The European Fusion Research Programme: Positioning, Strategic outlook and need for infrastructure towards DEMO

Part II. Facilities

Input to the Facilities Review Panel
prepared by the EFDA Leaders, the EFDA Associates and F4E

6 May 2008

EFDA
EUROPEAN FUSION DEVELOPMENT AGREEMENT

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Executive summary

In a first document entitled "Positioning and Strategic Outlook" a programmatic analysis has been carried out in order to assess which are the main lines of development of the fusion programme needed to support ITER and prepare DEMO, with the objective of being able to start the construction of DEMO in about 20 years. Gaps in the long term programme and the main technical risks for the development of fusion power have also been highlighted in this analysis (plasma control issues, steady state operation, materials, tritium breeding/fuel cycle and overall reliability/availability). In order to structure the programme and address these risks, seven R&D Missions have been proposed with a set of associated objectives and milestones which constitute a consistent roadmap towards DEMO.

On the basis of these R&D missions and associated milestones, the needs in terms of technology facilities and magnetic confinement devices have been analysed. The goal is to support an efficient construction and exploitation of ITER and fulfil the above mentioned R&D missions. The analysis has been conducted in successive steps: (1) definition of a Core Programme, (2) analysis of how facilities can address this programme, and (3) proposal including an estimate of resources needed.

A Core Programme, in which the construction and exploitation of ITER constitute the main pillar, has been defined that includes:

- *among all the components of the fusion programme, those that address scientific and technical issues that must be successfully resolved to fulfil the requirements of the roadmap towards DEMO, and*
- *additional elements necessary to address gaps and risks identified in the "Positioning and Strategic Outlook" document.*

Having analysed how facilities can address the objectives of this Core Programme, the following proposal is made:

I- Programme mainly focused towards ITER:

The Core Programme requires, in addition to ITER construction and the related Broader Approach Projects (JT60-SA and IFERC super computer):

- *upgrades and additional technology facilities to support the European contribution to the ITER construction;*
- *in parallel to ITER construction: a strong Tokamak programme to support ITER and prepare its exploitation, comprising:*
  - *an extension of JET,*
  - *the continuation of key European tokamaks,*
  - *upgrades on existing tokamaks to address high priority risks (Steady State, Reliable Operation) by advancing tokamak physics,*
  - *the further development of collaborations, in particular with new superconducting tokamaks outside EU;*
- *in preparation of a satellite tokamak programme to operate in parallel to ITER exploitation and complementing JT-60SA:*
  - *preparing the future operation of at least one European tokamak in the 1-2 MA class (upgraded existing device(s)),*
  - *in order to reduce risks and fill programmatic gaps, launching European studies of a high current tokamak (in the range 5MA) and considering the FAST proposal as a possible option for such a device,*
  - *considering further joint use with Japan of JT60-SA beyond the end of Broader Approach;*

- *in the frame of a long term High Performance Computing (HPC) policy[1] for theory and modelling of fusion plasmas and materials:*
  - in addition to the IFERC computer provided under Broader Approach, and in preparation for its use, approving as soon as possible the proposed European HPC.

II- Programme mainly focused towards DEMO:

In addition to the successful exploitation of ITER and accompanying devices, the on-going DEMO technology programme (materials and blanket development and R&D for helium-cooled divertors) and the IFMIF EVEDA (Broader Approach), the preparation of DEMO requires:

- *considering an additional R&D programme to reduce risks in the IFMIF-EVEDA and thus prepare for a successful fusion materials development programme;*
- *starting without delay the preparation of the decision to site and build IFMIF;*
- *launching European DEMO Conceptual Studies with supporting R&D;*
- *increasing the funding for facilities needed for DEMO oriented technology R&D, in particular heating and current drive systems, in-vessel components, reliability and maintainability (remote handling and design of components), thereby addressing identified technical risks;*
- *preparing for a DEMO engineering Design Study that should follow the Conceptual Studies after about 8 years[2];*
- *increasing the resources for the Fusion Materials Science and Technology Programme, thereby addressing one of the main risks of the fusion programme (Materials);*
- *preparing for a possible Component Test Facility (CTF), and thus addressing identified risks (materials; in-vessel components for tritium-breeding; reliability/availability), with*
  - CTF feasibility studies in the frame of the DEMO conceptual design studies,
  - a CTF physics and technology programme (the upgrade of MAST would address the physics issues).

Moreover, the stellarator programme, as part of the concept improvement programme, must be vigorously pursued. This requires

- *completing the construction of, and exploiting, WENDELSTEIN 7-X;*
- *launching at a later stage, Stellarator Power Plant conceptual studies in the frame of the DEMO conceptual design studies.*

A resource loaded planning showing the cost and time scales for this set of proposals is provided on the next page.

In addition to this Core programme, it is highly desirable to keep a range of facilities i) to address critical issues in the base programme, ii) to strengthen the scientific and technical understanding important for the progressive development toward a viable and economic fusion reactor, iii) to generate and gather new ideas outside the mainstream thinking, which stimulate and foster the essential elements of innovation and creativity.

The programme outlined in this paper would provide a sound basis for the timely development of fusion power, with Europe in a leading position.

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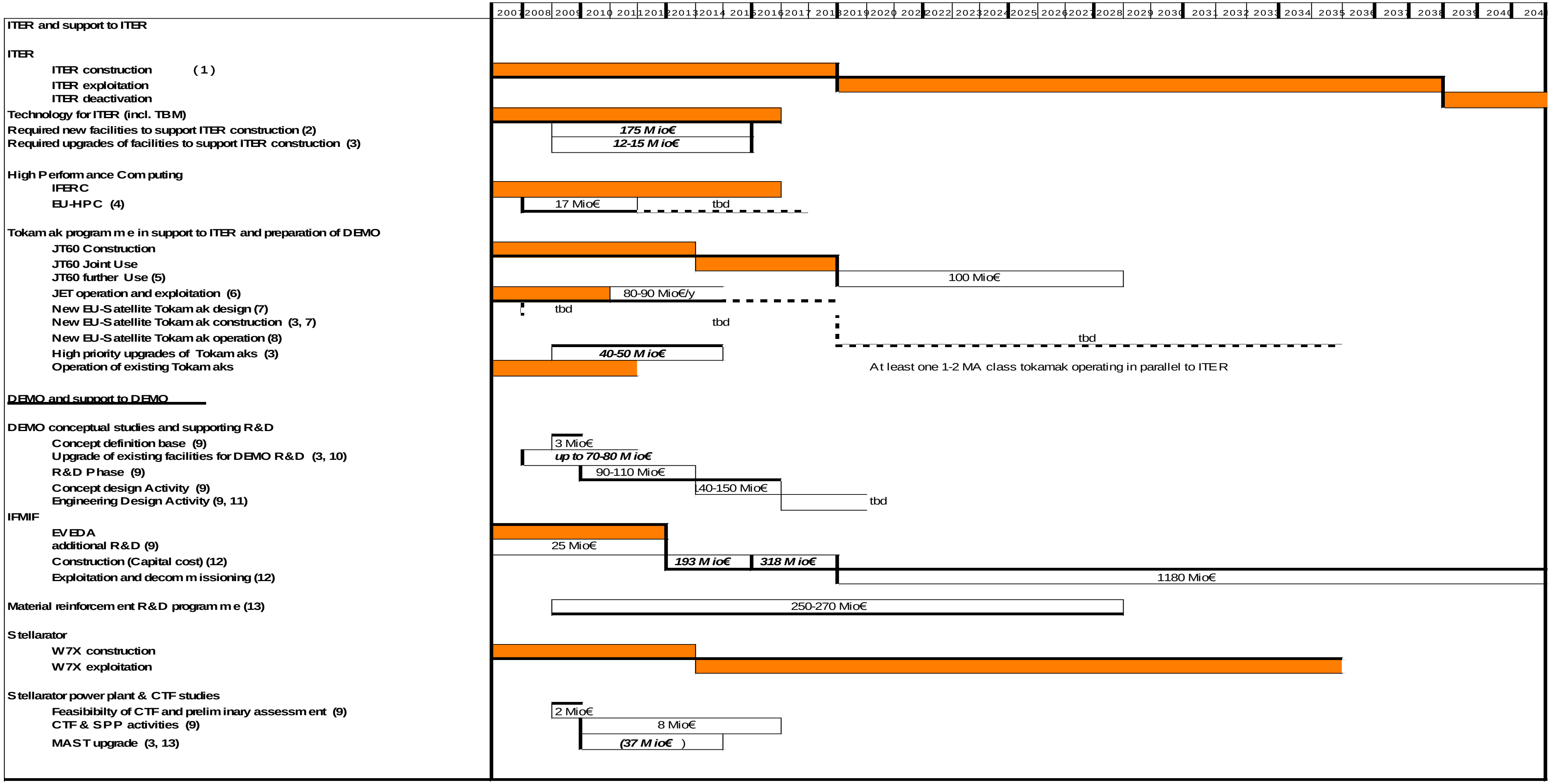
[1] *The first steps in developing this policy were recently made with specialised ad hoc groups; a long term policy deserves to be further developed.*

[2] *this will comprise a significant R&D programme that will require additional facilities and resources; however, this programme will only be costed once the Conceptual Study will be sufficiently advanced.*

RESOURCE LOADED PLANNING PROPOSED FOR THE EUROPEAN FUSION PROGRAMME



The red bars correspond to elements of the programme already decided

The white Bars are the elements proposed to complement the Core Programme

Figures in ***Bold Italics*** are for Investment cost only (2008 Euros)

Figures in normal font are for total cost (manpower, investment, operation or R&D when applicable) (2008 euros)

1 Planning under elaboration by the ITER Organisation
2 including the neutral beam test facility in Padova and the coil tests / EC contribution tbd
3 EC contribution up to 40% tbc
4 Cost of investment, operation and high level support; EC contribution 40% to investment and 50% to operation
5 Possible European contribution extrapolated from previous operation period (tbd between EU & JA)
6 Overall EC contribution 60-70 M (75% of operation; 40-100% of other costs)
7 Costs tbd; the capital cost of FAST, which is a possible option, is quoted by ENEA to be about 280 Mio€
8 Operating cost tbd, EC contribution up to 20% tbc
9 EC contribution tbd
10 Some of these upgrades could reduce the cost of the DEMO R&D programme mentioned under the next line
11 Cost of Engineering Design and related R&D will be an outcome of the Conceptual Design
12 EC contribution assumed to be 50% of total cost (F4E), construction under international collaboration
13 EC contribution 20-40% tbc
14 could be partially overlapping with high priority upgrades of existing tokamaks (above)

CONTENT

ANNEXES:

Chapter 1
Introduction

In a first document entitled “Positioning and Strategic outlook” a programmatic analysis has been carried out in order to assess which are the main lines of development of the programme needed to support ITER and prepare DEMO, with the objective of meeting the target set by the “Fast Track” approach, i.e. being able to start the construction of DEMO in about 20 years from now. The document highlighted the following programmatic needs:

(a) as ITER has an absolutely central role, as well as successful construction of the facility, it is essential to implement a consistent and focused programme in preparation of ITER operation and plan the European participation in the exploitation of ITER, which will dominate fusion research over two decades;

(b) to develop an integrated approach to DEMO design and necessary R&D, which should take into account at an early stage requirements on availability and efficiency and include an assessment of the feasibility and value of a Component Test Facility for the programme;

I to reinforce technology R&D with long term goals beyond ITER;

(d) to continue to explore alternative confinement schemes.

To achieve an efficient and focused implementation of this programme a set of seven R&D missions were proposed to be conducted in parallel to the construction of ITER and IFMIF and, in the longer term, through the exploitation of ITER and satellite devices:

M1: Burning Plasmas
M2: Reliable Tokamak Operation
M3: First Wall Materials & compatibility with ITER/DEMO relevant Plasmas
M4: Technology and Physics of Long Pulse & Steady State
M5: Predicting Fusion Performance
M6: Materials and Components for Nuclear Operation
M7: DEMO Integrated Design: towards high availability and efficient electricity production.

Milestones and detailed objectives relating to each R&D mission have been identified; the overall set of milestones and objectives constitutes a detailed R&D roadmap towards DEMO. These milestones are in Annex 1.

The purpose of the present document is to help define the infrastructure needed to implement this programme over the coming couple of decades. To this aim, the needs in terms of magnetic confinement devices and technology facilities are analysed against the goal of supporting an efficient construction and exploitation of ITER and fulfilling these seven R&D missions.

The logic of the document follows three successive steps:

⇒ *Definition of priorities (Core Programme) in Chapter 2,*
⇒ *Analysis of how facilities can address these priorities in Chapters 3 to 6 (including opportunities under international collaborations),*
⇒ *Overall proposal including an estimate of resources needed in Chapter 7.*

Details and supporting documentation (in particular a set of “fiches” for the main facilities) are provided in Annexes.

Chapter 2
Priorities for the fusion programme

2.1 A Programme supporting ITER and aiming at DEMO construction

The needs of the fusion programme to fulfil the "fast track" requirements are analysed below. This programme includes the construction and exploitation of ITER, the construction and exploitation of IFMIF, the DEMO conceptual and engineering design studies with supporting R&D, a programme on alternative confinement and the accompanying programme required to complement and support ITER and IFMIF and successfully achieve these objectives. The goal is to provide the scientific and technical basis required for a decision on DEMO construction in about 20 years.

