser welding and cutting tool

The EU HT has completed the manufacturing and testing of a cutting and welding tool able to operate through a 30 mm module penetration. Main results:

- The EUHT has been successful in welding and cutting the branch pipes, as required.
- Several welds were performed. Metallurgical inspections show good weld quality.
- Cutting has also been performed successfully using the same tool (pulsed YAG laser).
- Additional tests were performed in order to have a substantial dross recovery. The tube cutting was performed in two steps. First, an incomplete but deep kerf is made in order to keep the slag inside the tube. Second, the final cut (0.5 mm) is performed. With this technique most of the slag/dross is kept with the tube that is removed, but the cut surfaces are not suitable for re-welding.
- Additional tests have been performed with a 1mm misalignment of the welding and cutting tool with respect to the pipe axis. The quality of the welding and the cutting operations is considered satisfactory also in this case.

3. Additional achievements expected by July 2001

The L-4 main achievements expected by July 2001 are:

- The development and application of lower cost techniques and materials e.g.:
    - ✓ Use of Be/Cu-alloy alternative joining technologies (e.g. brazing)
    - ✓ Verification of proposed manufacturing methods for reducing costs (e.g. HIP with less stringent requirements, use of alternative cutting methods, use of anti-diffusion barrier for Be armour castellation).
    - ✓ Assessment of the use of CuCrZr with the reference manufacturing technologies.
    - ✓ Use of Be plasma spray
    - ✓ Experimental verification of electrical straps and insulation coatings

- Demonstration of a comprehensive fabrication method for the ITER-FEAT improved module design by fabrication of new prototypes. Separable FW panel prototypes and large size shield block mock-ups are being manufactured with the new manufacturing techniques by the EU and JA HT.

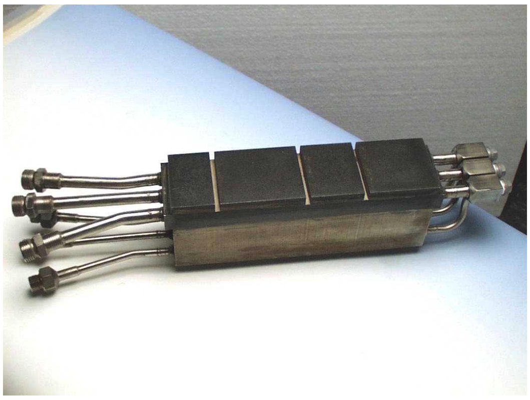


Fig. 2.1 EUTH FW Mock-up with HIPed Be Tiles

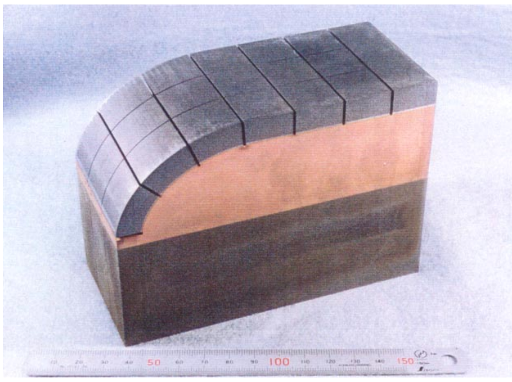


Fig. 2.2 JAHT FW Mock-up with HIPed Be Tiles

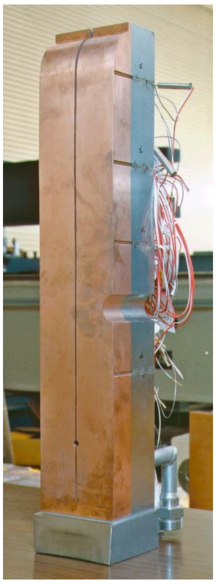


Fig. 2.3 JAHT ITER-FEAT FW - Mock-up (option B)

Fig. 2.4 JAHT ITER EDA FW/Blanket Prototype

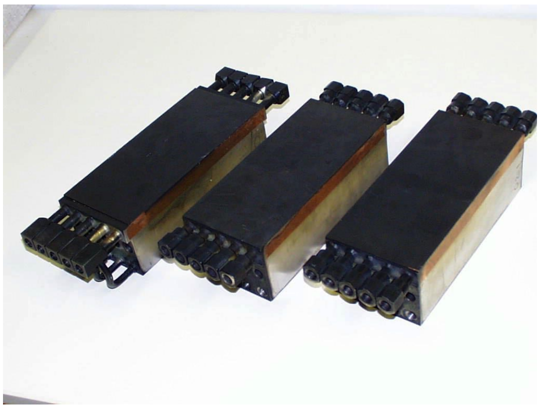


Fig. 2.5 EUHT FW Mock-ups subjected to Long Thermal Fatigue Testing

Fig 2.6 EUHT EUHT Shield Block Prototype

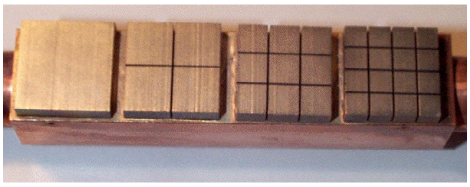


Fig. 2.7 RFHT Mock-up with Be Fast Amorphous Brazing

Fig. 2.8 RFHT Flexible Cartridge Prototypes

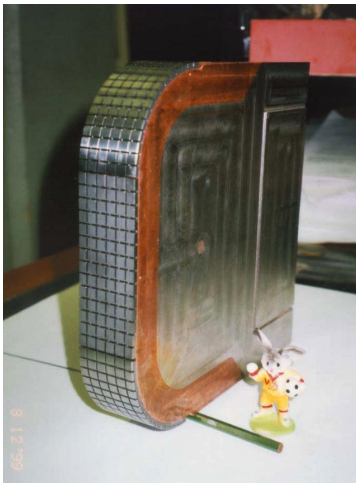


Fig. 2.9 RFHT Port Limiter Mock-up

Fig.2.10 Flexible Cartridge After Buckling Tests

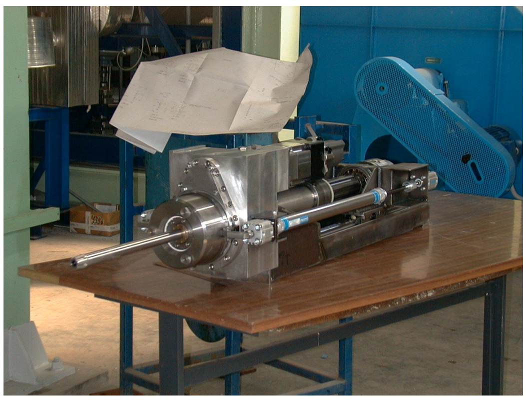


Fig. 2.11 Remotized Hydraulic Bolting Tool

3.5 Divertor Cassette Project (L-5) Progress Summary

At the beginning of the project it was identified that in order to avoid the problems associated with silver transmuting to the vacuum incompatible element cadmium, silver free joining techniques needed to be developed for the armour to heat sink joints of the plasma facing components. Initially, three candidate armour materials were proposed for the divertor (carbon-fibre composite, beryllium and tungsten), although it soon became clear that beryllium was unsuitable for use with the anticipated high heat flux. The achievements made so far for the tungsten and carbon armours are summarised in figs 1 & 2, and highlight the impressive progress made towards meeting, and in some cases exceeding, the ITER-FEAT goals. Future armour joining R&D will focus on adapting the manufacturing process and design for use with the reference Cu for ITER (CuCrZr), simplifying the design in order to improve reliability and reduce costs, and finally, developing reliable NDE for the joints. Other R&D will study designs that mitigate the problems of tritium co-deposition and dust management.

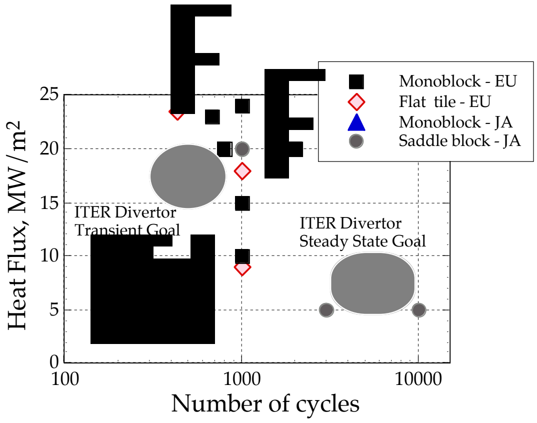


Fig 1 The results of the high heat flux tests of the small and medium scale CFC/Cu mock-ups with Ag-free brazing alloys.

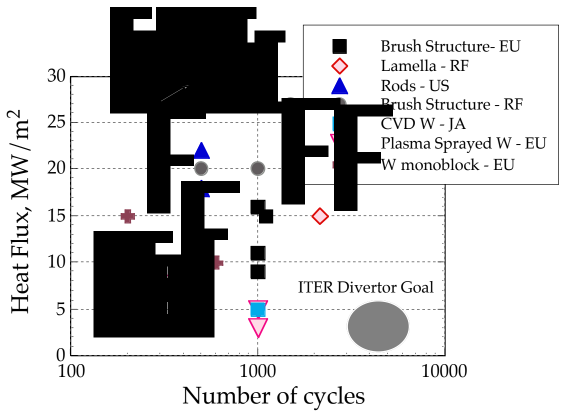


Fig. 2 High heat flux test results of the small scale W/Cu mock-ups.

The EU have built and tested a medium scale vertical target mock-up, which uses the armour and heat sink geometry proposed for the ITER-FEAT reference design. The reference design (fig.3) uses a CfC monoblock near the strike point employing a swirl tape in a 10 mm bore, 12 mm OD tube and W macrobrush in the upper target/baffle region. Under high heat flux testing in the Le Creusot e-beam facility, the mock-up sustained 1000 cycles at 15 $MW.m^{-2}$ on the W macro-brush armour and 2000 cycles at 20 $MW.m^{-2}$ on the CfC armour. Finally, the CfC armour was shown to survive > 30 $MW.m^{-2}$ in a CHF test. The CuCrZr alloy is preferred over Dispersion Strengthened Cu (DS Cu) for the heat sinks of the high heat flux components, because its post-irradiation fracture toughness is much higher. Hence, the EU is now concentrating its efforts on adapting the manufacturing cycle for the reference design to employ low temperature hot isostatic pressing (HIPing) at ~ 500°C, a process that will avoid over-ageing of the precipitation hardened alloy, but brings added manufacturing complexity.

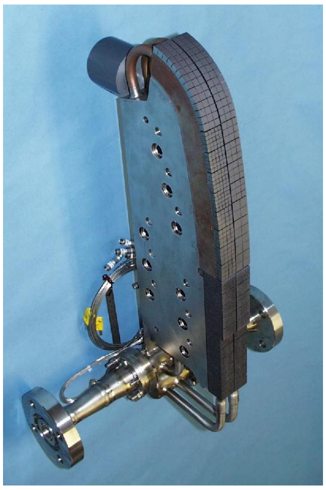


Fig. 3 The EU vertical target prototype

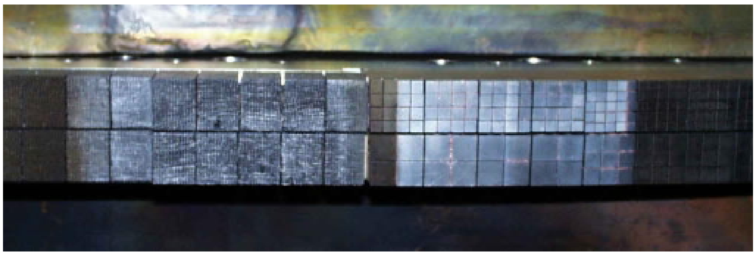


Fig.4 Detail of EU vertical target prototype after high heat flux testing

In line with the recommendation of TAC-13 design simplifications are being sought that can both improve the reliability of the components and bring cost savings. To this end the JA HT have instigated a programme of work to investigate the viability of using an annular flow configuration in conjunction with a CfC monoblock. If successful this design would eliminate the need for the return pipes that are a compromise between providing the flexibility to take the thermal expansion of the monoblock and sustaining the e-m loads during plasma disturbances. The JA HT has tested a 2-D CfC monoblock design (fig. 5) on a tube with a large enough diameter to accommodate the annular flow (triplex tube, 21 OD, 15 ID, inner & outer skin pure Cu, DS-Cu core). Part of the target was cycled at 5 MW.m^{-2} for 3000 cycles, and a few tiles exhibited elevated temperatures characteristic of large CfC to Cu joint flaws. A further two tiles survived 20 MW.m^{-2} for 1000 cycles (fig. 6). Unarmoured, annular flow mock-ups have been built and flow and CHF testing of these is scheduled for the next few months.

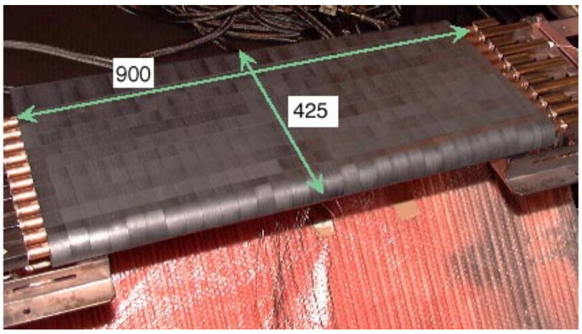


Fig.5 JA HT monoblock mock-up

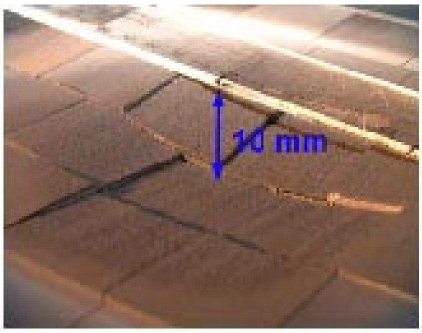


Fig.6 Detail of monoblock mock-up after HHF testing, showing erosion broadly in line with predictions

The RF used a fast brazing technique for the W and beryllium armoured mock-ups as a means of maintaining most of the good mechanical properties of the CuCrZr alloy. This

method passes current through the mock-up to ohmically heat the components (fig.7). The tungsten tiles (44 mm x 44 mm) used a pure Cu interlayer and the beryllium joints were brazed without interlayer. During heat flux testing the W armour survived 1000 cycles at 5 $MW.m^{-2}$ and the Be armour 1000 cycles at 1.5 MW.m-2 (fig.8). The RF is now developing a method, which uses conventional high temperature furnace brazing but with cooling achieved by introducing gas via the component coolant channels. If successful, it is expected that a fast-quench will achieve near optimum mechanical properties in the CuCrZr and at a lower cost than using HIP-ing.

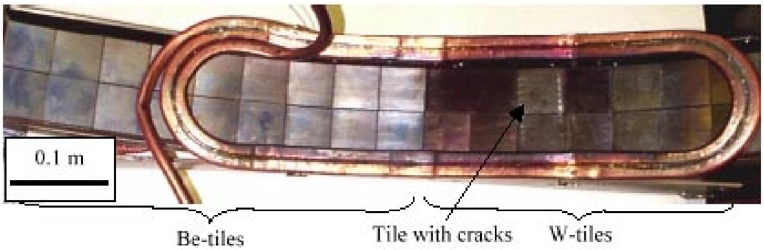


Fig. 7 RF mock-up in the "fast" brazing ohmic heating rig.

Fig. 8 W and Be armoured mock-up after heat flux testing.

Integration mock-ups from US, JA and RF were installed on the US cast steel, fabricated cassette body mock-up of the inner divertor channel (fig. 9). Meanwhile, the outer divertor channel, integration mock-ups from EU and RF are being installed on the EU weld fabricated cassette body (fig.10).

Development of suitable NDE for the brazed joints is important, if reliable components are to be guaranteed. Ultrasonic testing in JA and EU, and thermographic testing using pulses of hot water in the EU (SATIR facility at CEA) have shown promising results.

In summary, the progress made during the EDA has demonstrated the feasibility of building divertor cassettes capable of handling the high heat flux anticipated in the divertor. The programme to the end of the EDA is aimed at demonstrating reliability, and developing less expensive manufacturing methods.

Fig.9 Inner channel cassette body flow test in US

Fig.10 Outer channel of cassette body assembly under

Steady state goal and transient goal:
Code evaluations indicate that the normal operating mode of a partially detached plasma can be expected at the envisaged operating density and scrape-off layer (SOL) power, leading to a steady-state heat flux ≤ 10 MWm^{-2} on the divertor target. In the case of off normal events, in particular when the full power conducted into the SOL transiently lands on the vertical target (loss of partial detachment), the predicted peak heat flux is ~20 MW m^{-2}. These two heat load conditions are defined as steady-state goal and transient goal" I Fig.3.5-1. During a very short plasma transient such as ELM or disruption, a pulsed energy flux between 10 and 100 MJm^{-2} would be deposited in a very short period, e.g., 1ms. This heat load is simulated by a pulsed plasma discharge or electron beam. Erosion due to 20 MJm^2 has been studied with CfC irradiated at 0.4 dpa. The erosion rate is about 2 times higher than that without irradiation but no disruptive damage has been observed.

3.6 Blanket Remote Handling project (L-6) Progress Summary

1. Objectives

The main objectives of the Blanket RH Project is demonstrate,

(1) proof of principle of the use of a transport vehicle on a monorail including development of a large transporter system, having a large unsupported span, for the installation and removal of blanket modules,

(2) Proof of principle of the adopted remote handling scenario involving the handling of in-port assembly, use of double-seal doors as well as pipe welding and cutting equipment.

All remote handling equipment was installed in JAERI-Tokai and basic functions of IVT system such as rail deployment, module replacement, and module transfer was confirmed by the end of July 1998. The main objectives have been achieved satisfactory. Now, additional R&D planed for the extended EDA are under going as follows;

(1) Verify the feasibility of the mechanically attached module replacement
(2) Verify the feasibility of sensor based control
(3) Verify the feasibility of the rail deployment of the cask based IVT for ITER-FEAT
(4) Verify the feasibility of rescue operation

2. Achievement up to May 2000

2.1 Installation of Test Platform

Various equipment, a full-scale remote handling equipment (IVT), simulated reactor structure such as maintenance sells, ports, back plate, two types of modules (one is for welding attachment; the other for mechanical attachment), and control systems were installed on the test platform by the end of February 1998 (see Fig.2.1). The full-scale remote handling equipment is composed of a vehicle manipulator with 13 degrees of freedom, multi-joint rail for deploying into 180-degree in-vessel region, rail deployment and storage device at 0-degree port, rail support and module receiver at the 90-degree port, rail to rail docking device at the 180-degree port, and measurement and control systems.

2.2 Integrated Performance Tests

Integrated performance tests, such as rail deployment, module replacement, and module transfer using receiver, was carried out under manual and/or semi-automatic modes from the control room.

(1) loading test of vehicle manipulator

In order to verify structural integrity and measure the mechanical characteristics of the manipulator including all mechanisms such as a traveling, telescoping and rotation, the loading test was performed prior to the module handling demonstration. In this test, the loads are applied in radial, toroidal or vertical direction to the manipulator axis or to the gripping axis, and the displacements of various positions of the mechanism were measured as a function of the applied load. In the vertical direction, the measured average stiffness is about 12 kN/mm du to flexibility of the rotating mechanism while the stiffness of the manipulator arms and telescoping mechanism is over 100 kN/mm which roughly meets the design value. The radial and toroidal measured stiffness is about 800 N/mm which is also dominated by the rotating mechanism since the stiffness of manipulator arms and traveling mechanism is over 30 kN/mm equivalent to the design value. The measured backlash in the vertical, radial and toroidal direction is 2 mm, 8 mm, 3 mm, respectively. These values can be represented from clearance and backlash of gear mechanisms. As a whole, it has been verified that the fabricated manipulator and end-effector satisfy the module handling capability and the mechanical characteristics including nonlinear behavior of the mechanisms are quantitatively evaluated under the design loads.

(2) Rail deployment/storage test

The first trial of rail deployment and storage tests were carried out by manual and semi-automatic modes to demonstrate the mechanical feasibility of the integrated equipment, and to obtain the suitable control parameters, such as positions and speeds of respective rail deployment mechanisms during rail deployment and storage. This trial includes measurement of the deformations of the rail during deployment, especially vertical and horizontal deformations of rail at the 90 degrees port. The operation time for rail deployment and storage was about four days each, including measurement, adjustment and teaching operation. The vertical deformation of rail at the 90 degrees port results exceeded the specification of 100mm. This is due to mainly the backlash of the vehicle fixing arm which attached to the front part of the rail deployment device at the 0-degree port. It has also verified that the guide and docking mechanisms at the 90 and 180-degree ports can correct the deflected rail to the correct position and fix the rail as the toroidal ring structure.

(3) Module replacement test

After deployment of the rail, the module replacement test was carried out using the vehicle manipulator on the rail. The replacement operation was performed under manual and semi-automatic mode from the control room to assess the replacement procedures and mechanical feasibility. The measured positioning accuracy was about 1 mm as repeatability. The vertical

deflection of the vehicle manipulator under loading of the module weight during removal from the back plate was measured to be around 60mm that was less than the target value of 100mm. Simulating the expected deflection and compensating for the deflection (Fig.2.3) solved the problem of dynamic deflection during module removal. The load condition at removal were based on the opposite load conditions encountered during module installation, i.e., the respective position of the mechanisms of the vehicle manipulator were pre-defined, based on those of the loaded manipulator during module installation.

(4) Module transfer test

After removal of the module from the back plate, the module was transferred from the vehicle manipulator to the receiver located at the 90 degrees port and transported through the port my manual and semi-automatic modes from the control room.

(5) Radiation hard component development and test

R&D for development and test of radiation hard component are under going in other R&D task. The results of Gamma irradiation test relating with IVT L6 components are shown in Table 2.1. Gamma irradiation dose rate in the vacuum vessel at 10^6 sec after reactor shut down is about 2KGy/hr.

Table 2.1. Gamma Irradiation Test Results

Component	Hardness of dose
Electric motor	10-80MGy
Position sensor	10-30MGy
Image fiber	2-5MGy
Strain gauge	10-80MGy

3. Additional Achievement Expected by July 2001

3.1 Mechanically Attached Module Replacement

A blanket module of mechanical attachment type is under testing at the blanket maintenance test stand. The module installation/removal test was successfully performed by using teaching play back procedure under keeping 0.25 mm of positioning accuracy condition with an intrinsic compliance of manipulator. Next step is to test with an advanced control system using information from sensors to detect positions/reaction forces, because male-female structures of the module requires a good accuracy of positioning less than 0.25 mm of clearance in both side. The results of this test will be used for the sensor based control system design and R&D.

3.2 Sensor Based Control

The previous motion control units are changed to the universal control system based on VME bus system which can be further extendable easily for the future R&D. This advanced control system by using an information of position/reaction force detection sensors (including position adjustment system) and Graphic user interface (GUI) system for real time observer of IVT motion in place of viewing camera will be used for future handling testing.;

- Preliminary test or sensor based control to detect the blanket position
- Modifications/improvements of gripper
- Installation of motion control system to the controller
- Integration test of blanket handling

3.3 Cask Based IVT

Conceptual design of rail hinge structure, rail hinge-locking mechanism, rail joining mechanism, and rail joining procedure were completed. The most important issue is rail links connection with high accuracy. To guarantee sufficient accuracy of rail joint, once each rail links are joined by using hinge-locking mechanisms in a remote handling cask, and then rail links are jointed with assembling hinges. After that, rail hinges are unlocked and rail is deployed in a vacuum vessel as basically same as FDR (RH cell based IVT rail deployment manner). An R&D item for feasibility confirmation of cask based IVT is focused on rail jointing procedure in a port/cask.

3.4 Rescue Operation

The rescue operation is required in case of trouble on IVT system. Then, R&D of rescue operation of IVT using rescue tool are planed. In this R&D, the rescue operation on typical case will be demonstrated by using the full scale mockup model of IVT.

4.Summary

The required handling operations of IVT were performed till July 1998 and the basic feasibility of blanket handling test was successfully demonstrated together with the collection of the required database.

Till end May 2000, based on these data base, additional R&D resulting from design improvement such as cask based IVT (which requires rail connection in VV port) and mechanical attachment concept of blanket, and advanced R&D such as rescue scheme demonstration and full automatic (sensor based) operation are planned and already undergoing.

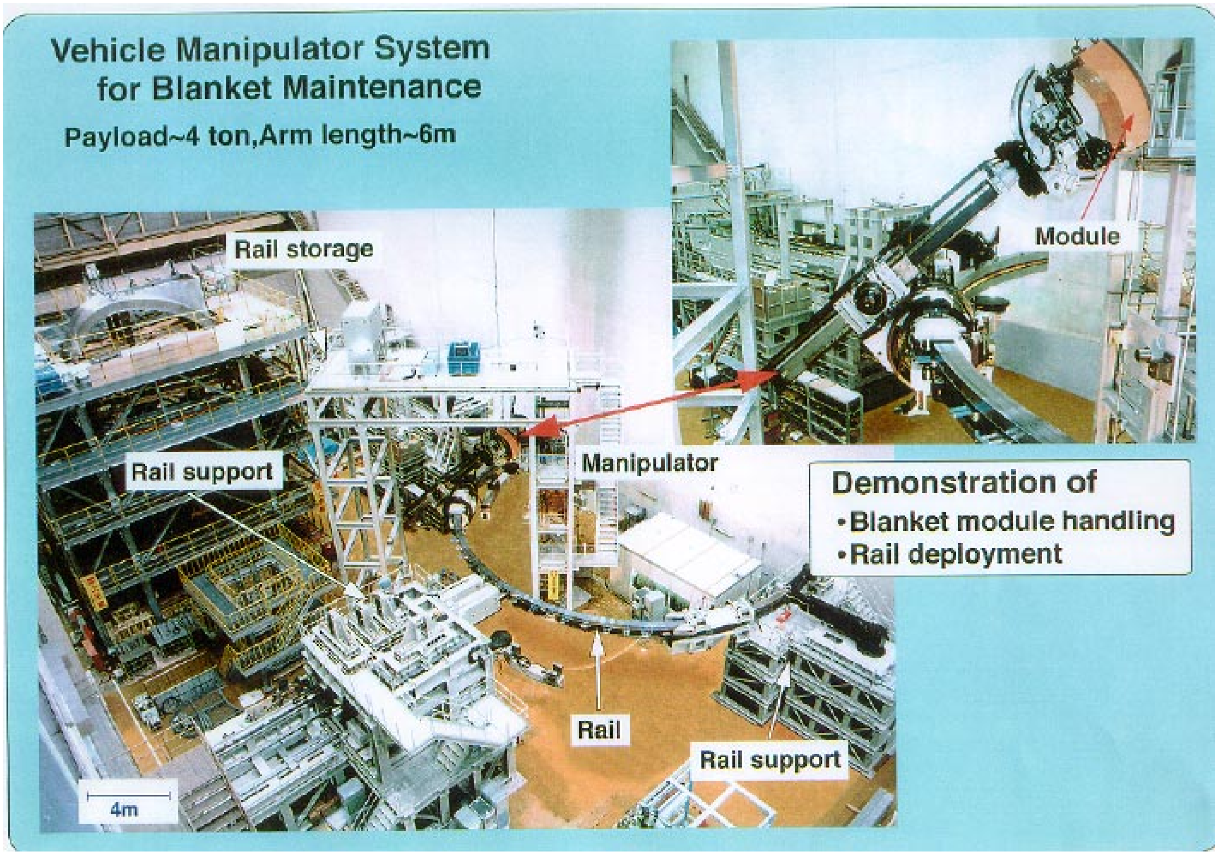


Fig. 2.1 Blanket Remote Handling Test Platform and Full-Scale Manipulator

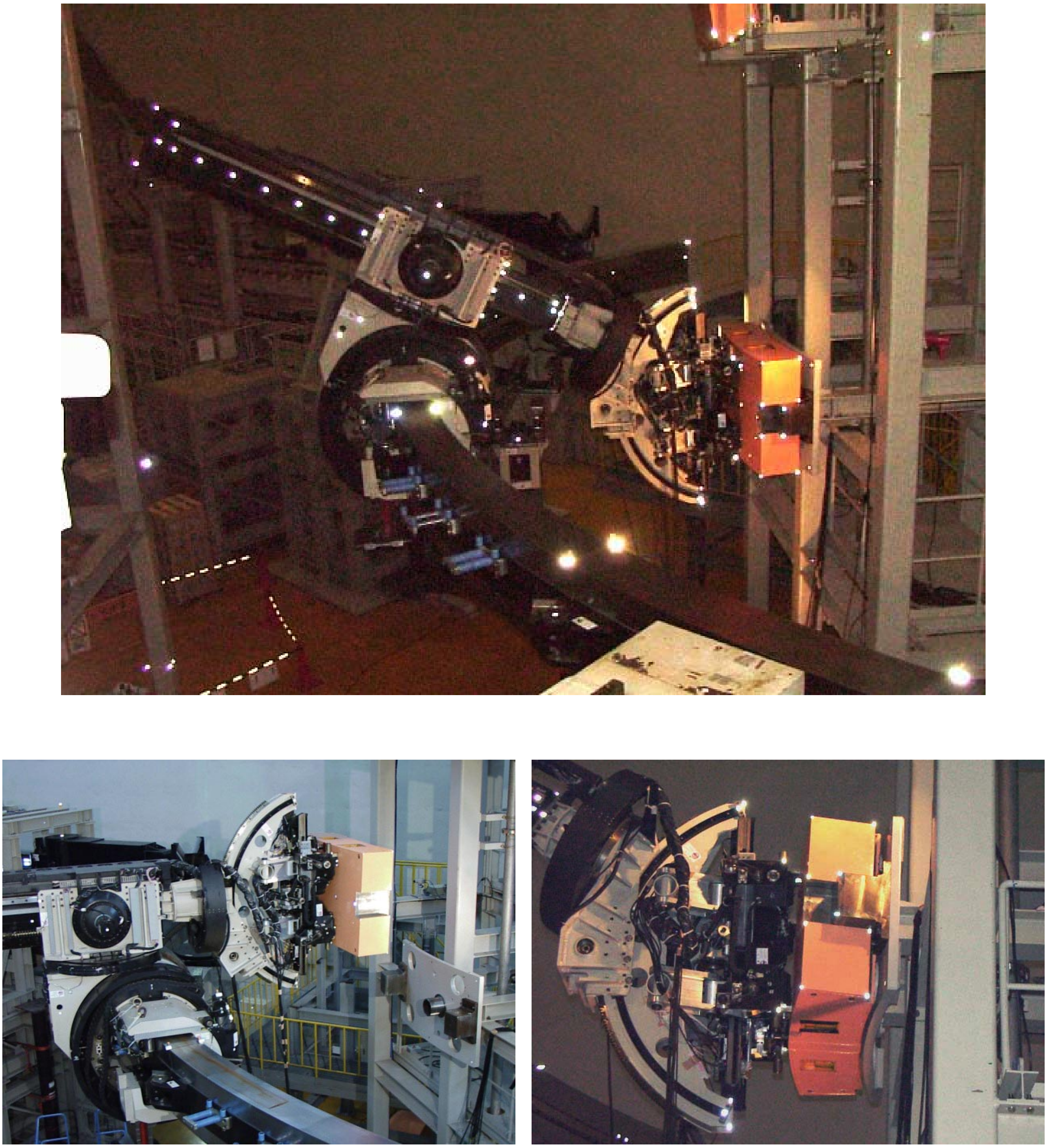


Fig. 2.2 Vehicle Manipulator under Operation

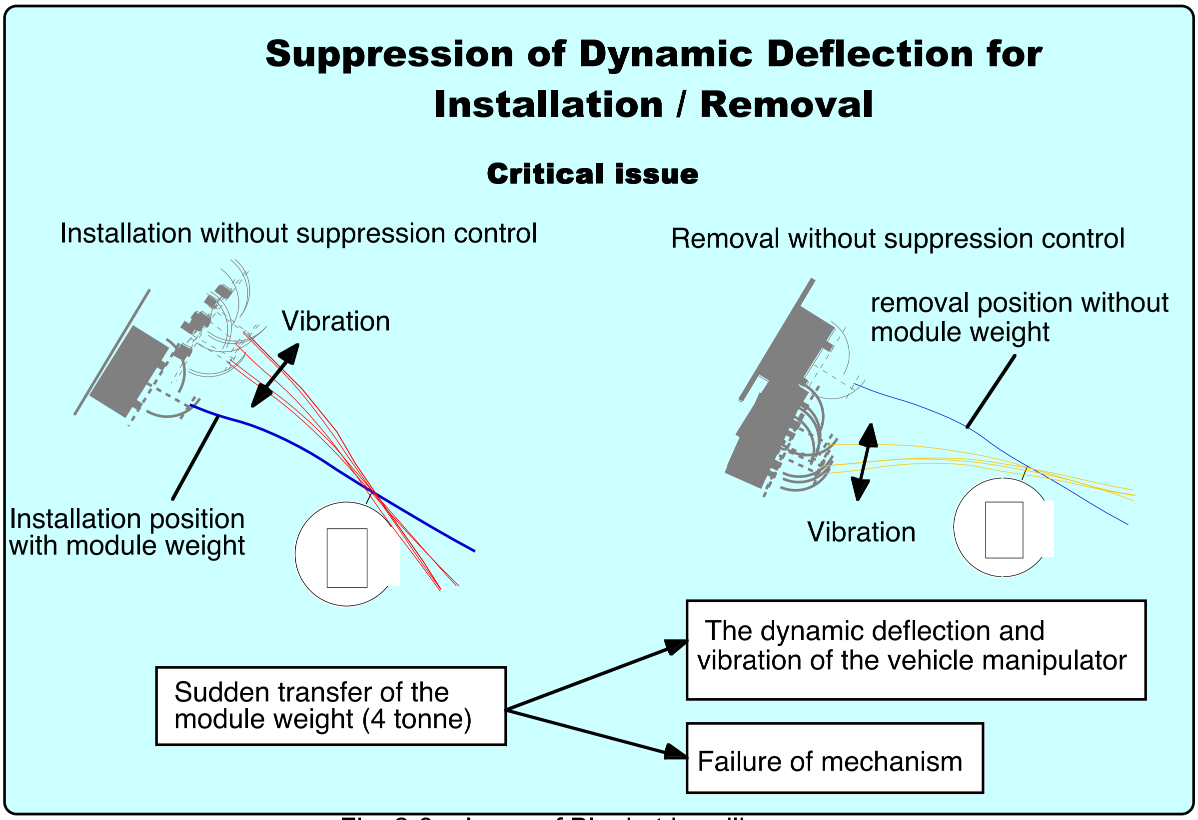


Fig. 2-3 a Issue of Blanket handling

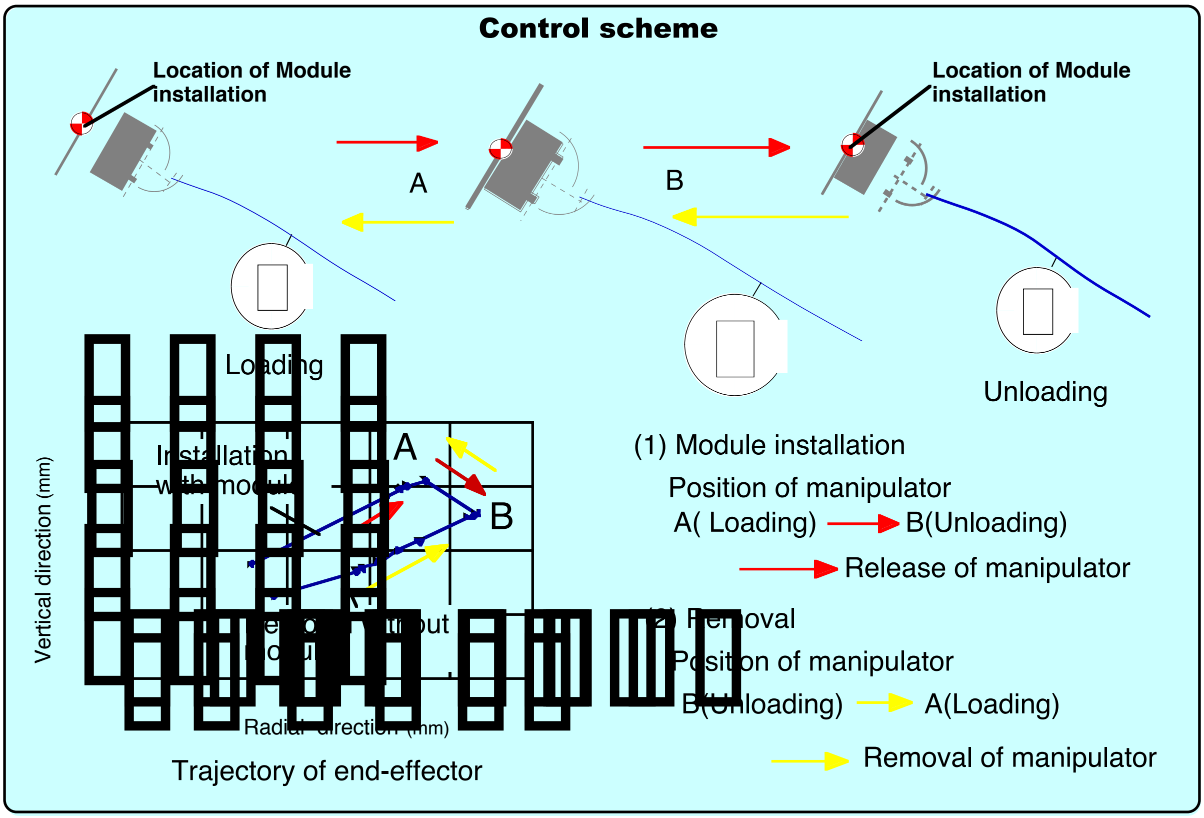


Fig. 2-3 b Control scheme

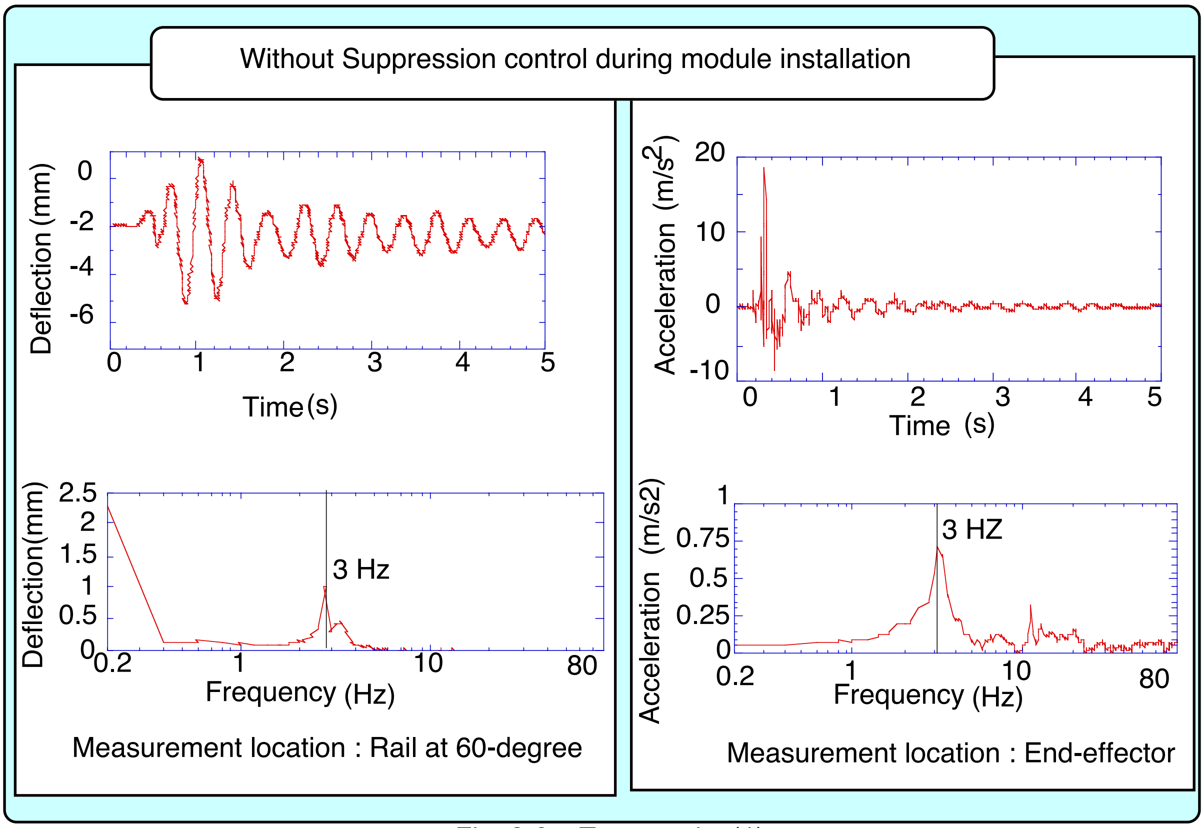


Fig. 2-3 c Test results (1)