To achieve this, a Core Programme is proposed below that includes:

- among the components of the fusion programme, those that address scientific and technical issues that must be successfully resolved to fulfil the "fast track" requirements, and
- additional elements necessary to address gaps and risks identified in the "Positioning and Strategic outlook" document.

This Core Programme contains the elements needed to fulfil all the milestones developed under the seven R&D Missions and listed in Annex VII of the "Positioning and Strategic outlook" document. These milestones are listed again in Annex 1 below, with the main facilities/resources required for their achievement. This programme also aims to address the main technical risks for the development of a fusion reactor, as identified in the above mentioned document, namely:

- plasma control issues (disruptions and power transients),
- steady state operation,
- materials including plasma facing materials,
- tritium breeding/fuel cycle and
- overall reliability/availability.

The elements which constitute the Core Programme are listed in a generic manner under section 2.2. This analysis is further developed for each set of facilities (magnetic confinement devices, technology facilities, high power computing) in chapters 3 to 6.

The analysis presented in this document proceeds as follows:

- this Chapter 2 highlights the elements constituting the Core Programme which are needed ***in addition to those presently foreseen/funded in the fusion programme***, because they ***address programmatic gaps or risks identified in the "Positioning and Strategic outlook" document.***
- in the subsequent chapters, the way facilities can address the objectives of the Core Programme will be analysed. As a result, a second category of requirements, to address ***gaps in resources/facilities,*** will be identified; in some cases ***redundancies*** will also be highlighted.

A complementary programme is also mentioned under section 2.3. This is highly desirable to support the Core programme with the aim of i) addressing critical issues in the Core Programme as they arise, ii) strengthening the scientific and technical understanding important for the progressive development toward a viable and economic fusion reactor and (iii) generating and gathering new ideas outside the mainstream thinking, which stimulate innovation and creativity and may open promising options or provide unexpected solutions to pending issues.

Overall it is essential that all elements contribute to maintain a lively, creative and sufficiently flexible and diverse programme, a necessary condition to be able to address unforeseen risks and to develop a long term vision.

2.2 Requirements of the Core Programme

This "Core Programme" comprises the objectives of the programme that must be successfully achieved to fulfil the "Fast Track" requirements (DEMO construction ready to start in about 20 years), including elements required to address programmatic gaps or risks identified in the "Positioning and Strategic outlook" document.
The requirements to achieve the Core Programme are listed under two categories, the ITER-related and the DEMO-related objectives[3]. These requirements are analysed against the Milestones/Objectives related to the seven R&D Missions and listed in Annex VII of the "Positioning and Strategic outlook" document. These milestones/objectives are shown in blue in the present chapter.

2.2.1 ITER

ITER obviously constitutes the main pillar of the Core Programme. In order to focus in the next sections on the needs of the programme in complement to ITER, the ITER experimental programme itself is briefly addressed here.

ITER will provide a huge step: it will explore a combination of physics parameters never attained and very close to those foreseen on DEMO and Power Plants; it will also provide an integrated demonstration of a number of technologies applicable on DEMO. However some gaps will remain to be addressed both in physics and technology, as pointed out in the gap analysis provided in Annex IV of the "Positioning and Strategic outlook" document. Table 2.1 summarises the capability of ITER to address the seven R&D Missions and points out which elements require further R&D to prepare for DEMO. The programme accompanying ITER in physics and technology should aim at addressing these additional R&D requirements.

The European contribution to the exploitation of ITER (which is expected to be at the level of 40 – 50% of the total) will be provided by research scientists from the fusion research institutions ("Associations") in member states who will have gained their experience and expertise from working on JET and other facilities in Europe and abroad.

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[3] *Indeed a number of elements in the first list contribute to DEMO, not just for the fact that ITER is meant to provide a significant contribution to DEMO, but key devices, such as satellite tokamaks, are meant to complement ITER in preparing the physics basis for DEMO.*

Table 2.1: capabilities of ITER to address the seven R&D Missions

Strong capabilities

Medium capabilities

	ITER (15 MA)	Comments	Further R&D required to prepare DEMO[4]
7-R&D Missions			
1 Burning Plasma		*Realisation and control of DT Burning plasma with power amplification factor Q=10 corresponding to at least 2/3 of self-heating by alpha particles from fusion reactions: burning plasma physics in conditions never attained; demonstration of burn control methods (fuelling and pumping, diagnostics and heating systems).*	*Operation at Q=30-40 (although ITER might address this and will have DEMO relevant core fast particle pressure)*
2 Reliable Tokamak		*Control of MHD phenomena (NTMs, ELMs, etc.); exploration of operational limits in reactor grade plasma; demonstration of essentially disruption-free operation close to stability boundaries.*	*Operation at higher normalised pressure (β) plasma normalised density, i.e. closer to operational limits for higher efficiency*
3 Wall and Plasma		*Operation with metallic wall at power densities approaching the DEMO regime (assuming C-free operation and ultimately all W divertor and first wall); edge and pedestal conditions close/similar to those expected on DEMO; divertor operation with high fraction of radiated power.*	*Operation at higher P/R and power density, higher absolute radiated power.*
4 Steady-State		*Achievement of very long pulse operation (possibly steady state conditions) at Q=5 would provide demonstration of physics and a number of techniques required for steady state operation.*	*LHCD not yet included on ITER.* *Full current drive capability (high βN).* *Full MHD control at high βN.* *H&CD technologies (higher efficiency, energy or frequencies; cw operation).* *Control with more limited diagnostics and actuators.* *Further developments of control systems.*
5 Predictive capability		*Unprecedented range and combination of physics parameters, at or very close to DEMO values in burning plasma conditions allowing almost full validation of physics and modelling towards DEMO.*	*Testing a range of physics (transport, instabilities, etc.) at higher β and βN*
6 Nuclear materials & components		*Tritium breeding blankets: Test Blanket Modules to be tested during first experimental phase of ITER.* *Operation of components in relevant nuclear and plasma environment.*	*Full T breeding blankets and T self-sufficiency.* *Materials (structural, functional).* *Plasma-facing components with integrated characteristics (materials, power loads, neutron loads, maintainability).* *Key component features tested to high n-fluence (e.g. welds)*
7 DEMO integrated design		*Licensing, planned and unplanned maintenance especially remote handling in a reactor environment.*	*High temperature operation of in-vessel components.* *High reliability of all auxiliary systems.* *Blanket maintenance and larger scale remote handling applicable on DEMO.*

[4] *Chapter 3 lists the specific technology facilities required to prepare DEMO. Annex 5 presents the key physics parameters and compares the range of parameters foreseen on present day tokamaks, satellite tokamaks, ITER and DEMO*

2.2.2 Core Programme: ITER-related objectives

- **Support to ITER Construction**

The very first priority is a successful construction of the ITER device, which is essential to ensure that the foreseen scientific and technical programme can be undertaken. ITER components are sometimes at the forefront of technologies; to support the construction of the advanced components an adequate R&D programme is necessary all along the construction period. In addition this programme should help guaranteeing the maintainability and availability of the device.

In order to support the procurement of items due to be provided by Europe, a number of facilities are required to qualify fabrication technologies and/or test components. These facilities relate to superconductor strand and conductor, magnet cold tests, low & high heat flux component, Electron Cyclotron Resonance Heating (ECRH) system, Ion Cyclotron Resonance Heating ICRH system, Neutral Beam Injector (NBI) system, Diagnostic systems, Divertor Remote Handling (RH), NBI RH, Transfer Casks, Test Blanket Modules (TBM), Cryopumps, Tritium (fuel cycle) system, safety and port plug testing. Technical testing of ITER diagnostics and related components for functionality and lifetime will require port plug test facilities with special equipment as well as demonstrations/validation on existing magnetic confinement devices.

The set of required technology Facilities is detailed under Chapter 3 and Annexes.

- **ITER key issues related to first phases of operation**

A successful, rapid and cost-effective start of ITER operation, in about 10 years from now, will be much facilitated if the issues listed below are successfully addressed in advance.

In relation to R&D Mission 2, Reliable Tokamak Operation (M2), operation strategy and control methods (conditioning, disruptions, ELMs etc.) need to have been fully developed.

This requires a broad dedicated experimental programme and developments on tokamaks: wall conditioning capabilities such as ICRH with metallic wall materials; MHD control tools such as ECRH/ICRH for NTMs, sawteeth as well as control coils and pellet pacing for ELM control in ITER relevant range of density and collisionality; ITER shape for the study of disruptions, operational limits and pulse management; sensors (IR, magnetics etc.) and actuators (PF system, control coils, fuelling, H&CD etc.) applicable on ITER; disruption mitigation tools applicable on / extrapolable to ITER; theory and modelling.

In relation to R&D Mission 3, First wall materials & compatibility with ITER/DEMO relevant plasmas (M3), diagnostics for T retention, erosion, dust etc. as well as T & dust control, mitigation and removal techniques need to be developed and qualified before being built and used on ITER; a consistent understanding of phenomena and data base need to be available to support the choice of and make available first wall materials for ITER DT operation (ITER presently foresees using W and Be only in that phase of operation); a benchmarked predictive capability for all relevant aspects of plasma-surface interaction needs to be available to support ITER experimentation.

This requires dedicated experimental programmes on Tokamaks (in particular with metallic plasma facing materials and high values of P/R; DT experimentation would provide useful information for T retention; remote handling tools could be helpful for in situ detritiation and dust removal demonstrations), plasma devices simulating the divertor and plasma-surface interactions in well-defined conditions, specific technology facilities (T and dust removal techniques including RH developments, plasma facing components testing), theory and modelling (high performance computing).

- **High priority actions to prepare experimentation on ITER**

Meeting the objectives listed below by the start of ITER operation would greatly support/facilitate the achievement of ITER's objectives.

In relation to R&D Mission 1, Burning Plasmas (M1), a burning plasma predictive capability has to be developed in view of benchmarking against ITER experiments. Burn control methods in preparation for the main ITER objective have to be developed, to guide achieving a Q=10 D-T burning plasma.
This requires dedicated experimental programmes on tokamaks (studies of fast particle physics, at relevant values of normalised plasma pressure and with dominant electron heating in ITER relevant dimensionless parameter space, and in particular studies of non-linear processes with acceleration by ion cyclotron resonance heating (ICRH) and NBI; diagnostics for plasma waves and fusion products etc.; ITER-relevant combination of heating, diagnostics and control systems for development of burn control methods; He ash control), DT experimentation, theory and modelling (high performance computing, see also Mission 5).

In relation to M3, integrated plasma scenarios compatible with high-Z materials need to be fully developed. Furthermore, hydrogen/deuterium inventory & other data related to erosion, dust, etc. need to be available to support the licensing for DT operation on ITER, i.e. in about 15 years from now.
This requires very significant dedicated experimental programmes on Tokamaks in particular with combinations of plasma facing materials and plasma configurations as foreseen on ITER, with the highest possible integration of ITER relevant densities and collisionalities as well as high P/R; divertor tokamaks equipped with ELM control tools and core impurity control tools such as ECRH and ICRH. This would benefit from DT experimentation (possible isotopic effects). Linear plasma devices, theory and modelling should also bring essential contributions.

In relation to R&D Mission 4, Technology and Physics of Long Pulse & Steady State (M4), plasma scenarios need to be developed and qualified for ITER long pulse operation (ITER Q=5), Lower Hybrid Current Drive (LHCD)[5], need to be ready for application on ITER, coupling of ICRH and LHCD has to be optimised and demonstrated in conditions which extrapolate to ITER. Furthermore, synergy between ECCD and LHCD has to be assessed as a possible option for raising the CD efficiency.
This requires dedicated experimental programmes on tokamaks (ideally with the whole set of relevant conditions: ITER shape, all four H&CD methods, diagnostics for profile control and for internal barrier studies, wall materials, long pulse, RWM control; alternatively an integrated tokamak programme combining efforts from several devices not necessarily all with the full set of required conditions), development and testing of steady state diagnostics, development and testing of steady state fuelling techniques, test of an ITER-relevant LHCD coupler on a divertor tokamak, and facilities for the development/improvement of H&CD technologies.

In relation to R&D Mission 5, Predicting Fusion Performance (M5), some high priority scientific questions need to have been solved/progressed significantly, aiming at a validated 'numerical tokamak' to be used in planning and analysing ITER discharges.
Such a numerical tool needs a spectrum of concepts, devices, theories and numerical simulations, each addressing a set of issues, and allowing a reasonable level of flexibility and integration. Thus it is critical

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[5] *LHCD is the most efficient external technique for driving current in tokamak plasmas and the most promising for controlling current profiles to achieve advanced modes of operation capable of steady state operation; however due to concerns about coupling and survivability of the launcher, it had not been included in the first set of heating and current drive methods on ITER; following recent experimental results obtained on JET, Tore Supra and FTU, ITER is now showing an interest in LHCD; the LHCD PAM coupler, a concept able to sustain high power loads, has been successfully tested on FTU: a missing demonstration is the use of this PAM coupler on an ELMy plasma (e.g. on JET or ASDEX-Upgrade)*

that the portfolio of experiments provides sufficient diversity to validate the models. The experiments, theory and codes should work very closely together; for example, in the near future it will be possible to make full torus images of the plasma turbulence that could challenge the numerical models. Some key scientific issues need to be addressed forcefully, in particular, a predictive capability for the H-mode pedestal is highly desirable.
This requires High Performance Computer(s), ideally in the PetaFlop range, dedicated experimental work on magnetic confinement devices with plasma configurations and parameters allowing extrapolation to ITER with accurate diagnostics (e.g. for pedestal physics: edge profiles and turbulence), and theory & modelling.