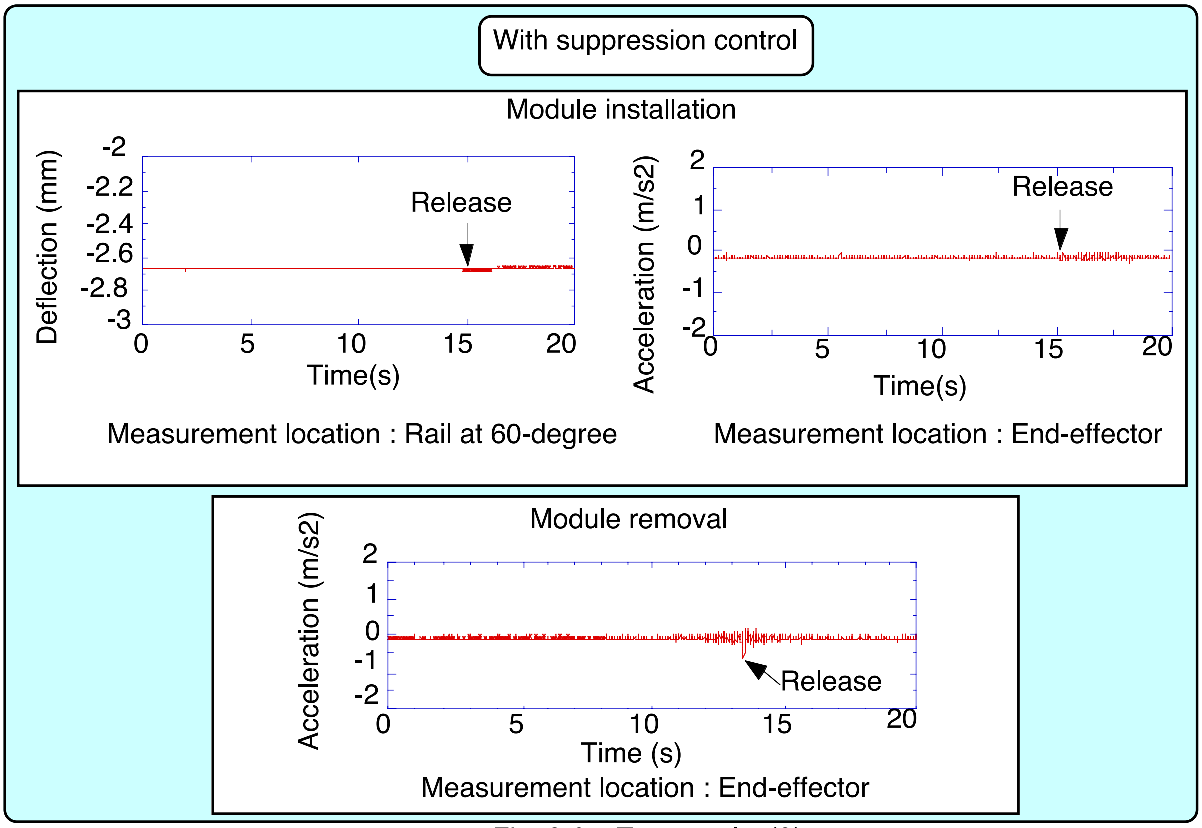


Fig. 2-3 c Test results (2)

Cask based IVT

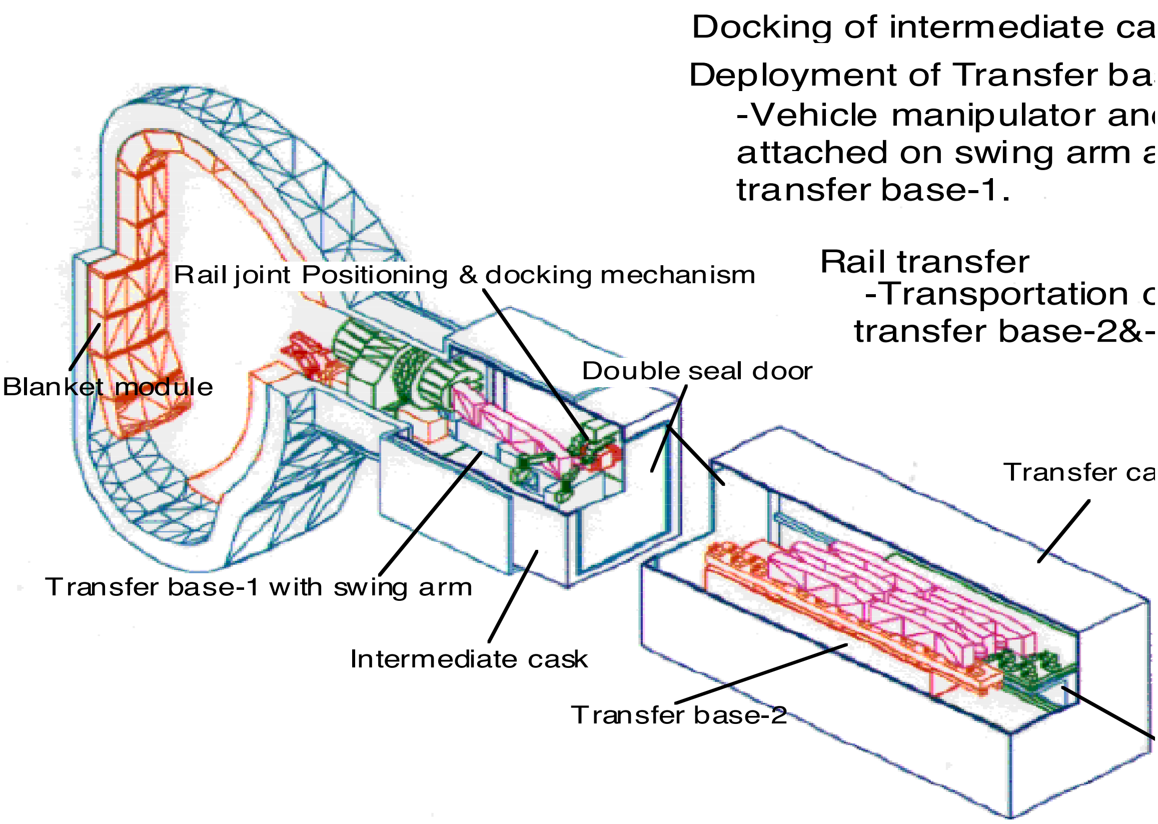


Fig 2-4 Concept of Cask Based IVT

3.7 Divertor Remote Handling Project (L-7)

1. Objectives

The divertor handling development (Project L-7) has two objectives, namely:
(1) To demonstrate the feasibility of divertor cassette replacement (in-vessel operation)
(2) To demonstrate the feasibility of divertor cassette refurbishment (hot-cell operation)

2. Achievement from August 97 up to May 2000

2.1. Introduction

Two experimental facilities have been built: the Divertor Test Platform (DTP) to achieve Objective (1), and the Divertor Refurbishment Platform (DRP) to achieve Objective (2). Both facilities are located in Brasimone (Italy). They have been developed by the EU Home-Team, with contributions from the JA Home-Team for the DTP. Besides, alternative components for divertor cassette replacement have been tested at JAERI.
Work for the L7 Project was completed during the EDA however, work within these facilities is continuing in order to fully exploit their potential contribution to ITER R&D.

2.2. Divertor Cassette Replacement

2.2.1. Main achievements based on the 1998 ITER design (from August 97 to July 98)

The DTP based on the 1998 ITER design includes the following prototypes & equipment:
- The cassette toroidal mover.
- The radial mover system, including tractor, central, second and standard cassette carriers.
- The RH auxiliary equipment including the cassette connections and the plug handling.
- A set of RH bore tools (welding, cutting and inspection) for 6 inch pipes.
- The DTP models and mock-ups including the platform proper (72° sector of the lower vacuum-vessel), the experimental mock-ups (dummy cassettes, supports, rails and plug).
- The experimental hall.
- Control room remote from vacuum vessel mock-up
- The DTP test and measuring equipment.

In addition, a bi-directional fork mechanism and electro-mechanical jacks for the bi-directional cassette toroidal mover and second cassette carrier have been tested at JAERI.

The main results are:
- All the components have been designed and constructed on time and to the specifications.
- The tests in nominal condition have been successfully performed first at the supplier's, and then at the DTP. In particular, these tests have demonstrated the ITER cassette handling concept, the integration inside the vessel, the accuracy of cassette positioning, the adequacy of the nominal gaps, the pay-load capabilities of the RH movers, and short handling times. The design has been incorporated in the EDA Final Design Report.
- Technological data have been obtained through basics testing: friction, locking systems, wheels, rails and rack-and-pinion.

2.2.2. Further achievements from August 1998 to May 2000

As planned, further results have been achieved in relation with the 1998 ITER design:
- Successful testing of movers in limit conditions: misalignment and gap accommodation.
- Estimate of RH intervention time: based on the DTP experience, the RH time analysis shows that the total duration of the in-vessel RH activities (i.e. excluding pipe cutting/welding, port handling and cask operations, etc.) for the installation of 15 cassettes would result in approximately 32 hours.
- Minor hardware improvements: carriers wrenches.
- Improvements of the remote control and achievement of full remote operations, involving the integration of: a supervision control system, a sensor-based control, a faster central cassette carrier controller and graphical assistance.
- Integration to the DTP supervisory system and successful testing of a 3D computer simulation model of the DTP using "TELEGRIP".
- Completion of a preliminary failure mode event analysis (FMEA).

In addition, the ITER-FEAT design requires new R&D, and preliminary results are available:
- Bore Tool Systems: A Bore Tool System for the cutting and welding of the curved divertor cooling pipes has been designed based on the assumption of 4" internal diameter pipes with a minimum bend radius of 400 mm. System testing expected before July 2001.
- Viewing and metrology system: Design work on a laser based viewing and metrology proof of principle system (IVV) suitable for deployment through penetrations at divertor level started in January 2000. System testing expected before July 2001 (N.B. deployment system not included in this work).
- IVP: Preliminary design work on a demonstrator robot manipulator (IVP) for access through in-vessel viewing penetrations and allowing first wall inspection started in January 2000. System testing expected before July 2001.

2.2.3. Other R&D needs related to the ITER-FEAT divertor maintenance not yet included

The timing of the new ITER-FEAT design and the time limits of the EDA extension could not allow to adequately cover the full supporting R&D for the Divertor maintenance. New R&D actions will be required after July 2001. It may be in particular the case in the following areas:
- Cassette handling: radial cassette handling in a cantilevered manner, cassette toroidal mover accommodating the new boundary conditions, umbilical management, in-vessel manipulator arm, new cassette supports and rails. An on-going design task will provide output by the end of 2000.
- Deployment system of the in-vessel viewing & metrology probes (on-going JCT design).
- Rescue operations.
- Further technological tests.

2.3. Divertor Cassette Refurbishment

2.3.1. Main achievements based on the 1998 ITER design (from August 97 to July 98)

The DRP based on the 1998 ITER design includes the following prototypes & equipment:

- The mock-ups of the divertor cassette, outer vertical target and alternative PFC attachments systems (shear keys and multi-links).
- The 3D metrology system for cassette measurement.
- The experimental hall including an heavy manipulator (100 kg), 2 light manipulator arms (20 kg), a bridge crane, cassette and target transporters, and the shear key insertion, locking and extraction RH tools (water and electro-mechanical devices).
- The test rigs for the shear keys development tests.
- In-bore cutting and welding device.
- The auxiliary equipment for the test and measuring equipment.

The main results are:

- The prototypes have been completed according to the original specification and time-plan. The experience gained in this phase was incorporated in the ITER final design.
- In general the DRP has demonstrated that all the critical operations during the refurbishment of the divertor are feasible and that the required precision in the positioning of the plasma facing components can be achieved. The equipment involved can be considered representative of the real system to be used in the hot cell.
- The feasibility of the cassette measurement was demonstrated by the metrology system (accuracy and repeatability of less than 0.2 mm).
- The complete cycle of insertion and removal of the outer target, including the locking of the keys (but excluding cooling pipe disconnection/connection) has been repeated several times. This has allowed the implementation of several improvements to make the operations easier and more reliable. The whole cycle can be completed in less than 6 hours.
- The fabrication of the prototypes has shown the large impact of tight fabrication tolerance on the cost of the keys, encouraging the development of the cheaper multi-link concept.
- Technological data has been obtained through basic testing: friction, heat evacuation.

2.3.2. Further achievements from August 98 to May 2000

As planned, further results have been achieved since August 98:

- Shear key attachment tests: tests to optimise the PFC removal and replacement processes were successfully completed in June 1999 involving near fully remote operations.
- 3D metrology: an upgrade to the system started in March 2000 to allow “line” or “continuous” measurements to be obtained.
- Further technological tests on shear key sub-assemblies: repeated insertion/extraction of wedges, articulation tests, vibration tests, load/deflection test (completion in June 2000).
- Development of a cheaper PFC-to-cassette attachment, the “multi-links” involving Al.Br pins and stainless steel housings and links: requirements, design, analysis, optimisation, expansion process and testing are on-going. The mechanical tests indicate a significant strength margin with respect to the requirements. Electrical tests will take place soon.
- “Multi-link” mock-ups and tooling are being used since November 1999 to develop RH methods and procedures. Design, fabrication, assembly and commissioning of prototypical tooling for PFC removal and replacement with the “multi-link” attachment scheme have been carried out. This tooling includes water hydraulic pin expansion tool, an electrically driven pin removal drill, and upper and lower target alignment tools.

2.3.3. Other R&D needs related to the ITER-FEAT divertor refurbishment not yet included

The successful achievement of the EDA extension R&D on cassette refurbishment will allow pursuing the development towards more hot-cell activities:

- Optimisation of remote handling cycles in hot-cell: cassette and PFC movements, manipulators & specialised tools, man-in-the-loop versus automated operations.
- Testing of pipe welding and inspections with RH tools.
- Simulation of rescue operations.
- Contribution to the definition of the hot-cell layout.
- Contribution to the possible refurbishment of diagnostics components.

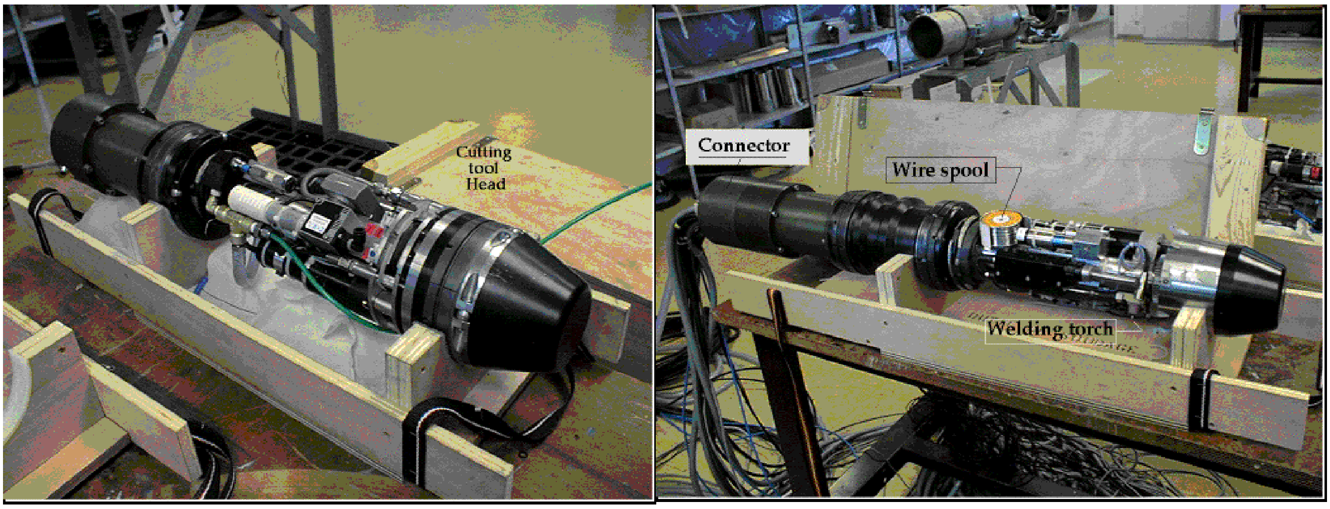


Fig. 2.1 Pipe Cutting and Welding Tools

Fig.2.2 Top view of the Divertor Test Platform (top left), Central Mover (top right) Duct Vehicle (bottom left) and Cassette Toroidal Mover (bottom right)

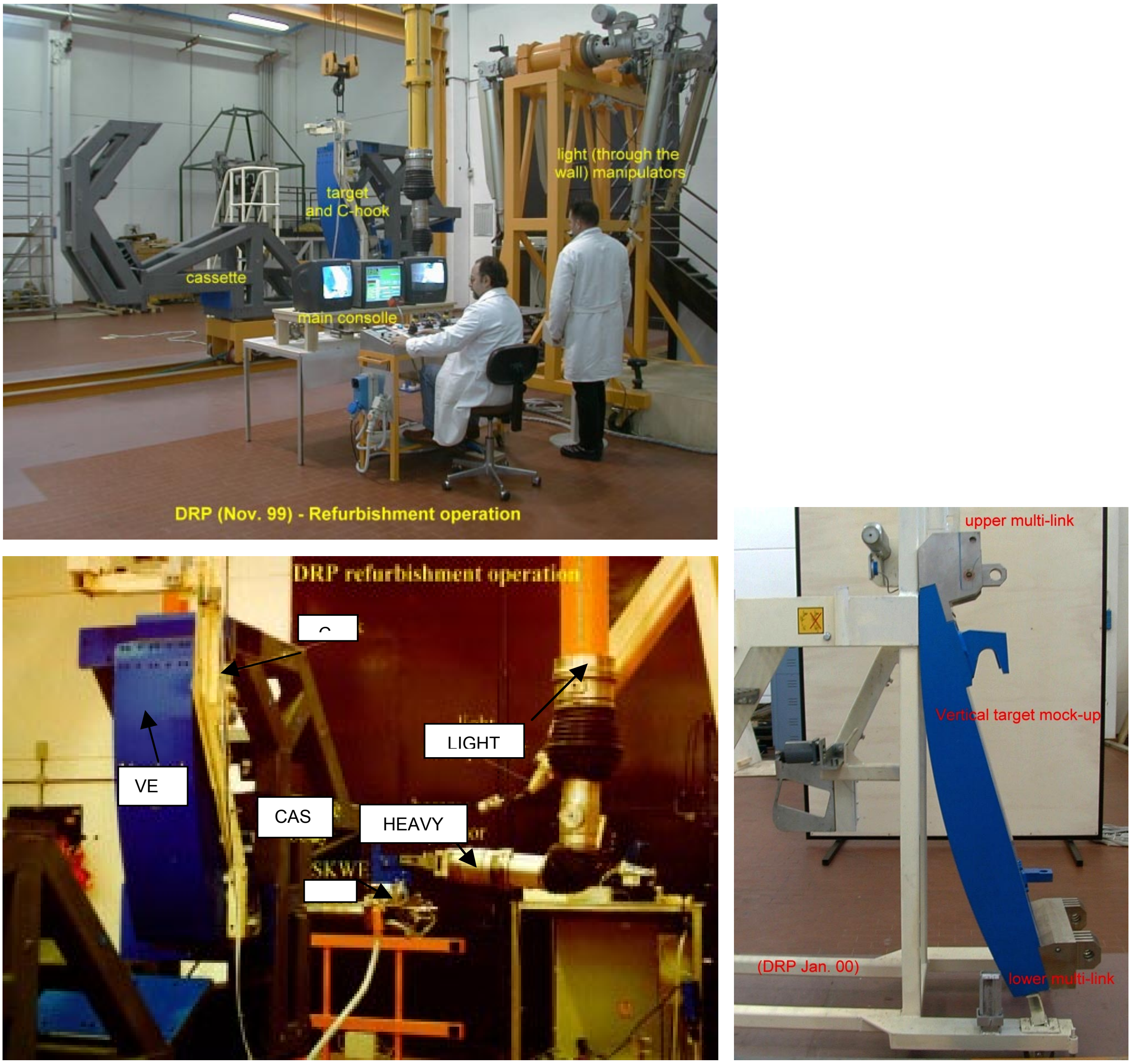


Fig 2.3 Overview of the Divertor Refurbishment Platform (top), Heavy and Right Manipulators at operation of the DRP (bottom left) and Divertor Target Mock-up after remove (bottom right)

3.8 Fuelling and Vacuum Pumping System R&D Progress Summary

1. Introduction

The R&D programmes that were initially implemented to support the design of the 1998 ITER have required little redirection to adapt them for ITER-FEAT. However, the fuelling programme that was originally being conducted solely by the USHT has had to be reorganised and is now being undertaken by the RFHT. While some of the basic R&D conducted in this earlier programme will useful a considerable amount of work will be needed to bring the programme back to the level that was reported in the FDR for the 1998 ITER design.

2. Fuelling System

2.1. Programme Objectives

The primary objectives of the work to be undertaken on the fuelling system are:

- to develop a tritium compatible pellet injection system of proven reliability to satisfy plasma fuelling requirements,
- to develop the design parameters for a curved flight tube to allow HFS pellet launch, and
- to conduct experiments to establish/ confirm the design parameters of the gas injection line needed to satisfy plasma fuelling response requirements.

2.2. Achievements up to May 2000

- Pellets were fabricated and the mechanical and thermal properties of D_2, DT and T_2 pellets measured. Pellet extrusion and injection experiments were conducted on D_2, DT and T_2 for pellet sizes up to 10 mm diameter. These experiments showed that pellets of this size could be fabricated and delivered at speeds up to 1 km/s and that the extrusion forces needed were ~1.5 and 2.5 times higher for DT and T_2 respectively when compared with D_2.
- A long pulse multi-piston extruder was fabricated and tested to demonstrate that the required extrusion rate, equivalent to 100 Pa m^3/ s, could be achieved. This type of extruder was at this time considered as the baseline for the pellet injector, however, towards the end of the EDA a screw extruder, developed within the RFHT domestic programme was being evaluated as a potential alternative to the piston extruder.
- A preliminary experimental investigation was conducted on curved flight tubes following the successful implementation of HFS pellet launch on ASDEX. While the results obtained, it showed that pellet speeds in the 4-500 m/s range could be achieved for pellets < 3 mm with the pellet remaining intact, and the speed for 10 mm pellets was limited to < 300 m/s. It was also found that the flight tube radius had a significant influence on the limiting speed before pellet fracture occurred. The pellet fracture speed observed for difference flight tube radii correlated well with a simple model that

had been developed for speeds < 4-500 m/s. However, above this speed the model broke down and large errors were recorded.

- Gas injection experiments were performed to demonstrated that the response rate needed to satisfy plasma performance requirements could be meet at flow rates up to 200 Pa m^3/ s. Maintaining the gas injection line lengths to < 20 m (typically required for installation) will allow a response times of < 1 seconds to 63.2% of demand to be achieved.

All the proceeding activities were performed by the USHT and as was noted above the RFHT has now assumed responsibility for this pellet injector R&D programme with this programme re-launched in December 1999. At this time the opportunity was taken to make some minor changes to the design parameters of the system to better reflect the requirements of ITER-FEAT together with the experimental results and experience gained in the initial part of the program.

- The first prototype of a tritium-compatible pellet injector has been designed, fabricated and is being prepared for tritium testing. The injector consists of a repetitive single-stage light gas gun integrated with a pellet generator based on a continuous solid hydrogen isotope screw extruder. The light gas gun is being used for expedience in these tests, the purpose of which is to examine the performance of the extruder. Successful tests have already been completed using D_2 demonstrating continuous injection periods over 100s. Work has now commented on the design of a full scale 6 mm extruder conforming to the requirements for ITER-FEAT.

2.3. Applicability of the R&D Programme to ITER-FEAT

The R&D programme that is in place is directed towards the development of a tritium compatible pellet injection systems that is completely relevant to the ITER-FEAT design parameters.

- Deep fuelling by pellet injection at fuelling rates up to 50 Pam/s are required, and to limit plasma density perturbations to <10% the pellet size is limited < 6 mm. Injection frequencies of up to 10 Hz are required for the smaller pellet sizes with a goal to increase this frequency to 50 Hz. These design parameters form the basis for the design of the hardware that is being developed in this R&D programme.

- The implications of HFS launch has had a positive impact on the requirements of the pellet delivery system allowing the pellet velocity to be reduced from 1,500 m/s to 500 m/s with a target of 1,000 m/s. However, the use of HFS launch has resulted in the need to develop the necessary technology for curved flight tubes, and this is a key element of the pellet injector programme.

- The experimental programme conducted to confirm the design parameters for the gas injection delivery lines established analytical has been satisfactorily concluded and remains relevant to ITER-FEAT.

2.4. R&D Programme up until July 2001

- Work will continue towards the development of major components of the pellet injector with system integration of the development unit occurring towards the end of this period. In parallel with this activity experimental work will commence on the curve flight tubes with design parameters being established during this time.

3. Pumping

3.1. Programme Objectives

The primary objectives of the work conducted on the vacuum pumping system are:

- to develop a tritium compatible batch regenerating cryogenic pump of proven reliability to satisfy plasma exhaust requirements,
- to confirm the performance predictions for a roughing system using multi-stage roots blowers,
- to evaluate the advantages of alternate roughing and high vacuum pumps to those selected for 1998 ITER design and now adopted for ITER-FEAT, and
- to conduct experiments to establish the feasibility of using of spiking for the identification of leaking water cooled circuits and the subsequent development of such methods if feasible.

3.2. Achievements up to May 2000

- Substantial testing has been conducted on charcoal coated cryogenic panels of reduced size but similar to these to be used in the cryogenic pump being developed. These tests included the investigation of pumping speed, gas load capacity and the adhesion of the charcoal layer following repetitive thermal cycling and tritium exposure. Tritium testing was conducted on small test coupons of ~ 40 mm in diameter.
- Following the successful demonstration of the pumping performance and adhesion tests on the reduced size panels a 50% full size model pump has been designed and built and installed in the new cryopump test facility. This pump has now completed acceptance testing and is in the process of undergoing a full range of performance tests over the complete plasma exhaust operating envelope.

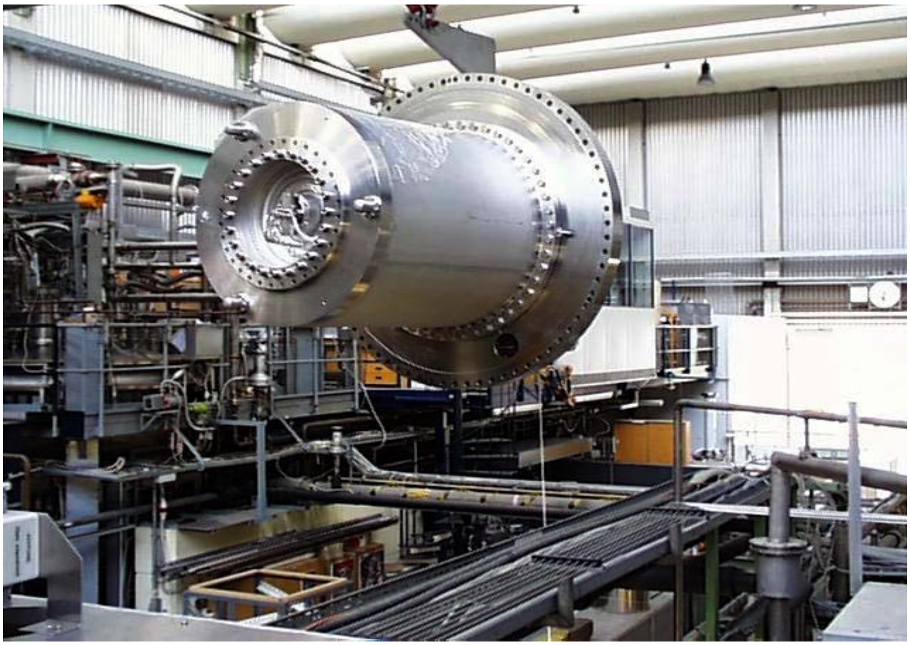


Fig. 3.1 ITER model pump on the way to the test facility TIMO

- The performance of scroll pumps has been investigated. These pumps were considered as a possible replacement from the piston pump/roots blower combination selected for the 1998 ITER and retained for ITER-FEAT. However, the performance of these pumps when operating in a roughing mode has been shown to be extremely poor due to high blank-off pressures especially for light gases. While multi-staging does improve the blank-off performance, multiple pump trains operating in parallel would be needed to meet the throughput requirements of the cryopumps during regeneration.

- Test were performed on a compound mechanical pump/roots blower combination nominally 25% of the 3,500 m^3/hr capacity needed for cryopump regeneration. The test results obtained confirmed that the model used to predict the pumping speed performance for a multi stage pump/roots blower combination was reason and could be used to develop the time line of the cryopumps during fast regeneration. The majority of these tests were conducted with light gases the most critical from a performance stand point.

- Test were conducted on a small scale cryomechanical pump which provided a speed of 0.25 Pa-m^3/s at 0.50 Pa. This pump is basically a compound molecular pump operating at cryogenic temperatures. The test results obtained would indicate that a pump scaled up to ITER size could satisfy the plasma exhaust requirements but the ultimate pressure achievable was less than would be needed for plasma breakdown at the start of the shot.

- Following an initial design assessment of alternate high vacuum pumping schemes to minimise tritium inventory a mechanical based pumping system using helical grooved pumps was selected as a potential option to cryopumps. Since pumps of existing design were unsuitable because of poor performance for light gases, a 2-stage helical grooved pumping system (GX150M and GX30M in series) was developed. These pumps incorporated magnetic bearings and the pumping speeds were optimised not only for H_2, D_2 and He, but also for N_2. The first stage GX150M was designed to have a large pumping speed for H_2, whereas, the second stage GX30M was a designed for high compression ratio. The GX30M showed a compression ratio of ~ 40 for H_2 at a backing pressure of 700 Pa, 100 times larger than the maximum backing pressure of helical grooved pumps of existing design. The 2-stage pumping system was shown to exhibited a maximum pumping speed of 0.19 m^3/s and an ultimate pressure of 10^{-3} Pa for H_2.

- Preliminary experiments were conducted to assess the mobilisation of dust during an up to air event, and a preliminary assessment of the filter needed to prevent the mobilisation of dust into the pump ducts completed.

- Following the positive results of initial design studies into the effectiveness of spiking as a potential method for the identification of leak of water-cooled circuits an R&D programme has been recently launched to confirm feasibility. While this task is proceeding no progress can be reported at this time.

3.3. Applicability of the R&D Programme to ITER-FEAT

- The R&D programme that is being conducted to develop the cryopump needed for plasma exhaust is complete relevant to the design parameters of ITER-FEAT. The test

results obtained to date would indicate that the pumping speeds and capacities needed to satisfy plasma exhaust requirements could be met. Thermal cycling and tritium exposure testing has demonstrated the resilience needed for the charcoal bonding materials and the process for its application.

- The roughing pumps in a multi-stage pump/roots blower combination are used to regenerate the cryopumps during regeneration. During a fast regeneration sequence, needed for long pulse operation, accurate prediction of the pump-down time, which is limited to < 75s, is essential. Due to the criticality of this pump-down in the regeneration time line substantiation of the analytical model used was needed, especially for light gases. This validation was provided by the tests conducted on the multi stage pump/roots blower combination at 25% ITER scale.

- The experimental programmes conducted on alternate pump types for roughing and high vacuum applications have clearly demonstrated the limited choice that is available when pumping speed, especially for light gases, over the pressure ranges needed is considered. These bench mark tests clearly demonstrates that at this time no viable alternative exists to the pumping system initially selected for 1998 ITER and now adopted for ITER-FEAT. The programmes undertaken to substantiate the performance of these pump options, especially for light gases, was essential in confirming the selection of the pumping system that had been made. The disadvantage of using throughput pumps is further compounded by the added conductance losses of the filters needed to protect the pump close running clearances from dust, and in the case of the cryomechanical pump by the added complexity of cooling of the pump duct and pump to cryogenic temperatures.

- Dust transport has a significant impact on the design and selection and sizing of equipment for the torus pumping system. This work has allowed nominal filter sizes to be developed, which are needed for system layout purposes, and allow estimates of pressure drop and conductance to be made for system sizing.

- The initial design phase to evaluate the potential use of spiking has shown that both soluble chemical tracers and gas tracers could work in the ITER environment. However, since the proposed method is a new one, it has not been possible to be precisely sure how sensitive the technique will be in practice and further experimental work is necessary to define the potential and limitations of this technique. The initial phase of the leak detection R&D programme will be a feasibility assessment to assess the viability of the proposed spiking method for the location of leaks in water-cooled circuits and to develop the necessary background information to define a detailed R&D program that will allow the development of this technique if successful for ITER.

3.4. R&D Programme up to July 2001

- Completion of the performance testing of the model pump and development of the design parameters of the full scale pump required for ITER-FEAT and the completion of further tritium testing on representative cryopanels.

- Completion of the proof of principle experimental program for spiking for both gas and aqueous solutions.

3.9 Tritium System R&D

1.Introduction

Throughout the EDA a considerable R&D programme has been maintained to develop processes, equipment and instrumentation for the tritium plant. The main results of recent work are described hereinafter.

2. Objectives and Achievements

2.1 Development of Tritium Measurement Instruments

(1) Objectives

This task is intended to provide experience and demonstration of the operation of a simulated ITER fuel cycle including plasma exhaust processing and cryogenic isotope separation and ZrCo based storage system, with real-time measurement systems composed of remote laser Raman analysis, micro gas chromatographs and other instruments. Particular interest is focused on the behaviour of ZrCo storage beds for rapid supply, recovery, and accountancy of the fuel to/from ITER-FEAT tokamak.

(2) Achievement up to May 2000

The following results have been obtained.

(i) For the rapid recovery, ZrCo beds could store almost full capacity of hydrogen gases within a few minutes without any difficulty even though the temperature of the ZrCo hydride itself was increased by hydrogenation.

(ii) For the rapid supply, the storage beds are heated at operational temperature while tritium supply requires additional heat for dehydrogenation. The test beds do not have enough heating power to meet the full supply requirement (200 Pa.m3/s) with one bed. A supply of nearly 20 Pa.m3/s was demonstrated for one bed.

(iii) For rapid accountancy, using in-bed gas flow calorimetry, accuracies of ± 1.5 % within 24 hours, and ± 3.5 % within 12 hours were obtained.

(iv) A relatively strong isotopic effect of the hydride beds was observed that manifested itself in a wide variation in isotopic composition of the evolving mixed hydrogen species during unloading of beds. This is an important effect that is under further study.

(3) Implications for ITER Design

ZrCo beds are being used in the design of the ITER T-Plant for the storage and delivery system. Rapid recovery and supply is essential for inventory and cost minimisation.

The above R&D results show that rapid recovery of tritium using hydride beds is feasible for ITER. However, for rapid supply the observed speed is not yet sufficient. Heat transfer to the ZrCo hydride powder needs considerable improvement.

(4) Programme up to July 2001

Work on rapid recovery, and especially rapid delivery are being continued, the latter aiming at improving the capacity of heaters and their thermal coupling to the ZrCo powder. The isotopic effect is also being further quantified.

2.2 Development of Decontamination Technique for Highly Tritium Contaminated Wastes

(1) Objectives

Establishment of effective techniques to remove tritium from selected highly contaminated tritium processing components and optimisation of tritium decontamination processes.
Investigation of tritium re-deposition (identification of tritium compound formation and rate constant, re-deposition rate of different tritiated species, etc).

(2) Achievement up to May 2000

Establishment of effective techniques to remove tritium

Using samples of artificially produced co-deposited layers UV light exposure released large quantities of hydrogen, carbon and hydrocarbons without production of tritiated water. It is concluded that the combination of heating and UV irradiation is an effective way to decontaminate tritium from the surface of co-deposited materials.

Investigation of various decontamination methods

Various decontamination methods such as purging with three different moisture concentrations, UV irradiation, oxygen baking and oxygen RF-plasma exposure have been investigated. It was found that UV irradiation is the most effective, although not by a very large margin as compared with dry nitrogen purging, of the methods used in this study. On the other hand, oxygen RF-plasma exposure removes tritium from 2-dimensional CFC-samples ten times more effectively than oxygen baking at the same oxygen pressure.

Investigation of tritium re-deposition:

To identify the reaction products and process in the reaction of T_2-CO 1:1 mixed system, infrared adsorption spectroscopy was applied. Spectral measurements of the reaction products showed the reaction products to be aldehydes (RCTO), alcohols (ROT) and carboxylic acids (RCOOT) in solid phase and tritiated water and methane in gas phase. Tritium accumulated in the condensed reaction products and was estimated to move into the condensed phase from the gas phase at about 5% per day.

(3) Implications for ITER Design

Effective techniques for the removal of tritium from FW components are a pre-requisite for successful ITER operations. Hence this work, that has started on artificial samples, will be widened to include materials from tokamak vacuum vessels. In case the techniques prove effective, they should be developed for the specific use in ITER.

Dry decontamination methods, that avoid the generation of tritiated water, are planned to be used in ITER on components that have been contaminated with tritium. These include not only exchanged parts and components from the tritium plant, but also remote handling equipment, etc. The methods that have been investigated are rather straightforward and simple and achieve surface decontamination factors of an order of magnitude. The results have shown the importance of smooth surfaces to minimise contamination. This is of particular importance for components of remote maintenance equipment that require hands-on maintenance.

Experiments of tritium re-deposition showed the occurrence of solid, tritiated products. The results indicate that the removal of tritium from mixed hydrogen/carbon layers, using

oxidation could have a severe drawback by depositing tritiated solid material along the extraction lines. It is clear, therefore, that this phenomenon should be fully understood before any co-deposit layer removal by oxidation should be applied in ITER.

(4) Programme up to July 2001

Actual tiles used in a tokamak will be exposed to the UV laser. Further parametric tests on decontamination of artificial co-deposited layers will be performed using UV laser, oxygen baking and oxygen RF-plasma exposure.

Some materials including elastomer O-rings, CFRP (carbon fibre reinforced plastic), electrical cables, optical fibre and lubrication grease will be used in tritium decontamination experiments.

The reaction mechanisms of the T_2O-CO_2 1:1 system will be further investigated.

2.3.3 Tritium Plant Safety Enhancements

(1) Objectives

This scope aimed at tritium inventory reduction in key systems, the development of advanced tritium measurement systems and monitors and reduction of tritium in waste materials.

(2) Summary of the Current R&D Status

Development of a plasma exhaust processing system for all tokamak off-gases has focussed on a catalytic gas phase process. The required high decontamination factors (DF's) of the three process steps (front end permeator, main impurity processing loop, and final polishing step) were determined individually and it was demonstrated that the overall DF of 10^8 specified by ITER can be achieved. In addition the first two process steps, that is the front-end permeator and catalytic impurity processing, have been investigated on semi-technical scale with tritium. In another series of tests, detritiation factors achievable by isotopic swamping in a batch type mode process were measured as well as in a counter current mode using a combined permeator/catalyst component.

(3) Implications for ITER Design

The exhaust system tested with the counter current polishing step is now the ITER reference system and the basis for the ongoing detailed design. The results of the testing are therefore in full applicable.

2.4 Isotope Separation Development

(1) Objectives

The goal of this task was the evaluation of a range of improvements to reduce the ISS inventory.

(2) Achievement up to 2000

Characterisation of cryogenic distillation packings for ISS was completed. This work included determination of HETP (Height of Theoretical Equivalent Plate) for candidate packing materials, and of maximum throughputs attainable before the onset of flooding conditions.

(3) Implications for ITER Design

The obtained ISS data, due to the specific configurations used, may not be fully applicable to

the re-designed ISS for ITER FEAT.

2.5 Long Term Integration Test of Tokamak Exhaust Processing System

(1) Objectives
Development of reliable processes and components for tokamak exhaust processing.

(2) Achievement up to 2000
The test facility upgrade, which involves the addition of a technical scale counter current isotopic exchange reactor in a separate glovebox, has been completed. The facility is fully commissioned for tritium experiments. A new catalyst has been developed for the isotope exchange component. In addition to these activities in the tritium laboratory in Karlsruhe, a counter current isotope exchange reactor unit has been supplied to the Active Gas Handling Plant at JET which will permit cross-checking of results.

For the dynamic response tests of cryogenic distillation columns, the modification of an existing test column at Ontario Hydro Technologies is well advanced and the test programme has been agreed.

(3) Implications for ITER Design
The upgraded facility is expected to deliver the final process data for the detailed design of the exhaust processing system. That design effort is now underway. The ISS for ITER FEAT will be operated predominantly in a dynamic mode. The results of the dynamic tests are therefore fundamental for demonstrating the feasibility of this approach as well as for the selection of the control configuration.

(4) Programme up to July 2001
Testing of the exhaust processing system will be carried out over the full range of parameters. The ISS dynamic test programme will be started imminently and should be completed by end 2000.

2.6 Development of Process Analytical Techniques

(1) Objectives
Development for the ITER tritium plant of a range of analytic instruments to support the T-Plant operations

(2) Achievement up to 2000
A laser Raman spectrometer has been tested and shown high sensitivity. Tritium monitors based on gas and solid scintillation principles have been fabricated for installation in tritium handling facilities for performance evaluation under conditions relevant to ITER applications over extended periods.

(3) Implications for ITER Design
Following successful testing, the above instruments will be adopted for analysis and monitoring function in the ITER tritium plant.

2.7 Development of Methods for Ex-situ Recovery of Tritium from Plasma Facing Components

(1) Objectives
Investigation of a range of techniques to promote the release of tritium from graphite and CFC tiles, and tritium assay of the detritiated samples to quantify the residual tritium levels.

(2) Achievement up to 2000
Depth profiles of the tritium concentration in a total of ~250 samples have been determined. Data obtained on the JET tiles showed that the in-vessel tritium inventory is mainly within co-deposited layers near the divertor region, whereas the amount of tritium stored on tiles throughout the remaining volume of the machine is a relatively small percentage of the total.

(3) Implications for ITER Design
The results of the work conducted by both the EU and the JA HT are expected to be fundamental in deciding on the final strategy for recovery of tritium from co-deposited layers in the ITER machine.

2.8 Tritium Recovery from Solid Wastes

(1) Objectives
Development of techniques to verify, on a technical scale, decontamination factors achieved in screening experiments and to quantify residual tritium concentrations.

(2) Achievement up to 2000
A strategy for the routine regeneration of molecular sieve beds used for atmosphere detritiation systems, with beds containing a total of several tonnes of molecular sieve material have been processed and returned to service.

Tritium components removed from test rigs during maintenance operations are being categorised according to size, weight, material, and tritium concentration and stored prior to detritiation (where appropriate) and disposal.

(3) Implications for ITER Design
The data obtained in this task will be directly applicable to defining ITER waste management strategies.

2.9 Characterisation of ZrCo Metal Hydrides

(1) Objectives
This work is of particular importance for the defining the operation conditions of the hydride beds.

(2) Achievement up to May 2000
It has been demonstrated that disproportionation of ZrCo can be avoided by choice of suitable operating parameter combinations. A significant result of this is the need to use pumping to deliver tritium from the beds to avoid simultaneous exposure of the hydrided material to high temperature and pressure. In addition it has been demonstrated that if disproportionation should occur, the intermetallic compound can be reproportionated at temperatures of around 600C, which are compatible with the mechanical design of the bed and its containment system. Investigation of time evolution of gas composition during delivery of gases of mixed isotopes is under way for the compositions 50/50 and 10/90 DT, as a basis for design of the ITER storage and delivery system.

3.10 Power Supply R&D Progress Summary

1. Introduction

The purpose of the R&D performed on Power Supplies for the "1998 ITER", during the EDA, is reported in Section 2. The results obtained so far are reported in Sections 3 and 4. The assessment of the results and their comparison with the original R&D goals and with the ITER-FEAT specifications, respectively, is done in Section 5. Finally, the additional achievements, expected by July 2001, are reported in Section 6.

2. Background Information

The ITER coil power supply systems are provided with switching networks and discharge circuits the main purpose of which is to initiate plasma current, and to discharge the energy stored in the coils when a quench is detected. For both applications DC circuit breakers rated normally at 45-60 kA steady-state current and 10-15 kV voltage are required to commutate current from the inductive (coil) circuits to discharge resistors. In addition, make switches with similar ratings are used, in the switching networks, to reduce the resistance and, hence, to control the loop voltage during the plasma initiation phase, and for protection of the power supply components.

DC switches with the required characteristics could not be found on the market since the combination of high DC currents and high voltages is not normally encountered in industrial applications. Also in industrial applications the operating time is not of great concern. Therefore, an R&D programme has been established with the overall aim to demonstrate feasibility of the DC switches with the parameters required for ITER application. In particular, prototypes of the following commutating devices had to be developed, manufactured and tested:

- Current Commutating Units (CCUs) for multiple operation in the Switching Network Units (SNUs) and the Fast Discharge Units (FDUs);
  The CCU is normally composed of a bypass switch, which carries the steady-state current, and a DC circuit breaker rated at pulse operation but able to interrupt the current and transfer it to a discharge resistor with the help of a counterpulse capacitor.
- Explosively actuated circuit breakers (pyrobreakers) to be used as back-up circuit breakers in the FDUs;
- Multiaction Make Switch;
- Explosively Actuated Make Switch to be used for protection, in case of component failures.

The task to the EU Home Team (T333E) included the development of a mechanical bypass switch (BPS) and of a vacuum circuit breaker (VCB) to be connected in parallel. The current commutation from the first to the second switch is due to the arc voltage. Both switches are based on industrial prototypes with lower current and/or voltage ratings.

The RF Home Team, in the framework of the R&D Task D334R, was assigned to develop and investigate the following switches:

- a multi-action mechanical bypass switch with arcless commutation to be used in a CCU together with a thyristor circuit breaker;
- pyrobreakers rated at 60 kA and 170 kA (for the CS power supply system, not relevant for ITER-FEAT);
- an explosively actuated make switch;
- a fast make switch.

It was anticipated that this work would be done on the basis of the large experience in the field of high current commutating devices available at the Efremov Institute (RF).

A description of the main results of the R&D activities obtained to date is given in the following sections.

3. EU Home Team Task

3.1 Development and Testing of the Mechanical Bypass Switch with Arc Commutation (60 kA)

This switch is foreseen as the bypass switch able to carry the steady-state current in the discharge circuits. The Mechanical Bypass Switch has been developed from an industrial high current, low voltage switch (~ 2 kV) produced by Ritter (Germany). The necessary improvement of its high voltage withstand capability was obtained by increasing the open contact, clearance and creepage distances. The decrease in the operating time (from 0.5 s to 0.25 s) was achieved by modifying the driving system.

The performance of the switch was first characterized with factory and laboratory tests in 1998. After this cycle of tests some minor modifications were made to improve its performance. A second test cycle was made at JET in the autumn of 1999: 1000 pulses were performed with current commutation (at 66 kA) to the parallel vacuum circuit breaker.

The Mechanical Bypass Switch passed the life tests without any remarkable problem. Minor improvements were identified and agreed with the supplier.

3.2 Development and Testing of the Vacuum Circuit Breaker (60 kA)

This switch is foreseen as the circuit breaker able to commutate the current in the discharge circuits. The Vacuum Circuit Breaker has been developed, by Siemens (Germany), from a device used initially for industrial AC applications and then modified for use in fusion laboratories to divert DC currents up to 50 kA. For the ITER application, the diameter of the electrodes has been increased from 100 to 125 mm and the vacuum chamber inner diameter (screen) increased from 154 to 174 mm. Each vacuum circuit breaker contains two bottles connected in series to increase reliability.

The performance of the switch was first characterized with factory and laboratory tests (RFX, Padova) in 1998. Among other tests, 1600 pulses with current commutation at 66 kA were performed successfully. However it is worth noting that the expected $I^2t = 8.7 \times 10^9\ A^2s$ had to be reduced to $2.3 \times 10^9\ A^2s$ to achieve successful current commutation: the I^2t capability drops dramatically with the current increase.

4. RF Home Team

4.1 Development and Testing of the Mechanical Bypass Switch with Arcless Commutation (60 kA)

This switch is foreseen as the bypass switch able to carry the steady state current in the switching networks. The Mechanical Bypass Switch (MBS) with arcless commutation is connected in parallel with fast thyristors that act as circuit breakers. The MBS consists in reality of two switches connected in series: the first two opening contacts commutate the current to two fast thyristors, they limit the arc voltage and therefore the erosion to the electrodes. The second two opening contacts open without current and sustain the high voltage.

Several steps have been made in the development. Firstly a 40 kA prototype was built and tested successfully. The experience gained allowed the construction of a second 60 kA switch. All planned tests have been performed: minor modifications have been made in steps to improve further the performance. Among other tests, 2000 pulses with current commutation at 66 kA were performed successfully; moreover 7000 switching operation (open, close) were done without current.

4.2 Development and Testing of the Pyrobreaker (60 kA)

This switch is foreseen as back-up circuit breaker in the discharge circuits. It is a very reliable, single action, component triggered with a pyrocharge. It consists of two parts, each triggered with separate explosive charges: a multigap pyrobreaker able to interrupt the current and a disconnector able to withstand the high voltage. Their development is based on the large experience existing at the Efremov Institute in this field.

All planned tests have been performed: minor modifications have been made in steps to improve further the performance. Finally, 30 pulses with current commutation at 66 kA were performed successfully.

4.3 Development and Testing of the Fast Make Switch (60 kA)

This switch is foreseen as the make switch able to carry the steady state current in the switching networks and for protection of the power supply components. It consists of two parts: the first is an industrial triggered vacuum spark gap with a turn-on time of less than 10 μs, and the second is a mechanical make switch able to close under relatively low voltage (~ 1 kV).

Several steps have been made in the development. Firstly a 40 kA prototype was built and tested successfully. The experience gained allowed the construction of a second 60 kA switch. All planned tests have been performed: minor modifications have been made in steps to improve further the performance. Among others, 2000 make (close) operations with current at 66 kA were performed successfully; moreover 7000 switching operation (open, close) were done without current.

4.4 Development and Testing of the Explosely Actuated Make Switch (60 kA)

This switch is foreseen for protection of the power supply components. This Make Switch was based on a prototype developed for a non-ITER application and improved to include water-cooled channels which allow an increased steady-state current capability up to 60 kA.

All planned tests have been performed: minor modifications have been made in steps to improve further the performance. At the end, 30 make (close) operations with current at 66 kA were performed successfully.

5. Assessment

The R&D goals, defined during the early phase of EDA, included the interruption of 170 kA, the current in the layer wound CS. The only reasonable solution for the multi-action switches was their parallel connection. Tests were foreseen but delayed and the decision to abandon the layer wound design allowed the cancellation of these tests. A 170 kA pyrobreaker was designed: it consisted of two parts, the first was successfully tested in 1998; later also this development was abandoned.

All other goals were achieved and therefore this R&D programme could be considered completed. However, the design of the toroidal field coils has been modified and the nominal current increased to 68 kA. In the past the commutation current of the switches (66 kA) has been considered 10% higher that the coil current (60 kA). If the same approach is maintained the new commutation current of the switches should be 75 kA.

6. Additional Achievements expected by July 2001

New R&D Tasks were agreed with both EU and FR Home Teams in 1999. During the Task Review Meeting, Garching, March 2000, it has been agreed, with both Home Teams, to test the switches already developed at higher currents, aiming to 75 kA. The results will allow the assessment of the margins.

Moreover, the complete operation of the current commutation circuit has been included in both the EU and RF Tasks already agreed. The current will initially flow in the Mechanical Bypass Switch, then it will be commutated to the circuit breaker (either the vacuum circuit breaker or the thyristors) and finally to the resistor. In this way all components will be tested in real conditions: in particular the real recovery voltage will be applied to the Mechanical Bypass Switch.

3.11 IC H&CD System R&D Progress Summary

1. Objectives

The target of the Ion Cyclotron H&CD R&D program is to validate the IC launcher design and to demonstrate the launcher component feasibility.

2. Achievement up to May 2000

The R&D program included:

- The construction of a full scale, single strap ITER prototype antenna including housing, Faraday Shield and current straps.

- A program of low power RF tests, including the antenna network analysis and RF B-field mapping. The purpose is the electrical characterisation of the IC antenna, the comparison of the current/voltage distributions against the one predicted by the models, and the measurement of the RF E-fields, to identify field concentration and to optimise the components geometry.

- A program of High Voltage with the purpose of assessing the voltage stand-off of the geometry of the prototype

- The development of the essential matching system components:

  d1) - the vacuum pre-tuner
  d2) - the vacuum window
  d3) - the VTL all metal support.

Point a), b), c) and d3) of the program have been performed on time before July 1997.
Point d1) and d2) of the program are being executed and expected to be completed before July 2001.

a) Construction of the Prototype

The ITER IC antenna prototype is shown in Fig 2.1.
Details of the construction are to be found in R&D Task ITER/US/98/IV-RF-01 (T361 US) final report.

b1) Vectorial Network Analysis Measurements

The vectorial network analysis measurements include:

a) Time Domain Reflectometry (TDR) impedance plot of the assembled antenna Vs electrical length with and without vacuum feed through.
b) the module of the reflection coefficient in the frequency range of perfect match and a Smith chart representation of the antenna complex impedance

c) An electrical model of the structure having the purpose of assessing the parameters of a transmission line antenna model used in the following to interpret the antenna electric measurements.
d) Parameters scan to optimise the position of the feeder tap.

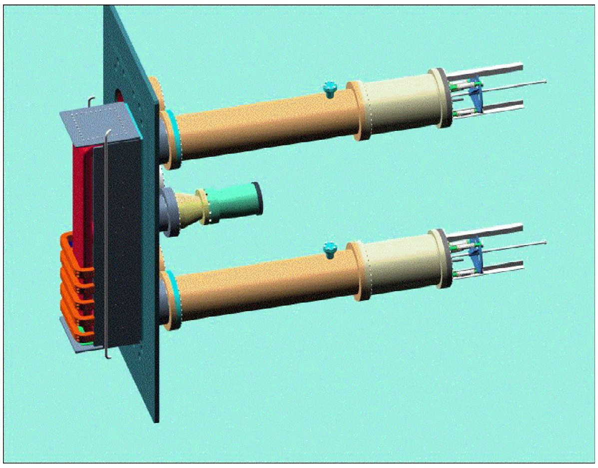


Figure 2.1 ITER IC Antenna Prototype

b2) Magnetic Field Measurements

3D magnetic field scans of the oscillating magnetic field have been performed using an automated scanning device. 2D magnetic field scans for all B components have been obtained at a distance of 45 mm from the Faraday Shield. Data are available with and without Faraday Shield.

The comparison between measured and calculated values (with the method outlined above) shows a position for the antenna septum two centimetres in front of the current strap rather than 4-cm behind (actual geometrical position). With this corrections the decay of the field components are in good agreement with the computed ones.

The B-field measurements are also in agreement with a 3D magneto static model of the current strap, including the box and the FS, performed with the ARGUS code.

These measurement indicate that the electrical modelling of the antenna is correct and give confidence to the models of plasma loaded antenna

c1) Power Tests

The power tests were performed in vacuum conditions in the RFTF (ORNL) device, using a power source tunable from 40 to 80 MW MHz and capable of an output power up to 1 MW. As foreseen in the program, the vacuum tuners, ceramic feed-through and all-metal supports (expected to be delivered to ORNL by EU and JA HT for Phase 2), were substituted by temporary equipment.

After bake-out of the antenna at 200 °C and conditioning procedures high power tests were carried out reaching standoff voltage of 60 kV for 1 sec were obtained. There is evidence that

this limit is not due to the antenna itself but to de-gassing of the (temporary) ceramic feed through.

The power tests provide indications on the behaviour of the ITER operation of the IC antenna. A 60 kV stand-off in vacuum provides a strong indication for a reliable operation on plasma up to 45 kV, which is the upper limit of ITER IC antenna operation.

d3) Development of an All-metal Vacuum Transmission Line Support.

The purpose of the development was to provide the IC Array vacuum transmission (VTL) line with one or more all-metal mechanical support (AMS).

The development of a full scale prototype of Vacuum Transmission Line including two AMS is described and specified in the final report of the technical annex of R&D Task G 51 TT 05 96-01-05 FJ (T238.1). The VTL length was optimized so as to minimize the overall input VSWR in the band 40-70 MHz, with a minimum at 60 MHz, when loaded by a 30 _ load.

A program of a) low and b) high power tests was executed on individual support(s) and on the overall assembly, to demonstrate VSWR performances over the frequency range and voltage stand off performances.

a) Low power tests:

Determine optimum length of coaxial between two supports with network analysis

Frequency:	40 - 70 MHz
Interval of length analyzed:	1m ≤ l ≤ 2.5m
Step: 0.25 (0.125 around the optimum length):	VSWR <2

b) High power tests:

Determine max. operating voltage in the following conditions:

Output power of the generator:	~ 100 kW
Frequency	60 MHz
Pulse length	≤ 5 sec
Target $_{max}$	≥ 40 kV

2. Fabrication

Full size mock-ups of AMS were designed with dimension based on the low power measurements and on the mechanical analysis presented in R&D Task G 51 TT 05 96-01-05 FJ (T238.1).

The support has comparatively small dimensions: 370 mm in width, 280 mm in length, and 520 mm in height and weights ~ 80 kg. The design requires vacuum capability; low RF losses and high voltage stand off. All the parts of the AMS are welded and a standard vacuum flange with metal gasket is used as interface to the coaxial line. Vacuum leakage was tested with Helium leak detector to < $1.8x10^{-9}$ Torr l / sec.

All RF facing surfaces are plated with 20 µm copper to minimise RF losses. Edges are rounded to at least R=3 mm and smoothly finished for high voltage operation. Detailed design drawings are available on paper and electronically in Mini Cad format.

An optimised section of VTL including 2 AMS is shown in Figure 2.2

Fig 2.2 Prototype of Vacuum Transmission Line with All Metal Supports

3. Low Power Tests

Network analysis of two AMSs connected with 30Ω coaxial line was performed in the frequency range 40-85 MHz, to determine and optimise the frequency band of the assembly. The S11 coefficient was measured for various distances between the two AMS (125 mm, 250 mm, two of 500 mm and 1300 mm coaxial lines and combinations of the above were used). The optimum distance was found to be 1925 mm with a max. VSWR = 2 at ~70 MHz, in accordance with previous estimates.

4. High Power Tests

High power tests were performed at the frequency of 60 MHz, using a driver stage of a JAERI RF power source converted from its original frequency range 102 - 131 MHz. A RF voltage in excess of 40 kV could be applied with ~ 110 kW of RF power, by using the standing wave developed between a stub tuner and a short circuit, with the AMS located at the voltage maximum and the overall assembly matched to the transmission line impedance. Directional couplers (calibrated by calorimetric method) were used to measure forward and reflected power in the high voltage section between the stub and the short circuit. The overall system was evacuated to ~ 10^{-6} Torr with a 150 l/sec turbo molecular pump, whereas the coaxial components between RF source and the vacuum boundary were filled of SF_6 gas at 300 kPa. 50/30 Ω impedance transformers were used to match the AMS characteristic impedance (30 Ω). Arc light could be detected from the ceramic feed through or through the AMS. One of the viewing ports was used to measure the temperature of the inner parts of the AMS using infrared thermometry. Prior to the power test, the vacuum section of the test circuit was baked at ~ 100°C for two weeks.

The achieved RF voltage standoff at the AMS was 55 kV for 1 ms, 45.5 kV for 0.1 sec, and 37 kV for 1 sec.

The voltages were not limited by the AMS mock-up but by arcing at a ceramic feed through for ~ 1 sec cases, causing gas emission, with a peak vacuum pressure >1 x 10^{-5} Torr in ~1 sec pulse, against a base pressure of 2.5 x 10^{-6} Torr. The position of the feed through was set near the peak of the standing wave in order to avoid multi-pactor discharges.

.

The voltage was limited by the RF generator capability in short pulse conditions. In the 55-kV/1 ms case, the generator output power (110 kW) was the limiting factor.

The stand-off test should be compared with the nominal operating voltage of the VTL: 11 kV. It is clear that the support is expected to reliably operate in ITER.

5. Additional Achievement Expected by July 2001

1. The EU Home Team is developing a Vacuum Pre-tuner under an R&D agreement lasting on May 2001. A call for tender specification has been issued and tender-related formalities are being implemented. A high power test program is foreseen in the Task Agreement.

2. A new vacuum window design of the vacuum window will be performed, to suit the new ITER-FEAT requirements. Results will be available in May 2001.

3. Modelling activity and mock-up development will take place in order to assess the performances of the multi-segment IC antenna proposed for ITER-FEAT. Results will be available on May 2001

In addition, it is noted that a proposal for the construction of an ITER-like antenna has received phase 1 approval within the program of JET upgrades to be implemented starting on 2004. A detailed engineering design is being prepared for the final (phase 2) approval.

3.12 EC H&CD System R&D Progress Summary

1. Objectives

The overall objective of the EC H&CD System R&D is to provide a 170 GHz, 1 MW, CW (test performed in 10 sec), high efficiency (50%) and reliable gyrotrons and windows reliably withstanding to a pressure of 0.2 MPa in case of in-vessel LOCA. (Possible improvement of output power of 2MW/tube and frequency tunability)

2. Achievement up to May 2000

1. R&D of a window

1) A fabrication of a CVD diamond window has been established. A CVD diamond disc has high thermal conductivity (2000 W/mK) and low tanδ ($2\sim4 \times 10^{-5}$ at 145 GHz). Because of this, 1 MW steady state RF window is now available.
2) A high pressure test proved that a diamond disc whose diameter of 70 mm and thickness of 2.25 mm could withstand to pressure of 1 MPa, which is much higher than the expected maximum internal pressure of 0.2 MPa in the vacuum vessel.
3) A diamond disc irradiated by neutrons up to fluence of 10^{21} n/m^2 shows no degradation of dielectric loss (3×10^{-5} at 145 GHz) at operation temperature of 100˚C and no change of mechanical strength (Ultimate bending stress of about 400 MPa) except thermal conductivity (~50%reduction). The location of the window will be located at the fluence of 10^{20} n/m^2 , where the reduction of thermal conductivity is negligible (~6%reduction).

2. R&D of a gyrotron

1) The maximum output power obtained until now in 170 GHz is as follows; 1 MW, 1 sec (RF) limited by a window. Improvement is under investigation. 0.5 MW, 8 sec (JA) limited by unnecessary RF mode in a tube. Improvement has been done. Test will start soon again.
2) A high efficiency of 50 % was demonstrated by a 110 GHz gyrotron with collector potential depression (CPD) and the same improvement is applying to the present 170 GHz, 1 MW gyrotorons.
3) Maximum output power of 2.2 MW for 1 ms (at 1.2 MW up to 15 ms limited by the DC power supply) at 165 GHz was accomplished by a coaxial cavity gyrotron (EU). The high power generation capability, step tunability (134~170 GHz) and high efficiency of about 50% with a single depressed collector of the coaxial cavity gyrotron were proved (Fig.3.12-3).

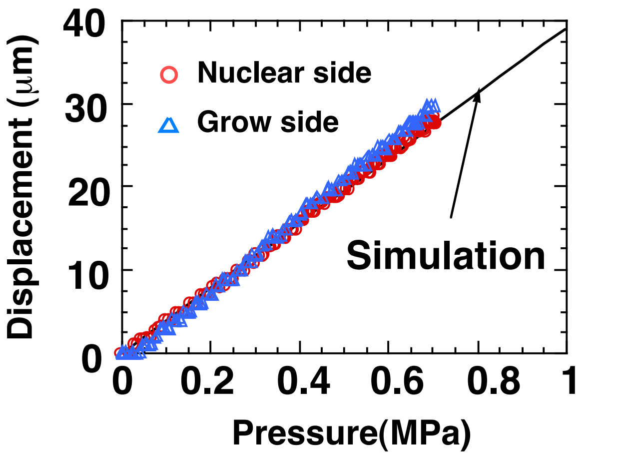
3. Additional Achievement Expected by July 2001

1) Output power of a gyrotron and transmission power of a diamond window: 1 MW, 10 sec at 170 GHz. (The operation condition becomes quasi-steady state in 10 second.)
2) Output power of a coaxial cavity gyrotron: 1.5 MW, 0.1 sec at 165 GHz.

10 atm. Resistive Diamond Window (Fig. 2-1)

Diamond Window

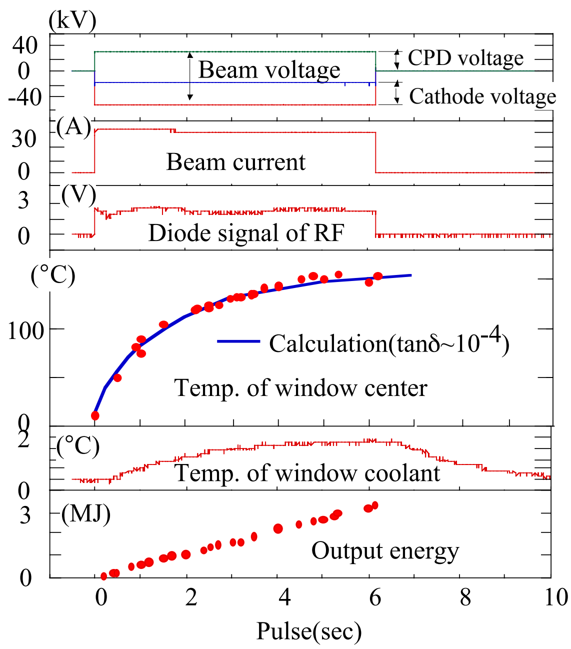
Displacement of disk Center vs pressure

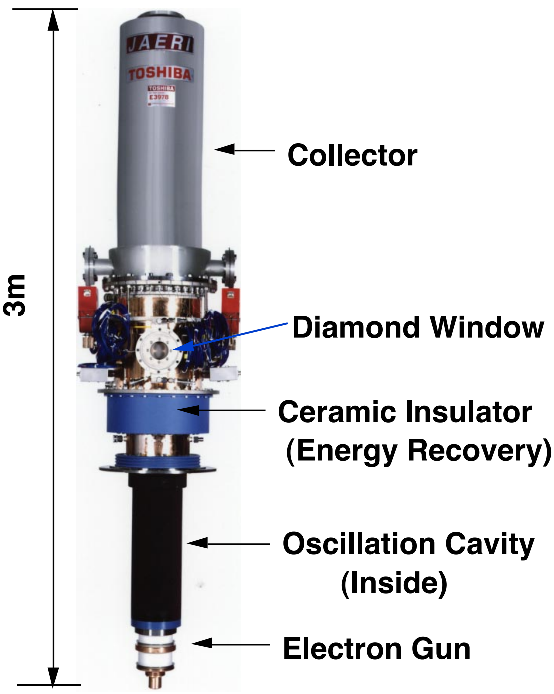


Young module		Poisson ratio	
Diamond	: 1100GPa	Diamond	: 0.1
Inconel	: 200GPa	Inconel	: 0.35
Al. braze	: 70GPa	Al braze	: 0.30

170GHz long pulse gyrotron (Fig. 2-2)

0.52MW-6.0sec Oscillation





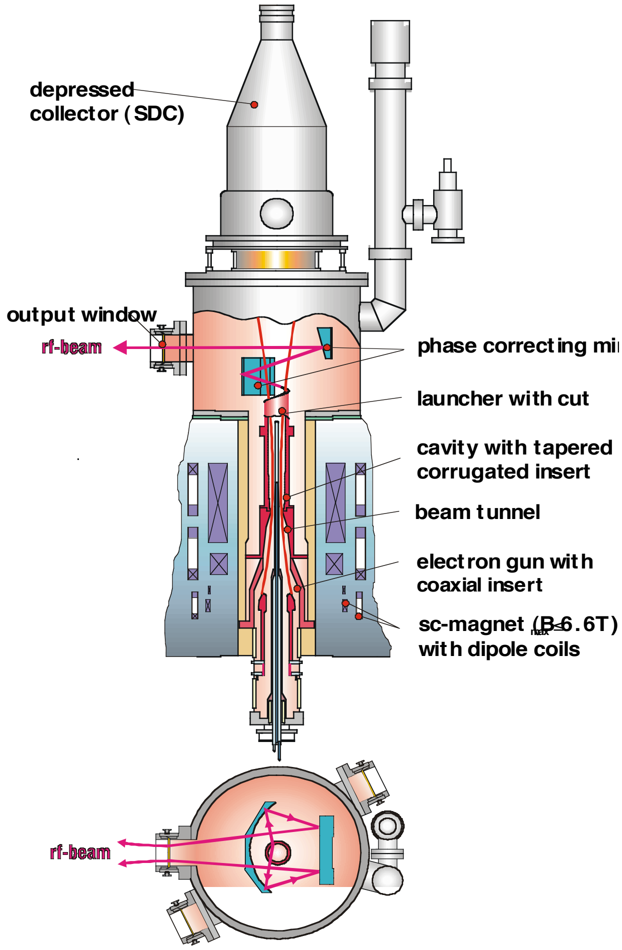
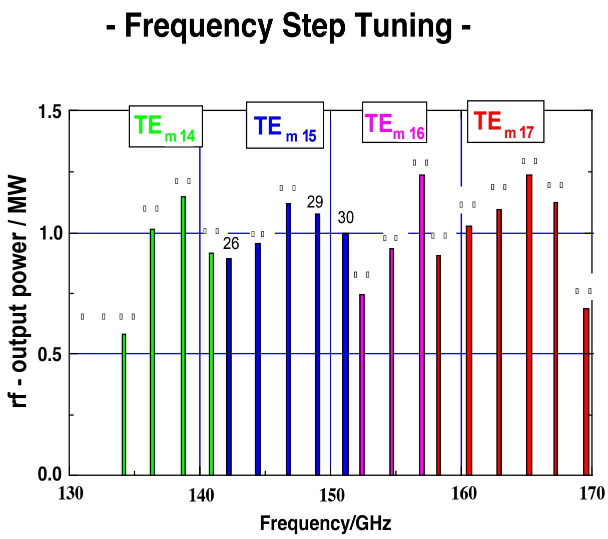


Fig. 2-3 165GHz Coaxial Cavity Gyrotron (above) and Frequency Tuning (below)



Operating Parameters : $I_b \cong 50$ A; U_c, B_{cav} and R_b adjusted individually. U_c, B_{cav} and R_b optimized for maximum rf-output power at $TE_{31,17}$, $TE_{32,17}$, $TE_{30,16}$, $TE_{28,15}$, $TE_{27,14}$.