In relation to R&D Mission 6, Materials and Components for Nuclear Operation (M6), the reference materials for the Test Blanket Modules (TBM) need to be qualified, in particular optimised varieties of the structural material EUROFER. Qualification has to include the development of manufacturing technologies and the validation of design rules.
This requires irradiation facilities, pre- and post-irradiation testing facilities (hot cells) and a materials sciences and modelling programme (High Performance Computing).

2.2.3 Core programme: Support to ITER experimentation and preparation for DEMO: satellite tokamak programme

The purpose and requirements of the satellite tokamak programme, in parallel with ITER construction and operation, are presented in Annex 5. The objectives are as follows:

- to optimise the concepts used in the ITER exploitation in conditions relevant for a reactor and to contribute to the consolidation of the ITER design choices;
- to contribute to the advancement of the physics understanding by extensive plasma diagnostics and modelling tools;
- to complement ITER in the testing of innovative technologies that are not yet foreseen to be tested on ITER itself;
- to contribute to filling the gap between ITER and DEMO in the development of robust regimes of operation characterised by more advanced plasma parameters, especially the investigation of regime sustainment compatible with high fusion gain operation with a minimum number of actuators and sensors, which would have beneficial effects on e.g. the capital cost and the cost of electricity of a reactor.

Addressing the Missions 1-4 requires a range of parameters and technological areas which are presented in Annex 5.
Addressing Mission 5 requires a set of devices with the capability of investigating a range of physics parameters relevant for an extrapolation based on theoretical understanding and advanced numerical simulation tools.
Major tokamaks (Ip 4-5MA or higher) accompanied by smaller more flexible devices (1-2 MA class) should address, in particular, high performance operation with advanced current profile and MHD (resistive wall modes) control, steady state operation, operation with metallic plasma facing materials (W is today's preferred option), development of a relevant set of diagnostics, control and H&CD techniques. In addition, flexibility in operations and enhancement is crucial for investigating different operating conditions, testing new components and improving the diagnostic and control capability.

As will be shown in Chapter4, this programme requires one or more JET-class tokamaks (current 4-5 MA or higher), accompanied by smaller more flexible devices (1-2 MA class) operating in parallel to ITER.

2.2.4 Core programme: DEMO and longer term-related objectives

In complement to a successful exploitation of ITER and of its accompanying devices, the objectives listed below need to be met to allow fulfilling the "Fast Track" goal of DEMO being ready to build in about 20 years.

- **Towards DEMO Engineering Design**

In relation to R&D Mission 7, "DEMO Integrated Design: towards high availability and efficient electricity production" (M7), a DEMO conceptual study, with significant involvement from industry, needs to be launched soon. A well-developed DEMO conceptual study should be available in about 10 years.
Additional programme element to address an identified gap: *This requires a DEMO design study group, no yet part of the fusion programme, with industry contributions gradually growing during the conceptual studies and ensuring "buildability, reliability, operability and maintainability", for example in the development of concepts for key in-vessel components.*

In relation to M4, DEMO H&CD and other steady state related requirements should be sufficiently defined so that relevant R&D can be launched in about 10 years.
In relation to M6, a preselection of the DEMO divertor and blanket concepts allowing launching R&D, should be made in about 10 years.
Additional programme element to address an identified risk (steady state operation): *This will require new/upgraded technology R&D facilities, some of them to be defined in 5-10 years (Heating and Current drive systems; high-temperature He facilities; conductor, cable structural materials, and coil testing for low temperature and high temperature superconductors; tritium-compatible vacuum pumping; tritium recovery; plasma facing components). Further details are given in Chapter 3 and Annexes.*

In relation to R&D Missions M1-5, a first set of elements constituting the DEMO physics basis, in particular for long pulse/steady state plasma scenarios, should be available in about 10 years.
Additional programme element to better address an identified risk (steady state operation): *This requires a dedicated programme on existing tokamaks with some upgrades (advanced modes of operation with H&CD methods, diagnostics and control methods applicable on DEMO, resistive wall mode studies etc.) and JT-60SA.*

In relation to R&D Mission M7, the feasibility of the proposed maintenance procedures for DEMO should be confirmed by R&D in about 15 years.
Additional programme element to better address an identified risk (overall reliability/availability): *This will require a significant programme on Remote Handling, to be specified when the DEMO concept is sufficiently developed and therefore to be launched at a later stage.*

- **Towards DEMO Licensing and Construction**

These objectives need to be achieved generally a few years before or just before DEMO construction can be decided and licensing granted. All information required for DEMO licensing and construction must be available in about 20 years.

In relation to M1-M4, the DEMO physics basis must have been confirmed, with feedback plasma control methods developed and qualified, appropriate diagnostics selected and qualified, plasma operation strategies and burn control capability demonstrated, no more than 2 H&CD methods selected.

The major contribution will come from ITER exploitation, complemented by joint exploitation of the satellite tokamak programme (in particular JT60-SA should be jointly used beyond the initial 5 years foreseen in the Broader Approach Agreement) and a substantial programme on other tokamaks. Qualitatively new information from ITER will be in the areas of α-particle heating and burn control, which can only be adequately studied there. This programme would be strengthened by a European satellite tokamak (see section 2.2.3).

In relation to M5, a fully validated numerical tokamak shall be available.
This requires High Performance Computer(s), validation of theory and modelling on ITER and other magnetic confinement devices. In particular, for designing DEMO, the numerical tokamak must be completely validated, including in presence of α-heating and burn control. By this time, also a quantitative understanding of the edge pedestal is mandatory.

In relation to M6 and M3: structural, functional and plasma facing materials need to be qualified and materials related data must be made available for licensing.
This requires the IFMIF EVEDA to be completed according to plan, the construction and operation of IFMIF decided without delay so that the 1st IFMIF Campaign can be completed a couple of years before all elements required to decide DEMO construction are available.
This requires also other irradiation facilities (fission reactors; multi beam facilities), activated materials testing facilities / hot cells and a significant materials development, science and modelling programme (with high performance computing facilities), as well as dedicated PFCs test facilities.

Additional programme elements to better address an identified risk (materials, M6):
- *It is proposed to **minimise risks on IFMIF** by reinforced testing / qualification in preparation of construction. Four actions have been identified (see Annex 2).*
- *It is proposed to **accelerate the Fusion Materials Development Programme** by increasing the human resources and by intensifying the experimental programme (see Annex 3).*

In relation to M6, the divertor and breeding blanket concepts will have been selected and designed.
This requires:
- *the successful performance of the conception, construction and testing programme of the TBMs on ITER, with supporting technology facilities (blanket component testing, helium flows, etc.).*
- *successful experimentation with the divertor on ITER, with materials applicable on DEMO, and*
- *testing facilities for divertor and blanket components.*

In relation to M6: a strategy for clearance and recycling of DEMO activated material needs to be defined.
This will require, in parallel to DEMO Engineering Design, recycling techniques to be developed and tested on specific facilities.

In relation to M4, R&D on DEMO relevant H&CD and other steady state related items should be completed.
The detailed R&D requirements will be defined during the DEMO conceptual design phase. In the meantime some DEMO-oriented R&D can be already conducted (e.g. towards long pulse operation, improved robustness, simplification, increased efficiency etc.).
This will require, in parallel to DEMO Engineering Design, new/upgraded technology R&D facilities (in particular H&CD test facilities and industrial developments to develop simpler, more reliable and more efficient H&CD systems).

- **Alternative Confinement/Stellarator**

The stellarator constitutes an alternative and back-up to the tokamak. Back-up, because the magnetic field is intrinsically steady state, and alternative, because the stellarator operation promises higher availability (less-demanding plasma control and disruption-free) and lower re-circulating power. Stellarator R&D therefore ***addresses identified risks (plasma control, steady state and overall reliability)*** and should be vigorously pursued to prepare for possible improved concepts for DEMO or beyond. However stellarators may require more complex technologies, in particular for magnets, vacuum vessel and in-vessel maintenance.
Several physics issues are shared with tokamaks and there will be mutual benefit in developing physics understanding, on core plasma phenomena as well as on plasma-wall interaction and on steady state technologies. Furthermore stellarators will be the only devices on which to study plasma wall interactions at high density on long timescales (density operation a factor 2 to 3 higher than on tokamaks at the same field).

Long term milestones have been identified as follows:
In relation to M4: the optimised stellarator configuration should be confirmed in about 10 years and the steady state features of the stellarator in about 20 years.
In relation to M5: a benchmarked numerical stellarator should be available in about 15-20 years.
In relation to M7: the engineering feasibility of stellarator power plants should have been studied in 20 years or earlier.

This requires the timely start of W7-X according to plan (2014), a second stage of operation with full steady state capability as well as close cooperation with other stellarators worldwide, in order for stellarators to converge towards a single concept line, similar to what has been achieved for tokamaks in the 1990's.
Additional programme element to address an identified gap: *stellarator power plant engineering studies need to be conducted; this would be best done by the team in charge of DEMO design, where all competences will be available.*

- **preparing for a possible Component Test Facility (CTF)**

In the "Positioning and Strategic Outlook" document, a possible Component Test Facility has been identified as an option to be studied since it could strengthen the development of fusion power by conducting a number of tests in support to DEMO in-vessel components, **thereby addressing identified gaps and risks (nuclear components, reliability**). Indeed there is likely to be a continuing need for such a facility through DEMO's lifetime and beyond, just as fission development has had a continuing need for analogous facilities.
It was pointed out that two main issues had to be addressed before the validity of this approach could be confirmed: (1) the technical feasibility of such a compact device; (2) the scientific feasibility of the most promising approach, the spherical tokamak (the potential showstoppers being the ability to operate in very long pulses and the handling of very high power density in the divertor– both issues that also have to be addressed for DEMO).
In preparation of decision making on a possible CTF (M7), the following two Milestones have been defined: CTF feasibility study available and R&D on Spherical Tokamaks completed within about 10 years.
Addressing these issues requires two set of actions, conceptual studies CTF feasibility study to be conducted in the frame of the DEMO conceptual design activities and R&D in relevant conditions on spherical tokamaks.

2.3 Complementary programme elements that support the Core Programme

This is constituted of complementary elements which are highly desirable to support the Core programme with the aim of providing a capability to i) address critical issues in the Core programme as they arise, ii) strengthen the scientific and technical understanding important for the progressive development toward a viable and economic fusion reactor, iii) generate and gather new ideas outside the mainstream thinking which stimulate and foster the essential elements of innovation and creativity. These new ideas may open promising options, provide unexpected solutions to pending issues and guarantee a faster path toward ITER operation and DEMO development. In general, these elements are considered an essential ingredient for an EU programme designed to address the problem of toroidal confinement as a “grand” experiment, based on a spectrum of concepts, devices, theories and numerical simulations, each addressing a set of issues, and allowing a reasonable level of flexibility and creativity.
This underlying programme provides further experimental and modelling capability to address issues in the Core Programme. Moreover the need for risk minimization strongly calls for keeping a reasonable level of flexibility in the programme provided by a portfolio of skills and resources, which could be focused on unexpected issues. This portfolio contributes significantly to the renewal of generations of physicists and engineers via education and training.

Such a portfolio exists within the European fusion research programme, both in physics and in technologies.:

- Alternative confinement devices such as the ***reversed field pinch*** (RFP) offer a number of opportunities. The RFP is an alternative concept, which provides a complementary approach to the tokamak. A number of results are relevant to tokamaks, including scientific and technological issues which can contribute to the successful operation and exploitation of ITER. This applies for example to active feedback control of MHD instabilities, magnetic self-organization issues (like current transport), link between particle, energy and momentum transport and MHD stability, density limits, fast-particle physics. In the case of active feedback control of MHD instabilities, the RFP community developed techniques anticipating tokamak needs.
- Other examples come from ***smaller tokamaks*** on which new ideas, sometimes unconventional, can be explored at low cost and with high flexibility. A recent example is the exploration of liquid metal limiters.
- The capability to address fusion relevant issues with ***new/emerging technologies*** must also be explored. For example, the materials R&D presently includes prospective studies of advanced materials like SiC/SiC. The potentialities for high temperature supra conductors are also explored.

Finally, this complementary programme also includes ***socio-economic studies (Socio-Economic Research on Fusion, SERF),*** which were launched in 1997 following the recommendations of the 1996 Fusion Evaluation Board set up by the Commission and chaired by Prof. Sergio Barabaschi. These studies have provided over the past ten years a very useful set of information (i) on the perception of fusion and public acceptance, (ii) on economical aspects of fusion (energy scenarios and models have been developed to study fusion as a part of the future energy system; external costs of fusion power production have been estimated and compared to those from other sources of energy; spin-offs have been analysed).