3.13 NB H&CD System R&D Progress Summary

1. Objective

There are two key R&D's: the development of the ion source and the accelerator. The objectives are as follows:

For the ion source R&D,

- Long pulse (~ 1,000 s) operation of the source with high D^- ion current density (extracted current ≥ 200 A/m^2) at low operating pressure (< 0.3 Pa).

For the accelerator R&D,

- Acceleration of high current negative ions (~ 0.1 A D^-, at > 200 A/m^2, or ~ 0.5 A H^- at > 280 A/m^2) up to 1 MeV, with the ITER relevant beam optics of 3 - 5 mrad divergence.
- Demonstration of the 1 MeV ITER relevant beam for 100 - 1,000 shots of > 2 s.

The design change in ITER-FEAT is an adaptation of Vacuum Insulated Beam Source (VIBS), instead of Gas Insulated one. The VIBS bushing is the interface between the insulation gas (transmission line) and the ITER primary vacuum (around the beam source), instead of gas to gas interface in GIBS. The objective of this R&D is the development and validation testing of key technologies.

2. Achievement up to May 2000

2.1 Negative Ion Source

The KAMABOKO source (Fig. 2.1) R&D, which is a prototype of the ITER ion source, is being carried out by collaboration between the EU and the JA Home Teams. The source achieved the H^- ion current density of 300 A/m^2 at the pressure of 0.25 Pa at JAERI in 1995, and the D^- current density of 200 A/m^2 at 0.35 Pa at CEA Cadarache in 1996 (Fig. 2.2). However, both experiments were in the short pulses.

The key issue for long pulse operation is to maintain the temperature of the plasma grid, the grid illuminated by the source plasma, at an optimum (~ 300°C) temperature for the long pulses to maximize the negative ion production on the Cs coated grid surface. Long pulse operation has identified all the "weaknesses" of the source and also of the test bed, and some major modifications have been required. At present the ion source was operated reliably for 1,000 s at the design parameters: 1,000 s pulses in H^- were achieved in 1999. However the current density is still modest (100 A/m^2 H^-).

2.2 1 MeV Accelerator

1. H^- ions (25 mA) were successfully accelerated with the MultiAperture, MultiGrid (MAMuG) accelerator up to 1 MeV by the JA Home Team in late 1997. The R&D is now aiming at beam acceleration of higher current. So far 180 mA H^- ion beams were accelerated reliably up to 900 keV. The ITER relevant optics has been confirmed at the energy level of 700 keV.

2. The SINGAP (SINGle APerture SINgle GAP) accelerator of the EU HT has accomplished acceleration of 860 keV, 43 mA H^- ion beams with the ITER relevant beam optics properties by 1997. Further, the EU HT has tested acceleration of D^- ions and succeeded to produce 630 keV, 106 mA D^- ion beams. The small divergence angle of ITER relevant level is also confirmed in the experiment.

2.3 High Voltage Bushing

1. Two bushings are being developed: One is for VIBS bushing (mainly for SINGAP at present) and the other is to isolate the transmission line from the power supply. Insulation material of the VIBS bushing is ceramic or porcelain due to the foreseen radiation environment. In order to hold the voltage in vacuum (inner surface of the bushing) several insulator rings are piled up to form an insulator stack with the centre conductor at 1 MV. The R&D is in progress at CEA Cadarache (EU).

2. An electrostatic design has shown that a prototype of slightly smaller dimensions than required for the ITER-FEAT can be built. This component will fit in the available space on the SINGAP test bed (Fig. 2.3). An insulator ring used in the prototype bushing is shown in Fig. 2.4, its height is 94 mm, and in its first tests the highest voltage held without breakdowns was 300 kV. Only 111 kV are required for the SINGAP experiment and the ITER-FEAT design.

3. For the HV bushing to be placed between transmission line and HD deck the JA HT developed a one piece, dish shaped, insulator with many metal conductors penetrating through the insulator. Silicon impregnated epoxy resin is used, since low radiation is foreseen on it. The insulation gas is present on both sides of the bushing; therefore relatively high electric field on the surface is acceptable. Fig. 2.5 shows the mockup bushing. Its diameter is 90% of the one required for ITER. The mockup bushing was tested up to 900 kV without breakdown for >1,000 s, with SF_6 gas at 3.5 Bar (Fig. 2.6). The achievement of 900 kV by the 90 % reduced mockup is directly applicable to 1 MV required for the full scale bushing.

3. Additional Achievement Expected by July 2001

1. To test the KAMABOKO source with Deuterium, the test bed control system has been modified for remote control: 1,000 s D^- operation is the target of this experiment.
2. The following is expected from the R&D on the accelerator:
   - Increase of the beam current up to 0.5 A (280 A/m^2 H^-) by MAMuG and 0.1 A (200 A/m^2 D^-) by SINGAP at 1 MeV,
   - Increase the pulse length of the beam up to > 2 s,
   - Demonstration of about 100 - 1,000 pulses, with the 1 MeV beam at the prescribed current, current density and pulse length.
3. Development of brazing technology of ceramic/porcelain to joint metal flanges.

4. Other R&D Activity

4.1 The JT-60U Negative Ion Source

The negative ion source of JT-60U neutral beam injectors has semi-cylindrical "KAMABOKO" shape with its dimensions of 1.22 m long and 0.64 m in diameter, which is similar to the source designed for the ITER NB system (1.6 m long and 0.9 m in diameter). The JT-60U source has been expected to achieve rated current (22 A) with the current density ~ 130 A/m^2 and the pressure < 0.3 Pa. However, the status of the JT-60U source is still 85 A/m^2 for D^- ions (110 A/m^2 for H^-) at 0.22 Pa, this is 65% of the rated value.

A recent measurement of the JT-60U beam profile near the accelerator exit revealed a low beam output from the top and bottom part of the grid. The reason for this is considered to be the non-uniform plasma production in the ion source. For production of uniform plasma inside the source the filament power supply was modified in JT-60U so as to control each filament group.

4.2 Radio Frequency Source

IPP Garching has investigated several versions of the negative ion sources driven by radio frequency (rf): The latest one is called as "Type VI". This rf source has achieved so far the H^- current density of 200 A/m^2 at 0.7 Pa as the lowest pressure. It is observed that the presence of Ar gas in the discharge (20% of the H_2 pressure) increases the H^- ion current density, more than the Cs effect with the plasma grid at the optimum temperature.

4.3 Plasma Neutralizer

The RF Home Team has already reached the design targets of the existing plasma neutralizer, PNX-U (Table 4.1). From the latest results of the plasma neutralizer experiment, the following improvements can be expected if the plasma neutralizer is adapted to the ITER NB injector with minor changes of the design:

- The injection power could be increased to 22.3 MW/injector due to the higher neutralization efficiency (80% instead of 60% expected with the conventional gas neutralizer of the reference design).
- The gas flow from the plasma neutralizer could be reduced by a factor 3-5. This is an advantage in the case of continuous operation.

However, the extrapolation from the present results to the plasma neutralizer for ITER NB is large, see Table 4.1. Moreover the plasma neutralizer will add significant complexity to the injector and possible reduction of reliability. Design effort is in progress on the PNX-SU (Super Upgrade), as a good intermediate stage between the PNX-U and the PN-ITER.

Table 4.1 Summary of the PNX-U achievement

	PNX-U		PNX-SU	PN ITER
	Achieved	Design		
Axial length, l (m)	2.5		≤ 2	3
Volume, V (m^3)	0.5		~ 1.5	10
Line density, nl (x 10^{18} n_e/m^3)	1.8 (Ar plasma)	1.4	5	7
Ionization degree	0.25		0.3	> 0.3
rf power, P_{rf} (kW)	50 x 2		50	500
rf frequency, (GHz)	7 (Klystron)		7 (Klystron) + 24 (Gyrotron)	24 (Gyrotron)
Conductor type	Normal conducting (Cu)		Super conducting (NbTi)	
Magnetic field, (T)	0.36	0.5	1	

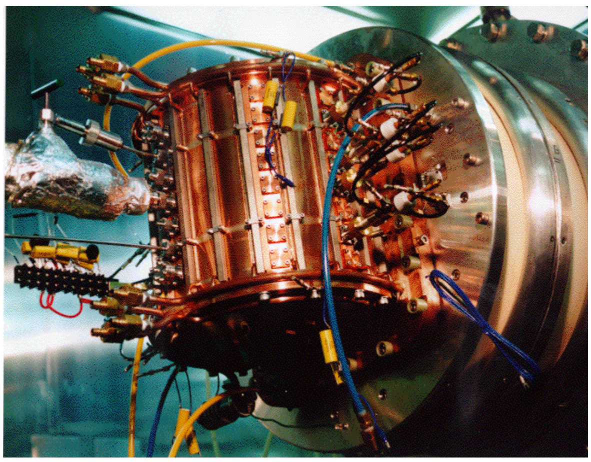


Fig. 2-1 KAMABOKO source as a prototype of the ITER negative ion source.

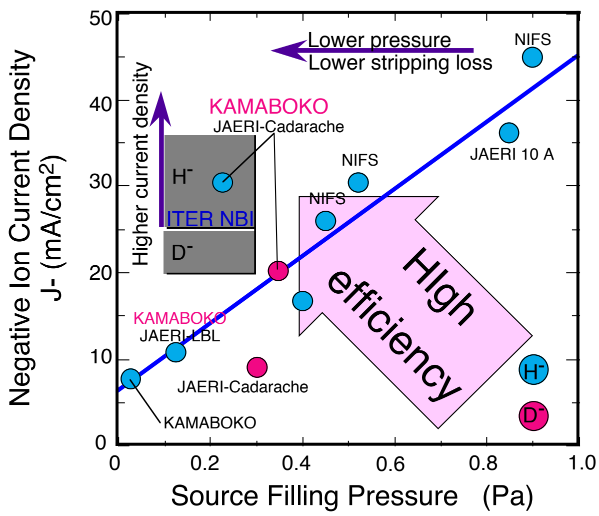


Fig. 2-2 Achievement of the KAMABOKO source.

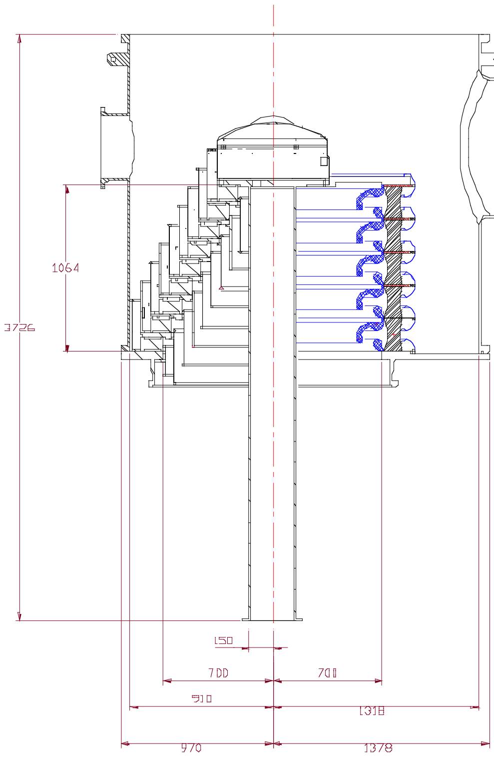


Fig. 2-3 A 1 MV Ceramic SINGAP Bushing - Prototype for ITER NBI.
Left: original epoxy bushing with present 9 stage, Right: section of prototype insulator/screen assembly.

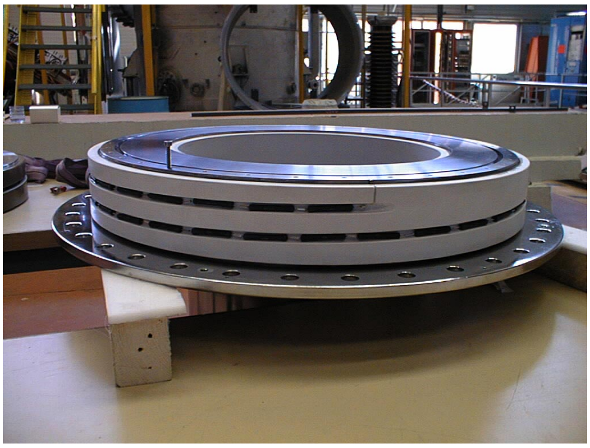


Fig. 2-4 A porcelain insulator ring used in the prototype bushing (CELAREP c120).

Fig. 2.5 Mockup of the HV bushing II.

Fig. 2-6 Test facility for the mockup bushing, located between two SF_6 tanks.

3.14 Diagnostics R&D Progress Summary

1. Objectives

The diagnostic R&D programme has two major components: (a) irradiation tests on some prototype diagnostic components and on candidate materials that could be used in diagnostic construction; and (b) development of new diagnostic components where the performance of existing ones is not adequate. The overall goal is to provide the data necessary for the design and construction of the individual diagnostic systems that will be used on ITER. To satisfy some of the measurement needs, developments in diagnostic physics, and development of some new diagnostic techniques, are required. These developments are carried out through the voluntary physics arrangements under the direction of the ITER Physics Committee.

2. Progress Up To May 2000

2.1 Irradiation Tests on Materials and Components for Diagnostic Construction

Many existing diagnostics techniques for magnetic fusion experimental facilities especially tokamaks can be applied to ITER with some modification. However, irradiation effects, which are not important in the present facilities, can be a critical issue in applications on ITER. A substantial component of R&D on diagnostics has therefore been devoted to the study of irradiation effects on key components. The selection or improvements of components have been made on the basis of the results obtained. The radiation levels experienced by the components depend on the location and the principal effects that have to be considered depend on the material and the function. In Fig.2-1, examples of typical locations of key components are shown with life times of these components derived from the R&D programme. Some results of the irradiation tests are summarised in Table 2-1.

The principal materials and components examined in the irradiation effects programme are ceramics, insulators, wires and cables, mirrors, windows, optical fibres, and bolometers. The properties of concern include electrical resistance, dielectric loss, optical absorption and emission, reflectivity, as well as mechanical and thermal properties. The main results are as follows.

Ceramics, insulators, and wires/cables. Several diagnostics, for example magnetics, will have components mounted in the vacuum vessel and these components will use ceramics, insulators and wires. The influence of irradiation on the electrical properties of candidate ceramics and insulator materials (Al_2O_3, Wesgo, and Vitox) is therefore important and has been extensively investigated. The important physical effects are radiation-induced conductivity (RIC) and radiation induced electrical degradation (RIED). In the case of in-vessel cables for low signal levels, the additional problem of radiation induced electromotive force (RIEMF) must also be considered.

RIC has been studied for many years and an extensive database exists together with a sound theoretical understanding. The data shows that the effects of RIC can be rendered negligible

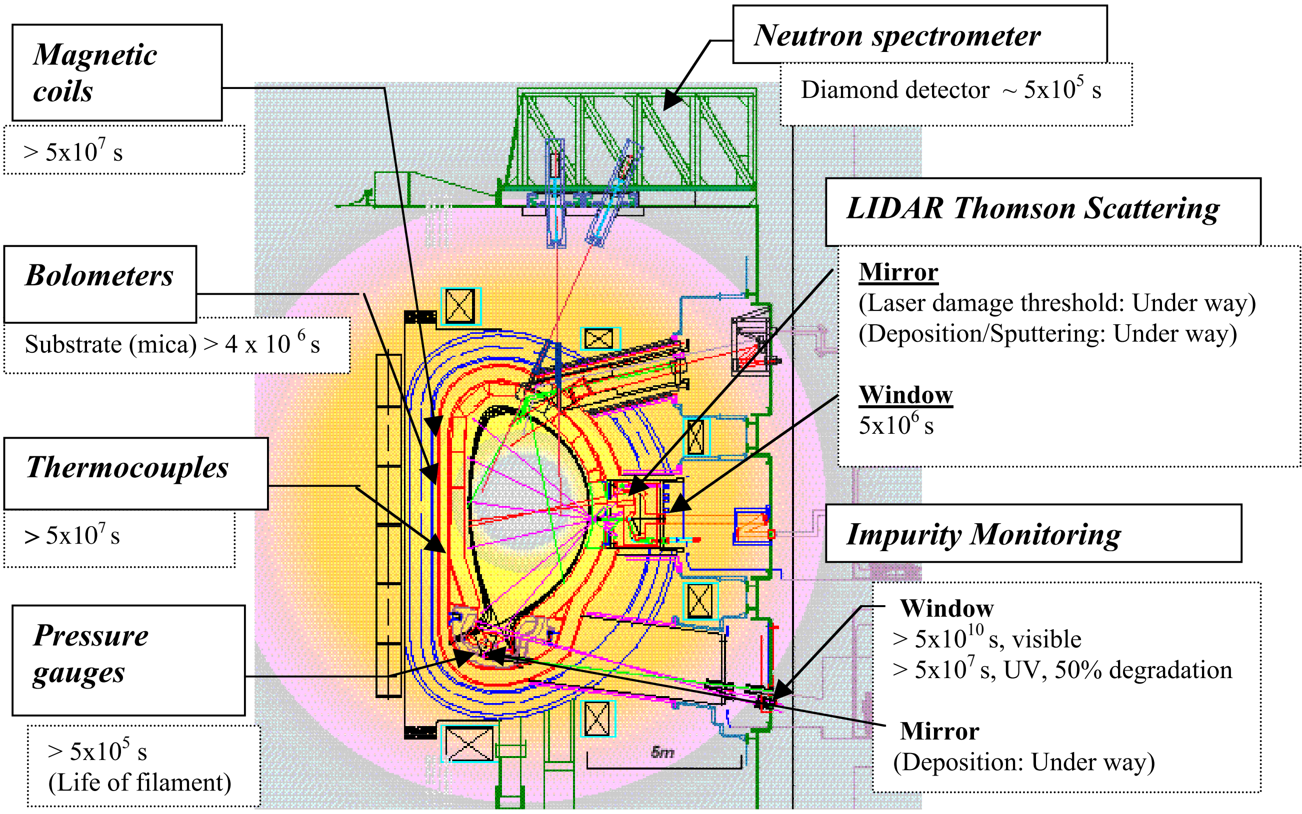


Fig.2-1 Life time of some key diagnostic components at 0.5 MW/m^2 or 450 MW. The fluence of 0.3 MWa/m^2 gives 2 x 10^7 s burn at 450 MW. The life times are mainly limited by radiation effects except where shown with ( ).

by careful choice of materials. On the other hand, the mechanism of radiation induced electrical degradation (RIED) is still not understood, but an extensive database has been established. The effect has been found to occur only under very precise conditions (typically electric field > 50 keV/m applied when the temperature is in the range 175±25°C or 600±50°C) and so can be avoided by design. RIEMF is also not presently understood. It has been observed to occur in experiments in which MI cable and prototype magnetic coils have been irradiated in test reactors. In general the observed RIEMF is current driven, and the generated current is in the range of nanoamperes. In the experiments with magnetic coils the asymmetric component of the induced EMF is of the order of microvolts and appears to be influenced by several parameters including cable diameter (ratio inner/outer), insulator type (alumina, magnesia), reactor power level, and cable configuration, although no clear parametric dependencies have emerged. Since the magnitude of the effect is small it is difficult to ensure that other effects - for example, thermoelectric effects and grounding problems - are not causing systematic errors in the measurements. Although the effect is small it could be significant in some applications, for example magnetic pick-up coils where long integration times are involved, and so it is important that the experiments are continued until it is understood.

Mirrors and reflectors. For many diagnostic systems the plasma facing optical element will be a mirror. The lifetime of these first mirrors is therefore a key parameter. The mirrors will be subject to intense UV radiation, neutron heating, particle fluxes arising from charge

exchange atoms (CXA) (typically $2x10^{19}$ particles/m^2/s with energies up to several keV), and will be subjected to the deposition of material eroded from the divertor, first wall and shield structure. Probably the most important effects are the CXA fluxes, which can lead to erosion, and deposition. Mirrors of several metals (Be, Cu, SS, Mo, Ta, W) with different microstructure (polycrystal, film, single crystal) have been bombarded for long periods (≤15hrs) by deuterium ions of energy 0.07 to 1.5 keV and the optical properties of the mirrors (specular and diffuse reflectivity and planarity) have been measured. The principal conclusion is that suitably chosen metal film mirrors mounted on a metal substrate can have a good resistance to the CXA flux. For example, Rh film mirrors of thickness ~10 μm mounted on Cu can be used in locations where the CXA flux onto the mirror surface will not exceed $2 \cdot 10^{18}$atom/m^2s (~1/10 of the CXA flux to the first wall). Moreover, single crystal molybdenum and tungsten mirrors can maintain good mirror quality even when subject to CXA flux similar to that at the first wall. On the other hand, even very thin layers (h≥ 10nm) of a contaminating film can seriously reduce the reflectivity. Mitigating methods (baffles and shutters) as well as potential cleaning methods (e.g. low energy discharge cleaning, laser cleaning) are therefore being investigated in current work.

Windows. The principal properties of concern for diagnostic windows are the radiation induced absorption, which has an instantaneous and a permanent component, and radioluminescence. Mechanical damage is not expected to arise because the windows will be placed in radiation fields well below the values at which mechanical deformations will occur.

Measurements have been performed on several materials including sapphire, and crystalline and amorphous quartz. The results show that suitable window materials are available for passive diagnostic systems operating in the wavelength ranges ~400 nm to 5 μm; for example, the radiation induced absorption and radioluminescence are very low in KU1 amorphous quartz. Suitable methods exist for bonding these materials to metal and so the window problem for these diagnostics is essentially solved. At shorter wavelengths (< 400 nm) both effects are enhanced and further work is required to find the optimum window material. At very short wavelengths (< 200 nm) the absorption is very high even in the absence of irradiation and direct coupling is necessary. Active diagnostics that employ high power lasers have a much lower tolerance to absorption since in this case even a very small absorption (< 5%) can lead to unacceptably high power deposition. Further work is required to optimise the windows for such systems. At longer wavelengths (>5 μm) there are two possibilities, ZnSe and diamond. ZnSe is a relatively soft material and a special method of making metal/window seals with a large diameter of 100 mm is being developed. For microwave diagnostics, several suitable window materials exist and no problems are anticipated.

Optical fibres. Because the optical path in the material is much longer, radiation induced absorption and radioluminescence are even more significant in optical fibres. At high levels of irradiation mechanical damage (embrittlement) can also occur. A substantial programme has been performed in which the optical properties of potential fibres have been measured at relevant radiation levels. The results have shown that optical fibres cannot be used inside the vacuum vessel. However, the effects are sufficiently low at the radiation levels expected outside the bioshield that fibres can be used there at visible and longer wavelengths. In the intermediate region (the cryostat), it is highly desirable and may be possible to use fibres especially at infrared wavelengths but more work is required to determine the optimum material and the precise magnitude of the optical properties. A round robin experiment is in progress to attempt to resolve some discrepancies observed in the different experiments.

Table 2-1 Results of Irradiation Tests on candidate Materials for Diagnostic Components

Diagnostic components	**Materials Tested**	**Accumulated effects**	**Dynamic effects**
Ceramics (electrical insulators)	Single crystal sapphire and polycrystal alumina (Al_2O_3)	3 dpa in helium gas (RIED: No catastrophic degradation)	10^4 Gy/s (RIC: < 10^{-6} S/m)
Wires /Cables	MI-cables: SUS, Inconel (sheath)/MgO, Al_2O_3 (insulator)/ Cu, Ni (centre conductor)	1.8 dpa (RIED: No catastrophic degradation)	10^4 Gy/s (RIC: < 10^{-6} S/m) 10^3 Gy/s (RIEMF: < 10V)
Windows	Fused Silica/Quartz (400-1200 nm)	10^{-3} dpa (Transmission; 5% degradation: 8 mm^t)	Radioluminescence: 10^7 photons /Gy.Å.steradian.cm^3 at 410 nm
	Sapphire (800-5000nm)	0.4 dpa (Transmission; No degradation: 1 mm^t)	Radioluminescence: 10^{10} photons /Gy.Å.sterarian.cm^3 at 410 nm
Optical fibres (Visible region)	Pure silica (core)/F doped (clad)/Al jacket (RF KS-4V)	10^7 Gy (under gamma) (Transmission: 2-2.5 dB/m)	Radioluminescence:
(Visible region)	Pure silica (core)/F doped (clad)/Al jacket (JA F-doped)	$1x10^{-2}$ dpa (Transmission: 20 dB/m)	Radioluminescence:
(IR region)	Pure silica (core)/F doped (clad)/Al jacket	1 dpa (Transmission: 10 dB/m)	
Mirrors /Reflectors	First mirrors: Metal (Cu, W, Mo, St.St., Al)	40 dpa (Cu) *1 (Reflectivity: No degradation) 0.1dpa (Mo) (Reflectivity: No degradation)	
	First mirrors for LIDAR: Single coated (Rh/V, St.St)		
	Dielectric mirrors: (HfO_2/SiO_2, TiO_2/SiO_2)	< 10^{-2} dpa *2 (Flaking, Blistering)	
	LSMs *3: (Mo/Si, W/B_4C and W/C)	< 10^{-2} dpa (the shift of the peak reflectivity to shorter wavelength)	
	X- ray crystals: (Ge, Si, SiO_2, Graphite)	10^{-2} dpa	

***1 Imitation experiment using Cu^+ ions of 1 or 3 MeV**
***2 Partially damaged, Dielectric mirrors are used as second mirrors.**
*3 LSM (Layered Synthetic Microstructures): in well-shielded location and temperature control.

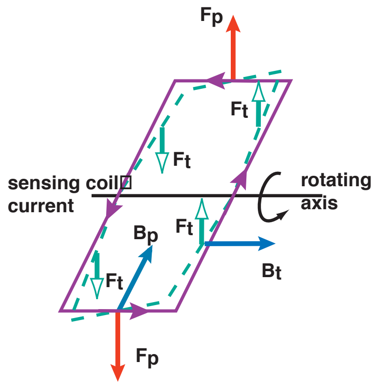
Bolometers. The first part of this programme has been to determine the radiation hardness of existing bolometers and candidate bolometer substrate materials. The JT60 type bolometer Which utilises a gold absorber on a polyimide substrate was found to be insensitive to gamma

radiation but the resistivity is expected to change significantly under neutron irradiation. JET type bolometers use mica as the substrate material and this is expected to be relatively radiation hard. Tests on mica have been performed in the JMTR and no significant changes in the physical properties have been observed up to a dose of 10^{-2} dpa. Bolometers based on mica may therefore be suitable in ITER for at least the initial years of operation. In a joint EU/JA investigation, an in-situ irradiation test of a 'JET' type bolometer employing a mica substrate is planned to start in October 2000. Alternative potentially radiation hard substrate materials, Al_2O_3, AIN, CVD diamond, natural diamond, KU1 and Si_3N_4, are also under investigation. A new type of bolometer based on the ferroelectric effect has also been proposed but is not yet under development.

1.2 Development of new diagnostic components and techniques

The principal components developed in this programme are a steady-state magnetic sensor, enhanced NPA detector, neutron activation with fluid flow, bubble chamber detector, diamond neutron detectors, and an active optical alignment system.

Steady-state magnetic sensor. For very long pulses, the classical inductive method for measuring the poloidal magnetic field will become subject to unacceptable systematic errors arising from integrator drift. A method of measuring static magnetic fields is therefore required and a promising technique is being developed. A jxB device measures the field by observing the torque on a current loop (Fig. 2-2). The torque is measured by a balanced load cell (four strain gauges in a Wheatstone bridge arrangement). The initial work has shown the basic viability of the technique. Current work is concentrating on establishing the linearity and sensitivity. Radiation tests are planned.



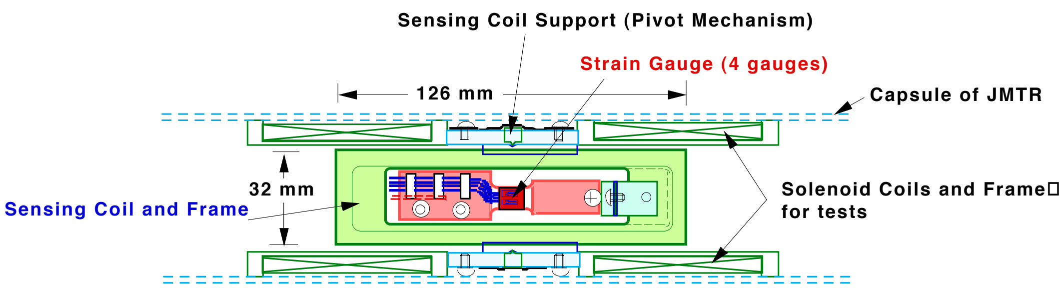


Fig. 2-2. Principle of Operation and Prototype Implementation of the Load Cell Type Steady State Magnetic Sensor

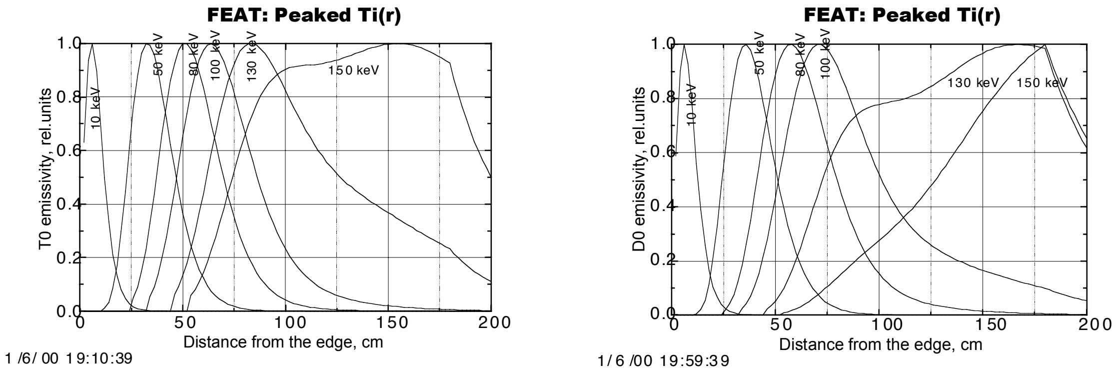


Fig. 2-3. Calculated Emissivity Functions for D and T and Typical ITER-FEAT Conditions Showing that with the New NPA Detector Measurements of n_d/n_t Should Be Possible for r/a ≥ 0.6.

Enhanced NPA detector. The NPA will be the main diagnostic for measuring the fuelling ratio, n_d/n_t. In order to make measurements in the plasma core, it is necessary to measure particles emitted with energies up to 100 keV (Fig. 2-3), but particle detectors presently employed in NPAs cannot be used at high energies because of high sensitivity of the detectors to neutron and gamma radiation. An upgraded NPA detector employing an accelerator unit is therefore being developed. The accelerator has been designed, constructed and tested with satisfactory results. The next step is to develop the analyser dispersion system and to establish its compatibility with the accelerator.

Neutron activation with fluid flow. The neutron activation technique with encapsulated foils is an established method for measuring the time-integrated global neutron source strength. By using a flowing fluid it is in principle possible to make time resolved measurements. In this case water in a flowing loop is exposed to the neutron source in one location and gamma rays from the decay of ^{16}N are measured remotely (Fig. 2-4). The technique has the advantage of requiring very simple hardware near the tokamak. Measurements with an experimental loop of similar size to that which would be needed on ITER have shown that a time resolution of 0.05 s can achieved. The delay in the measurement is about 0.9s.

Bubble chamber detector. The measurement of the confined alpha particle population will be important on ITER and in principle it can be measured by analysing the 'knock-on' tail on the neutron spectrum. The technique requires a neutron detector with a very sharp energy threshold and bubble chambers are a good candidate. An initial feasibility assessment has been carried out with promising results. Water and liquid CO_2 are proposed as the prospective working liquids and it is proposed to using flowing liquid to obtain the required time resolution.

Diamond detectors for neutron measurements. Diamond detectors are a possible device for measuring the DT neutron spectrum. They have the advantages of high energy resolution coupled with small size and high radiation resistance. In principle both natural and synthetic diamonds can be used. Thus far the performance of the natural diamonds is superior but the

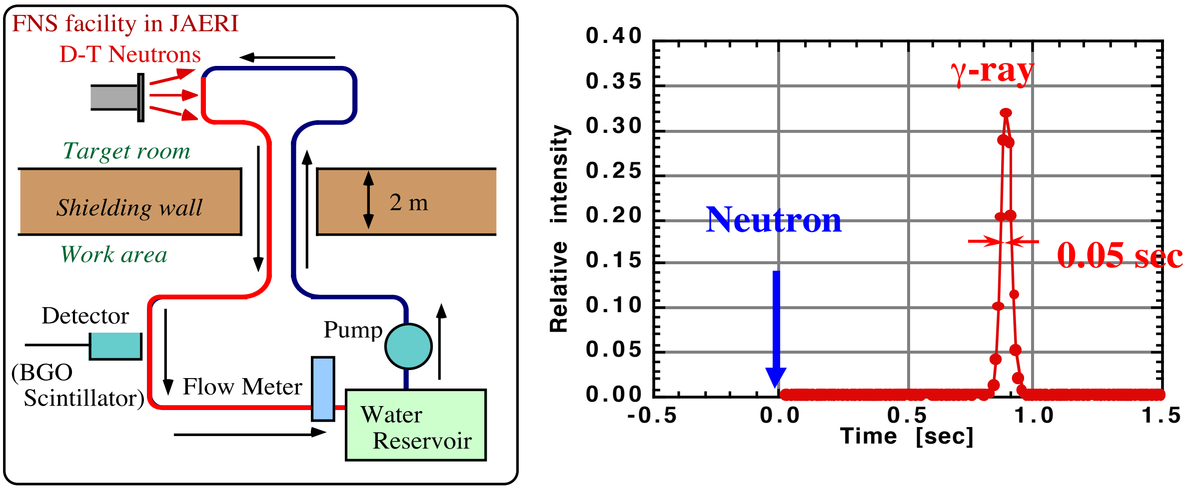


Fig. 2-4. Schematic of the Experimental Test of the Fluid Flow Activation Technique and some Initial Results.

synthetic diamonds have the potential advantages of higher availability and more reproducible performance. In this programme the performance of both natural and synthetic diamonds is investigated. Radiation induced defects in the diamond could degrade the performance and current work is concentrated at investigation this possibility in natural diamonds and exploring possible restoration (e.g. annealing) techniques.

Optical alignment system. The optical paths of ITER diagnostics run through many relay mirrors, lenses and other optical components which are mounted on different supports. The supports are will be subjected to differential movements. A prototype active alignment system suitable for application on ITER is therefore being developed. The system is being develop specifically for the divertor impurity monitoring system but the techniques and hardware could potentially be applied to other optical diagnostics.

3. Results Expected By July 2001

The main results expected in these topics by July 2001 are summarised in Table 3.1.

Table 3-1. Main Results in Diagnostic R&D Expected by July 2001

Topic	Main results expected by July 2001
Ceramics, insulators, wires/cables	Systematic studies of irradiation effects on ceramics and insulators will be continued including studies of RIED, RIC, dielectric properties, mechanical properties, thermal conductivity and tritium diffusion. For these effects sufficient data should be available for component design. Results of a more detailed investigation of RIEMF in MI cables will be available.

Topic	Main results expected by July 2001
Mirrors and reflectors	Results on the effects of deposited materials, including the effect on specular and diffuse reflectivity. Sufficient data should be available to permit a choice of almost every diagnostic first mirror.
Windows	Data will be available to support the selection of a window material and mounting method for every diagnostic.
Optical fibres	More precise information on the potential performance of optical fibres in the intermediate (cryostat) region. Development of more radiation resistant fibres.
Bolometers	Data to enable the choice of the bolometer type for initial ITER operation.
Steady-state magnetic sensor	Completion of radiation tests including in-situ measurements of linearity and sensitivity.
Enhanced NPA detector	Completion of development of enhanced detector enabling measurements of neutral particles up to an energy of 100 keV.
Neutron activation with fluid flow	Full characterisation of prototype assembly including comparison with calculations and determination of the accuracy of neutron flux measurements by this method.
Bubble chamber detector	Completion of development and evaluation of prototype flowing fluid device.
Diamond detectors for neutron measurements	Completion of the investigation of the effects of radiation induced defects and exploration of possible restoration techniques.
Optical alignment system	Full evaluation of prototype system for divertor impurity monitor.

3.15 Safety related R&D Progress Summary

1. Objectives

Safety related R&D up to the end of 1998 was primarily oriented to understanding safety issues and investigating ITER related phenomena to obtain the basic data and develop models and tools for ITER safety assessments. Since 1998, the safety related R&D has the objective of verification and validation of data, models and computer codes used in the safety assessments.

Tasks Validating Data used in Safety Assessments:

Task	Objective
Beryllium-steam reaction studies	Perform Be-steam integral validation testing on first wall mock-ups simulating the ITER first wall with a Be protection layer.
Tritium mobilisation from co-deposited (soft) carbon films	Obtain data for D/T mobilisation from co-deposited carbon films in air and steam for samples from tokamaks and produced in plasma test facilities.
Dust source term studies	Investigate dust characteristics, chemical reactivity of Be-dust with water/steam and air, hydrogen isotope behaviour (accumulation, permeation, desorption) in plasma facing materials, and consequences of chemical interaction between carbon elements and Be-dust.

Tasks Validating Computer Codes used in Safety Assessments:

Task	Objective
Integrated ICE facility for analysis validation	Demonstrate that the pressure-suppression system and its design parameters are adequate for mitigation of Ingress of Coolant Events (ICE) validate safety analysis codes and methodologies.
Corrosion code validation experiments	Experimental validation of the PACTITER code, which is used to model the transport of activated corrosion products, under representative ITER conditions.
Thermohydraulic code validation experiments "EVITA"	Simulate physical phenomena occurring during water/steam into the cryostat, in particular condensation on a cryogenic surface. Experiments should provide data on asymptotic ice layer thickness to validate local thermo-hydraulic effect correlations and models under water/steam and air ingress into the ITER cryostat.
MAGS validation	Validation of the module CRESJ (Contact Resistance Strand-Jacket) in the MAGS code system by experiments on the contact resistance between superconducting cables and their jackets.

A related task is provides the necessary computer analysis and code support for validation.

2. Achievement up to May 2000

Safety related R&D prior to 1998 had the objectives of understanding the issues, gathering data on source terms, modelling underlying phenomena, and developing and applying analytical tools for the safety analysis of ITER.

Fundamental to the safety assessment of a nuclear facility is understanding the hazards involved and in particular the radioactive source terms and mechanisms for their mobilisation. Considerable effort was focused on gaining this understanding. R&D on oxidation driven mobilisation of activated materials (combined with decay heat estimates and analytical modelling) showed that this would not be an issue for ITER. Dust from existing machines and disruption simulations were characterised. Tritium implantation, permeation, de-sorption, co-deposition and mobilisation were investigated and models supported by underlying data developed for use in safety analysis.

An important issue for water-cooled plasma facing components with beryllium is the beryllium-steam reaction that can occur during accidents when there is an ingress of coolant into the vacuum vessel. Several tasks gathered basic data on the beryllium-steam reaction kinetics and its dependence on temperature, form of beryllium, surface state, etc. In addition, analytical models were developed to be able to predict hydrogen formation and the effect of the exothermic reaction heat which could lead to a self-sustaining reactions. These models and data were used in the NSSR-2 analysis.

Understanding decay heat is important in accident sequence modelling as it is a mechanism to mobilise activation products and possibly damage confinement boundaries. Tasks related to the validation of decay heat predictions included irradiation of stainless steel and copper by 14 MeV neutrons. The uncertainties of 5-10% from these validations are incorporated in the bounding decay heat estimates used in safety analysis.

The importance of activated corrosion products for occupational safety was recognised and experimental and modelling efforts directed at prediction of corrosion product activation and transport in order to be able to predict radiation levels around heat transfer system components. Beryllium handling also raises potential occupational safety issues. Tasks investigated beryllium handling at existing facilities and some additional R&D on Be monitoring and waste handling, particularly the issues related to mixed activated-beryllium wastes. Guidelines and good practices were identified and provided to relevant designers.

A significant effort during the pre-1998 R&D effort was the development of analytical tools to undertake the safety analysis. These included development of new computer codes and modification of existing ones for fusion applications. By 1998 we had a set of analytical tools capable of analysis of transient behaviours of the plasma-wall interactions for accidents, thermal hydraulic behaviour of the heat transport system under accident conditions including discharge into vacuum and onto cryogenic surfaces, temperature and pressure dependent chemical reactions for ingress of coolant into the vacuum vessel, and aerosol, dust and tritium mobilisation and transport. In addition analytical tools were developed and underlying failure rate data were collected to permit plant sequence analysis to be undertaken. These analytical tools and data were essential for the safety assessment of ITER completed for NSSR-2.
Preliminary results from current R&D have been obtained to validate data used in safety assessments. For example, beryllium emissivity factors over a wide temperature range and

different surface oxidation degrees have been obtained using different types of samples. Be emissitivity is important in calculating the temperature of in-vessel components under accident conditions where radiation cooling is assumed. Preliminary results of tritium and deuterium releases from co-deposited layers are available which confirm the assumptions in GSSR are bounding.

Results to date have not identified issues such as new phenomena to be included in the safety assessments or non-conservative assumptions.

For code verification and validation, the set up of current experimental facilities is in hand. Results for the water coolant ingress into the vacuum vessel obtained by the MELCOR, INTRA computer codes and the ISAS system of computer codes have been compared with the test results of the integrated ICE facility (Figure13.5-1). Analysis of ICE tests and code calculations has shown that calculated pressures are in a good agreement and calculated vapour temperatures show a reasonable agreement with the test.

Validation and related benchmarking activities to date show reasonable agreement between the predictions of the safety analysis codes against each other and against experiments.

3. Additional Achievement Expected by July 2001

The final set of results from validation of data is not expected until mid-2001. At that time it will be possible to collate the complete set of data acquired by safety-related R&D and make cross comparisons among the experiments and compare with analysis results to identify uncertainties, margins and gaps in knowledge. Further R&D is likely to be needed in key areas especially mobilisation and transport of in-vessel tritium and dust under accident conditions expected for ITER. Related issue is validation of predicted decay heat under ITER neutron spectra to validate decay heat analyses.

Computer code validation activity is under way and results will be available by mid-2001. In addition to the ICE results mentioned above, the MELCOR and MAGS code models for the water coolant ingress into the cryostat will be validated against the results of the EVITA facility (simulation of water/steam condensation on cryogenic surfaces). Experiments simulating activated corrosion product generation and transport will be performed using the CORELLE facility and compared with the PACTITER code results.

Verification and validation of computer codes used for safety analysis needs to be related to uncertainties and margins in the analysis results. When the current round of validation experiments and benchmarking activities and ITER-FEAT safety analysis results are available, it is planned to identify the codes used, the accidents sequences they were used in, the phenomena that the codes model that significantly affect the results, and the experiments and benchmarking available. This will provide a systematic view of computer code validation specifically for fusion and may identify specific gaps where additional R&D is needed.

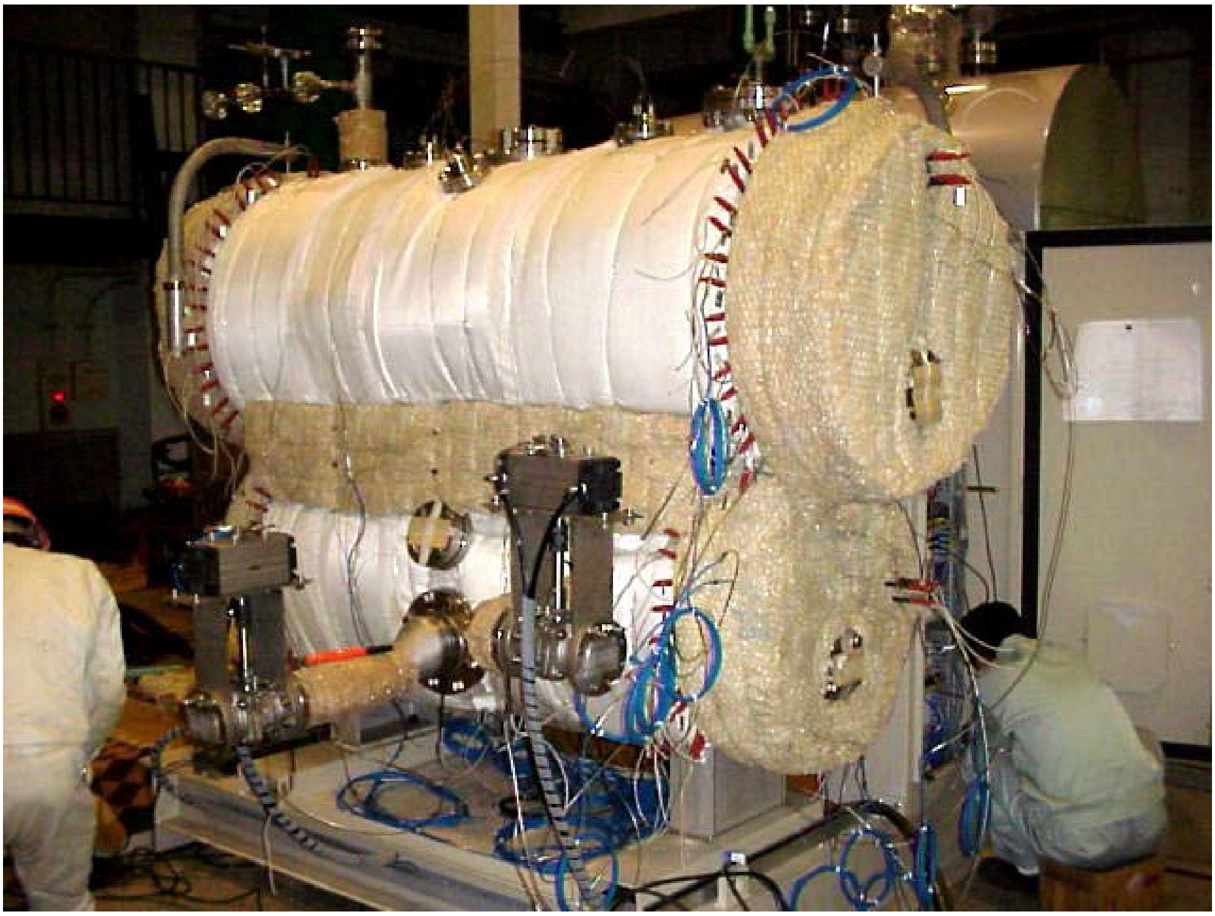


Figure 3.15-1. Assembled ICE facility

3.16 Structural and Plasma Facing Materials R&D Progress Summary

The rationale of materials selection, their properties, effect of component manufacturing cycle, environmental effect of the material behavior and irradiation effect on the material properties are given in the reference ITER documents:
- Materials Assessment Report (MAR),
- Structural Design Criteria (SDC) and
- Materials Properties Handbook (MPH).

The main objectives of the recent R&D are the following:
- justification of materials grades and specific requirements to the materials ensured from the components design,
- full characterization of selected materials (including effect of components manufacturing cycle), and
- providing required database for the components design.

Materials R&D for in-vessel components are performed for the support of ITER design, and are included both in L4 and L5 Projects. Some of the data are generic, and can be used both for divertor and for the first wall modules development. The other ones are performed in support of specific components design (for example, bolt materials or Ti alloys for the flexible cartridge).

The R&D comprises the structural materials (SS 316L(N)-IG, CuCrZr-IG, CuAl25-IG, Inconel 718 and Ti-alloys), plasma facing materials (Be, W and CFC) and joints between SS/SS, SS/Cu alloys, Cu alloys/Be, Cu/W and Cu/CFC.

The main results (received during 1999 and May 2000) of R&D tasks for materials used for ITER in-vessel components design are given below.

Cu alloys Characterization.

The investigations were focused on the CuCrZr-IG and CuAl25-IG alloys proposed for the ITER PFC components.

Achievements:
Tensile, fatigue, fracture toughness, crack nucleation, creep-fatigue, creep crack growth, microstructure and electrical resistivity have been studied.
The design allowables Sm, Se and Sd have been assessed for both alloys on the bases of available experimental data. Those stress intensities should be used for the structural analysis of material to prevent fracture due to flow localization in materials with low work hardening capability (due to irradiation effect). The results are presented in figure 2.

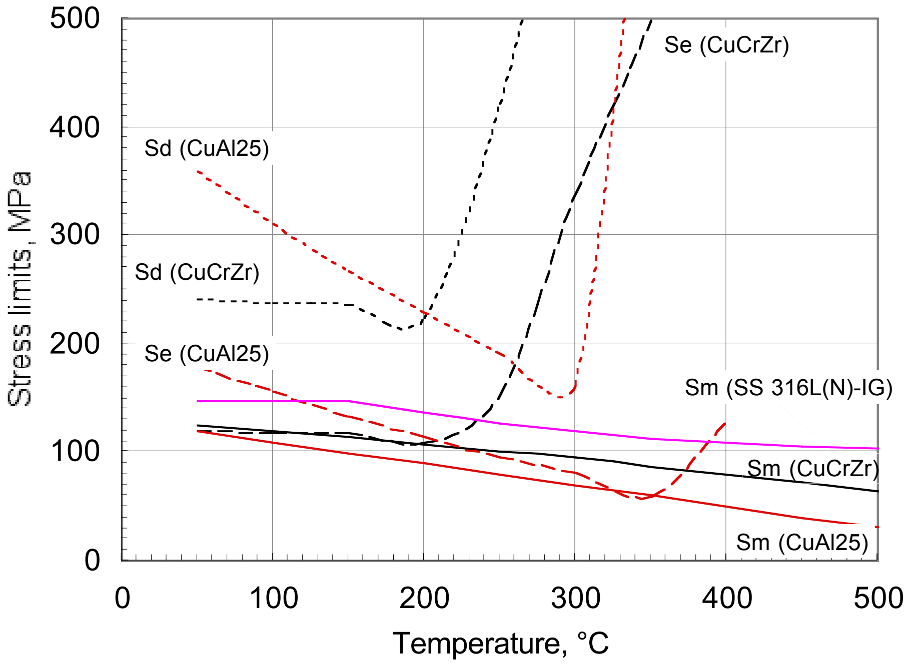


Figure 2. Specification of Stress Intensity Limits for Cu Alloys

Issues:

* CuAl25-IG

- Low fracture toughness of CuAl25-IG is confirmed by various types of experiments, but the crack growth is stable and fracture is ductile. However, the initiation of crack nucleation does not appear to be instantaneous.
- Preliminary results of creep-fatigue interactions and creep crack growth experiments suggest that the time dependent cyclic phenomenon may play an important role in determining the component's lifetime.

* CuCrZr-IG

- Mechanical properties and thermal conductivity are sensitive to heat treatments (hence to joining technology).
- Fracture toughness of CuCrZr-IG decreases with temperature, but remains considerably higher than in the case of CuAl25-IG.

<u>Cu/SS joints Characterization.</u>

Achievements:

Generally good bonding of both CuCrZr and CuAl25 alloys to SS was achieved. CuCrZr joints require quenching with the following age heat treatment to obtain optimum properties.

Generally, the results obtained for the joints lead to the following relationship of fracture toughness:

- base material > joint sample > irradiated joint sample,
- CuCrZr joints >>CuAl25 joints,
- lower temperature data > same at high temperature.

However, results are geometry dependent, and the samples do not generally fail at the interface.

Successful thermal fatigue tests of first wall mock-ups were reported (EU), under the following conditions: 0.75 MW/m^2 @ 30,000 cycles, 5 MW/m^2 @ 1000 cycles.

Understanding of mechanical behavior of joints, however, is complicated. The main factors affect the Cu/SS joint properties;

- mechanical properties of each of the base alloys;
- difference in mechanical properties of base materials;
- microstructure of the joint interface (joining method), diffusion, phase transformation.

Irradiation effects.

Last generation (EU, JA, RF) of HIPed CuAl25/316L(N) and CuCrZr/316L(N) joints demonstrate high level of strength and satisfactory (non-zero) ductility after irradiation to 0.4 dpa at 150°C and 300°C and to 2 dpa at 200°C. Low cycle fatigue tests of Cu/SS joints irradiated to 0.4 dpa at 150°C and 300°C demonstrate that low-cycle fatigue life of specimens decreases by 20-30%.

Numerical calculation demonstrates that irradiation increases the concentration of stresses near to the joint line during bending tests (primarily at 150°C irradiation and testing)

Critical issues:

- A possibility exist to use baking to recover the ductility of irradiated Cu alloys in the SS/Cu joints. This should improve the plastic behavior of joints.

SS and SS/SS Joints Characterization.

Taken into account that most of data for wrought SS 316L(N)-IG is available, the investigations were focused on the SS properties characterization after manufacturing cycle (HIPing) and irradiation effect on the material with the proposed HIP cycle.

Summary of the last results on irradiation tests.

Irradiation effects on tensile properties are summarized in Table below.

Material	Irradiation, 4-5 dpa	Irradiation ~ 10 dpa
Wrought SS	Hardening up to 200% Saturation of hardening	Strain to necking < 1% Loss of strain hardening capability
Solid HIPed SS	Hardening up to 280% Saturation of hardening	Strain to necking ~6-8 % Retains capability to strain hardening
Solid HIPed SS/SS joints	Same as base metal	Same as base metal
Powder HIPed SS	Hardening up to 270% Saturation of hardening	Strain to necking ~3-4% Low strain hardening capability
TIG welds	Hardening up to 80% Saturation in hardening	Strain to necking < 1% Loss of strain hardening capability

Irradiation by dose up to 5 dpa of wrought SS results in minor changes of the fatigue life time (strain controlled) of 316L(N) SS. HIPed SS exhibited slightly lower life time after irradiation by dose of about 3.8 dpa than unirradiated SS.

HIP bonded SS/SS joints and powder–HIPed SS seems more sensitive to irradiation degradation of the fatigue resistance in comparison with wrought SS.

The fracture toughness of SS decreases up to 5-6 times after ~4 dpa irradiation and slow decreasing with higher doses of irradiation. The conservative degradation rule suggested by the EU HT is to use reduction factor of 0.1 at doses exceeding 1 dpa.

Investigations did not reveal susceptibility to SCC of irradiated by dose up to 10 dpa of the 316L(N)-IG SS both wrought cast materials and weld joints.

Critical issue:
Irradiation effect on the material and SS/SS joints after multiple HIP cycle.

Be and Be/Cu Alloys.

The selection of the reference grades as S-65 VHP and DShG-200 is confirmed. This selection was based of the excellent behaviour of these grades at the thermal fatigue-thermal shock conditions. The on-going R&D program is focused on the study of the performance of these materials at ITER operational conditions, especially neutron irradiation effect on the mechanical properties and thermal shock/thermal fatigue resistance and thermal erosion of Be (with and without neutron irradiation). The EU and RF Home Teams mainly perform this activity.

Main achievements:
- Neutron irradiation leads to strengthening and to decreasing of the ductility (Figure 3) and fracture toughness; (For expected ITER conditions (damage ~ 1 dpa, irradiation T ~ 200-300°C) it seems that the total elongation could be still in the level ~ 1%, (more activity is still needed)).
- S-65C Be grade showed slightly less thermal erosion in comparison with other grades and weak dependence on neutron irradiation (at least for fluence < 0.3 dpa).

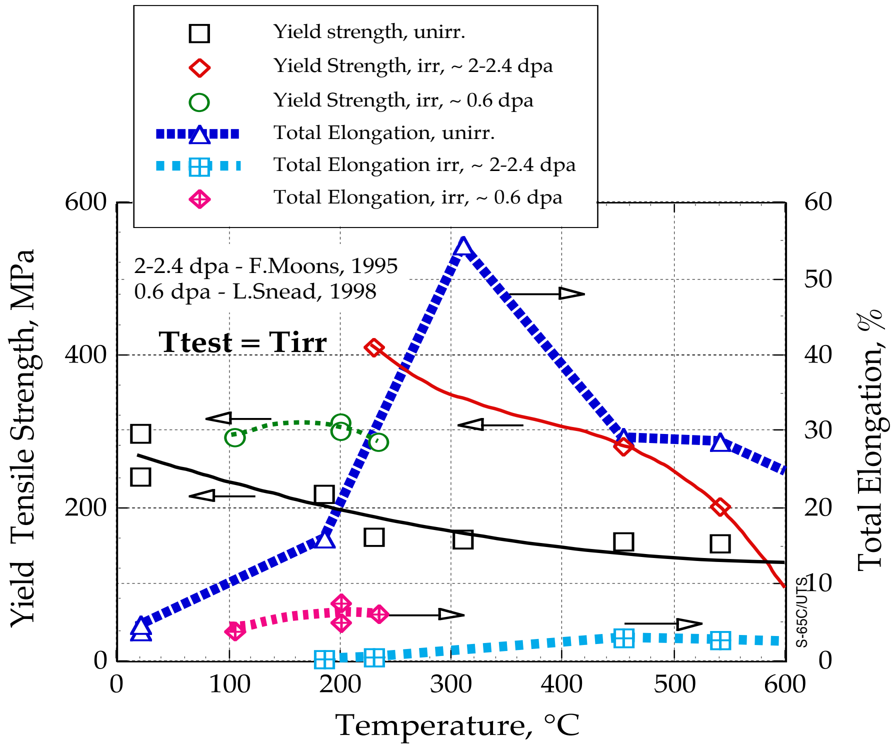


Figure 3. Irradiation Effect on Tensile Properties of Be.

Further activities:

- Further investigation of the irradiated Be under plasma disruption, VDE and consequent thermal cycling are still required. It is important that irradiated Be should have ~1000-1600 appm He and ~1-1.6 dpa.
- Plasma sprayed Be also has to be included in the further investigations, because it seems that this method could have some advantages (to be confirmed) for manufacturing of the first wall.

Optimization of the Be/Cu joining technologies is in the progress. Based on the previous results the following technologies are promising for the first wall components:

- HIP with Ti interlayer, EU HT;
- Brazing with CuInSnTi ("fast brazing", RF HT) and CuMnSnCe alloys (EU HT).

The high heat flux tests of the small-scale mock-ups are confirming this selection, also at least for use of the CuMnSnCe alloy the resistance to neutron irradiation has been indicated.

Further activity includes:

- Optimization of the technology and high heat flux testing (different brazing alloys, brazing temperature, Cu alloy heat sink, tile geometry, brazing time etc.) – RF HT;
- Neutron irradiation and HHF testing of HIPed Be/Cu joints for first wall application, EU and RF HTs,
- Development of the HIP technology with reduced temperature (~ 550-620°), JA HT.

<u>W and W/Cu Joints.</u>

Up to the end of EDA the pure sintered W and W-1%La_2O_3 have been proposed as reference. The ongoing activity is focused on the characterization of these materials as well as

comparative study with other industrial grades and optimization of the W/Cu joining technology.

Main recent achievements:

- The results of the thermal shock testing of the different W grades (Fig. 2) shows that weight loss of the W-1%La_2O_3 is significantly higher than for pure W (JA HT).
- The results of the thermal shock testing of neutron irradiated W demonstrated increasing the cracks formation in W, however, preliminary, thermal erosion was not increased (EU HT).
- Thermal fatigues tests at 15MW/m^2 revealed no difference between pure W and 1%La_2O_3 (EU HT).
- Systematic study of the different W grades at thermal fatigue and disruption simulation has been performed by the RF HT. All tested tungsten grades show good thermal fatigue performance at proper armor design up to heat flux of 15 MW/m^2. Rolled tungsten with texture orientation perpendicular to heat flow direction shows tendency of delamination at heat fluxes higher than about 5 MW/m^2. Disruption simulation tests show that nearly all tungsten grades suffer from surface cracking except for single crystal tungsten. Thermal fatigue test at 20 MW/m^2 of mock-up after disruption simulation demonstrated that there is severe crack formation in almost all materials except for W-5Re-0.1ZrC alloy and W-0.02 Re single crystal alloy.
- W armour 10x10x10mm together with casting technology as reference option is recommended for highly loaded PFC components.

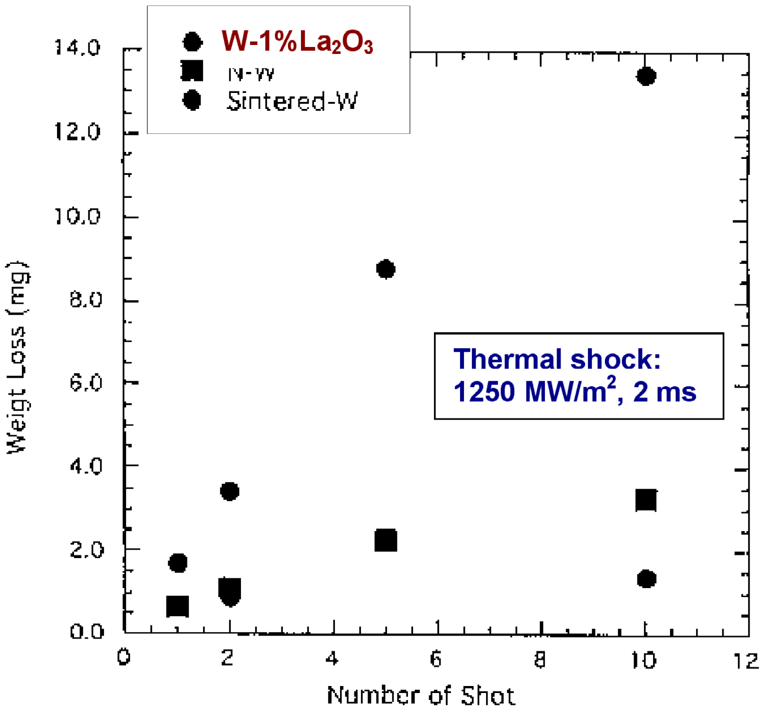


Figure 4. Thermal Shock Tests of W.

These results show that for current design of the W armoured divertor components the pure sintered W could be selected. In case of the use of W in the strike point more activity is needed.

Further activity includes:

- Study of the neutron effect on the W properties (EU HT);

- Optimization of the W/Cu joining technology as:
  - development of the W rods hot pressing technology (JA HT);
  - neutron effect on the W/Cu joints (EU and RF HTs).

CFC, and CFC/Cu Joints.

Recommended materials are 3D CFC Sepcarb® NB31 (EU) and NIC-01 (JA), doped CFC Sepcarb-inox NS31 (CFC with 10%Si) is also under study.

Main recent achievements are:

- The results of the thermal shock testing of the unirradiated and irradiated CFCs have been reported: it was shown that neutron irradiation leads to increasing the thermal erosion of ~1.5-2 times for CFC, EU and JA HT;
- Preliminary disruption simulation tests shows that (energy density – 15 MJ/m^2, t – 40 - 50 μs) shows that Sepcarb-inox NS31 has lower thermal erosion, EU and RF HT;
- HHF testing of the irradiated components demonstrated the excellent performance, EU and JA HTs.

Further activity includes;

- study of the neutron irradiation influence on the properties of reference CFCs, mainly influence of the low irradiation temperature on the thermal conductivity ;
- neutron effect on the CFC/Cu joints with reference technology, EU and RF HTs.

Ti Alloys Irradiation Tests.

The R&D activity on Ti alloys was focused on the investigation of the effect of the irradiation on the tensile, fatigue and fracture toughness properties. The following material grades have been tested:

- (α+β) - alloy, Ti-6Al-4V,
- (α) – alloys, PT-3V (RF HT) and Ti-5Al-2.4Sn (EU HT).

Achievements:

Tensile and fatigue properties have been investigated before and after irradiation (EU HT).
The properties measured before irradiation are satisfactory.
Preliminary results of investigation show that the yield strength of all Ti alloys increases on irradiation. The value of irradiation hardening is different for different materials, irradiation temperatures and material structure (for α+β-alloy). The increase of yield strength of Ti-5Al-2.4Sn α-alloy and of Ti-6Al-4V α+β-alloy at T_{irr}=T_{test}=50°C is ~120 MPa and.~170MPa, respectively. The increase of yield strength at T_{irr}=T_{test}=350°C is about 70-75 MPa for Ti-5Al-2.4Sn and 370-410 MPa for Ti-6Al-4V. Irradiation hardening in much stronger in the α+β Ti-alloy. The irradiation hardening of the α+β alloy increases with the increase of irradiation temperature. The elongation decreases significantly due to irradiation. However, for all temperature range the uniform elongation is above 5% and total elongation is above 10-12%.
After irradiation, the fatigue endurance is better in Ti-5Al-2.4Sn alloy (results only after p-irradiation at high strains). An asymmetry of the cyclic stress has been observed for all test conditions.

Critical issues:

- Choice of a suitable heat treatment (in the case of α+β alloys) that (a) should be compatible with the real component geometry and manufacturing cycle, and (b) should provide optimal combination of properties (strength, ductility, fracture toughness and radiation resistance;
- The chosen materials should be characterized in the actual dimensions of mill products.

Radiation Induced Stress Relaxation of Inconel 718.

High strength Inconel 718 is proposed for the bolts of the primary wall and first wall modules fastening. Radiation induced stress relaxation is one of the critical issue for the bolts feasibility. Two methods were used for in-pile stress relaxation measurement, pressurized tubes and pre-stressed tensile uni-axial loading.

Achievements:

Irradiation by dose from 0.3 to 1.0 dpa and PIE is completed.
Material is characterized in unirradiated state. Results of stress relaxation investigation are presented in the figure 1.
The following conclusion can be made:

- Bolt for flexible cartridge fastening: dose of irradiation ~ 0.02-0.1 dpa
  => Stress relaxation will be ~ 80-90% from initial pre-load for the end of life,
  So, required pre-stress should be ~ 800 Mpa;
- Bolt of the primary wall fastening: dose of irradiation ~ 1.1-2.9 dpa
  => Stress relaxation will be ~10-40% from initial pre-load for the end of life.
  Intermediate tightening of bolts will be required for the bolted primary wall option.

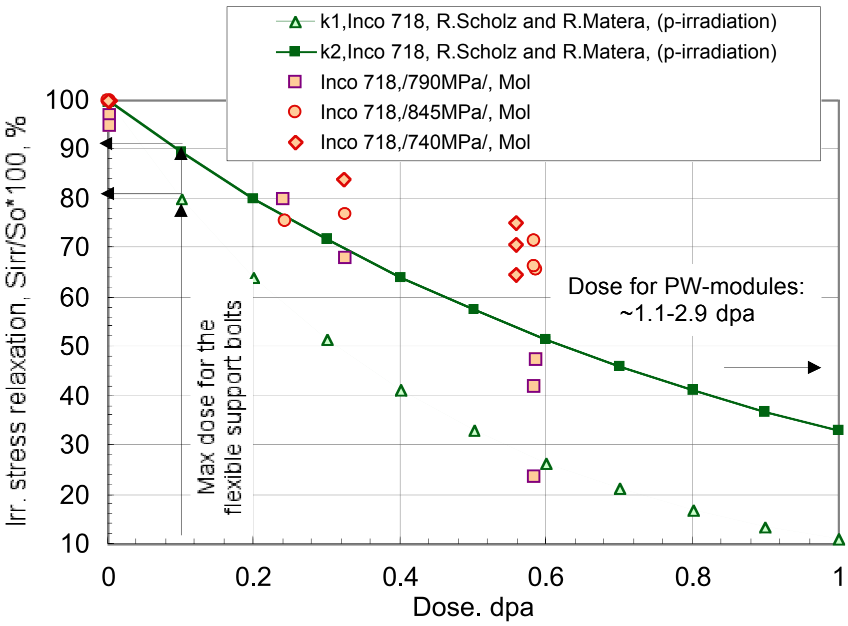


Figure 1. Results on Radiation Induced Stress Relaxation (EU HT).
(Note: $k1=1.2*10^{11}$, $k2=0.6*10^{11}$)

Ceramic Insulator of Tests.

Ceramic insulators are proposed to use for the limiter plates insulation and to provide electrical insulation of the first wall modules. The electrical resistivity changes due to irradiation have been studied within the R&D for diagnostic components. The objective of this investigation was to study the structural integrity of irradiated coatings under impact and compression loads and ability to maintain the electrical resistance after these loads.

Achievements:

Specimens with ceramic insulator coating have been manufactured and tested before irradiation. These are Al_2O_3 and $MgAl_2O_4$ with different substrates.

Coatings manufactured by the JA HT have good mechanical properties in unirradiated state. Preliminary impact tests have been performed. The 12 kgf weight falling from 20 mm did not disintegrate of the coating.

Investigations of the ceramic coating properties before irradiation have been completed by the RF HT. The available data show that break-down voltage is above 2500V, electrical resistance ~ $10^{10}\Omega$ and compression strength >200 MPa.

RECORD OF THE ITER MEETING

Tokyo, 19-20 January 2000

……….

8. The Meeting accepted the ITER-FEAT Outline Design Report (Attachment 6), taking note of the TAC Report and recommendations (Attachment 7) and agreed to transmit the reports to the Parties for their consideration and domestic assessment.

9. In light of TAC recommendation 2 (2), the Meeting agreed that the Outline Design Report provides the basis for continuing design work by JCT and Home Teams.

10. Recognising the importance to optimise a single agreed design, the Meeting asked the Director and JCT to interact with the Parties during the course of their domestic assessments. The Parties should keep the Director informed of the findings of their domestic assessments with a view to optimising a design for approval, following TAC review, at the coming ITER Council Meeting, in the context of the planned Joint Assessment.

……….

EUROPEAN DOMESTIC ASSESSMENT

OF THE

ITER-FEAT OUTLINE DESIGN REPORT

OPINION OF
THE CONSULTATIVE COMMITTEE FOR THE EURATOM SPECIFIC RESEARCH AND TRAINING PROGRAMME IN THE FIELD OF NUCLEAR ENERGY (FUSION)
- CCE-FU -
ON THE EUROPEAN DOMESTIC ASSESSMENT OF THE ITER-FEAT OUTLINE DESIGN REPORT

Brussels, 11 July 2000

INTRODUCTION

The ITER Meeting held on 19-20 January 2000 in Tokyo accepted the ITER-FEAT Outline Design Report (ODR), taking note of the TAC Report and recommendations and agreed to transmit the reports to the Parties for their consideration and domestic assessment.

In order to provide the CCE-FU with a technical basis for its Opinion, at its meeting on 14 January 2000, the CCE-FU gave the responsibility for organising the domestic assessment to the Fusion Technology Committee (FTC) under the Chairmanship of Mr. R. Andreani. Upon Mr. R. Andreani's proposal, the FTC agreed that the following persons would be in charge of preparing the assessment:

- 'Physics': Mr. Robinson, with support by Mr. Kaufmann;
- 'Engineering': Mr. Andreani, with support by Mr. Vetter, Mr. Komarek (new European member of TAC) and an expert from EFET to coordinate industry participation.