Chapter 3
Addressing the seven R&D Missions: existing and required facilities for technology R&D

Successful implementation of the DEMO – oriented fusion development programme hinges on: the timely achievement of ITER construction and operation; the success of the accompanying programme; the timely design (including validating R&D), construction and operation of IFMIF and the ITER Test Blanket Modules; and the availability and exploitation of adequate testing facilities to develop and qualify specific technologies and engineering design solutions that cannot be tested in ITER and IFMIF.

A first document entitled 'Positioning and Strategic Outlook' provided a programmatic vision and framework, with seven R&D Missions and many associated Milestones, which would be used to analyse the need for particular facilities and other resources. Such an analysis is presented, for technology facilities, in this Chapter. It summarises the analyses, in the form of mappings from the missions to the main required generic means of execution, and from those generic means to the requirements for key technology facilities – with identification of areas where there are key gaps in capacity or surplus capacity, areas of broad balance between requirements and capacities, and areas needing further detailed consideration (see Tables 3.1-3.3).

A more detailed presentation of the mappings is provided in Annex 4.

3.1 Development of the Required Generic Means of Execution from the Missions and their Milestones

Table A4.1 in Annex 4 presents the results of the first stage in the analysis, developing the Missions and their Milestones into the principal Required Generic Means of Execution, in the various fields of technology. Note that objectives to be achieved by toroidal plasma devices other than ITER, and their associated systems, are excluded from this Table, as these matters are covered in other Chapters and annexes.

As the construction of ITER is a *sine qua non* for the achievement of all of the Missions, the key technology issues associated with the European contribution to ITER construction are collected together at the head of Table A3.1, followed by the other elements. A more compact version of Table A3.1, displaying the key points, is given below as Tables 3.1(i) and 3.1(ii). In these Tables, 'IR' signifies 'ITER requirement', and 'MR signifies 'Mission requirement not satisfied by ITER'.

Table 3.1(i)
Required Generic Means of Execution, related to ITER Construction*

Mission	Description	Required Generic Means of Execution
Construction of ITER (for all Missions)	Testing and qualification of key components.	IR1: Strand & cable structural materials testing. IR2: Conductor testing. IR3: Magnet testing IR4: Low & high heat flux component testing. IR5: Electron Cyclotron Resonance Heating (ECRH) system testing IR6: Ion Cyclotron Resonance Heating (ICRH) system testing IR7: Neutral Beam Injector (NBI) system testing IR8: Cryopump system testing IR9: Port plug testing IR10: Tritium (fuel cycle) system testing
	Remote handling.	IR11: Divertor Remote Handling (RH) IR12: Neutral beam injector RH IR13: Transfer Casks
	Licensing.	IR14: Safety – related testing IR15: Dust and tritium measurement and removal techniques
	Other	IR16: the ITER Test Blanket Modules and programme

*based on present EU procurement sharing

Table 3.1(ii)
Required Generic Means of Execution for Technology R&D towards DEMO

Mission	Description	Required Means
3	First wall materials and compatibility with ITER/DEMO relevant parameters.	MR1 Plasma wall interaction simulators. MR2: Plasma facing component testing. MR3: High performance computers for materials modelling. (See Chapter 5) MR4: Facilities for non-irradiated materials characterisation. MR5: Plasma facing materials irradiation. MR6: Facilities for post-irradiation examination.
4	Physics and technology of long pulse and steady-state.	MR1 – MR6, plus: MR7:Test Beds for H&CD systems (improvement of long pulse capabilities and efficiency)
6	Materials and components for nuclear operation.	MR1 – MR5, plus: MR8: Neutron irradiations of structural and functional DEMO materials. MR9: Charged particle beam irradiations of model alloys and EUROFERs. *(MR10 not used)* MR11 Facilities for further blanket developments (beyond ITER TBM) MR12 Tritium facilities (including vacuum pumping) MR13 Helium loops with relevant parameters. MR14 Test facilities for the qualification of mock-ups and prototype components. MR15 IFMIF construction and test programme. MR16 Facilities for the qualification of in-vessel maintenance procedures[6].
7	DEMO integrated design: towards high availability and efficient electricity production.	All the above, plus: MR17 Facilities for the further development and qualification of H&CD[7] systems MR18 Facilities for the qualification of Balance of Plant components. MR19Facilities for further strand and conductor development, test facilities for advanced model coils[7]

3.2 Development of Key Facility Requirements from the Required Generic Means

Table A4.2 in Annex 4 presents the results of the second stage of the analysis, developing the Required Generic Means of Execution (IR1 – IR17 and MR1 – MR19) into the requirements for key facilities, and including comments on where there are clearly identified key gaps in capacities or surplus capacity,

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[6] DEMO-Design-dependent

[7] Realization of these facilities depends on whether the technology becomes relevant for the DEMO design

areas where assessment shows there is broad balance between requirements and capacities, and areas where further careful study will be needed before coming to firm conclusions.

Tables 3.2a and 3.2b, below, are summarising extracts from Table A4.2, for ITER- and DEMO-relevant needs respectively, showing the main clearly identified key gaps in capabilities.

3.2a New Facilities to fulfil ITER needs

The prime elements of the Core Programme (see Chapter 2) include the construction of ITER and all necessary supporting actions. A successful European contribution to ITER construction, including the Test Blanket Modules, requires setting-up the set of missing facilities identified in table 3.2a. For this purpose, the Table includes a preliminary assessment of costs. It must be noted that there is still ongoing discussion with the ITER Organisation to determine the additional testing facilities to be built on-site (e.g., magnet cold tests, RH test stand, vacuum laboratory, port-plug test facilities to test primarily RF Heating and Current Drive Systems, etc.).

Table 3.2a: Key Gaps in Technology Facilities (ITER Needs)

Required means	**New facilities or upgrade of existing facilities eligible to ITER credit**	**Rough Estimated Costs[8] (M€)**
ITER related		
IR3: Magnet testing.	TF-PF windings cold tests (at 4K with low levels of current)	50
IR4: Low & high heat flux component testing.	Beryllium-compatible HHFT for series production and acceptance tests of ITER FW panels	3
IR7: NBI system testing	Neutral Beam test Facility, which is planned to be built in Padua.	~100
IR9: Port plug testing	In addition to dedicated ITER facilities of port-plug testing, the need for supporting facilities in the parties is being assessed.	< 10Meuro[9]
IR 11: Divertor Remote handling	Divertor Test Platform (DTP2). Allows simulation of divertor in-vessel maintenance operations using prototype divertor RH equipment (Movers, end-effectors and tooling) and in a full scale mock-up of ITER divertor region.	5
IR13: Transfer casks	Transfer cask transport and docking operation and in-cask operation.	<2
IR15: Dust and tritium measurement and removal techniques	In-Vessel Viewing System (IVVS) test facility. To test and verify visual and metrology capabilities of a full scale prototype of the IVVS (deployer + probe+ vacuum chamber + access duct) in vacuum and temperature conditions close to the real ITER ones.	~5

[8] This is the total cost of investment (not the Euratom contribution); it does not include operation and testing costs.

[9] Cost depends on the final technical specification and functional testing requirements of this facility, which still need to be agreed upon, and will ultimately depend on the availability and characteristics of port-plug test facilities on-site

3.2b Upgrades to fulfil ITER needs

In addition to the necessary facilities shown above, a number of additional upgrades of important facilities in the Associations, are recommended to minimise risks in the Core Programme, namely: *(i)* to maintain adequate testing capability in important areas where there are still R&D development needs; (ii) minimise possible risks arising from tests currently planned in non-EU facilities (such as high heat flux tests of divertor components) prove difficult or slow etc; and (iii) to deliver components that meet technical specifications. These include[10]

IR4b: High heat flux component testing

- JUDITH / HML (FZJ, Germany) – a few M€ (out of 10 M€ proposed by FZJ)
- GLADIS (IPP Garching, Germany) – 1 M€
- Electron beam FE 200 for HHFC testing, (CEA and AREVA-NP, France) 1 M€

IR5: ECRH system testing.

- ECRH Launcher structural test facility (FZK, Germany) – 2M€

IR8: Cryopump system testing

- TIMO2 cryo vacuum test bed (FZK, Germany) 2 M€

IR11: Test Blanket Module (TBM) testing

- EBBTF (ENEA, Italy) - 1.2 M€

IR14: Safety – related testing

- HYDEX (FZK, Germany) – 1.3 M€

3.2c New/upgraded Facilities to fulfil short term DEMO needs

These are listed below in Table 3.2b, together with a preliminary estimate of required cost. It should be noted that this list includes only facilities upgrades > 1 M€ and does not include upgrades for which funds have been already committed

[10] It should be noted that only facilities upgrades > 1 M€ are mentioned here and upgrades for which funds have been already committed are excluded

Table 3.2b: Key Gaps in Technology Facilities (short term DEMO Needs)

DEMO related	Comments on key gaps in present capabilities	Examples of possible upgrades of existing facilities and tentatively foreseen costs
MR1: Plasma wall interaction simulators.	ELM/disruption simulation facilities are desirable, and not fully/reliably covered by existing international collaborations.	• Magnum PSI (FOM, The Netherlands) – 3 M€ • Plasmatron vision (SCK CEN Belgium) – 1.1 M€ • TechnoFusion laboratory in Spain 8 M€ [11]
MR3: High performance computers.	Needed for multi-scale modelling of materials bombarded by plasmas and/or neutrons.	See Chapter 5
MR12: Tritium (fuel cycle) facilities (including vacuum pumping).	Facilities for single integrated tests of processes and components for tritium recovery from large blanket-related purge and coolant helium streams. If water cooling were re-instated in the blanket programme, analogous facilities would be required. The continued availability of tritium laboratories (though currently provision is adequate) is essential.	• JET Active Gas Handling System could provide relevant vacuum pumping and process information during further D-T experiments on JET (see 4.2.8) • Tritium Laboratory (FZK; Germany) – 8 M€ • FML Hotcells (FZK, Germany) – 1M€
MR13: Helium loops with relevant parameters.	See MR11. Also HELOKA TDM for helium-cooled divertor development.	• HELOKA HP (FZK, Germany) - 16 M€
MR15: IFMIF construction and test programme.	Elements to minimise risk on IFMIF have been identified in Annex 2. The programme requires n-sources with adequate spectrum and facilities to develop liquid lithium technologies (e.g., fluid dynamics modelling and validation, impurities monitoring and purification techniques, corrosion).	• See Annex 2 for risk minimisation on IFMIF EVEDA • Frascati Neutron Generator (ENEA, Italy) – 7 M€[12] • TechnoFusion laboratory in Spain – 5 M€ [13] • TechnoFusion laboratory in Spain – 19 M€ [14]
MR16: Facilities for the qualification of in-vessel maintenance procedures	A major remote maintenance laboratory.	Requires new facilities, which are DEMO-design-dependent, probably 10-20 M€

[11] The multi-purpose test facility, TechnoFusion currently planned to be built in Spain, which should include a plasma-wall interaction laboratory, would also be useful and add important information in this area.

[12] Acquisition of a 30MeV cyclotron to product IFMIF-like neutrons

[13] TechnoFusion would include a liquid metal Laboratory with a Li loop .

[14] TechnoFusion would include a multibeam laboratory for materials irradiation..

3.3 Development of Key Facility Requirements from the Required Generic Means: Surplus Capacity

From Table A4.2, it is clear also that there is surplus capacity in some areas, as summarised in table 3.3.

Table 3.3
Surplus Capacity in Technology Facilities

IR1:	Strand and cable structural materials testing.
IR5:	Low power gyrotron test beds
MR4:	Facilities for non-irradiated materials characterisation.
MR8:	Neutron irradiation, prior to IFMIF, of 'Eurofers' (but note that irradiations to high levels of displacement damage are dependent on continued access to reactors like Jules-Horowitz (F); the proposed Pallas (NL) in Europe)
MR11:	Facilities (other than tritium, helium and remote handling facilities) for further blanket development.
MR14:	Test facilities (other than those referred to above) for the qualification of mock-ups and prototype components.

Table A4.2 also identifies a number of areas where further study is needed before coming to firm conclusions on gaps/redundancies in the near term, and areas where it is premature (by five to ten years) to come to conclusions on future requirements.

Chapter 4
Addressing the seven R&D Missions: existing and required European magnetic confinement devices

The capability of the magnetic confinement devices to address the 7 R&D Missions in preparation of and in support to ITER exploitation, and in preparation of DEMO, are analysed in this chapter. It is very important to recognise the need for a diverse range of tokamaks, with varying sizes, aspect ratios and plasma shapes, to maximise understanding of fusion science and to optimise the choice of DEMO which will probably not be simply a scaled up version of ITER. At present, the European fusion programme benefits from a coherent set of devices, in particular small, medium size (~1 MA) and large (> 3MA) tokamaks, able to address complementary issues and spanning a wide range of physics parameter to extrapolate to ITER. These devices are well equipped with heating systems, diagnostics and other auxiliaries to address efficiently physics and operational issues in preparation of ITER.

Over the coming ten years, important contributions to the 7 R&D Missions are expected to be made, in particular with ITER/DEMO relevant plasma facing materials on two European devices with ITER-like shape (W and Be on JET and W ASDEX Upgrade). The programme would benefit from additional DT experimentation on JET as well as from some investments to upgrade specific technical capabilities on existing tokamaks, which all remain to be decided.