It was agreed that, in addition, expertise was to be drawn from scientists and engineers from the Associations, as appropriate.

OPINION

The CCE-FU:

- having had an in depth exchange of views on the basis of the Report from the FTC (doc. EUR (00) CCE-FU 7/3.2) and having heard the outcome of the ITER meeting of 30 June 2000 on the domestic assessments of the ITER-FEAT ODR

- endorses the findings of the FTC on the European Domestic Assessment of the ITER-FEAT Outline Design Review, and expresses the opinion that:

    - **The machine, ITER-FEAT, the design of which is presented in the Outline Design Report and accompanying document (Technical Basis for the ITER-FEAT Outline Design) successfully responds, in broad terms, to the requirements set by the Special Working Group, established by the ITER Council, in response to Task No. 1 and adopted by the Council.**

- **The parameters chosen represent convergence towards a coherent design, based upon preserving adequate margins within the cost target against the new objectives and yet retaining flexibility to exploit advances in physics understanding.**

- **ITER-FEAT can meet its objectives of extended burn in inductive operation with power amplification Q >10 at the reference operating values of Ip=15MA, Paux=40MW, thus providing 400 MW of fusion power. The margins to achieve this objective are in the range of 15-25% . The design also has sufficient flexibility to explore hybrid scenarios with long pulse capability (> 2000 seconds), and scenarios aiming at demonstrating steady state operation with the ratio of fusion power to input power for current drive of at least 5, provided further confinement enhancement can be achieved.**

- **Although most of the components of ITER-FEAT are still in a preliminary design stage, the new design appears suitable to be developed into the final design of a machine capable of achieving the objectives set by the ITER Council.**

- **The target cost for the realisation of ITER-FEAT was set by the ITER Council at about half of the 1998 ITER cost estimates. The present preliminary analysis provides confidence that this target will be reached. By the end of 2000, the detailed cost estimates will be provided from the detailed design specifications and manufacturing studies in the industries of the three Parties. It should furthermore be noted that the Balance of Plant is, to a large extent, site dependant and will therefore remain uncertain until a Host Country has been chosen.**

- **The extensive safety analysis work carried out for ITER FDR is applicable to ITER-FEAT. Tritium inventory and release problems are reduced in ITER-FEAT with respect to the FDR.**

- notes with satisfaction that the ITER Meeting on 30 June 2000 has endorsed the assessments and recommendations of the TAC Report and approved the ODR as updated following domestic assessments and as outlined to TAC as the basis for preparation of the Final Design Report.

EU DOMESTIC ASSESSMENT OF ITER-FEAT ODR: PHYSICS

OVERALL ASSESSMENT

- ITER FEAT performance predictions benefit from the reduced uncertainties arising from the reduced value of normalised beta and the more favourable predictions of the power necessary to access regimes of improved confinement and particularly from the reduction in density normalised to the Greenwald density into regions covered by the confinement database for existing ITER geometry devices.

- The parameters chosen represent convergence towards a coherent design , based upon preserving adequate margins against the new objectives within the cost target with a shallow minimum at intermediate aspect ratios than at lower or higher ones and yet retaining flexibility to exploit advances in physics understanding.

- ITER-FEAT can meet its objectives of extended burn in inductive operation with Q >10 at the reference operating values of Ip=15MA, Paux=40MW, and has sufficient flexibility to explore hybrid scenarios with long pulse capability (2000s), and scenarios aiming at demonstrating steady state operation with the ratio of fusion power to input power for current drive of at least 5, provided further confinement enhancement can be achieved. Margins exist to achieve the performance objectives typically in the range of 15-25% by increasing the density and current within the achieved normalised β and density parameters and safety factor , q ,on existing ITER geometry devices. A smaller but adequate margin of 10% exists to achieve the approach to ignition at high currents with q less than 3, albeit for limited pulse durations.

- For the divertor an integrated approach has been achieved consistent with inductive operation scenarios which limit the power loading to the divertor targets to less than 10MW/m**2 (to be compared with 20MW/m**2 achieved with carbon & tungsten target tiles in the R & D programme) and with acceptable He concentrations as demonstrated on existing devices. SOL scaling, C migration and minimising T retention , and a reduction in ELM amplitude whilst retaining the required confinement enhancement, are key R & D requirements. Divertor compatibility with the approach to steady state scenarios is more severe and remains to be demonstrated. Partial detachment, reduction in ELM amplitude particularly at high normalised densities, as observed in ITER geometry experiments, gives confidence that an acceptable solution can be found.

- The position control capability is fully acceptable. It is capable of controlling significant excursions in inductance and β at increased elongations (c.f. FDR) at acceptable power levels given the near up-down symmetric coil system and internal support rings for the blanket modules. The two ECCD MHD control systems offer the flexibility both for NTM control and sawtooth stabilisation ,and the external coils can be used for both error field control and RWM stabilisation.

- The H & CD systems have increased importance in ITER-FEAT compared with the FDR, and high reliability and continuous operation together with coupling into narrow SOLs and ELMing plasmas for IC & LH systems are key R & D requirements.

- A comprehensive list of diagnostics has been proposed with a focus on current profile control but with large extrapolations from present systems requiring an early focus on the necessary R & D.

- The key elements of R & D in the short term in priority order are:- the population of the database at sustained high normalised densities for the ITER reference equilibrium parameters, and the related problem of the reduction in amplitude of ELM's, increase in pressure arising from stabilisation of NTM's, the development of techniques to minimise the tritium retained in co-deposited layers of carbon and the avoidance and mitigation techniques for disruptions including the sensitivity to the production of runaway electrons.

ITER FEAT
Outline Design Report European Assessment
Engineering

1. General Assessment

The machine, ITER FEAT, which design is presented in the Outline Design Report and accompanying document (Technical Basis for the ITER FEAT Outline Design) responds, in general terms, to the requirements set by the Special Working Group, established by the ITER Council, in response to Task No. 1 and adopted by the Council.
Although most of the components of ITER FEAT are still in a preliminary design stage and also the cost of the machine (reducing the cost to about 50% of the cost of ITER, as given in the Final Design Report, was one of the key objectives) is given so far as a simple extrapolation of the ITER FDR cost estimate, the new design appears suitable to be developed into the final design of a machine capable of achieving the objectives set by the ITER Council within close to the cost limits.

Extensive interaction between the Joint Central Team and the Home Teams of the parties has taken place through the work of a joint "Concept Improvement Task Force" and an "Integration Task Force" where fundamental physics choices and many aspects of the design have been discussed.

In the course of the European assessment, a series of questions aiming at better clarifying the rationale behind a number of choices made by the JCT have been submitted to the Project Director. To most questions detailed answers have been provided in advance of a meeting held in Garching on May 22, 2000, where technical discussions with the JCT have also taken place.

The assessment has been conducted by Dr. R. Andreani, Chairman of the Fusion Technology Committee assisted by Dr. J.Vetter, also member of the FTC, and by Dr. P. Komarek, European member of the ITER Technical Advisory Committee, by Dr. J.M. Bottereau of CEA EURATOM Association, who has taken part in the Garching meeting, and by members of EFET. Industry had been requested to look in particular into aspects of design related to manufacturing procedures and to cost drivers.

2. Magnetic Structure

The magnetic structure of the machine has evolved in the sense indicated in the past by the European Home Team. The choice of a wedged solution for the Toroidal Field magnet has allowed to adopt a Central Solenoid (CS) made by a number of coils.
This provides a more viable and safe manufacturing solution for the CS, eases transportation problems, already reduced by the size of ITER-FEAT, and allows greater flexibility in controlling the plasma shape.

The design of the TF coils has not been frozen yet. The JCT appears oriented to maintain the radial plate configuration adopted for ITER FDR , in spite of slightly higher costs, emphasizing strongly the electrical insulation issue. The manufacturing process of this configuration has been demonstrated through the construction of the TF model coil and has to be validated by the model coil operation soon. The JCT has been encouraged to continue investigating cost effective manufacturing methods.

The CS pancakes will use the square cable in conduit conductor already developed. The question of the use of Incoloy as a jacket material has been raised again. This material is still

presenting problems (a satisfactory weld filler has not yet been fully qualified). On the other side, also the possibility of using a stainless steel, capable of standing the heat treatment needed to form the Nb_3Sn superconductor, without losing part of its fatigue performance, is still under investigation.

For the TF conductor jacket the JCT has now chosen stainless steel. For the square CS conductor, three alternatives have been presented as far as the jacket is concerned: a thin titanium liner and a steel jacket made by two semiconduits welded together, an Incoloy jacket or a steel jacket. Final choice will depend on results of the running material qualification tests.

Altogether, the design and the related R&D of this basic component of the machine, is well advanced. The CS model coil testing is fully confirming the performance expected according to the design assumptions for the Nb_3Sn superconductors but confirmation of the relevance of the joints in respect to the new CS design has still to be made. For the Poloidal Field coils, made of NbTi, a limited R&D programme is being launched.

Concerning the question of the choice of a solution with coils internal to the vacuum chamber for plasma control, the position of the project is that the technical risk involved would be too high and is not compensated by a large enough gain in allowable plasma elongation.

3. Vacuum vessel, first wall/blanket, divertor

The decision of removing the backplate is reasonable in the case of ITER where temperature difference of the coolants in the blanket and in the vacuum vessel is rather low (temperatures are both in the range 100 – 150 ^oC) and no separation is needed between blanket and vacuum vessel which acts as the first containment.

As a consequence there is a reduction in cost but an increase in the complexity of the vacuum vessel, which has to support directly the blanket modules and an increase of the safety issues.

However the on site assembly time of the vacuum vessel has been reduced by going to nine 40^o sectors. This fact would require additional capabilities in the machine shop (one sector weights more than 530t) and would complicate the transport. But it would allow better control of most of the fabrication procedures and would reduce the number of field welds. A good part of the critical problems posed by the manufacturing and installation of such a complicated structure has been covered during the EDA. New possible methods of fabrication (forging or casting of components or hiping of parts of the vacuum vessel) are under investigation with the scope of reducing the cost. The design appears well under way.

Use is being made of the housings for the blanket modules supports acting as connecting ribs to reinforce the double walled vessel.

As far as the blanket is concerned, the size of the modules has remained as in the FDR (so their number has substantially decreased) but they have been simplified making them flat. In addition a separable first wall has been adopted and this appears a rather important improvement concerning machine operation and reduction in the amount of waste produced. Typically, after dismounting blanket modules from the machine, only the first wall elements would be replaced in a hot cell. A further improvement which needs to be pursued is a solution whereby first wall elements could be removed and replaced in situ without removing the entire module.

No change in armour material is envisaged. An extremely important problem is the tritium retention in graphite in the divertor region. Considering the very large uncertainty on this issue, an in depth investigation on JET is mandatory.

4. 17 MA Operating scenario

The 17 Ma , 700 MW operating scenario has been investigated in more detail by the JCT. The conclusion of the analysis are the following: burn duration based on the available magnetic flux is of the order of 170 s; PF coils and related power supplies are capable of providing the needed plasma control within their operating limits; the TF magnet system is capable of 20,000 cycles at 17 MA vs. 30,000 at 15 MA although some additional analysis is required concerning fatigue performance of the insulation of the poloidal shear keys in the bottom part of the TF coils. Nuclear heating of the coils corresponding to the larger neutron production can be dealt with by increasing the helium flow rate within the limits allowed by the presently designed cryogenic system.

5. Long pulse operation

Steady state operation of the TF magnets is possible both at 15 and at 17 MA plasma current as far as the cryogenic system is concerned. Limitation is set by the present design of the cooling towers and water collection system and is only related to cost. The allowable length of continuous operation depends on atmospheric conditions and is located at present between 2,000 and 4,000 s. While there are no new feasibility issues and only relatively minor cost implications for the Fuel Cycle in accommodating long pulse operation, an additional tritium inventory of several hundred grams appears to be necessary.

6. Balance of plant

The layout of the plant has been substantially reviewed in order to reduce the footprint of the buildings and the length of needed interconnections, and also to maximise the reuse of buildings for different purposes at different stages of the project. The significant improvement achieved is shown from a reduction from ITER-FDR to ITER-FEAT of 50% in the cost of the BOP, which is more than one would expect from simple scaling.

7. Cost

The estimate of ITER-FEAT cost, as presented in the Outline Design Report, is only a preliminary rescaling of the ITER FDR cost (1998). The result (~4 BEURO) exceeds the target of 50% of ITER FDR cost estimate.
However this estimate, made by simple rescaling, does not take into account possible improvements in the design and fabrication processes of reduced size equipment.

With reference to the study of the European Union "Intermediate Aspect Ratio" machine (EUI), and the costing exercise made by European industry, significant savings appear to be possible in several areas. Nevertheless the cost of the Balance of Plant is, to a large extent, site dependant and will therefore remain uncertain until a Host Country has been chosen.

A valid cost estimate of ITER FEAT will be obtained only after engineering detailed studies have been worked out to provide specifications for an industrial cost estimate. This cost analysis would be performed by the end of the year 2000.

JAPANESE DOMESTIC REVIEW

ON THE ITER-FEAT OUTLINE DESIGN REPORT

March 29, 2000

The Fusion Council

Atomic Energy Commission of Japan

1. Introduction

1.1 Background

The Engineering Design Activities (EDA) of the International Thermonuclear Experimental Reactor (ITER) had been conducted under the cooperation among the four Parties; Japan, the European Union (EU), the Russian Federation (RF), and the United States (US) since July 1992. The objectives of ITER would be to demonstrate controlled ignition and extended burn of deuterium-tritium (D-T) plasmas, with steady-state as an ultimate goal, to demonstrate technologies essential to a fusion reactor in an integrated system, and to perform integrated testing of the high-heat-flux and nuclear components required to utilize fusion energy for practical purposes.

In June 1998, the ITER Final Design Report (ITER-FDR) was completed, and the activities were to have proceeded to the construction phase. However, for financial reasons in participating Parties, it became difficult to shift into the construction phase. Therefore, the four parties (three parties at present due to the US withdrawal in July 1999) agreed to extend the EDA for three years. During this extended period of EDA, the cost reduction has to be explored by re-examining the technical objectives and by setting new technical guidelines, in order to increase the possibility of ITER construction.

On the basis of these new technical guidelines, the ITER-FEAT Outline Design Report, developed by the ITER Joint Central Team led by the ITER Director in collaboration with Japan, EU, and RF Home Teams, was submitted to the ITER meeting in January 2000, and was distributed to each Party for their domestic reviews. The Fusion Council of Japan has received the ITER-FEAT Outline Design Report, and has requested the Technical Subcommittee of ITER/EDA (subsidiary to the Fusion Council) to review the following documents from the technical viewpoints:

- ITER-FEAT Outline Design Report
- Attached document to the ITER-FEAT Outline Design Report (Technical Basis for the ITER-FEAT Outline design)
- Report of the ITER Technical Advisory Committee (TAC) Meeting

This report summarizes the results of the domestic reviews carried out by the Technical Subcommittee of ITER/EDA.

1.2 Position of ITER-FEAT Outline Design Report

ITER-FEAT is designed on the basis of the new technical guidelines, employing the technology which has been established through the R&D during the past EDA, or that being established by the current R&D. The ITER-FEAT Outline Design Report is the first milestone in the design activity for ITER-FEAT, and should be the basis for the forthcoming detailed design work that will lead to the ITER-FEAT Final Design Report.

1.3 Technical Objectives of ITER-FEAT

The technical objectives set in the new guidelines are as follows:

(1) Plasma Performance

- The ratio of fusion power to auxiliary heating power should be at least 10. The possibility of controlled ignition (Q_∞) should not be precluded.
- An inductive pulse flat top capability during burn should be 300 s to 500 s.
- Steady-state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5 should be demonstrated as the aim of ITER.

(2) Engineering Performance and Testing

- Technology essential for a fusion reactor such as superconducting magnets, remote maintenance devices, etc., should be demonstrated.
- Testing for nuclear engineering devices, high heat-flux components, and blanket modules for DEMO reactor should be carried out. The requirements include:
  Average neutron flux ≥ 0.5 MW/m^2, and
  Average neutron fluence ≥ 0.3 MWa/m^2.

1.4 Viewpoints of the Review

The review of the ITER-FEAT Outline Design Report has been conducted primarily from the following viewpoints, taking the previous domestic review of the ITER-FDR and the new technical guidelines for ITER-FEAT into consideration. In addition, technical issues, to be clarified in the forthcoming activity, have also been discussed.

(1) Does the Outline Design satisfy the technical objectives for ITER-FEAT?
(2) Is the Outline Design based on the technology R&D having been conducted so far or on the database being currently established?
(3) What are the key points that need consideration in the Detailed Design?

2. Assessment

2.1 General Assessment

ITER/EDA Technical Subcommittee judges that the plasma performance defined in the ITER-FEAT ODR is appropriate for fulfilling the technical objectives on the whole. However, the Subcommittee notes that the complete scenario for steady-state plasma operation with $Q \geq 5$ is yet to be developed at present.

For the ITER-FEAT engineering performance, the Subcommittee judges that the engineering performance is consistent with the plasma performance and operational scenarios, and that neutron irradiation fluence of 0.3 MWa/m^2 can be achieved based on the results from the technology R&D in the Home Teams.

The safety design is basically directed toward ensuring necessary confinement capability against radioactive materials, while utilizing inherent safety characteristics of fusion and keeping flexibility as an experimental device. The Subcommittee judges that this approach is appropriate.

The estimated cost and schedule are judged to be appropriate on the whole. The Subcommittee, however, encourages that further cost reduction should be pursued without deteriorating the technical credibility in achieving the technical objectives.

In conclusion, since the viewpoints (1) and (2) described in Section 1.4 are satisfied, the Subcommittee judges that ITER-FEAT ODR is a sound basis to proceed to further detailed design, and that continuous work should be advanced toward completion of the Final Design Report.

Regarding the viewpoint (3) in Section 1.4, the Subcommittee has summarized recommendations for the detailed design activity in Sections 2.2 through 2.6 for physics, technology, and other areas. The Subcommittee expects that these recommendations be sufficiently considered in the forthcoming detailed design.

The Subcommittee expresses its sincere respect to the ITER Director, the ITER Joint Central Team, the Home Teams, and the Physics Expert Group for their significant endeavor devoted to the ITER-FEAT Outline Design Report.

2.2 Assessment of Physics Area

The Subcommittee judges that the plasma performance defined in the ITER-FEAT ODR satisfies the new technical objectives, presented in Section 1.3. It is noted, however, that the complete scenario for steady-state plasma operation with Q≥5 is yet to be developed under the present database. The Subcommittee expects that in the forthcoming detailed design, a consistent design among the plasma performance and tokamak device components should be pursued from the following viewpoints:

(1) Limit of Operational Parameter Region

- It is highly rated that the feasibility margin of plasma operations has been discussed.
- Plasma parameters in the vicinity of their operational boundaries for JT-60, JET, etc., have been adopted as operational parameters in ITER-FEAT. Therefore, it is important to evaluate the validity of these plasma parameters. Although confinement deterioration near the Greenwald density has been already evaluated, on the basis of such an evaluation, the feasibility in the margin for plasma parameters of q(95%), κ (ellipticity), and δ (triangularity) attained simultaneously, together with those of $P_{L/H}$, n/n_G, and β_N, should be assessed particularly in the vicinity of their operational boundaries.

To ensure consistency among all items presented in the ODR, a unified set of plasma parameters should be chosen as a reference set. Through evaluation on each parameter margin from the reference set, it should be confirmed that the confinement performance, operation limits, controllability, divertor performance, fuel purity, etc., fulfill the corresponding requirement. This could allow more detailed discussion on the feasibility study of the technical objectives for ITER-FEAT.

(2) Scaling Law of Energy Confinement Time

The scaling law of energy confinement time employed for the ITER-FEAT design has been derived from the experimental database excluding the data for small tokamak devices, which is different from the case in the ITER-FDR design. The background reason for this should be clarified. A probabilistic analysis applied to the item (1) mentioned above should be similarly applied to different scaling laws. It is important that three-dimensional space of the parameters (H-factor, fusion power, and Q value) be introduced in this probabilistic analysis. In addition, it is encouraged to investigate the cause of the dispersion in the experimental data points of energy confinement time.

- For research on D-T burning plasmas in ITER-FEAT, it is expected to explore the possibility to demonstrate a self-ignited plasma even in a short pulse discharge. Studies on the basis of probabilistic analysis are encouraged for this exploration.
- It is highly rated that a study of the plasma operation with Q=10 has been progressed. Furthermore, in order to reach a perspective on a DEMO through only a single step of ITER, it is important to investigate possible plasma operations with much higher Q-value under the assumption that confinement time would be improved more than predicted. For this purpose, the study regarding the scaling law on the basis of probabilistic analysis is encouraged.

(3) ELMy H-modes

- Design reliability has been significantly increased from that of ITER-FDR, where plasma operation beyond the Greenwald density limit has been assumed. Since some experimental observation shows an increase of heat flux onto the divertor plates during the period of ELMy H-mode with the increase of the triangularity, further discussion on the control of ELMy H-modes is important.

- Since one of the crucial constraints on the plasma performance in ELMy H-modes is considered as neo-classical tearing modes (NTM), the evaluation on the unstable region for the NTM is important. Further study on the possibility of control and suppression of the NTM with electron cyclotron resonance heating and current drive is expected.
- Since high plasma current is necessary for an ELMy H-mode plasma with high performance, the sawteeth instability would be more enhanced. Consequently, further evaluation of the following influences by the sawtooth activity is expected: the influence of the sawteeth oscillation on energy confinement time, that on the ELM activity, that on the NTM. In addition, the evaluation of the influence on the sawtooth activity caused by Alpha and high energetic particles is expected.

(4) Internal Transport Barrier Mode

- ITER-FEAT is designed aiming at a steady state operation with Q=5 by realizing a high confinement state based on internal transport barrier (ITB) modes. Although ITER-FEAT is to be operated in the ELMy H-modes, the achievement of the ITB modes could be also expected. It is important to study the profitability of the ITB modes together with confirmation of a steady state operation, a problem of helium ash accumulation, etc. Further intensive study is expected to increase reliability in prediction of confinement performance of ITER-FEAT by using experimental data on the ITB modes in DIII-D, JT-60, and JET.
- The neo-classical tearing modes (NTM) and resistive wall modes (RWM) are the crucial instabilities to limit the improvement of energy confinement of the plasmas in the ITB modes. Hence, it is necessary to study the region where the instability is excited, and to discuss the methods of its feedback control.

(5) Mitigation of Toroidal Field Ripple with Ferritic Steel

- Since it is essential to mitigate Alpha particle loss caused by the toroidal field ripple, the installation of ferritic steel for its mitigation should be considered.
- It is predicted that installation of ferritic steel may influence plasma MHD characteristics (such as changes in the beta limit and probability of disruption), and hence its assessment should be conducted.

(6) Divertor Function

- It is highly rated that the two-dimensional model analysis on the divertor plasma has been progressed to such an extent that the comparison with experimental data becomes possible. However, it is necessary to show its applicability to an ITER-class plasma. Furthermore, it is important to discuss the compatibility of the divertor function with high density and high confinement operation of the main plasma.
- It is necessary to show that the increase of the particle exhaust by using gas puffing from the upper region is acceptable without changing the HH factor, PL/H, etc., even with neutral particles in the divertor region.

(7) Diagnostics and Control Systems

- It is highly rated to show the categorized diagnostics according to their priority in experimental importance. It is necessary to examine if the proposed diagnostic system sufficiently guarantees sound and stable plasma operations.
- For plasma control, it is important to establish the criteria by which the preparation of the control systems (systems needed from the beginning of the project, and those needed in the later stages), through discussion of the following items:
  - Coverage of control candidates and evaluation of their necessity and sufficiency.

- Survey of control methods and evaluation of their reliability.
- Specification of required heating power for each control method and location of its actuators.
- Clarification of couplings among the multiple controls.

(8) Fuel Supply / Ion Density

- To demonstrate high density plasmas close to the Greenwald density, high field side pellet injection is regarded as a promising method. Since the pellet size and velocity for ITER-FEAT should be designed to be larger and faster than those for the existing tokamak, it is necessary to assess pellet injection for ITER-FEAT quantitatively, and a scenario of fuel supply should be discussed with employing both pellet injection and gas puffing.
- In addition, it is important to clarify a model for the prediction of helium density and impurity ion density in a plasma under the major operation modes.

2.3 Assessment of Engineering Area

The Subcommittee judges that the concepts of the engineering components required from the viewpoints of the plasma performance and operation scenarios are appropriate. However, it is necessary to describe the feasibility of the design concepts of the major components such as a conductor of the superconducting magnet, its structure, and the blanket, having been changed since FDR-ITER, on the basis of the results from the current R&D.

Previous to the forthcoming detailed design activities, it is necessary to describe the load conditions and material properties for the components in a normal operation condition, and for the safety components in an abnormal condition as well. In addition, the criteria for evaluating the validity of the analysis and design should be clarified. In particular, the following points should be clarified in the detailed design.

(1) Engineering Constraints on an Operation with Plasma Current of 17.4 MA

(i) Disruption Conditions

It is necessary to establish the operation scenarios at plasma disruptive instabilities including a vertical displacement event (VDE) in a nominal operation (Ip=15 MA). On this basis, it is important to identify the design conditions for planning pulse operations with larger plasma current (17.4 MA).

(ii) Integrity Evaluation of In- and Ex-Vessel Components

On the basis of the established operation scenarios in disruptive instability including VDE mentioned above, the electromagnetic forces dynamically induced in the in- and ex-vessel components should be systematically evaluated. In particular, for the components (such as a vacuum vessel, etc.) related to the safety functions, it is necessary to clarify that their integrity can be preserved through the structural analysis taking account of the operation phase. In addition, regarding the allowable number of operation, determined by the integrity of the tokamak components, it is necessary to perform a detailed structure design with specification of the load conditions, material properties and design criteria.

(2) Superconducting Coil

Geometrical Support Concept of Toroidal Field (TF) Coils

For the TF coil structure, there are two winding pack options, i.e., a thin jacket circular conductor with a radial plate and a conductor with a square steel jacket. Winding pack concept should be finally determined through further comparison of their structures and performances. For the wedge support structure entirely contacting with the inner side of the

TF coil, it is necessary to verify the validity and feasibility of the TF-coil support concept including the illustrative fabrication procedures. On the basis of the existing manufacturing technology, it is also necessary to develop the assembly and fabrication procedures for TF coils.

Central Solenoid (CS) Coil

It is necessary to clarify the performance of CS coil system and the feasibility of winding concept on the technological basis having been achieved or being currently achieved. It is also necessary to verify the technical margin on design parameters of the structure. In particular, for winding by six conductors in hand ("pancake winding" structure), it is important to evaluate the feasibility of this structure concept in the existing fabrication technology. For the support structure of the CS coil hanging from the TF coils, it is necessary to ensure consistency with the requirements for various advanced plasma operations, and to show the structural validity with specification of the load conditions including electromagnetic forces estimated during the operation.

(3) Blanket Structure

In ITER-FEAT, a concept of the blanket module, directly attached onto the inner wall of the vacuum vessel, is adopted. Regarding this attachment concept, it is necessary to show the integrity of the vessel structure under the electromagnetic force and heat loads during the various operational modes. In addition, on the based of the stress analyses with specifying the design concept, the support structure and the operating conditions, it is also necessary to show that the installation of the breeding blanket modules are never to be excluded.

(4) Blanket Cooling Concept

There are two options for the blanket cooling concept, i. e., one with cooling channels inside the vessel structure between the two walls, and the other with cooling channels behind the blanket modules in a vacuum region. In both options, it is necessary to confirm their structural feasibility. In particular, for the former one, it should be confirmed to satisfy the structural design standards having been developed up to now. For the later one, it should also be confirmed to satisfy the structural design standards by evaluating the neutron shielding capability, the consistency with in-vessel components and the electromagnetic forces. On the basis of the detailed analyses, it is necessary to compare the options for the blanket cooling concept and the cooling channel structure employed previously; and the reference cooling concept should be selected as soon as possible.

(5) Support Structure of Vacuum Vessel

Regarding the support structure concept of the vacuum vessel, it is necessary to pursue the design consistent with the tokamak assembly procedures. In particular, for the load conditions, it is necessary to evaluate the combinations of the seismic and electromagnetic loads induced by disruptive instability including VDE, and TF fast demagnetization. On the basis of the load conditions evaluated, the feasibility of the support structure should be clarified.

(6) Evaluation of Neutron Fluence

It is necessary to clarify the primary factor that determines the limitation of the maximum neutron fluence in the current design. To keep an engineering feasibility of operations with higher neutron fluences in future, the main parameters of the breeding blanket system, its operational conditions and its applicability to the current design should be evaluated.

(7) Plasma Facing Materials

Through continual consideration on the plasma facing materials, it is important to evaluate the performance of the materials under the assumed operation conditions, in addition, the tritium inventory in the materials should be continually evaluated.

2.4 Safety

The safety design of ITER-FEAT is based primarily on the principle of 'defense-in-depth' with consideration of the mission of ITER as a fusion experimental device and of the inherent safety characteristics. This approach is judged to be appropriate.

It is necessary in future to clarify such parameters as tritium inventories and others that are important for safety regulation and to evaluate the validity of the analysis in the process of safety analysis. It is also necessary to adequately confirm integrity of the confinement boundaries for radioactive materials in the future design.

2.5 Cost Estimates

It is important to make efforts toward reducing the construction cost of ITER-FEAT to 50% of that in the ITER-FDR.

2.6 Schedule

The present schedule is judged to be reasonable. It is desired that, with the progress of design, consideration on testing and assembling of components should be appropriately reflected in the schedule and that the schedule should be properly performed review as processes for safety regulation become clearer. For the schedule of technology R&D, further efforts are essential so that their results can be reflected in the coming final design.

3. Additional Recommendations

- It is necessary to clarify the differences in the schedule and in the facility design between ITER-FDR and ITER-FEAT due to alteration of their guidelines. Since this implies that the differences would appear in the operational region attainable in a D-T burning plasma, and in certainty of performance prediction, it is important to compare their evaluated differences.
- It is necessary to discuss the validity of the physics design and the feasibility of the engineering design in common. For example, the following items are to be discussed.

  - It is necessary to show the compatibility between the plasma performance and the divertor functions for heat removal and helium ash exhaust, and the feasibility in the divertor hardware.
  - It is necessary to verify the design validity of both the diagnostics and the control systems for ITER-FEAT. In particular, since the plasma parameters for the nominal operation of ITER-FEAT is close to the limit of plasma operational region, plasma controls with high reliability are necessary.
- The JCT physics team should provide the criteria evaluating the importance in physics issues by utilizing the present database.
- The ITER/EDA Technical Subcommittee requests that the design status reflecting the recommendations in this review report on the ITER-FEAT ODR should be accordingly reported to this Subcommittee.

COMMENTS BY THE HOME TEAM OF THE RUSSIAN FEDERATION ON THE TECHNICAL BASIS FOR THE ITER-FEAT OUTLINE DESIGN

RF HOME TEAM COMMENTS
on the
«Technical Basis for the ITER-FEAT Outline Design»,
presented by Joint Central Team

In April-May the discussion of the Outline Design Report materials for the ITER-FEAT was organized in Russia. The discussion was held by three leading institutes - Kurchatov Institute (plasma physics, safety, auxiliary heating and diagnostics), Efremov Institute (electrophysical systems and engineering structures) and RDIPE (blanket) with participation of independent experts from leading RF institutions and enterprises involved in the ITER project.

On the whole the project has been highly appreciated. Despite very short time given for its preparation it appears to be sufficiently consistent.

Nevertheless, the Russian specialists (independent experts including) have made some remarks and recommendations with the aim to improve the Project.

COMMENTS ON DIFFERENT PROJECT SYSTEMS

I. Plasma Physics, Auxiliary Heating, Plasma Diagnostics

Plasma Physics

. **1.** ***Ion cyclotron frequency range***

1.1. It is important to stress the particular role of the **ICRF** system in the prolonged (several years) non-active ITER operation phase (mainly the hydrogen plasma) permitting **several viable problems to be solved.**

In this period the favorable possibility exists to **model the behaviour of alpha-like energetic ions population** in H(He-3) scenario (small amount of resonant He-3 ions in the bulk hydrogen plasma, with mass ratio being the same as for the D/T plasma) at effective temperatures and populations close to classical alpha-particles in the D-T phase with the aim to investigate:

- **classical or non-classical** power transfer of energetic «alphas» to electrons and ions (numerous statements about the «classical» behaviour in TFTR and JET are not convincing because of a low level of alpha power and non-reactor-relevant amount of alpha population in these experiments) due to possible collective phenomena (TAE mode).

- **alpha-like particle transport**

- the effect of **collective phenomena,** e.g. MHD instabilities like **TAE,** and other modes **on alpha-particle transport** even in the non-active phase. This provides a possibility to modify (enlarge) the ICRF complex in case of anomalous alphas losses (if any).

In our opinion it is necessary to include some of these statements in ODR Section 1.2.5.4 (Summary of H/CD system capability).

1.2. Mode-converted off-axis ICRF CD

This scenario is becoming even more attractive in the ITER-FEAT in comparison to the large ITER due to **a smaller major radius** of the machine. Recall that the mode conversion efficiency depends on the radial distance between the «heavy minority hybrid resonance - cut-off» pair. This length is proportional to the major radius R and minority ions density, so the conversion efficiency (due tunneling of Fast Wave) even at a large «minority» content (tritium) rises by a factor of 1.5. The diffraction effects increase additionally the conversion efficiencies (results of full wave STELION code).

In the non-active ITER phase **the mode-converted CD efficiency** additionally rises strongly due to much smaller «heavy» ion content can to be used and position of resonance to be located **at far inside** outer plasma parts. In this far inside plasma parts the population of trapped electrons is significantly decreased, thus increasing additionally the CD efficiency (in comparison with the ECRF CD method which can not work in the far-inside plasma region because of parasitic absorption on the second or third harmonics).
We stress that the TFTR experiments with small and moderate amounts of tritium have made clear the current profile modification in the ICRF scenario.
The typical scenarios H(5-10%He4) or H(5-10%D) are very promising **for NTM stabilization** with excellent localization (see our report made at the H/CD ITER Meeting, Cadarach, September 1999).

Finally, we conclude that ODR statement in Section 1.2.5.4 «IC off-axis current drive is not viable in ITER-FEAT because the mode conversion process yields a very low efficiency under relevant plasma conditions» **is erroneous** and definitely must modified.

1.3. In Chapter II. Section 7, page 10, in subsection I.7.2.3.1 «Operating scenarios (ICRF)» the statement «A large experimental data base is available for **D** majority at the He-3 fundamental in the **«reversed»** minority schemes» is misunderstanding in claiming the «D majority» for reversed schemes. More correct would be to say: «**H** majority», with the rest of the sentence remaining unchanged (because He-3 is «light» minority for the bulk deuterium plasma!). Really, such H(He-3) scenarios were really studied many times (Asdex-U - recently).

2. *LH system*

It is well-known that the LH frequency range method has many unsolved physical and technical problems. Among physical problems there are the following:

- **coupling problem** (epy plasma with specified parameters must be very close to the antenna mouth («grill»)) is very difficult for large machines and especially for ITER (in Asdex tokamak incorporation of 2(!) mm graphite frame completely eliminated coupling of RF power to the plasma). Nevertheless, in TS with a plasma-grill gap of ~ 10 cm the coupling was attained (with a «local plasma branch» formed

between the grill and the plasma). It is very difficult to predict f such a «branch» in the divertor ITER plasma.

- **«spectral gap» problem**, when imposed waves were very fast (~ c/2) and the plasma did not have such resonating electrons, nevertheless, the driven current was effectively created. The physics of this phenomenon was not understood clearly. The opinion that in ITER with its very hot plasma «such gap problem» will be absent is not an answer, because something might happen again, e.g. generation of very slow- own waves with reduced CD efficiency, etc.

There are also **technical problems**, as, for example:

- At planned 5 GHz frequency the **antenna (grill) has a very delicate multi-structure** (toroidally several mm) of small waveguides with thin walls (2 mm). What would happen with such an antenna during a single plasma disruption is easily predictable. Bump limiters can eliminate coupling (see above). Erosion also can be a problem for such LH antenna.
- Possible is the generation of relativistic electrons on the plasma boundary followed by a probable carbon/beryllium «blum» of local spots on the first wall (JT-60 experience, 1988 ITER CDA meetings) or ~ cm holes creation at divertor plates in Asdex, Alcator C, problems with metal and low Z material limiters.