In the longer term, while Europe is expected to bring a major contribution to stellarator R&D with the W7X project, foreseen to start operation in 2014, there is a risk that the European contribution to the tokamak R&D decreases significantly since (i) the European tokamaks having all been built in the 1980’s and 90’s, a number of these devices will probably no longer operate in parallel to ITER (either due to ageing or obsolescence) and (ii) no superconducting divertor tokamak exists or is presently foreseen to be built in Europe. These weaknesses will be partially compensated by joint EU-Japan experimentation on JT60-SA and could be further compensated by strengthening international collaborations. However **the question of a tokamak programme in Europe in parallel to ITER exploitation is among the most acute questions to be addressed** (the requirements for the “satellite tokamak programme” are described in Annex 5).

The analysis made in the present chapter shows that addressing the 7 R&D Missions requires a broad experimental capability. For the longer term, the present set of planned devices worldwide does not address all programmatic requirements. In particular there are weaknesses in the ability to address Missions 1 and 3 (this is further illustrated in chapter 6). With respect to Mission 1, JT60SA could be usefully complemented by a high current device providing further data in a range of fast particle physics parameters closest to ITER and DEMO (see plots in Annex 6). Addressing the development of plasma scenarios compatible with DEMO relevant wall materials (Mission 3) requires that one or more divertor tokamaks of sufficient size (possibly a few MA) equipped with tungsten plasma facing materials are available worldwide; this device should also be able to operate in the same range of Power/Radius as ITER and, if possible, DEMO to develop relevant divertor operation. One option for Europe would be to negotiate a second phase of JT60SA with tungsten plasma facing materials and upgraded heating power. In any case, if another high current tokamak was to be built in Europe or elsewhere, it should directly address DEMO needs by avoiding C as plasma facing material. Finally, and since JT60SA is the only high current (> 3 MA) device presently foreseen after JET stops operation, the worldwide tokamak programme shows a relative weakness in the range of physics parameters which can be achieved to support ITER and prepare DEMO physics and plasma operation (as illustrated in the plots shown in Annex 6). This weakness would be overcome by **a second high current tokamak operating in parallel to ITER**.

4.1 Overview of Magnetic Confinement Devices

4.1.1 Tokamaks

Magnetic confinement fusion research requires very demanding plasma conditions. The plasma density (n_e), temperature (T) and confinement time (τ_E) in a fusion power plant needs to satisfy the Lawson Criterion, $n_e \tau_E > 1.5\ 10^{20}\ m^{-3}$ s, at an optimal core plasma temperature of around T=15-25 KeV. Tokamaks proved, so far, to be the most efficient devices in approaching these conditions. Around 30 tokamaks of various sizes are in operation in the world today (medium size means plasma current in the range ~1 MA and large size plasma current > 3MA), of which 9 devices are European:

- JET is the flagship Tokamak of the European fusion programme. It has the unique capabilities of using Tritium and Beryllium and it has a comprehensive remote handling system. The most important auxiliaries include NBI, ICRH and LHCD heating and current drive, including the test of an ICRH ITER-like antenna. JET can run plasma configurations with a shape very similar to that foreseen on ITER. The change of the first wall and divertor materials with the same mix as ITER (Be and W) is under way, together, with a number of specific diagnostics, heating and control enhancements.

- ASDEX-Upgrade is a medium size Tokamak now operating with a full tungsten (W) wall, currently the choice of first wall materials for a Tokamak-based demonstration power plant (DEMO). It has versatile heating systems with ample heating power, including NBI, ICRH and ECRH; and can run ITER plasma shapes.

- Tore-Supra is a medium size Tokamak with a circular cross section. It is the only operating device in Europe equipped with super conducting magnets and the only device in the world equipped with a full set of actively cooled plasma facing components, therefore, capable of long pulse operation (6 minutes achieved). It is equipped with ICRH, LHCD and ECRH heating systems.

- TCV is a tokamak with a versatile plasma shape capability, allowing to access the whole ITER shape range and beyond, and a powerful ECRH heating system, capable of detailed physics studies in a broad operational domain with dominantly electron heated plasmas.

- MAST is a low aspect ratio Tokamak, capable of large plasma pressures relative to the magnetic field pressure (β) and can run with ITER plasma shapes. For heating, current drive and start-up it has Neutral Beams and the Electron Bernstein Wave form of ECRH. It is the only spherical tokamak in Europe.

- FTU is a Tokamak with a circular cross section capable of high magnetic field operation, equipped with Molybdenum (Mo) plasma facing components and a liquid Lithium target. It operates routinely at plasma densities close or higher than in ITER and it is equipped with IBW, LHCD and ECRH heating systems.

- TEXTOR is a Tokamak with a circular cross section, equipped with a dynamic ergodic divertor and a toroidal belt limiter, specialised in plasma wall interaction studies using PWI test facilities with comprehensive diagnostic tools. It is equipped with ICRH, NBI and ECRH heating systems.

- COMPASS is a small Tokamak experiment capable of H-mode operation with ITER-shaped plasmas for detailed physics studies. This device has recently been moved from Culham to Prague and will start operation at the end 2008.

- ISTTOK is a very small device capable of AC Tokamak operation for several cycles. It is testing an innovative concept of Liquid Gallium Limiter.

This forms a coherent set of devices, with small, medium size (~1 MA) and large (> 3MA) tokamaks, allowing to address complementary issues and span a wide range of physics parameter to extrapolate to ITER.

A worldwide survey of the main Tokamak parameters is shown in Table 4.1. Devices under construction or for which construction has been approved are shown in Table 4.2. A further analysis of the opportunities under international collaborations is made in Chapter 6. An overview of the capital investment, operating cost and other resources related to the European Magnetic Confinement Devices is given in Table 4.3.

Table 4.1: Overview of parameters of main Tokamaks worldwide

Device	Operation Since	Country	Configuration	Steady-state capability	Ip MA	Bt (T)	R_0 (m)	a (m)
JET	1983	EU	Divertor		4-5	4	2.96	1.00
JT-60U	1991	Japan	Divertor		3	4	3.40	1.00
D III-D	1986	US	Divertor		1-3	2.1	1.66	0.67
Tore Supra	1988	France	Limiter	SC Magnets, Actively cooled first wall	2	4.2	2.4	0.75
FTU	1990	Italy	Limiter		1.6	8.0	0.93	0.30
ASDEX-Upgrade	1991	Germany	Divertor		1.6	3.1	1.65	0.50
MAST	1999	UK	Divertor		1.4	0.52	0.85	0.65
NSTX	1999	US	Divertor		1.4	0.6	0.85	0.61
EAST	2006	China	Divertor	SC Magnets	1 (1.5)	1.7	3.5 (4)	0.4
CMOD	1993	US	Divertor		1	6.0	0.67	0.22
TCV	1992	Switzerland	Divertor		1	1.54	0.88	0.25
TEXTOR	1981	Germany	Limiter / ergodic divertor		0.8	3.0	1.75	0.47
HL-2a	2002	China	Limiter		0.5	2.8	1.64	0.40
HT-7	1993	China	Limiter	SC Magnets	0.3	2.0	1.22	0.27
COMPASS	Soon	Czech Rep.	Divertor		0.35	2.1	0.56	0.2

Table 4.2: Main Tokamak parameters of devices under construction or for which construction has been approved. All these machines have D shape divertor plasmas and are superconducting.

Device	Foreseen Operation	Country	Ip (MA)	R_0 (m)	Bt (T)	a (m)
ITER	>2016	International	15	6.32	5.3	2.02
JT-60SA	2014	JA (in collaboration with EU)	5.5	3.16	2.7	1
KSTAR	soon	Korea	2	1.80	3.5	0.5
SST-1	soon	India	0.2	1.10	3.0	0.20
T15	2008	Russia	1	2.4	3.6	0.42

Table 4.3 Capital investment, operating cost and other resources related to the European Magnetic Confinement Devices*. Investments on JET are/were made with 80%/100% Euratom funding, and operation is paid 75% from Euratom funding and 25% from a Joint Fund from Associates. Devices in Associations and upgrades have been/are funded at up to 40% by Euratom and operation at 20%.*

Device	Association	original investment and upgrades (in Mio€)	date of investments and upgrades	actualised cost of investments and upgrades in Mio€) 2007	ongoing upgrades (Mio€)	proposed upgrades (Mio€)	period of proposed upgrades	Operation (costs of overheads not necessarily included in the operation costs of all devices)		
								number of operation days /year	yearly cost of operation Mio€ [15]	manpower for operation ppy [16]
Tokamaks										
JET	EFDA	657.2	1977-2007		60.4			64	65-70	545
Tore Supra	CEA	177.5	1984-2007	240.5		25	2013-2014	70	19	100
ASDEX-Upgrade	IPP	220	until 2007		12.70	17.9	2009 – 2013	60	9.75	85
FTU	ENEA	138.8	1983-2000			6	2009-2011	70	2.5	35
TEXTOR	FZJ	132	1981-2003			2	2008-2012	75	4.9	11
TCV	CRPP			110		10	2009-2013	80	6	45
MAST	UKAEA		1996-today	46.3		37	2010-2014	90	6.5[17]	60
Compass	IPP CR	27.5[18]			3.9		2008-2010	100	0.85	20
ISTTOK	IST	1,75	1990-1991					180	0.175	4
Stellarators										
W7X	IPP	372	1997-2014			50	after 2015	N/A	N/A	120
TJ-II	CIEMAT	40	1995	60		4	2008-2011	55	2.6	20
RFPs										
EXTRAP T2R	VR	3.5	1991-1999	4.1				60 to 120	0.55	4
RFX	ENEA	71	1985-2004					155	2.5	25

In addition to Tokamaks, the European fusion programme benefits from other magnetic confinement devices, such as stellarators and reversed field pinches.

[15] including manpower costs

[16] excluding scientific exploitation, but including diagnostic maintenance and data collection

[17] cost of operation will be 8 Mio€ if upgrade accepted

[18] 7,5 re-installation upgrade (incl. New building),
original investment coming from UKAEA

4.1.2 Stellarators

Stellarators provide a promising alternative to the tokamak. Their main advantage is intrinsic steady state operation without the need of externally driven toroidal current, and thus no current driven instabilities such as the occurrence of disruptions. As a consequence higher availability and lower re-circulating power are expected in stellarator power plants. However, the development of the stellarator is at least one generation of device behind that of the tokamak: in Europe, W7X will address in the second half of the 2010's parameter ranges equivalent to those explored by large tokamaks (JET, JT60U and TFTR) in the 1990's. However, stellarators have to face additional engineering challenges which relate to the complex structure and geometry of the coils and vacuum vessel, as well as to a more limited access to in-vessel components for remote maintenance.

There are two stellarators operating or under construction in Europe, TJ-II and Wendelstein 7-X.

TJ-II is a medium size classical stellarator with a flexible "Heliac" configuration for studying basic stellarator physics such as role of magnetic topology (shear & rational surfaces), electric field effects (with specialised diagnostics like heavy ion beam and Doppler reflectometry) and β-limits. Recently equipped with 2 MW NBI and using Li coating for density control, TJ-II enters a phase where ballooning stability limits and new divertor concepts (flux expansion) will be tested, aiming at supporting the relaxation of design constraints (impact on coil complexity & blanket space) in reactor relevant stellarators.

Wendelstein 7-X (W7-X) is a large optimized stellarator, designed for steady-state operation and currently under construction in Greifswald, Germany. The so-called HELIAS (Helically Advanced Stellarator) configuration is based on quasi-symmetry and minimized plasma currents combining improved neoclassical confinement properties (of the thermal plasma and fast particles) with a viable divertor concept. According to the present plan W7-X should start operation with inertially cooled plasma facing components in 2014. A second stage of operation with full steady-state capability will follow after about 4 years.

Table 4.4: Major stellarator experiments worldwide

Device	Start of operation	Country	Configuration	Steady state properties	R_0 (m)	<a> (m)	B_0 (T)
W7-X	*2014*	Germany	Quasi-isodynamic (HELIAS)	SC magnets, actively cooled first wall	5.5	0.55	3.0
LHD	1998	Japan	Heliotron	SC magnets	3.7	0.6	3.0
TJ-II	1998	Spain	Heliac		1.5	0.2	1.1
NCSX	*2011*	US	Quasi-axis-symmetric		1.4	0.32	1.7
HE-J	1999	Japan	Heliotron		1.2	0.15	1.5
HSX	1999	US	Quasi-helical		1.2	0.15	1.35
H-1NF	1992 (upgrade 1997)	Australia	Heliac		1.0	0.2	1.0
U-2M/3M	1989	Ukraine	Torsatron		1.0/1.7	0.12/0.22	1.3/2.4

4.1.3 Reversed Field Pinches

The Reversed Field Pinch (RFP) is a toroidal configuration with high magnetic shear, where the edge toroidal magnetic field reverses its direction. The magnetic field is mainly produced by the current flowing in the plasma. The toroidal coils provide only the weak outer reversed toroidal field, which makes the use of superconductors unnecessary for possible reactor applications. The strongly helical path of the current in the plasma allows a higher current and hence greater ohmic heating than in a

tokamak, and should lead to thermonuclear temperatures without any additional heating. However, the performance achieved up to now on RFPs does not extrapolate to power plant application.
Two RFP experiments are presently operated in Europe. *RFX-mod (Italy)* is a medium sized RFP (2.0 m, 0.46 major and minor radii), with plasma current as yet up to 1.5 MA (nominal 2 MA), 0.5 s pulse length, and temperature up to 1.2 keV. RFX-mod is equipped with the best real time control systems for active stabilization of plasma instabilities (192 active independent coils) ever built in a fusion device. *Extrap-T2R (Sweden)* is also equipped with a similar real time controller (128 active independent control coils). The device is a smaller RFP of 1.24 m major radius, 0.18 m minor radius, 0.3 MA plasma current and 0.1 s pulse length, also working on active MHD mode control and is a flexible test bed for new control algorithms.
Theoretically the ohmic RFP magnetic configuration may be laminar and chaos-free. During the last 10 years, the RFX-mod experiment has been coming steadily closer to this state by improving the magnetic boundary, and by increasing the plasma current. A full exploitation of RFX-mod in the MA plasma current range is necessary to understand major parameter dependencies for confinement.

4.2 Addressing the seven R&D Missions: analysis of the capabilities of the present European tokamaks and capabilities required from future devices (Satellite Tokamak Programme)

The above listed European devices, in close cooperation with a number of international devices operating around the world, have advanced significantly the development of the physics and technologies required for the use of fusion as a power source. However, sustained fusion power production with a positive overall power balance requires further research and development. These research needs have been summarized in terms of seven research and development missions[19] which cover: burning plasmas, reliable Tokamak operation, first wall materials & compatibility with the relevant plasma conditions, technology and physics of long pulse & steady state, predicting fusion performance and materials and components for nuclear operation. The capabilities of the present European tokamaks and requirements for future devices (the so-called Satellite Tokamak Programme, i.e. tokamak programme in parallel to ITER operation aiming at supporting ITER and preparing DEMO) to address these seven R&D Missions are analysed below. Key background elements are also provided by:

- the capabilities of ITER to address the R&D Missions (Chapter 2, Table 2.1)
- the technical requirements specific to the satellite tokamak programme (see Annex5).

Further details on the capability of the various devices, in particular plots against key physics parameters, are given in Annex 6. The capabilities of the non-European devices to address the seven R&D missions are also discussed in Chapter 5.

4.2.1 Mission 1: Burning Plasmas

In the area of Burning plasma studies, the strong coupling between the plasma parameters and the plasma self-heating and the possibility of collective instabilities generated by the large fraction of fast ions need to be addressed. In order to consolidate the physics knowledge and prepare the operation of ITER and the achievement of its key objective (to produce and control a burning plasma with power amplification Q=10 or more), it is important to develop heating strategies on present day tokamaks in conditions where fast ions can be produced and studied, in the most relevant range of parameters, i.e. at

[19] Chapter 4: Seven R&D Missions for the European Fusion Programme, The European Fusion Research Programme: Positioning, Strategic outlook and need for infrastructure towards DEMO January 2008

low ρ^*_{fast}[20], substantial β and β_N (i.e. in pressure conditions relevant to ITER/DEMO/Reactors) and with a large fraction of plasma pressure from fast ions.
Present devices can also address individual aspects of burn control. This requires a set of heating systems to simulate alpha particle heating and external heating used for control. Many aspects require plasma scenarios in plasma configurations similar to those foreseen on ITER.

Capabilities of Present European Tokamaks until ITER starts operation
JET is the only device capable of DT operation and it also has a large enough plasma current to confine the 3.5 MeV fusion born alpha particles; it is equipped with a unique and comprehensive set of burning plasma diagnostics. Heating by alpha particles can be studied up to a fraction of 10-15% of total heating power (cf. 67% or more on ITER), but it should also be noted that following the JET heating system upgrades in 2010, the fast alpha normalised pressure (β_{fast}) in JET scenarios will actually reach or exceed values in the ITER standard scenario, enabling experiments on Alfvén Eigenmodes. JET is also the best equipped to perform fast particle experiments outside DT operation, due to its auxiliary heating systems and capability of developing plasma scenarios under the conditions closest to ITER and the capabilities of ICRH acceleration of He^4 and Deuterium-Tritium operation. With its neutral beam heating and low magnetic field due to its low aspect ratio, MAST provides also a good capability to address fast particle issues (with high β_N and $v_{fast}>V_{Alfven}$ as for alphas on ITER). ASDEX Upgrade has a powerful ICRH system and innovative fast particle diagnostics, including the innovative Collective Thomson Scattering. Useful pieces of information can be obtained on other tokamaks equipped with ICRH heating (Tore Supra, Textor).

Capabilities required from future Tokamaks in parallel to ITER operation (satellite tokamak programme)
In support to ITER and DEMO, fast particle experiments will be possible in JT60-SA with high power Negative Neutral Beam Ion (NNBI); however the parameter space of JT60SA is limited as shown in Annex 6. ***Therefore JT60SA would be usefully complemented by a high current device in Europe, providing further data at lower $\rho^*_{fast.}$ And higher normalised fast particle pressure.*** This would also allow addressing integrated burning plasma physics issues such as dominant electron heating by the fast ions and cross-scale couplings with plasma turbulence.

4.2.2 Mission 2: Reliable Tokamak operation

The primary goal of Mission 2 is to support ITER in the preparation for operation, ensuring that all plasma discharge phases can be achieved in a reliable manner, avoiding or mitigating any abnormal events and reaching the fusion target goal. On the longer term, this Mission will cover the definition of the criteria for DEMO operation.

This mission covers a number of operational issues, among which wall conditioning, pulse management (plasma breakdown, current ramp, termination strategies), plasma control tools for avoidance of stability limits, in particular prediction of disruption occurrence and mitigation, plasma performance optimisation (NTM control) and wall safeguarding (ELM avoidance/mitigation).
High performance operation on ITER and in preparation of DEMO will require operation close to the stability boundaries, in particular maximising the plasma pressure for optimal fusion performance. Therefore, active control of instabilities is needed for operation in optimal conditions and mitigation strategies are required. JET and ASDEX Upgrade operate in a regime of β_N and ν^* (while JET can also approach ITER ρ^*) to study the β-limiting instabilities for ITER and DEMO, so that important experiments can be carried out in the area of research devoted to control and mitigation of instabilities (although all ITER-like parameters cannot be attained simultaneously). Experiments on disruptions can be carried out in all tokamaks, but the problems connected to Vertical Displacement Events (VDEs) and

[20] Alpha particle Larmor normalised to the plasma minor radius

halo currents can only be studied in shaped cross-section. However, the ITER and DEMO ion Larmor radius normalised to the plasma minor radius (ρ_*), a key parameter for stability, can not be matched in present experiments.

Capabilities of Present European Tokamaks until ITER starts operation

This demanding and broad programme can benefit from contributions from most tokamaks having:

- wall conditioning tools applicable on ITER/DEMO (ICRH on JET, ASDEX-Upgrade, with metallic first wall materials as on ITER/DEMO; ICRH and ECRH on TEXTOR; ICRH on Tore Supra with carbon first wall but with a permanent magnetic field),
- plasma breakdown and start-up tools (ECRH on ASDEX-Upgrade, MAST, TCV, ECRH and LHCD on FTU and Tore Supra),
- control and networking systems connected to the appropriate real time actuators (such as error field correction/magnetic perturbation coils as on JET, MAST and ASDEX-U; ELM control/mitigation systems, e.g on MAST; ECRH with divertor plasma as on ASDEX-Upgrade, TCV, etc.; pellet injectors in presence of ELMs as on JET, MAST and ASDEX-Upgrade) and sensors (infrared cameras such as on JET, ASDEX-Upgrade, MAST, Tore Supra and TCV),
- disruption mitigation tools (gas jet as on JET, ASDEX-Upgrade(here in conjunction with disruption prediction), MAST, TEXTOR, Tore Supra and TCV, or ECCD on ASDEX-Upgrade, TCV, Tore Supra, FTU).

Operation close to operational boundaries in conditions approaching ITER ρ_* can be carried out on JET; this capability will be improved further after the ongoing auxiliary heating upgrade (maximum heating power in the range 45 MW against 30-32 MW today). However, JET lacks the Electron Cyclotron Resonant Heating (ECRH) capability, important for controlling some plasma instabilities using localised current drive. Therefore, these studies need to be done in other devices, the most relevant capability being offered by (i) ASDEX-Upgrade with its ample heating power and versatile ECRH capability and its planned installation of in-vessel plasma control coils and (ii) TCV with its high power versatile ECRH systems.

Capabilities required from future Tokamaks in parallel to ITER operation (satellite tokamak programme)

JT60SA will be able to approach ITER parameters, as JET today, and will be a key tool to support ITER in particular for the development of strategies to operate close to operational boundaries; furthermore it will be equipped with ECRH a powerful control tool for core instabilities. However JT60-SA will have limitations, e.g. no ICRH system, and, more importantly a non-reactor relevant carbon first wall (at least in its first phase). In any case, the programme is very demanding and ***addressing Mission 2 requires a broad experimental capability*** as illustrated above. This should also benefit from an adequate European satellite tokamak.

4.2.3 Mission 3: First wall materials & compatibility with the relevant plasma conditions

Since the materials that can be used in the Tokamak first wall are relatively limited, the integration of the plasma parameters with the characteristics of the plasma facing components is a key issue for fusion research. Carbon I has the good properties of high operating temperature and low Z, but it suffers from significant plasma erosion and forms co-deposits with Tritium; therefore, leading to high fuel retention. The use of Carbon surfaces will be reduced to the minimum, if used at all in ITER. Carbon is ruled out for DEMO, due to the fast degradation of thermal conductivity and mechanical stability under irradiation. Therefore operation with metallic walls must be forcefully pursued.

Another key issue to be addressed is operation with high radiated power fractions. This is needed to account for the fact that power fluxes on divertor targets are limited by technology in the range 10, possibly up to 20 MW/m^2.

An integrated development of plasma scenarios applicable on ITER requires a set of experimental conditions which cannot always be met simultaneously. Plasma scenarios require operation at or close to the ITER plasma shape with X-point, the divertor and first wall being equipped with metallic plasma facing materials. A key parameter for the relevance of the divertor operation is the ratio of heating power to the major radius P/R (see Annex 6). Operation with a full metal wall is also essential to study and demonstrate acceptable fuel retention, ahead of the ITER DT phase. Wall temperature is a key parameter related to phenomena leading to tritium retention; therefore another key parameter is operation at temperatures relevant to ITER (about 200ºC) and at higher temperatures as foreseen on DEMO. Integrated plasma scenario studies must be complemented by the development of Tritium and dust diagnostics and removal techniques, which would benefit from Tokamaks equipped with remote handling tools for in-situ demonstrations. To develop a full metal wall tokamak operation, a number of additional dedicated PWI questions must be addressed under defined conditions in plasma experiments (in addition to HHF tests), such as melt layer stability, surface morphology/stability at high particle fluences, bulk tritium migration and retention, mixed material formation .

Capabilities of Present European Tokamaks until ITER starts operation

Operations closest to the full set of conditions are met on ASDEX-Upgrade (W plasma facing components) and JET (ITER combination of Be-W plasma facing components when the ITER-like Wall (ILW) will have been installed in 2010). In a further DT experiment, JET is expected to reach P/R in the range foreseen on ITER. A full set of remote handling tools is available on JET, and Tore Supra and FTU have some remote handling capability. TEXTOR and MAST can access and study heat fluxes of ITER and DEMO relevance. TEXTOR provides PWI test facility for medium sized components. Capabilities to contribute to this mission are offered by the other devices, as shown in Table 4.5 below.

Capabilities required from future Tokamaks in parallel to ITER operation (satellite tokamak programme)

JT60SA will have a relatively good capability to address divertor operation issues with a P/R approaching ITER parameters. JT60SA should also have remote handling capabilities. However JT60SA will not be equipped (at least in a first phase) with a relevant set of first wall materials (carbon is foreseen both in the divertor and on the first wall) and will therefore not provide a fully integrated demonstration of plasma scenarios compatible with ITER/DEMO plasma facing materials. Furthermore, and as will be pointed out in Chapter 6, the other superconducting tokamaks starting operation or under construction in the world are equipped with Carbon plasma facing materials, except EAST, which might use W plasma facing components after 2010. Given the importance and difficulty of the task, ***it would be essential that one or more tokamaks of sufficiently large size and plasma current equipped with divertor, metallic plasma facing materials (W), relevant P/R and remote handling capabilities operate worldwide. In any case, if another high current tokamak was to be built in Europe or elsewhere, it should directly address DEMO needs by avoiding C as plasma facing material.***

It is important to point out that preparation of DEMO operation will also require devices with plasma facing operating at high wall temperatures (400-500ºC), a set of conditions which will not be available on ITER. The programme should also start addressing this point.