In our opinion it is necessary to say something about these problems (ways how these problems are planned to overcome in ITER-FEAT) in Technical Basis ODR report, even though some promising results on MHD stabilization are obtained on some machines.

3. List of tasks proposed for the self-consistent 1.5-dimensional transport simulation

3. 1. Low Hybrid Heating and Current Drive

To provide the steady state operation in the ITER FEAT it would be necessary to improve the confinement in respect to the H-mode. In present-day experiments such improvement is obtained in the Reversed Shear (RS) plasma configurations.

To create the Reversed Shear configuration in the ITER-FEAT plasma it would be necessary to use some efficient technique to generate non-inductive current near the edge. It was proposed to use LHCD for this purpose.

It is necessary to create the numerical model to describe the LHCD in a realistic manner. The necessary power and LHCD system parameters should be obtained with this model by the self-consistent 1.5D transport simulations.

3.2. Plasma Fuelling

In present day experiments it was obtained that pellet penetration from the high magnetic field side is much higher than from the low field side. Deep pellet fuelling could be attractive in ITER FEAT for particle profile control and increase of the discharge duration (due to the bootstrap current increase).

A realistic model (when available) should be incorporated into the 1.5D transport code to describe the deep pellet penetration. Optimization of the discharge scenario should be carried out.

3.3. Comparative Transport analysis

Several transport models are under consideration by the ITER Expert Group. All these model give similar accuracy to describe the present-day experiments.

It would be necessary to carry out the predictive comparative 1.5D transport simulations of the ITER scenarios at least for a few most predictive models to estimate possible operational space and its peculiarities at least for the basic scenario.

3.4. Auxiliary heating and current drive

In routine calculations of the ITER FEAT scenarios the auxiliary heating system parameters are supposed to change gradually and independently.

It would be necessary to clarify the influence of the real possibilities of the auxiliary heating systems on the basic plasma scenarios.

At present ICRH heating and current drive technique is considered as an alternative to the NBI. They are exepceted to transfer more power to ions than other techniques.

It would be necessary to carry out the comparative analysis of RF and NBI with the realistic system parameters to determine the real advantages and disadvantages of all of them.

3.5. In the most part of the present-day experiments plasma fuelling is caused by the particle sources distributed in the plasma core (such as NBI). Power sources are also distributed in the core. So, for this sort of fuelling the particle confinement time could be considered as similar to the energy confinement time. And plasma transport model accuracy at the edge has no too strong influence on the predictive simulations.

The reactor plasma is supposed to be fuelled by the neutral flux from the edge. For high density reactor plasmas the neutral penetration length (density gradient zone) is lower or about the pedestal width. So, the accuracy of the particles transport model at the edge plays the crucial role (particularly for the required plasma fuelling and for the analysis of the divertor capabilities.)

It would be necessary to carry out studies of the requirements for the divertor and plasma fuelling for different types of the edge density transport models.

4. Simulation of the plasma evolution at the initial discharge stage and during disruption.

4.1. Now the analysis of plasma shape control scenarios for ITER in minor disruption cases is being carried out with use of independent drops of β_p and l_i . But in reality these values depend on each other. In this case it would be much better to consider a drop of plasma pressure instead of β_p drop.

4.2. It is not correct to specify for the plasma current ramp-up analysis only one value of plasma internal inductance equal 0.85. Non-linear simulations show that during the plasma current ramp-up stage a plasma internal inductance value can change within 0.5 - 1.0 for different time moments.

4.3. It would be necessary to study the question of range value of angle of runaway electron incidence on the first wall during major disruption phenomena

5. On the ferromagnetic insertions effect on ITER plasma performance.

As it was noted in ODR, reduction of the toroidal field ripple amplitude by a factor of 2 will be necessary to protect plasma phasing components against extremely high heat loads produced by alpha particle ripple loss in reversed shear operation regimes. Such a reduction is supposed to be achieved by means of installation of the ferromagnetic insertions (FI) close to the plasma column. Preliminary analysis of the idealistic FI construction made at the Efremov Institute has shown that necessary reduction of the ***main harmonic*** of ripple perturbation can be realized in ITER-FEAT. However, first, due to the proximity of the FI to the plasma and, second, due to the deviation of the realistic FI construction from the ideal toroidal symmetry other harmonics of ripple perturbation will appear in the plasma. For alpha particle (and other super-thermal ion) ripple loss the most dangerous seems to be additional perturbation with toroidal number of 2N, (where N is the number of Toroidal Field Coils (TFC)). Apparently, it was the 2N additional ripple perturbation that was responsible for unexpectedly low reduction of the neutral beam ion ripple loss after installation of FIs in the recent JFT-2M experiments. On the other hand, there is a series of principal obstacles like a port existence, which will broke toroidal symmetry of the FI array and, as result, will produce the spectrum of harmonics with lower toroidal numbers. Besides, positioning of the ferromagnetic materials close to the plasma can affect also MHD stability of ITER discharges, in particular, development of the Resistive Wall (RW) and Neoclassical Tearing Mode (NTM) instabilities. Therefore, the following problems should be carefully examined:

1) **Toroidal and poloidal spectrum of the perturbative magnetic field created by realistic set of FIs in ITER-FEAT should be calculated**
2) **Ripple loss of fast ions should be estimated with allowance for the existence of additional ripple harmonics**
3) **Effect of the FIs on the NTM, RW and other MHD modes should be examined**

Estimations 2) and 3) then can be considered as criteria for further optimization of FI construction.

Auxiliary Plasma Heating

Remarks on ODR (Chapter II.7 Heating and Current Drive and Section II.7.2.2 – Electron Cyclotron System)

1. In the Introduction, Section II.7, the functions of the auxiliary heating and current generation systems at different ITER-FEAT operation phases are presented:

a) Pre-ionization and current start-up

b) Attainment of H-mode.

c) Heating of the axial plasma.

d) On- and off-axis CD.

e) Stabilization of MHD-modes, sawtooth oscillations and neoclassic tearing modes (NTM).

f) Control of total current profile to provide the regimes of enhanced confinement (inner transport barrier).

It is stated also that the fulfillment of these functions requires different systems of auxiliary heating to be combined:

a) neutral injection with power P = 30 MW

b) three high-frequency heating systems:

– ICRH
– ECRH
– LH

with a total power of 20 MW (with subsequent increase in power up to 100 MW).

But the analysis of the optimal choice of auxiliary heating and current generation systems for fulfillment of one or other specified function is complicated, since:

1.1 Many parameters to be attained for fulfillment of a specified function are not given, for example, the current profile and non-inductive current profile for attainment of the enhanced confinement and steady-state operation of the ITER FEAT.

1.2. It is not indicated, precisely what systems of auxiliary heating and current generation are planned to be used for fulfillment of the specified function, or the systems are assumed to be chosen during experiments on the ITER-FEAT.

1.3. It is not indicated, whether several systems will perform simultaneously the specified functions, for example the realization of the specified nonmonotonous current profile j(r) (and, hence, q(r)) by ECCD and LHCD.

1.4. If the system for heating and CD is assumed to choose in the phase of experimental ITER-FEAT operation, it would be desirable to analyze in ODR advantages and disadvantages of the systems to be used for fulfillment of a particular function and the problems to be solved to choose the system.

1.5. It might be well (for the analysis of the optimal choice of the systems) to separate the basic functions into different operation phases of the ITER-FEAT:

a) the main phase with achievement of the specified value Q;

b) operation with achievement of the steady-state conditions.

2. Remarks on Section II.7.2.2

2.1. Section II.7.2.2.1 and Table II.7.2.2-1 show that the A-type lead should provide various functions, some of which can be performed simultaneously.

Therefore, to analyze the possibility to perform the specified functions it would be useful to present the EC-system scheme stating the geometry, the distribution of toroidal and φ and poloidal θ angles over different waveguide lines with power distribution, the tasks to be accomplished by one or another part of the EC-system (subsystem).

Otherwise the uncertainties, as pointed out below, might arise:

2.2. As stated in Section 2.1 and Table 2-1, the upper port and B-type lead are intended to stabilize NTM. This separation between the A-lead through the equatorial port and B-lead through the upper port is made with the aim to control only one angle

(toroidal φ or poloidal θ) for the given lead. At the same time the leads for two ports, i.e. equatorial and upper, are given in Table 2-3 describing the NTM stabilization system. Still the question remains uncertain whether these two systems are alternative, between which the choice is to be made later, or they work simultaneously. In the latter case the total power of the NTM stabilization system is 55 MW, this falling outside the limits of the assumed increase in the total power up to P = 100 MW. The matter is that the NTM stabilization system should work simultaneously with the heating system, for which the neutral injection is assumed to be used.

2.3. In Table 2-2 describing the parameters of the A-lead (through the equatorial port) three functions are given, which are likely to be performed simultaneously, i.e. on-axis CD, off-axis CD (at r/a = 0.6) and on-axis H.

In this connection the following questions arise:

a) The experience of the preceding ITER project shows that it is not sufficient to indicate the off-axis CD at r/a =0.6 in order to analyze the possibility for realization of the given profile of the total j(r) or non-inductive $j_{CD}(r)$ current.

b) What is meant by the power value given in the last column of Table 2-2? Since the toroidal angles φ are different, it means that three functions indicated in the first column of the Table should be performed by different subsystems, and hence, it is necessary to know the distribution of power between these subsystems.

The above underlines the need for the EC-system scheme with separation into subsystems, with the parameters of these subsystems, the functions these subsystems should perform, stating the simultaneity of their fulfillment and distribution of power between the subsystems. In this case it is necessary to indicate also with what other heating and current generation systems these EC-systems should simultaneously work.

NBI System

NEUTRALIZER

Injectors of deuterium atom beams currently under development for ITER plasma heating and current drive are based on the idea of negative ion acceleration and further neutralization with a gas neutralizer. The replacement of the gas neutralizer (GN)by the plasma one should increase the NBI efficiency.

The **advantages** of PN over GN are associated with a higher neutralization efficiency at a lower target linear density:

- **High neutral beam power** can be obtained from a given accelerated ion beam power.
  The numbers are as follows: the **GN** maximal efficiency is **61%**, the **PN** maximal efficiency is **85%**. So the efficiency increasing coefficient will be 1.4 . It means that instead of 16.7 x 2 = 33.4 MW we could inject 33.4 x 1.4 = 46.8 MW. This is not so far from 50 MW that might add the third injector.
- **Lower power evolution to the residual ion dump.**
- **Lower gas flow into the beamline (by a factor of five).**

The main **disadvantage**: PN will complicate the injector, possible is reduction of reliability.

There is a **risk factor**: the extrapolation from the present results at PNX-U to PN-ITER is high. It means that an **intermediate step (PNX-SU)** is necessary to minimize the risk.

	PNX-U		**PNX-SU** intermediate step	PN ITER
	Achieved	Design		
Axial length, l (m)	2.5		? 2 m	3
Volume, V (m^3)	0.5		~ 1.5	10
Line density, nl (x 10^{14} cm^{-2})	1.8	1.4	13	20
Plasma density (x 10^{12} cm^{-3})	1 (~1.5 n_{cutoff})	0.6	7	7
Ionization degree	>0.25		0.3	> 0.3
power, (kW)	50		50	500
frequency, (GHz)	7 (Klystron)		24 (Gyrotron)	24 (Gyrotron)
Conductor type	Normal conducting (Cu)		Super conducting (NbTi)	
Magnetic field, (T)	0.36	0.5	1	

PNX-SU ought to be constructed as soon as possible. Its **objectives** are:

- to achieve plasma parameters close to those necessary for PN-ITER.
- to verify the analytical models physical data base used for the design of PN-ITER
- to check the PN-ITER design structure which will be as similar to that of PNX-U as possible.
- to gain technological and operational experience.

Cost estimate

PNX-SU : the device and two gyrotrons with associated power supplies: **k$582**
PN-ITER : Very preliminary and nonofficial estimation made by E. Di Pietro (JCT) **for two PN units 5.6 kIUA vs. 63 kIUA for the third injector.**

Residual Ion Dump (RID)

The reference design of the ITER NBI includes the RID device based on electrostatic deflection of the residual ions. Although the RID design is well substantiated physically, there is no engineering experience in operations of such an electrostatic device at full-scale parameters. It seems reasonable to consider an alternative design version of magnetically operated RID.

PLASMA DIAGNOSTICS

The ITER-FEAT diagnostic complex design is the result of the work of the Joint Central Team and Home Teams. It should be noted that the structure and purpose of the diagnostic complex has undergone the least changes as compared with other systems of the ITER Project.
The complex consists of 40 separate diagnostic systems designed to measure the spatial-time changes of the plasma parameters. The operation principles of various diagnostic systems are applied for various divisions of physics.
The diagnostic complex design was repeatedly discussed at meetings of the Physics Expert Group on Diagnostics and some subject-matter technical meetings.
The regular 12th meeting of the Physics Expert Group on Diagnostics was held recently in Russia.
On the whole, the principle of diagnostic complex design presented in the project, i.e. the composition, purposes, the main peculiar features and construction principles should be recognized as conforming to the purpose of the ITER-FEAT project.
But the realization of various diagnostic systems under the ITER-FEAT project, both already employed in experiments on tokamaks and newly developed for the ITER-FEAT project, might face considerable problems because of the specific character of the ITER-FEAT, first of all, considerable radioactivity.
Not completely understood is the phenomenon of radiation-induced EMF in the MI-cable, the problem concerning the first mirror has not been solved.
There are draft designs for arrangement of different diagnostic systems on the facility.
There are no engineering studies of the «input device» arrangement of most diagnostic systems on the vessel and ports of the facility.
The design versions for arrangement of the «input devices» should be elaborated concurrently with the development of the design of the vacuum vessel and ports of the facility.

Unfortunately, some questions, the answers to which are required when designing the ITER-FEAT diagnostic complex, are not embodied in the document. In particular, one of the main problems is the degradation of the optical properties of diagnostic elements located inside the vacuum vessel (mirrors, windows). The main processes causing the degradation are:

- Sputtering by charge-exchange atoms (CXA). The influencing factors are the composition, flow and energy spectrum of CXA; mirror material and its state (polycrystal, monocrystal, alloy); the possibility of chemical sputtering; and to a lesser degree the mirror temperature.
- Impurity deposit caused by re-sputtering of material from the port walls or the first wall. The mechanism is sputtering of the port walls or first walls by charge-exchange neutrals and transfer of material onto the mirror surface. The processes are influenced by the composition, flow and spectrum of sputtered atoms (ions); production of chemical compounds, i.e. molecules (of CH and other types); the mirror parameters determining the film adhesion, i.e. initial purity of the surface, temperature.
- Dust deposition and its modification on the mirror surface.

As seen from this brief examination of the degradation processes of optical elements, it is essential to define more particularly the design solution for:

1. structural materials and coatings of the first wall elements, local limiters, divertor plates;
2. the technology of the proposed processes for vessel cleaning between plasma discharges.

Besides, it is necessary to calculate temperatures inside the vessel elements both during the plasma discharge and vessel warm-up. It is particularly important for the diagnostic windows, since in case of a colder window the impurities from the walls will go to its surface as a result of warm-up even with the windows protected by shutters.
The systems of passive (shutters, etc.) and active (remote cleaning) protection of the diagnostic windows and mirrors, as well the systems for monitoring of the optical elements require development.

II. ELECTROPHYSICAL SYSTEMS

1. Magnet System

1.1. Comments on the ITER-FEAT Conductor Outline Design

1.1.1. No «technical basis» is provided to justify the conductor design criteria but a declarative specification only.

The criteria (ΔT=1K, h=1000W/m^2 K for Nb_3Sn, and (ΔT=1.5 K, h=600W/m^2 K for NbTi) cannot be recognized to be «usual» as stated in the outline design.

The scaling and verifiable parameters ((ΔT, h), used in the criteria, belong to the DC approach to the CICC stability analysis basing on the ideas of Hoenig and Dresner on the stability margin and limiting current. This approach is well developed and experimentally verified but for the case of equal current distribution among strands and subcables in CICC, what is relevant for the small-scale CICC at sufficiently slow ramping of the applied field.

For the large CICC exposed to the fast time varying magnetic field the effect of uneven current distribution among strands and subcables comes to the foreground as the reason for the Ramp Rate Limitation observed, e.g., in the DPC experiments. This effect should be better evaluated and is in focus of the on-going studies.

At any rate the «technical basis» for the design criteria can be the data on the operation limits obtained experimentally (POLO, DPC, ENEA 12 T Coil, CSMC, etc.), and their scaling-up and comparison with the criteria proposed for ITER.

1.1.2. At the NbTi R&D meeting in Garching it was agreed that the solder surface coating and CuNi barriers mentioned in Table II.1.3-8 (NbTi strand parameters) are to be additionally investigated. Ni plating was agreed as the reference design option.

1.1.3. «…Various combinations of operating current & field…» (Chapter II Section 1 Page 31) cannot explain the variation in strand Cu: non Cu and using the segregated pure Cu cores for the reference PF conductor, but the limitation on the protection dump voltage for the coils storing different energy can. Without last argument the use of fore rather different PF conductor designs looks strange.

1.2. Comments on the ITER-FEAT TF Coil Outline Design

The TF coil design with radial plates and thin-wall round conductor seems to be preferable due to its following valuable advantages:

- lower mechanical stresses in turn insulation;
- possibility of monitoring over the state of the double-layer turn insulation with SS foil in between;
- higher attainable level of the accuracy of manufacturing, determined by the tolerances of machining of the grooves.

1.3. Comments on Mechanical Structure of the ITER-FEAT Magnet

The considerable positive feature of the Magnet Mechanical Structure is the use of the uniformed supporting system. The support based on sets of flexible plates is very effective to provide the required elastic constraints for normal and abnormal operation of the Magnet.

2. Vacuum Vessel

The currently considered design of the ITER FEAT vacuum vessel differs essentially from the VV FDR bearing in mind the integrated cooling collectors of blanket modules, nevertheless the technical realization of the project seems to be feasible. It is recommended to continue the feasibility study of the VV design paying attention to the following:

- lack of integrated calculations of the VV cooling system;

- lack of experimental data confirming the correlations applied for calculations of the VV cooling system;
- vacuum vessel reliability and its certification as the first safety barrier bearing in mind the complications of its construction and resulting abundant welded joints on the inner VV wall and the impossibility to control them during operation.

3. Vacuum Vessel Thermal Shield

1. In the Report the facility disassembly procedure has not been considered. This might result at a later time in the necessity to reconsider the design of the main VVTS components, in particular, joining units and support structure.
2. For ITER-scale facilities it is problematic to establish conditions under which there will be no considerable degradation of the silver coating of the thermal shield during facility cool-down and operation. The availability of a great deal of insulation materials, threaded joints, constructions not undergone vacuum-technological preparation (preliminary degassing), a low degree of cryostat evacuation (10^{-3}-10^{-4} Pa) might cause hydrocarbons and water vapors to deposit on the thermal shield in the form of solid condensate films, ice and frost.
3. To confirm the VVTS reliability and to substantiate experimentally the design solutions applied in VVTS it is essential:

3.1. to define gas release from non-metal materials used in the cryostat volume;
3.2. to perform mechanical tests of shielding sector joining units;
3.3. to mock up VVTS segments with the aim to simulate the assembly-disassembly of the joining units by remote maintenance means;
3.4. to test the Ag coating for radiation resistance and embrittlement under conditions close to the operating ones;
3.4. to perform thermocyclic strength tests of Ag coatings adhesion under conditions corresponding to the operating ones (T=80-423K, neutron fluence 1×10^{19} n/cm^2).

4. 3.5 to work out technologies for recovery of coatings on damaged VVTS surfaces;
5. to perform experiments on defining the possibility of hydrocarbon films, ice, frost forming on the VV thermal shield.

4. Power Supply System

The presented material reflects the main changes of the power supply system in comparison to the EDA phase are reflected. These changes correspond in our view to the present-day current scenario, as well as to the design of the PF and TF coils. The results of the development of the power supply system are reflected in the existing schemes not presented in this report, as well as in lowering of the requirements for the level of power consumed by the facility. The preliminary analysis allows the conclusion to be made on a considerable cost reduction of the power supply system equipment in comparison to the ITER FDR.

5. Divertor

Considerable progress has been attained in developing the divertor since the FDR phase. Despite a significant reduction of the space allocated below the X-point a number of modifications in the FEAT-divertor configuration provided the maximum heat flux on the targets at a reasonable (≤ 20 MW/m^2) level. Another significant achievement is a reduction in the nomenclature and number of PFC components. More attention was paid to the private zone so as to provide more flexible design in respect to such safety-related issues as the control of dust accumulation and reduction of tritium retention in carbon-based films.

The promise to achieve 50% cost reduction seems to be quite realistic. Besides, the divertor design group has managed to reduce considerably the cost of the divertor by changing the cooling system parameters and by eliminating such components as the baffle.

One of the tasks of the ITER-FEAT tokamak divertor is to reduce the amount of tritium accumulated in PF coils. It is assumed that under ITER-FEAT conditions hydrocarbons produced by chemical sputtering of graphite tiles of the vertical targets will deposit on the liner located in the lower part of the divertor and produce on its surface amorphous hydrocarbon layers. These layers are produced at surface temperatures of <300°C and result in tritium retention. Therefore, it is planned to sustain a high temperature (800-1000°C) of the liner surface so as to prevent the growth of such films.

But the carbon-based films deposited on the limiter surface might be the films with a diamond-like structure. Layers with such structure were found on the inner structural elements of T-10 and DIII-D tokamaks.

Essential features of diamond-like film (DF) deposition in the gas medium were studied in detail by E. Kondoh et al. (Journal of Appl. Physics 73 (1993) 3041, where the dependence of DF growth rate on the gas composition and pressure and on the surface temperature was investigated. The largest growth rate (GR) of diamond-like films were observed for the gas mixture H_2+HC_4 (2%, CH_4 partial pressure p~1 Pa) at a surface temperature of 900°C. In this case GR amounted to ~1 μm/h. The main biomolecular reaction resulting in DF growth proceeds by Eley-Rideal scheme:

$CH_4 + S \rightarrow S\text{-}CH_3 + H$,

$S\text{-}CH_3 + H \rightarrow S\text{-}C(d)$, where S - the surface, C(d) - diamond-like carbon.

But in the plasma medium, with excited and chemical active molecules, the DF growth rate might exceed essentially the growth rate in the gas medium. To understand the DF deposition processes it is necessary to develop the modules describing adequately different interrelated processes of gas mixture plasma activity, heat transfer and re-deposition, chemical and gas kinetics. The development of these models is extremely complicated, and simplified models will not give correct quantitative and, conceivably, qualitative results. Consequently, of first importance in studying the DF growth processes should be experimental investigations in order to clarify the possibilities for a considerable (as compared with the films produced at low surface temperatures) tritium retention in DF.

It is proposed to perform experiments on operating facilities with the aim to define:

- DF growth rate on the inner surfaces of divertors;
- GR dependence on surface temperature, neutral hydrogen pressure, relative concentrations of different hydrocarbons, surface materials;
- percentage of hydrogen isotopes in DF and its dependence on the above parameters.

6. PF Magnetic Configuration and Scenario

The «Technical Basis for the ITER-FEAT Outline Design» has been analyzed and the conclusion has been made that the engineering solutions taken in this document correspond to the concepts of the key RF HT specialists on plasma current, shape and position control. However, attention is to be drawn to some misprints made in the document, namely, «Chapter II Section 1 Page 2», the ninth line from bottom, it is mentioned that two central solenoid modules (CSL1 and CSL2) are connected in series. Actually it is CS modules CSL1 and CSU1 that are connected in series.

7. Tokamak Assembly, Assembly Technologies

The main stages of the technological processes of tokamak assembly seem to be technically feasible, rational and cheaper in comparison to the ITER FDR version. In the Final Report the definition of the magnetic axis and magnetic data of the tokamak should be presented in more detail, as during assembly of the facility in the cryostat the need might arise to adjust the position of the assembled vacuum vessel (prior to its welding) depending on the results of magnetic measurements. The VV supports on the TF coils are arranged in the area not easy to access. The access for fastening or adjustment is possible only through the vertical gap in the middle plane between the external intercoil structures. This gap should be covered with joining plates of the intercoil structure, which are welded after completion of VV assembly. This stage requires further study.

On the basis of the materials presented in the Report by the Joint Team the detailed specifications have been worked out for design of the tooling necessary for tokamak assembly.

8. Cryostat and Cryostat Thermal Shields

8.1. Cryostat Design.

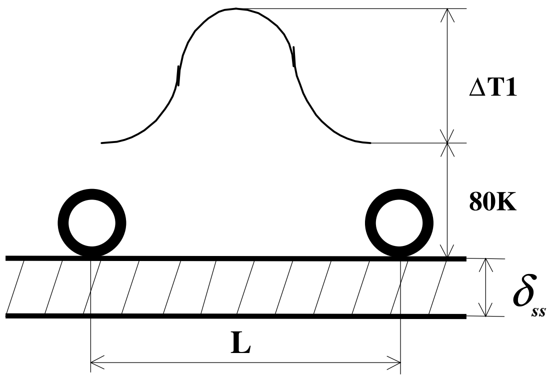
In case of helium cooling lines damage the pressure inside the cryostat will rise, that might cause its destruction. Hence, gaseous helium must be removed rapidly by means of the cryostat safety system. The information in FDR concerning the cryostat safety system is absent.

8.2. Design of Thermal Shields.

The reference design of the cryostat thermal shield (CTS) presented in ODR is based on the use of Ag-plated stainless steel panels (two stainless steel plates with a thin 0.5-mm-thick copper sheet between them in order to increase heat conduction and, hence, to increase the distance between the cooling tubes, as shown in Fig. 2). In this case the heat flux onto the electromagnetic system (~4 K) is sure to be lower than in the case shown in Fig. 1, but electromagnetic forces acting on the shielding panels will be higher by a factor of 5. The same effect, i.e. the reduction of the heat flux onto the electromagnetic system, might be achieved by placing an additional stainless steel screen, as shown in Fig. 3.

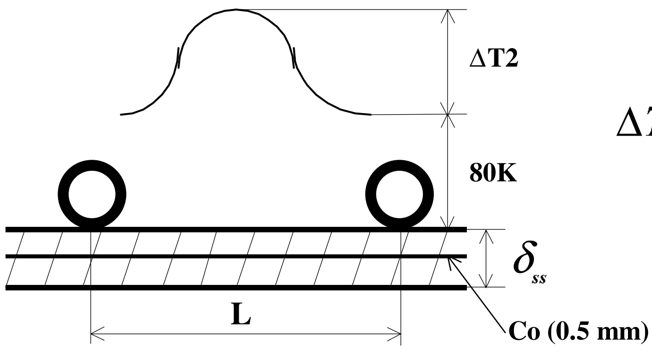
To reduce the cost of the external thermal shield (CTS) it might be reasonable to abandon silver-plating of the panels because of their oxidation, when in operation, resulting in an increase in their emissivity and to make only electropolishing of the surfaces.

Figure 1



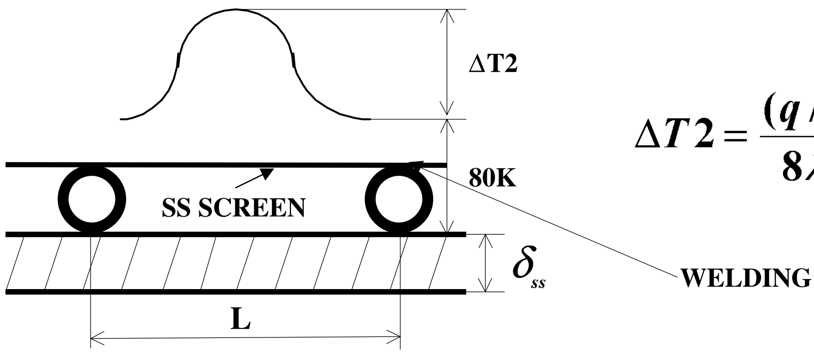
$$\Delta T1 = \frac{qL^2}{8\lambda_{ss}\delta_{ss}}$$

Figure 2



$$\Delta T2 = \frac{qL^2}{8(\lambda_{ss}\delta_{ss} + \lambda_{Co}\delta_{Co})}$$

Figure 3



$$\Delta T2 = \frac{(q/2)L^2}{8\lambda_{ss}\delta_{ss}}$$

9. ITER Fuel Cycle

The problems of fundamental importance for the Project have been elaborated in sufficient detail, they are based on the experimental data obtained on the JET facility and laboratory testing facilities at FZK and in Russia. No remarks on the submitted design materials.

10. Building and Service, Water Cooling System, Site Requirements & Assumptions, Site Layout Strategy, Tokamak Building, Plant Services

In the materials considered no fundamental differences from the previously elaborated documentation on FDR ITER have been found except for the decisions on the water cooling system of ITER-FEAT. The fundamental difference of the decisions is related to heat transfer from the vacuum vessel. The decision taken for the ITER-FEAT will cause warm-up of the vacuum vessel during accidents to temperatures close to the «bakeout» regime as a result of the loss of off-site power supply, seismic actions, etc. In the FDR project in emergency situations the ITER facility was cooled down and brought to the safe «cold» state. The temperature decrease in the FDR project was eliminated by the design solutions by the vacuum vessel cooling system.

The decision on VV cooling for the ITER-FEAT requires additional calculations for strength of VV construction. In so doing account should be taken of the combination of increased thermal loads arising at emergency cool-down and the loads arising during seismic actions of SL-2 level. This is associated with that in contrast to the «bakeout» regime the temperature at the VV input and output differs several fold, thermal stresses of the VV components are also different.

11. Safety

The safety objectives, principles and criteria are presented adequately, generic elements of the safety approach are also well presented. The contents of this section satisfy the objective of a top level document which deals with the requirements at the highest level.

There are also some comments on different tokamak systems.

1. Scenario of accident cryosorption pumping equipment thawing out in the case of a tightness loss in tubes for plasma facing component (primarily the divertor) cooling is not considered. The analysis of accident tritium release and cryogenic system integrity is absent.
2. The engineering analysis of characteristics of the reactor power control system, including the analysis of the reactor operation stability at partial power levels, is absent.
3. Terminological carelessness is apparent (e.g. the term «confinement» in the ODR is used now as a radioactivity confinement barrier, now as a safety function).
4. Tritium inventories and emissions stated in ITER-FEAT materials require additional studies and justification.

Vacuum Vessel

The presence of water for cooling (baking) both of the vacuum vessel and of the blanket modules that are placed inside the electromagnetic system, having the cryogenic temperatures, can lead to the damage of in-vessel elements in an emergency of water cooling system. To avoid negative sequences of such emergency the water

cooling system must include both the independent water removal system and the drying system.

Proper tokamak
During the operation the tokamak elements made of stainless steel (electromagnetic system, vacuum vessel, thermal shields, etc.) become magnetized due to thermal and electromagnetic loads. As T-7 and T-15 experience has shown, metallic dust formed during operation, deposit on the surfaces of elements, including electric insulating assemblies. The break-down of electric insulation can result in damage of the elements. Hence, it is necessary to control constantly the electrical shorts between the elements during the operation. The electrical shorts control system should be included into the ITER-FEAT design.

Analysis of possible emergencies
The emergency in one of the systems can result in damage of other systems. A separate chapter on the analysis of possible emergencies should be included into ODR.

III. BLANKET SYSTEM

General statement.

We support the principle solutions of mechanical attachment, primary modules and port-limiter. The comments on the mentioned components in order to improve the design, provide safety operation and reduce the cost are presented infra.

A. Updating of mechanical attachment unit design

1. It is necessary to introduce the locking of central bolt thread in order to provide the safety operation of flexible attachment unit. (This is verified experimentally, see RF Final report on Task T216+2R2).
2. To provide the fabrication errors compensation for vacuum vessel and modules (see Figs. 1-3). This compensation is necessary to eliminate the over-loading of the attachment elements under electromagnetic loads.

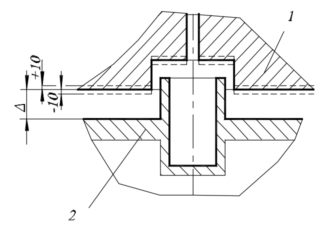


Fig.1. Axial errors ±10 mm. 1.- Module; 2.- Vacuum vessel

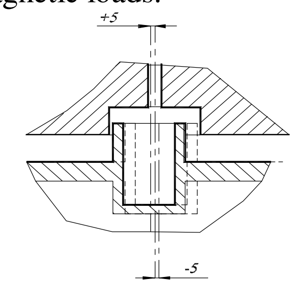


Fig.2. Lateral misalignments of axes Δ=±5 mm.

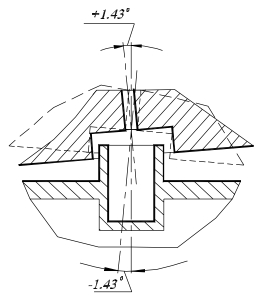


Fig. 3 Angular errors γ=±25 mrad (1.43^0)

3. We propose to introduce the place for wrench in the flexible element design for the mounting/dismounting carrying out.
4. It is necessary to provide the automatic contacting of matting surfaces without machining of the attachment unit details (see Fig. 4) for the effective cooling of the supporting flexible element.

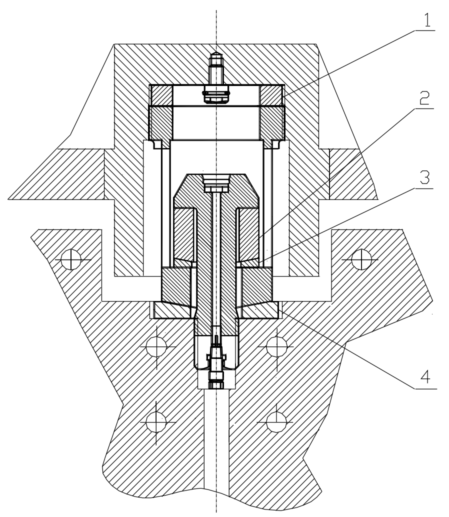


Fig. 4. Flexible self-positioning mechanical support.
1.- Compensation bush;
2.- Internal spherical washer;
3.- Spherical washer;
4.- External spherical washer.

B. Primary module.

1. The A and B options have a large volume of machining in the shielding part. It is reasonable to develop the option of cast shielding part with the built-in coolant pipelines and headers for the module cost reducing as a whole.
2. It is necessary to write precisely in the Blanket Technical Requirements for what purposes the first wall (FW) should be replaceable (for the repair, storage of relatively not large designs or another purposes).

It is reasonable to consider two cases for the FW repair.

<u>First case</u> – the repair changing of the FW from the plasma chamber side without the module dismounting.

<u>Second case</u> – the FW changing in the hot cell after the module dismounting.

The robot equipment is required in both cases. Both cases are identical from the point of view of the FW or module changing as a whole from the vacuum vessel.

It is necessary to note that the reasonability of the radioactive design repair or the old FW changing on the new one is in doubt from the point of view of the operation reliability of the repaired module. It is more reliable to change the module on the new one. In this case the replaceable FW design can be reasonable from the point of view of reducing the materials volume for storage.

The first wall option to be changed from the plasma chamber side has only one advantage in comparison with the FW attached to the shielding part on the module back. This advantage is the possibility to carry out the module accidental change with relatively minimum difficulties. In this case it is possible to remove the FW in vacuum chamber by the bolts drilling-off and to open (FW is removed) by any method the defected places through the shielding part and disconnect the module.

So, the design of replaceable FW should allow the changeability in the vacuum vessel without the module dismounting as a whole. The option A satisfies better this requirement but it requires the defined development taking into account the simplification of the connection unit on the coolant between the FW and shielding part (for example on the base of SMA using).

C. Port-limiter.

In order to reduce the port-limiter cost it is reasonable to develop the design option with the brazed bimetal panels as a typical product for this technology (in particular two bimetal panels are fabricated in RF by brazing in gas static device with low parameters).

D. General remarks.

It is reasonable to introduce the following explanations at the end of each section:

- the adopted changes relatively to the previous stage (ITER FDR)
- the verification of adopted changes (logic, advantages, quantity/quality comparisons) and achieved results.