4.2.4 Mission 4: Technology and physics of long pulse & steady state

This mission directly supports the second objective of ITER, which is to aim at demonstrating steady-state operation, with a target power amplification factor of Q=5. It also prepares for DEMO operation with full steady state operation. This is a challenging objective since the tokamak is not intrinsically steady state, which requires an integrated approach in the development of both the physics and technology aspects to provide an efficient current drive capability. Operation in long pulses is facilitated by high β_N operation, since it provides large bootstrap fraction and reduces current drive requirements. It is therefore important to develop an adequate plasma stability control strategy (Resistive Wall Modes)

using current profile control and active stabilization by means of control coils, for optimal plasma conditions at highest possible pressure. An important physics parameter for the demonstration of steady state operation is the ratio of the pulse length to the current diffusion time. Ultimately, full steady state is only reached when the plasma-wall interaction has reached steady conditions (temperature, particles recycling).
The optimisation of the current drive capabilities also requires further development of the auxiliary heating and current drive techniques, such as improvements to the Lower Hybrid Current Drive (LHCD) and Negative Neutral Beam Injection systems. In particular, adequate coupling of the LHCD waves in the various plasma conditions requires extensive research in Tokamaks, since the wave propagation depends on the specific scenario in question. Finally super conducting magnets is obviously a crucial technology for Tokamak steady state operation.

Capabilities of Present European Tokamaks until ITER starts operation

Unique pulse length and current profile control capabilities are offered by Tore Supra, the largest operating superconducting tokamak in the world and the only one equipped with actively cooled plasma facing components. However Tore Supra is a limiter tokamak with a circular plasma cross section and is not able to operate with an X-point or in the H-mode regime, as foreseen in ITER.
Integrated scenarios with long pulse capability are developed in the most relevant plasma conditions on JET (with its LHCD system; error field correction coils for RWM probing studies) and on ASDEX-Upgrade (although with presently limited capability of profile control), although for limited pulse lengths. This Mission, and in particular the exploration of high βN operation (required for achieving large bootstrap current), will benefit from the foreseen installation of control coils for active stabilisation of RWMs in ASDEX-Upgrade. The installation of LHCD capability on this device is also desirable, since it would provide an improved capability to develop plasma scenarios with current profile control and would also provide the possibility to test the LHCD coupler concept foreseen for ITER (so-called PAM) in the presence of an ELMy plasma[21]. Other tokamaks contribute to this Mission, in particular FTU (current drive studies with LHCD and ECCD at high density), TCV (electron transport barriers, high βN operation, ECCD). And MAST (NBCD including off-axis capability, high β_N operation, electron Bernstein wave current drive). The main aim of the proposed upgrade to MAST is to test steady-state regimes.

Capabilities required from future Tokamaks in parallel to ITER operation (satellite tokamak programme)

This area of tokamak research is very demanding and is likely to benefit from the largest effort worldwide. Furthermore, scenario development close to operational boundaries is not a prime mission for ITER and should be done in ‘satellite’ devices. A number of devices equipped with super conducting magnets are starting operation or planned to start in the near future, but with limited current capability (I_p <3 MA) such as KSTAR (South Korea), EAST (China), SST-1 (India) and T-15 (Russia). On a longer time scale, JT60-SA (I_p up to 5.5 MA) will provide a significant contribution to the field. It is foreseen to be equipped with active control coils; however it is limited in its H&CD capability (NBCD and ECCD, but no LHCD foreseen). ***No superconducting/long pulse divertor tokamaks exist in Europe, a weakness which could be compensated on one side by reinforced international collaborations (strong and extended participation in JT60-SA and collaborations with other superconducting devices) and from the other side by a European satellite tokamak (with adequate heating power and plasma facing materials to help address mission 3, and with very long pulse capability in order to allow addressing Mission 4 as well).***

[21] This PAM (“Passive Active Multijunction”) concept has been successfully tested on FTU but without ELMs and was proposed on JET in 2004 but the project was cancelled by lack of funding

4.2.5 Mission 5: Predicting fusion performance

The proposed devices and new devices under construction would not be able to close the gap between the JET operation space and ITER (see the plots shown in Annex 6). Therefore, extrapolation to ITER and from ITER to DEMO relies on basic theoretical understanding and modelling activities.
Extrapolation to ITER has shown the value of a step-ladder approach, with small size devices (e.g. Compass), medium size devices (e.g. ASDEX-Upgrade or DIII-D), large size devices (JET and JT60U) and geometrically dissimilar devices (e.g. MAST) providing a range of physics parameters to test theoretical hypotheses and models and/or scale empirically when no theoretical model is available. A variety of heating capabilities, diagnostics and other tools is also essential to develop experimentation on the full range of parameters required for testing theory.
While significant progress has been made in plasma physics and tokamak physics over the past decades, a number of areas still require significant efforts until a fully validated predictive capability, a "numerical tokamak", is available. H-mode pedestal physics has been pointed out as a top priority to be addressed, among several other areas of research.

Capabilities of Present European Tokamaks until ITER starts operation
All tokamaks contribute to physics studies which are complemented by experiments in other magnetic confinement devices such as Stellarators and reversed field pinches. Furthermore, low aspect ratio devices such as MAST extend the capability to test theoretical models. For example, it is important to study plasmas with different levels of electron and ion heating and momentum input. Some devices offer versatility for such studies, such as TCV with its present heating systems and proposed upgrades, ASDEX-Upgrade or Tore Supra. JET is the closest device to ITER, due to its size and large plasma current capability, therefore, the results from JET are crucial in preparation for ITER operation.
Exploration of difficult scientific issues requires adequately designed sets of plasma diagnostics, especially in the area of pedestal physics where spatial gradients are extremely high. A permanent and well coordinated effort in this area is therefore needed.
It is essential to upgrade computation facilities in line to the need for large scale simulations and model integration. This has to be accompanied by dedicated research in general fusion plasma physics theory to be incorporated into the modelling tools. This is further addressed in Chapter 5.

Capabilities required from future Tokamaks in parallel to ITER operation (satellite tokamak programme)
Although extrapolation from ITER to DEMO will represent a smaller step than from JET to ITER, a similar ***"step-ladder" capability must be retained when considering the full set of tokamaks worldwide*** in order to ensure an appropriate validation of theory. The large size device will be ITER, the medium size shall be "JET-class"device(s) (i.e. operating at Ip well above 3 MA) and the smaller size devices the 1 MA range tokamaks. Up to know only one device worldwide, JT60SA, is foreseen in the "JET-class", while several 1 MA-class devices are foreseen. ***The worldwide programme, and the European fusion programme in particular, would benefit from a second machine in the Ip ~4-5 MA or above category, complementing the scientific capabilities of JT60SA.***

4.2.6 Mission 6 & 7: Materials and components for nuclear operation; DEMO Integrated Design: towards high availability and efficient electricity production

One of the areas, which requires significant further development, is the operation of Tokamaks in a nuclear environment. ITER will be the first Tokamak to operate under a nuclear license, but still the overall activation of the ITER components will be small when compared to DEMO. The operation of a Tokamak in a nuclear environment poses a number of challenges. In particular, all components have to be engineered to be able to cope with the neutron fluxes during their life time. Complementary to a fusion irradiation materials test facility, a programme to test components, including a low Q tokamak (component test facility) is highly desirable as mentioned in Chapter 2.

A Tokamak based component test facility would address many aspects of a power station environment, i.e. it would simultaneously provide neutron wall loads with the exact fusion neutron spectrum; high heat fluxes; thermal cycling; thermal, electromagnetic and mechanical stresses; chemical erosion; etc. However, there are a number of challenges associated with the production of the required neutron fluxes (1 MW/m^2) for sufficiently long periods at high availability. In order to maximize the neutron wall load, a relatively compact device is required. A design based on a compact spherical torus has the advantage of providing the required neutron fluxes at moderate tritium consumption; but this approach still requires a detailed feasibility study and further physics and technology developments.

Another aspect related to an integrated approach to DEMO is the fact that technical constraints related to nuclear operation will limit the number of auxiliary systems (heating, diagnostics, etc.) and therefore put constraints on achievable plasma scenarios. The preparation of plasma scenarios with control tools applicable on DEMO (actuators and diagnostics) should start receiving a specific effort. The programme is expected to receive useful guidance from DEMO Conceptual studies, which would integrated all the scientific and technical constraints.

Programme on present and future tokamaks

A programme should already develop on present and future tokamaks to prepare plasma scenarios and operation in conditions applicable on DEMO. This goes beyond the present focus towards high βN operation, since development should also be guided by physics simulation taking account DEMO technical constraints.

The spherical tokamak concept offers an attractive option for a CTF. However potential show-stoppers should be addressed, in particular the capability to operate in very long pulses/steady state and ways to minimize/handle very high power fluxes in the divertor (typically 5 times the ITER power fluxes). MAST, with its proposed upgrade, could address these and also give valuable information for DEMO; the high power densities possible in spherical tokamaks mean that DEMO-relevant values can be achieved on today’s machines making them good test-beds for this aspect of DEMO.

4.2.7 Summary of the capabilities of the European tokamaks

The capability of the different European devices in addressing the programmatic missions 1 to 6 is summarized in table 4.5.

Table 4.5 Summary of the present capabilities of the European Tokamaks to address the 7 R&D missions The colour code is meant to highlight capability to address the Missions as follows:

Strong capabilities (in its parameter class)[22]

Medium capabilities (in its parameter class)

	3-5MA	**1-2 MA**						**< 0.5 MA**	
Mission	JET	ASDEX -U	Tore Supra	FTU	TEXTOR	TCV	MAST	Compass	ISTTOK
1 Burning Plasmas	DT operation ; Ip>3MA for α confinement; ICRH fast ions; He NB; n and γ diagnostics; Active MHD Spectroscopy	ICRH fast ions, fast ion loss and CTS diagnostics	ICRH fast ions		ICRH fast ions; CTS and fast ion CXRS		super-Alfvénic io ns simulate α-particles Active MHD Spectroscopy		
2 Reliable Operation	Large current disruptions; ELM control Coils; pellets for ELM control; LHCD and ICCD for NTM control; ICRH for conditioning; disruption mitigation	Pellet ELM control; ELM coils (2010 onwards); ECCD NTM control; ICRH for conditioning; disruption mitigation	ECCD; ICRH for conditioning; also with permanent field; disruption mitigation	ECRH MHD control	ELM Control coils ; disruption mitigation ; ECR and ICR for wall conditioning with magnetic field	ECRH	ELM Control coils. H-mode access. Disruption mitigation	ECRH Control coils	
3 Operation Compatible First Wall	Unique Beryllium capability; ITER-like wall (2010) with ITER-relevant plasma configuration; Full Remote Handling ITER-relevant wall temperature (200-300C)	Full Tungsten Wall with ITER-relevant plasma configuration	Actively cooled components; Remote handling under development ITER-relevant wall temperature (200-300C)	Molybdenum and Liquid Lithium Wall; Remote handling	Actively cooled&, heated PWI test facilities with unique diagnostic capabilities Wall temperatures up to 300C. Test of W limiters.		Flexible divertor tests at high heat fluxes. Unique ELM diagnostics		Liquid metal
4 Steady State Operation	LHCD for off axis current drive; Error field coils (RWM); Real-time control	Internal coils for RWMs (2010 onwards); Real-time control	Super Conducting magnets; actively cooled PFCs; LHCD for off axis current drive; Real-time control	LHCD for off axis current drive		ECCD	On- and off-axis NBCD Strong shaping		
5 Predicting Fusion Performance	ITER like shape; Closest to ITER range of parameters	ITER like shape, versatile heating (low/high momentum, ion/electron heating)	Electron Heating with low momentum input	Electron Heating with low momentum input	Ergodic divertor	ITER like shape; Electron Heating with low momentum input	ITER like cross-section at low aspect ratio breaking degeneracy in data; High β operation High res. diagnostics	ITER like shape	
6 Operation in Nuclear Environment	Tritium Capability Remote Handling								

[22] ***a dark green on a 1-2 MA tokamak is not equivalent to a dark green on JET***

4.2.8 Mid-term and long-term proposals

Table 4.6 Upgrades of Tokamaks/ new tokamak proposed by the Associations *(as mentioned above, cost of upgrades/new devices, except on JET, are funded at up to 40% under Euratom and for the rest by National governments from the involved Associations)*

Facility	***Assoc.***	***Proposed major Upgrades / new device***	***Main Purpose***	***Cost estimate and proposed time schedule***
ASDEX-Upgrade	*IPP*	*- Internal coils and stabilising shell* *- LHCD* *- possibility to upgrade Ip to 2.5 MA, major rebuild*	*- RWM and ELM control* *- test ITER-relevant LHCD coupler and vary q-profile for steady state plasma exploration* *- access parameter range closer to ITER*	*- 6.9 Meuro 2010-12* *- 11 Meuro, 2009-2014* *- not costed, longer term*
Tore Supra	*CEA*	*- ECRH 6MW* *- ICRH 1000 seconds*	*Core heating, fast particle physics, burn control, physics and technology of steady-state, long pulses at 10MW/m2*	*- 15 Meuro* *- 10 Meuro 2013-14*
TCV	*CRPP*	*- Gyrotron upgrade 1-3 MW* *- NB heating 2-3 MW*	*High density heating studies, fast ions, MHD study and control*	*10 Meuro (18 MFS) 2009-13*
MAST	*UKAEA*	*Upgrade includes 7.5MW additional NB, new divertors, new PF solenoid, new TF centre rod, pellets, 1MW EBW*	*Preparing the base for a spherical tokamak CTF* *Attain β and βN in excess of DEMO levels* *Attain divertor power density at DEMO levels*	*37 Meuro 2010-14*
FTU	*ENEA*	*Full power refurbishment, Real time steerable antenna, Collective Thomson scattering*		*6 Meuro*
FAST	*ENEA*	*New device, high field compact copper coil based.* *6.5 MA, 7.5T, R-1.8m*	***Satellite tokamak programme****: fast particle physics, W wall and innovative wall materials, closest to ITER parameters, dominant electron heating (ICRH, ECRH and LHCD)*	*280 Meuro* *6 years required for construction*

JET:

A decision has to be made soon concerning the near-term future of JET. The “ITER-like Wall” project, which consists in changing the first wall and divertor materials with the same mix as ITER (Be and W) will be installed from June 2009 onwards. Operation should restart at the end of Summer 2010. The JET programme is not financed beyond December 2010, while the exploitation of the ITER-like Wall would require a few years of operation. Furthermore the preparation of ITER would benefit from another DT experiment on JET.