O. G. Filatov,
RF Home Team Leader,

ITER MEETING
Moscow

28 June 2000

ITER - Progress in Design and Validating R&D
note by the Director

ITER - Progress in Design and Validating R&D

note by the Director

1 Introduction

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"*. Recognising the importance to optimise a single agreed design, the Meeting *"asked the Director and JCT to interact with the Parties during the course of their domestic assessments.....with a view to optimising the design for approval following TAC review, at the coming ITER Council meeting."*

This note summarises, for ITER Council, material presented in detail to TAC, in the report "Progress in Resolving Open Design Issues from the ODR", including:

- further investigation of Physics issues raised in the TAC report or in the course of the Parties' domestic assessments, and
- features of the ITER design which reflect a resolution of choice of options.

In addition, the note summarises and illustrates recent progress in the programme of validating technology R&D, as presented in the report to TAC, "ITER Technology R&D Progress Report".

2 ITER Physics

2.1 Highlights of recent progress.

The TAC report (Dec 99) and subsequent interactions with the Home Teams in the course of their domestic assessments indicated a number of questions and recommendations of areas for further physics investigation. The design progress report to TAC elaborates work undertaken by JCT and Home Team members and other experts of the Parties, under the following headings:

- sensitivity analyses of inductive (Q=10), hybrid (Q=5) and non-inductive (Q=5) scenarios to a range of assumptions or modelling;
- possible high Q (~50) and transient ignition with a short pulse heating;
- analysis of requirements for steady state operation;
- pedestal database used for ITER performance projections;
- divertor physics and, in particular, design robustness against ELM's;
- possible suppression of neoclassical tearing modes by ECCD.

The work reported has generally taken the form of further detailed modeling studies using the latest experimental results, and further developments and analysis of ITER databases. In some areas, work has included new approaches and new insights into important physics issues. Highlights include:

- recent recognition of the relation between the H-mode pedestal temperature and core energy confinement which has provided insights into the importance of high triangularity in achieving good confinement at high density; a model has been

developed to quantify pedestal energy content scaled from present experiments; nevertheless, a better understanding of ELM phenomena is needed;

- the sensitivity of plasma performance against density profiles (for the same average density) confirms the need for the development and appropriate modelling for pellet injection from the high field side.
- the novel approach of a dimensional extrapolation technique based on a system code applied to the ITER H-mode Energy Confinement Database tries to overcome the problem of hidden interaction among certain parameters, particularly when close to their limits.
- recent work in Divertor modeling has included validation of the B2-Eirene code against ASDEX-U and JET experimental results and an analysis of the effects of divertor geometry which supports the adoption of a V-shaped target configuration.
- modelling using modulated ECCD to stabilise neo-classical tearing modes suggests that NTM detection in the early stage of evolution allows the power requirements on the EC power to be eased; further experimental work is now required to verify the models used.

2.2 Continuing Physics R&D

The degree and quality of recent progress in ITER physics, bears witness to the effectiveness of the framework of voluntary collaboration established for ITER Physics activities, even if the need for more modelling and experimental work on key issues is recognized to strengthen further the basis for extrapolation.

To a large degree, physics issues raised by TAC or during the course of the Parties' domestic assessments have already been recognised through the ITER Physics Committee system and are included in the list of Urgent and High Priority Physics Research Areas that the Physics Committee has recommended to the Parties for their experimental programming. (see Table 1 below).

It is most important to continue a co-ordination of the Parties' physics efforts in order to maintain the focus on deepening and strengthening the physics basis for ITER in preparation for possible operation and exploitation of ITER.

Table 1 Urgent (Bold) and High Priority Physics Research Areas

Urgent: Essential to confirm the feasibility of the inductive Q=10 scenario for the draft Final Design Report of ITER-FEAT at the end of 2000

High: Information valuable for design of ITER-FEAT, especially for establishing a scenario for steady-state operation of ITER-FEAT

Research Areas	Issues
Finite-β effects	**Tolerable ELMs (dW/W<2%) with good confinement alternate to type-I ELMs** (e.g. type II, Type III+core confinement) **Stabilisation of neoclassical islands** and recovery of ß
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 Carbone Chemical sputtering and deuterium retention/cleaning methods
Diagnostics	Determine requirements for q(r) and assess possible methods that can be applied to ITER Determine life-time of plasma facing mirrors and optical elements(incl. Those in divertor) Reassessment of measurementrequirements in divertor region + recommendation of diagnostic techniques
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.
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).
H-mode power threshold	H-mode accessibility in ITER-FEAT , Data scatter
Density limit physics	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

3 Engineering Design - assessments and choices.

The report “Technical Basis for the Outline Design of ITER-FEAT Report” left a number of engineering design choices open for further analysis and evaluation. TAC comments and points raised in the course of the Parties’ domestic assessments have also been examined by the JCT and Home Teams.

The report presented to TAC sought to present the rationale for the design choices by making the balanced judgement between a wide spectrum of considerations, like operation performance, engineering margins, manufacturing costs, etc. Thus previous design options have converged to a reference design for all the major system, in particular:

1. for the **magnet system** in relation to:
   - support of TF coils loads, with additional pre-compression rings to improve the capacity to resist out-of-plane loads
   - conductor and winding design issues, maintaining the radial plate configuration for the TF coils and keeping two options for R&D related to material for the CS conductor jacket.
   - limits to elongation/triangularity

2. a separated manifolding for the **blanket coolant system,** limiting its interaction with the VV

3. for the **vacuum vessel** design
   - double wall vessel, reinforced by the cylindrical housings for the blanket modules attachments, in addtion to the poloidal ribs
   - analysis of VV load conditions and related structural assessments

4. **materials** choices for the **divertor** targets

5. **building/services** and **hot cell** design

4 ITER Safety

The extensive analysis base available in the ITER non-site-specific safety report (NSSR) is being used to improve the implementation of safety in the design. Specifically the confinement approach is being reviewed and refined to obtain a balance of safety requirements imposed on the systems with confinement functions, in terms of required levels of assurance, reliability etc.

Working from the NSSR, the safety design focuses on confinement as the key safety function whilst other safety-related considerations (such as heat removal, control of chemical or magnetic energy, control of coolant enthalpy etc) are analysed from the perspective of protecting confinement barriers. A "lines-of-defence" methodology is being used to provide a systematic way to obtain the required level of safety while balancing the requirements imposed on systems and components.

Appropriate design measures to control inventories and to reduce operational losses, are being pursued so that project release guidelines can be met without needing a tall stack. (A controlled, monitored release point will still be needed, the height of which would be set as necessary to satisfy the regulatory requirements of the host.)

The target for the current phase of ITER is to provide the Generic Site Safety Report (GSSR), which will document the safety assessment of the new design, as part of the final output of the ITER EDA. The GSSR is also intended to provide a basis from which to start preparing regulatory submissions for siting, subject to the further site-specific design adaptations and host country specific safety assessments that will be needed to obtain regulatory approval for construction.

5 Status of ITER Technology R&D

The present status of technology R&D in the main areas of joint work during the EDA was presented to TAC in a specific report; an overview is presented below. Selected illustrations of the projects are appended to this note.

The major technical challenges in ITER are:

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

The new design of ITER relies mostly on technical solutions, which have been qualified in the R&D programme launched previously for the 1998 ITER design. Major developments and fabrication have been completed and tests have significantly progressed. The technical output from the R&D validates the technologies and confirms the manufacturing techniques and quality assurance incorporated in the ITER design, and supports the manufacturing cost estimates for important key cost drivers.

The testing of models is continuing to demonstrate their performance margin and/or to optimize their operational use.

The realisation of major joint technology projects offers insights useful for a possible future collaborative construction activity. Valuable and relevant experience has already been gained in the management of industrial scale, cross-party ventures. The successful progress of these projects increases confidence in the possibility of jointly constructing ITER in an international project framework.

Significant efforts and resources (about half of the total) have been devoted to the Seven Large R&D Projects which cover all the major key components of the basic machine of ITER and their maintenance tools. All participants are to be commended for their dedication and co-operation. **The success of both the process and the outcomes merit recognition and wide dissemination throughout the Parties as exemplars of what focused international joint activities in science and technology can achieve**.

Central Solenoid (CS) and Toroidal Field (TF) Model Coils Projects (Ll and L2))

These two projects are working towards developing the superconducting magnet technology to a level that will allow the various ITER magnets to be built and to operate with confidence. The Model Coil Projects are intended to drive the development of the ITER full-scale conductor, including the manufacturing of strand, cable, conduit and terminations, and the conductor R&D in relation to AC losses, stability and joint performance. These Model Coil Projects also integrate the

supporting R&D programmes on coil manufacturing technologies, including electrical insulation, winding processes (wind, react, and transfer) and quality assurance. 29 t of Nb_3Sn strand, from seven different suppliers throughout the four Parties, has been produced and qualified. This reliable production expanded and demonstrated the industrial manufacturing capability for the production of the 480 t of high performance Nb_3Sn strand as now required for ITER.

For the CS model coil, the cabling and jacketing technologies and winding techniques have been established and all these activities have been completed. The next critical step, the heat treatment to react the superconducting alloy without degrading the mechanical properties of the Incoloy jacket, has been successfully completed. By using approximately 25 t of the strand, the inner module (US), the outer module (JA), and the insert coil (JA) were fabricated and assembled. In April 2000, the maximum field of 13 T with a cable current of 46 kA and magnetic stored energy of 640 MJ has been successfully achieved in the ITER dedicated test facility at JAERI. Pulse operation has been experienced under conditions more severe than during ITER-FEAT operation. The insert coil has been also tested at 13 T and more tests are ongoing. The size of the CS model coil (3.6 m in diameter and 2 m in height) is almost the same as a module (4 m in diameter and 2 m in height) of the Central Solenoid in the new design and the maximum field is also the same.

For the TF model coil, forging and machining of the radial plates have been completed. Cabling, jacketing, winding, reaction treatment and transfer of the reacted conductor in the radial plates have also been successfully demonstrated. The coil is fully assembled except for the final impregnation of the winding pack in the coil case, which is underway. All the work has been performed in EU. The coil is expected to be delivered to FZK, Karlsruhe in the summer of 2000. The Model Coil uses a cable similar to the full-size TF coil cable and the cross section of the TF model coil is smaller but comparable in size to that of the actual TF coil. The model coil will be tested first on its own and later in conjunction with the LCT coil in the TOSKA facility. With the LCT coil, a field of 9.7 T at 80 kA will be achieved. By comparison, the peak field and the operating current are 11.8 T and 68 kA in the new design of ITER.

In addition, a TF insert coil with a single layer will be tested inside the bore of the CS model coil test facility at JAERI at a field up to 13T. This insert coil will be completed in the RF this year.

A 1 km jacketing test, which exceeds the design requirements, has been separately demonstrated in the RF.

For the development of the manufacture of the TF coil case, large forged and cast pieces (about 30 t and 20 t respectively) have been produced in the EU. Investigation of the properties of the forging has revealed values exceeding the requirements of 1000 MPa yield stress and 200 $MPam^{1/2}$ fracture toughness, with low fatigue crack growth rates. The casting also shows properties adequate for the low stress regions of the case (yield stress about 750 MPa). Welding trials have demonstrated successful welding of the cast to forged sections.

For the case assembly welds, electron beam (EB) welding is planned for the first pass followed by submerged arc welding for the remainder, to minimise distortion. The welding processes have been qualified, and preparations for the final welding demonstration are underway.

Vacuum Vessel Sector Project (L3)

In the Vacuum Vessel Sector Project, the main objectives are to produce a full-scale sector of the ITER vacuum vessel for the 1998 FDR design including the equatorial port, to establish the tolerances, and to undertake initial testing of mechanical and hydraulic performance. The key technologies have been established and, in relation to manufacturing techniques, two full-scale vacuum vessel segments (half-sectors) have been completed in JA industry, using a range of welding techniques, within the required tolerances. At JAERI, they were welded to each other and the equatorial port fabricated in the RF was attached to simulate the field joint planned to be done at the ITER site during assembly of the machine. Remotised welding and cutting systems prepared by the US were also tested and applied; remotised tools for non-destructive testing will be soon experimented.

Blanket Module Project (L4)

The Blanket Module Project aims at producing and testing full-scale modules of the first wall and shield elements and full-scale, partial prototypes of mechanical and hydraulic attachments. The key technology has been successfully developed, tested and qualified.

- A range of crucial material joints such as Be-Cu and Cu-stainless steel have been successfully made by using hot isostatic pressing (HIP) and other advanced techniques inside each of the four Parties.

- A full-scale model module, without the attachments, has been completed in JA by using mainly forging and drilling.for the shield block manufacturing.

- The module attachments have been developed and tested in the RF.

- A full-scale module with attachments is under fabrication in the EU. The full-size shield block has been completed by using powder HIP. After the first wall is attached to this block, the module will be tested to confirm that it meets the requirements for anticipated loads, electrical insulation and remote handling together with the necessary accuracy of positioning.

- A port limiter mock-up with Be tiles has been fabricated by using a fast breezing technique in the RF.

- In parallel with these fabrications, heat cycle and irradiation tests have been performed for the base materials and the bonded structures and have demonstrated that the performance is well within the acceptable level.

Divertor Cassette Project (L5)

The Divertor Cassette Project aims at demonstrating that a divertor can be built within tolerances and withstand the high thermal and mechanical loads.

- A full-scale prototype of a half cassette based on the 1998 ITER design has been built by the four Parties. Plasma facing components shipped from JA and the RF were installed in the divertor cassette body fabricated in the US, and hydraulic flux and mechanical tests were performed at Sandia National Laboratory.

- Various components for high heat flux were fabricated and tested in the four Parties. High heat cycle tests show that CfC monoblock survives 20 MW/m^2 x 2000 cycles (EU) and W armours survive 15 MW/m^2 x 1000 s (EU / RF). A large divertor target mock-up with CfC attached to DSCu through OFCu has been successfully tested with 20 MW/m^2 x 1000 cycles from a large hydrogen ion beam with a diameter of 40 cm, a performance consistent with ITER operational needs.

- Irradiation tests have been also performed. For example, CfC brazed on Cu survived 20 MW/m^2 x 1000 cycles after 0.3 dpa irradiation at 320°C. Tests with pulse heat deposition simulating the thermal load due to disruptions have demonstrated erosion but no disruptive failure of CfC armours even with 0.4 dpa irradiation. (The average neutron fluence of 0.3 MWa/m^2 at the first wall gives 0.38 – 0.59 dpa on the CfC divertor target.)

Blanket Remote Handling Project (L6) and Divertor Remote Handling Project (L7)

The last two of the Large Projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions on reasonable timescales. These technologies should provide the flexibility needed for ITER to pursue its scientific and technical goals whilst satisfying stringent safety and environmental requirements. In this area, full-scale tools and facilities have been developed. Their testing will be extended over a long period of time including the ITER operation phase. This is necessary not only for developing the right procedures but also for optimizing their use in detail and minimizing the intervention time. Rescue procedures and equipment to recover failed equipment are also being developed. The facilities will also allow training of operators.

The Blanket Module Remote Handling Project aims at demonstrating that the ITER blanket modules can be replaced remotely. This involves proof-of-principle and related tests of remote handling transport scenarios, including opening and closing of the vacuum vessel, and of the use of a transport vehicle on a monorail inside the vacuum vessel for the installation and removal of blanket modules. At first, the procedures were demonstrated at about one fourth scale. Work is now in progress on a full-scale demonstration. The fabrication of the full-scale equipment and tools, such as a 180° rail, a vehicle with telescopic type manipulator, and a welding / cutting / inspection tool, have been completed in JA. The simulation of installation and

removal of a simplified, dummy shield blanket module of 4 t has been successfully performed by using a teach and play-back procedure. The dummy module was installed with only 0.25 mm of clearance between dummy keys and keyways using the intrinsic compliance of the manipulator. Integrated tests in a blanket test platform which simulates the full-scale structure of a 180° ITER in-vessel region are providing comprehensive validation of the remote handling system so as to allow completion of the detailed design of the components and the remote handling equipment. The real in-vessel operation will be done in a gamma field of 10^4 Gy / h. Key elements such as motors, position sensors, wires/cables, glass lenses, electrical insulators, periscopes and strain gauges have shown to survive tests at 10^6 –10^7 Gy.

In the Divertor Remote Handling Project, the main objective is to demonstrate that the ITER divertor cassette can be installed and removed remotely from the vacuum vessel and remotely refurbished in a hot cell. This involves the design and manufacture of full-scale prototype remote handling equipment and tools, and their testing in a divertor test platform (to simulate a portion of the divertor area of the tokamak) and a divertor refurbishment platform (to simulate the refurbishment facility). Construction of the necessary equipment and facilities has been completed and successful tests carried out with the remote handling transporters and tools procured in the EU, including a central cassette carrier from JA and a transporter from Canada. The system is based on a toroidal transporter that moves on the same rails to which the individual divertor cassettes are attached. This can move a cassette in front of a remote handling port from where the cassette is extracted with a radial transporter that is deployed from a transfer cask. Redesign of equipment is underway commensurate with design changes to the divertor cassette that were necessary for the reduced size of the new ITER.

Other R&D

In addition to the Seven Large R&D Projects, development of key components for fuelling, pumping, tritium processing, heating / current drive, power supply, diagnostics, as well as safety-related R&D have significantly progressed.

- For example, a tritium pellet injector has been tested with a total amount of 36 g T_2 and 28 g DT and ejection of a large pellet (10 mm) from a 80 cm radius curved guide tube has been successfully achieved with 285 m / s in the US. Further tritium pellet injector development is being continued in the RF.

- A full-scale cryogenic pump for DT, He and impurities has been completed and is under testing in the EU.

- A tritium processing system with 180 g T was successfully operated for 12 weeks in the US.

- Key components for the ICRF antenna and the transmission line have been developed and tested at a higher voltage than the expected operational voltage.

- Gyrotrons at 170 GHz have been developed and successfully operated at 0.5 MW x 8 s in JA and at 1 MW x 1 s in the RF. More developments are needed to reach

the objective of reliable CW operation at 1 MW per tube. Diamond windows capable of transmitting multi-megawatts at 170 GHz have been developed in the EU.

- Almost full-size negative ion sources and high voltage technology (1 MeV) have been developed for NB injection in JA and the EU. Final objectives of CW operation under specified performance have been only partially achieved, and more experimental improvements and tests are needed.

- Mechanical bypass switches and fast-make switches have been developed and successfully tested at 66 kA and explosively actuated circuit breakers at 66 kA and 170 kA at Efremov Institute (although 170 kA is no longer required for ITER).

- Irradiation tests of key components of diagnostics have provided values required for shielding of components and planned replacement. The unexpected radiation-induced emf (RIEMF) effect especially on measurement by magnetic probes is an important issue and is under study. Lifetimes of mirrors set near the plasma will be limited by deposition / sputtering and are under investigation.

- Safety-related R&D, such as characterisation of dust in tokamaks, tritium co-deposited with carbon, and experiments on steam-material reactions, has provided inputs for the key phenomena and data for ITER safety assessments. The current R&D emphasis is now on verification and validation of data, models and computer codes. Measurement and removal of radioactive and tritiated dust in the vacuum vessel are under investigation.

- Neutron shielding tests by using a 14 MeV neutron source in JA and the EU demonstrates that the accuracy of shielding calculation is within 10 %.

ITER
Engineering Design Activities

Illustrations
from the
Seven Large Fusion Technology R&D Projects

June 2000

L1—Central Soleonoid Model Coil

L1— The Central Solenoid outer module being placed outside the inner module which has already been installed in the vacuum chamber.

L1— CS model coil and CS insert installed in the vacuum chamber at the test facility in JAERI Naka. The preload structure (upper beams and tension rods), helium pipes and top of the coils are shown. Behind is the vacuum chamber lid.

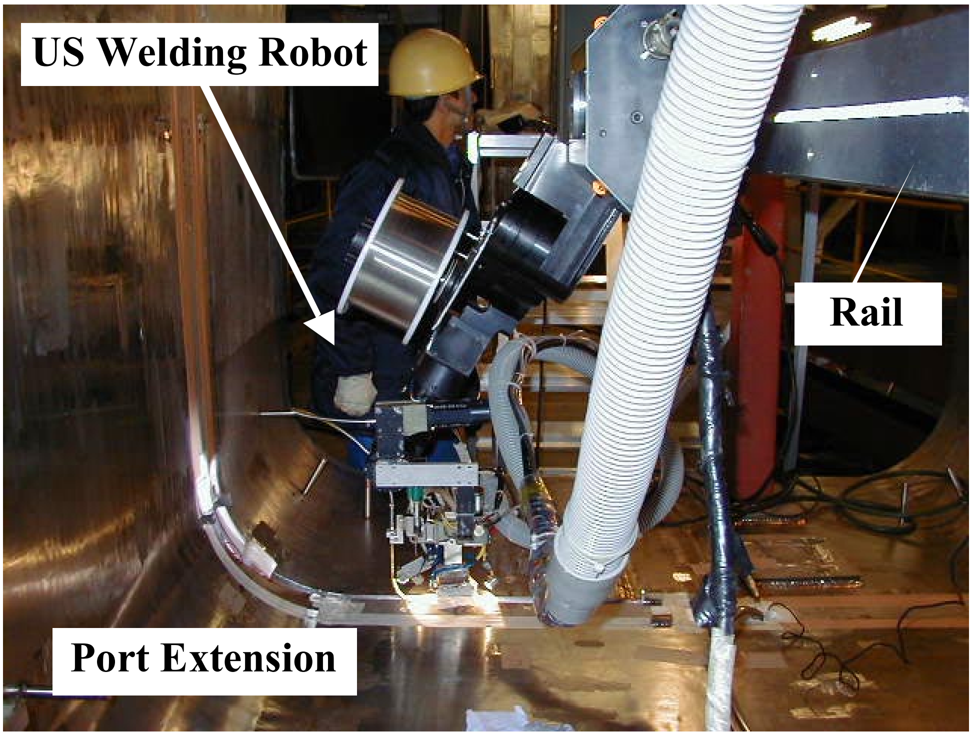
L2 — Toroidal Field Model Coil

L2— Winding TF model coil Conductor into Mould

L2 — Forging of Inner Leg Curved Section of TF coil case as Hollow Tube

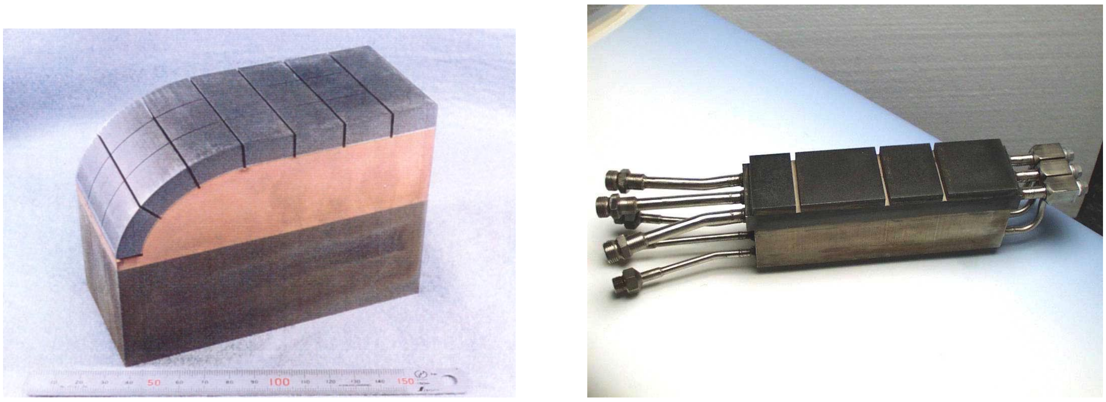
L3 — Vacuum Vessel Sector Model

L3 — Vacuum Vessel Port Extension, delivered and installed on the VV Full-scale Sector Model



L3 — Inner Shell welding demonstration on VV Full-scale Sector Model

L4 — Blanket Module

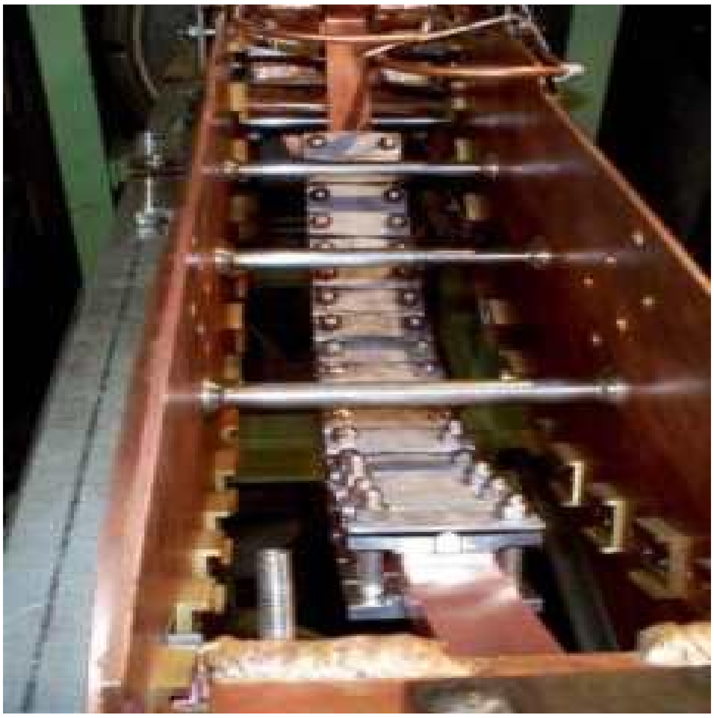


L-4 — Blanket module mock-ups

L4 — Flexible Cartridge Prototypes

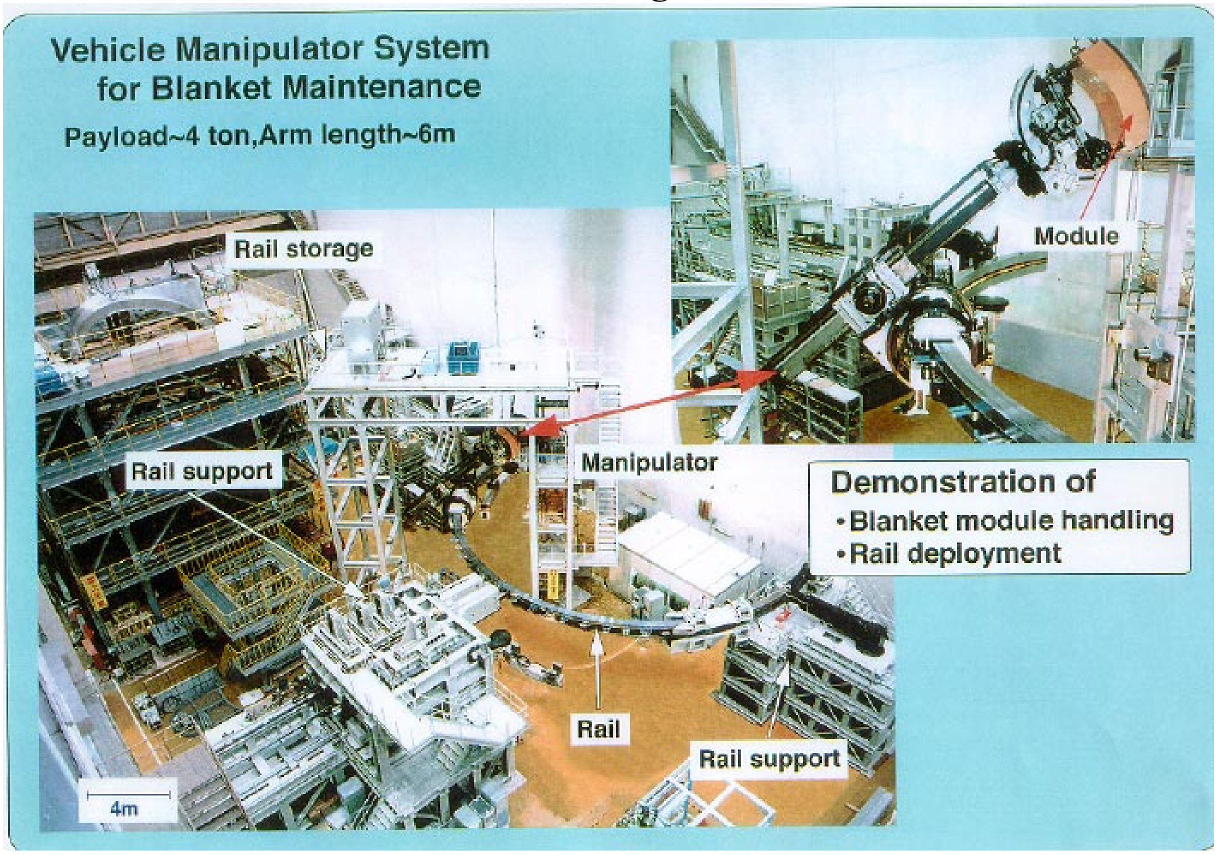
L5 — Divertor Cassette

L5 — Outer Channel of Cassette body

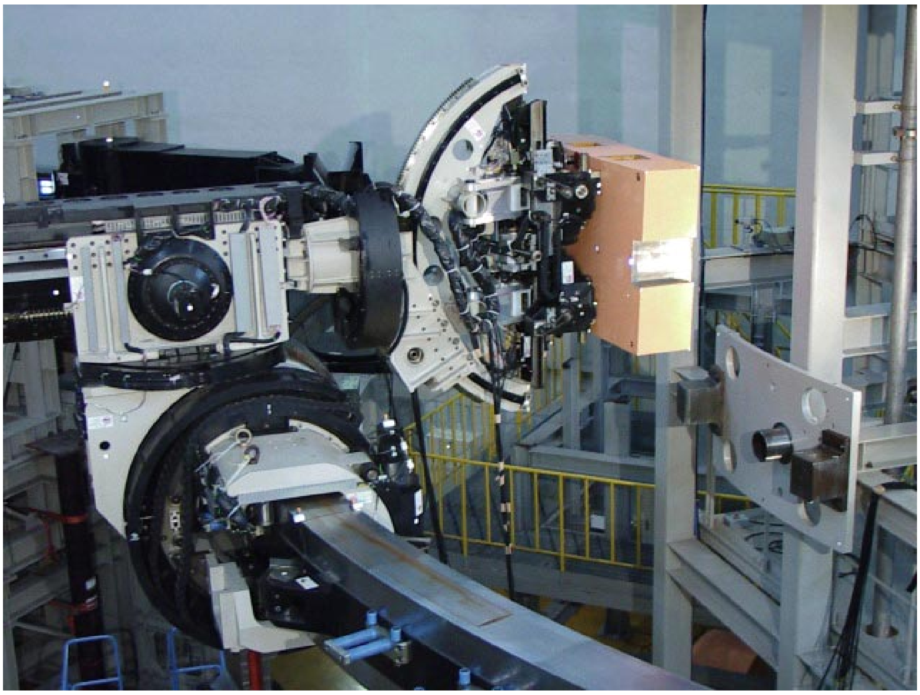


L5 — Mock-up in the “fast” brazing ohmic heating rig for brazing W and Beryllium armour to CuCrZr alloy

L6 — Blanket Remote Handling and Maintenance

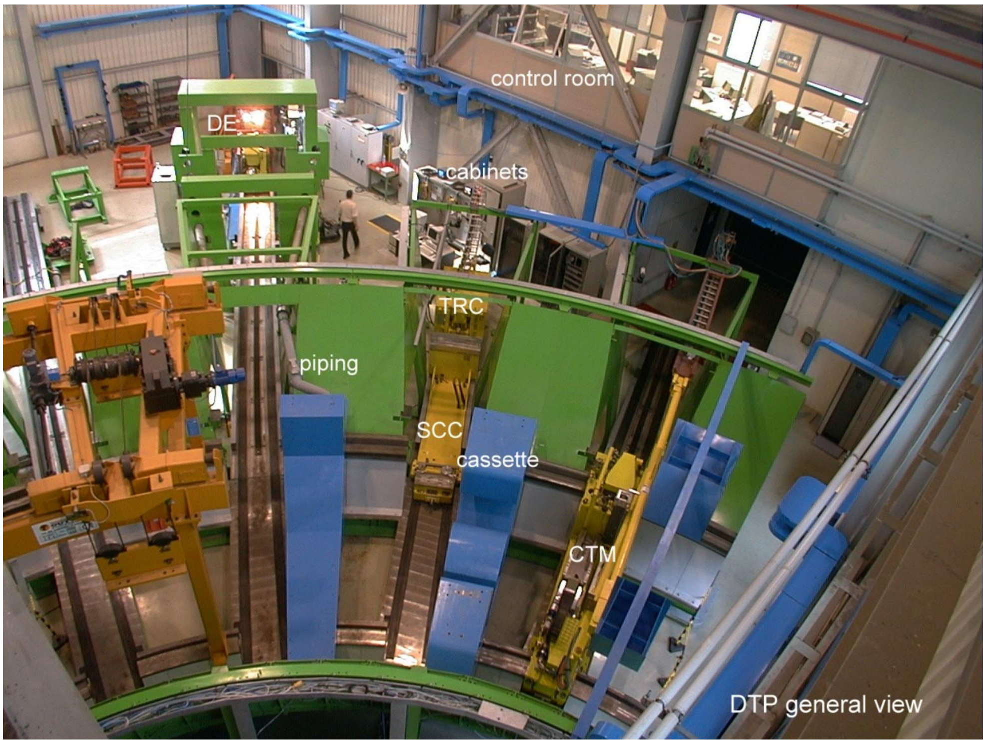


L6 — Blanket Remote Handling Test Platform and Full-scale Manipulator

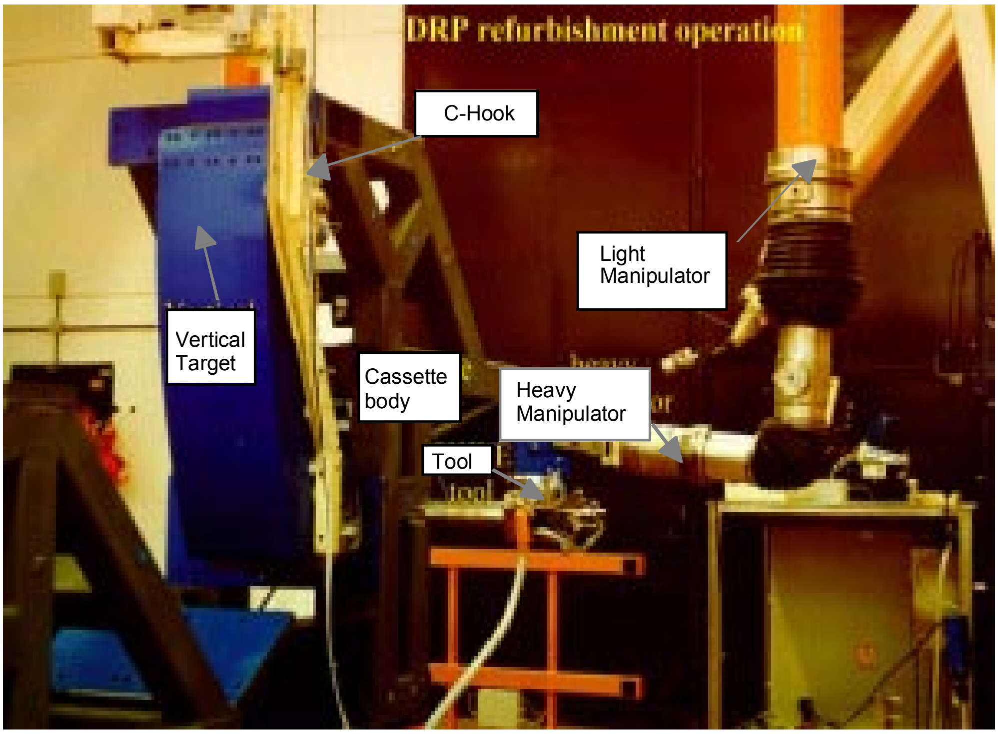


L6 — Vehicle Manipulator under operation

L7 — Divertor Remote Handling and Maintenance



L7 — Top view of Divertor Test Platform



L7 — Heavy and Light Manipulators at the Divertor Refurbishment Platform

RECORD OF THE ITER MEETING

Moscow, 29-30 June 2000

……….

3.1 Having noted the Director's presentation on Progress in Design and validating R&D for ITER (**Attachment 5**) and the presentation from the TAC Chairman, the Meeting:

i) endorsed the assessments and recommendations of the TAC Report (**Attachment 6**); and

ii) approved the ODR as updated following domestic assessments and as outlined to TAC, as the basis for preparation of the Final Design Report.

3.2 The Meeting congratulated the Director, JCT and Home Teams for their successful joint work to establish a single mature design for ITER consistent with its revised objectives.

……….

01-00322