Decision making is being prepared as follows: the EFDA Steering Committee, at its meeting of 10-11 March 2008[23], has recognised the scientific need for full exploitation of the ongoing enhancements and tritium operation, requested the EFDA Associate Leader for JET to prepare, and present to the EFDA Steering Committee a plan for the full exploitation of the on-going enhancements and tritium operation and requested the Commission to investigate the possibility of making adequate resources available within the fusion programme budget.

Proposed Upgrades and new tokamak(s):
The Associations have proposed a number of upgrades and one new device, which are presented in the "fiches" (see Annex 8). There is one proposal, FAST, for a new high current (>5 MA) tokamak to operate in parallel to ITER in complement to JT60SA.
These proposals are summarised in the table 4.6.

4.3 Addressing the seven R&D Missions: analysis of the capabilities of other magnetic confinement devices (stellarators and RFPs)

4.3.1 Addressing the seven Missions in Stellarator Research

- Mission 1: fast particle confinement and energetic particle mode in stellarator configuration. The main objective of stellarator research in this area is to demonstrate basic fast particle confinement which is compatible with the requirements of a fusion reactor (sufficient α-heating and low enough fast particle fluxes to first wall components). This also includes the study of fast particle modes in 3D magnetic field configurations. W7X, which is optimized for good fast particle confinement, will be equipped with a set of fast particle diagnostics and with ICRH to produce a population of fast particles. TJ-II is presently studying the effects of turbulence on fast particle stability.

- Mission 2 : Stellarator can support the mission on reliable tokamak operation in a number of areas, in particular wall conditioning. The requirement for wall conditioning applies to both tokamaks and stellarators. In W7-X methods such as low power RF heating will be tested and further developed.

- Mission 3: One of the main objectives of W7-X is to demonstrate divertor operation in a helical system with stationary heat fluxes of up to 10 MW/m^2. Based on the promising results from the predecessor W7-AS, W7-X will be equipped with an (helical) island divertor. Studies of alternative divertor concepts based on flux expansion are presently carried out in TJ-II. The plasma facing components of W7-X receiving high heat fluxes ($\geq$0.2 MW/m^2) will be covered with carbon. Assuming an operating regime can be established (e.g. the high density H-mode discovered in W7-AS) that overcomes the tendency of stellarators to accumulate impurities, high-Z materials (tungsten) as a first wall material should be tested at a later stage. An area where stellarators could in fact go beyond the objectives of ITER is high density (divertor) operation, as the Greenwald density limit does not apply to stellarators.

- Mission 4: The main advantage of stellarators is their capability to provide stationary plasma confinement without the need of external current drive and sophisticated control schemes to generate large bootstrap current fractions. Depending on the magnetic field configuration or optimization the rotational transform still varies with increasing beta (e.g. in NCSX the bootstrap

[23] This resolution was subsequently also adopted by CCE-FU in April 2008

current provides a significant part of the rotational transform while in W7-X the Pfirsch-Schlüter and bootstrap currents have been minimized). To fully exploit this steady state capability superconducting coils are required. The only two stellarator devices in operation or under construction which have super-conducting magnets are LHD and W7-X. In addition, after an initial experimental phase W7-X will be equipped also with actively cooled first wall elements for high power steady state operation.

A prerequisite for the development of steady state plasmas are suitable diagnostic and heating methods, including device control, data acquisition and analysis. For W7-X an ECRH system is being constructed which can deliver 10 MW over 30 minutes.

From the technology point of view both stellarators and tokamaks need superconducting coils for steady state operation. Although depending on the aspect ratio of the device and the bending of the conductors in the coils there are different requirements, the basic technology is comparable.

- Mission 5: Specific as well as more universal contributions are expected, such as the validation of a numerical stellarator, the physics of 3D effects (transport, etc.), edge physics and the comparison between stellarator and tokamak. For stellarators an important task is to establish a unified confinement scaling which up to now does not exist. In this context the understanding of the ratio of neoclassical to anomalous transport and their dependence on the level of neoclassical optimization has to be improved. This involves a comparison of as many different magnetic field configurations and optimization schemes as possible (comparison between devices and variations for a given device). In order to catch up with the tokamak development it is desirable to transfer in particular those results which are unique for ITER, such as fast particle physics, to stellarators. For this purpose suitable theoretical models, which include 3D magnetic field configuration, have to be (further) developed and validated. TJ-II is very well equipped with dedicated diagnostics allowing contributions to be made in the physics of turbulent transport (including momentum transport) and ballooning stability (using the possibility to modulate the magnetic well).

4.3.2 Addressing the seven Missions in RFP Research

During the last 10 years RFP research has made significant advances in understanding and improving confinement, has accessed regimes with plasma current above 1 MA and has reached the highest technological and scientific standards in the area of stability control with active coils. RFPs are now in a position to provide a contribution to three of the seven missions, to critical and unique tests of tokamak physics stretched to the extreme of low field and to fusion science generally, including scientific and technological issues important for ITER operation and full exploitation.

- Mission 2. RFP can contribute to the integrated development of plasma control tools in preparation to ITER operation, in particular for real-time control of MHD instabilities, which is a key point for reliable tokamak operation. Specific contribution concern real time actuators, the geometry and the coupling of active coils, and the design of mode controllers based on the physics of Tearing Modes, including the issue of tearing mode locking avoidance (as a tool to prevent disruptions in the tokamak).

- Mission 4. The RFP contribution to testing plasma control solutions, discussed for mission 2, is relevant also to mission 4. In addition, RFPs can offer a unique contribution to the physics and control of resistive wall modes (RWM) and to the development of advanced feedback models applicable to generic control systems. RFP experiments are an optimal basis for clean benchmarking in simple reference cases of MHD codes used to predict ITER RWM stability in AT scenarios and to study the interaction between RWM and tearing modes.

- Mission 5. The chances of building a reliable numerical tokamak increase if more physics and a broader parameter space are included. To this extent, RFX-mod can explore confinement at currents similar to those of large tokamaks, but with 10 times smaller B_T. Moreover, it can contribute to the mission with critical and unique tests of tokamak physics stretched to the extreme of low field. Examples are: (a) the origin of density limit: which is found very similar to that of tokamaks, but it is not disruptive. (b) turbulent transport in the core: to understand how electrostatic turbulence and transport behaves at strong magnetic shear and relatively large gyro-orbit. It will be a key tool in the validation of tokamak transport codes such as gyrokinetic codes. (c) edge turbulence and transport: with key contributions to the first principles understanding of pedestal transport properties. (d) Beta studies: since the nonlinear physics of what happens when different MHD stability limits are exceeded is general, (e) momentum transport, for which magnetic instability RFP explanations have made strong progresses. (f) Non-linear MHD, and the effect of stochastic magnetic fields, now key to several main tokamak subjects (e.g. ELM control).

Chapter 5
Addressing the seven R&D Missions:
High Performance Computing (HPC) facilities

The decision to build ITER in Europe has led EU to review its own resources and in particular to propose measures to have sufficient theory and modelling capability to adequately support and exploit the ITER project and design DEMO.
The complexity of plasma physics in magnetic fusion devices drives the need to develop appropriate physics-based models and state-of-the-art computer codes. Due to the fact that modelling codes should become more and more realistic, including various nonlinear effects as well as couplings across different spatio-temporal scales and even physics models, the requirements for High Performance Computing (HPC) resources are likely to increase rapidly. HPC facilities could lead to new scientific discoveries, and will certainly provide a better insight in key scientific issues and improve our ability to control tokamak and stellarator plasmas. Furthermore the development of materials toward DEMO and fusion power plant requirements requires a substantial effort in materials sciences supported by adequate modelling.

Two set of actions should satisfy the needs over the next 8 to 10 years:
- ***an Integrated Tokamak Modelling Task Force (for which a Gateway computer has just been built at ENEA-Portici) and a Fusion Materials Development Topical Group, were set-up and***
- ***currently significant HPC facilities, are being prepared with a 100 Tflops European HPC proposed to be financed in the near future and the IFERC computer (possibly 1 Pflops) is foreseen on the medium term in the frame of the Broader Approach with Japan (in the longer term 10 Pflops is likely to be needed to complete the numerical tokamak goal).***

The first steps in developing of a long term HPC policy were recently made with specialised ad hoc groups (see below under 5.2); a long term policy deserves to be further developed.

5.1 Review of the needs

The theory and modelling needs for ITER and DEMO can be divided into three main areas:

Supporting operation of ITER (R&D Mission 2)

In order to operate safely the ITER device a suite of numerical tools of different level of complexity is necessary. The ITER operational codes are not limited to plasma performance, but include also superconductor losses, dynamical stresses, plasma control, power supplies, heating systems, diagnostic systems, cooling systems, nuclear safety and radiation protection systems. The EU presently covers most items connected with ITER operation, however, part of the existing codes will have to be essentially redeveloped to satisfy the required QA for use during ITER operation**.** Efficient exploitation of ITER requires the creation of a common software platform and standards forcing compatibility of all Participant Teams codes. The same refers to definition of data to be shared by all codes used for ITER Operation.

Plasma physics (Integrated modelling; R&D Mission 5)

The ultimate purpose of integrated modelling is to provide magnetic fusion with tools able to simulate in a comprehensive fashion all relevant processes in a tokamak discharge and in the components of the confinement system. A complete set of simulation tools should be made available to enable design of “virtual tokamaks” in order to facilitate and secure the operation of present and future devices – just as “flight simulators” or “fission reactor simulators” do in their respective domains – as well as to train young people. Strategically, this means that integrated modelling aims at capitalising on the overall science and technology fusion know-how, as developed from first principles as well as from engineering activities. These integrated models are built-up from modules describing individual physics processes and plasma regions. Considering as a prominent example anomalous transport, the recent progress in computational power and the development of efficient gyrokinetic and gyrofluid codes enabled the realization of truly ab-initio modelling of plasma confinement in tokamaks. When fully developed, the simulation tool, in combination with data on component performance which can be used to optimise the availability, will also provide a means for optimisation of commercial operation of a power plant (e.g. maximise kWh production per wall lifetime cycle).

Materials research (R&D Missions 3 & 6)

Modelling materials response to radiation under fusion conditions is an extremely complex problem. Several orders of magnitude in time and length scales must be bridged in order to connect from the formation of defects, to the macroscopic changes in properties over time scales of the lifetime of the reactor, that is, from nanometers to meters and from picoseconds to years. The methodology currently used is based on a multiscale approach, where the most accurate, but computationally expensive, model is used to obtain basic parameters that are then transferred to other models that allow for larger systems and/or longer time scales. Such an approach requires the use of different codes with different computational requirements. They involve calculations based on density functional theory (DFT) and molecular dynamics (MD) simulations on a microscopic scale whereas long time evolution and global modelling requires the use of either kinetic Monte Carlo (kMC) or rate theory models together with discrete dislocation dynamics (DDD).
The European materials modelling program has been very successful in modelling radiation effects in pure metals in recent years. The next big challenge for the fusion materials community is presently the move to modelling microstructure evolution over the reactor lifetime for binary and ternary alloys. This can only be achieved with appropriate HPC resources.
Furthermore, the future design and licensing process of components using complex materials combinations requires a stronger linking of materials R & D and engineering, with a substantial modelling component needing HPC resources.

5.2 Steps taken to address the needs

In order to satisfy the needs, two set of actions have been taken: setting-up an Integrated Tokamak Modelling Task Force and a Fusion Materials Development Topical Group in Europe, and preparing adequate HPC facilities.

Integrated Tokamak Modelling Task Force

It has been recognised that first, a framework for theory and modelling (T&M) across the whole fusion community should be established, within ITER and beyond, and codes will need to be completely redeveloped to meet appropriate QA standards for the operation of ITER. In anticipation of these needs, Europe has set the Integrated Tokamak Modelling Task Force (ITM TF) under EFDA in 2003. Its objectives are fully aligned with ITER requirements and the Task Force works in close interaction with ITER and the international partners. It started defining standards and integrating codes. A first facility specified by the Task Force has recently been built at ENEA-Portici (the so-called "Gateway" computer which provides a platform for the development of the modelling tools and to access the Grid and various computing facilities).
Another role of the ITM Task Force is to ensure that the physics models implemented in the codes are validated on existing tokamaks, in order to turn them into reliable instruments of interpretation prior to the start of ITER operations.

Fusion Materials Development Topical Group

A Fusion Materials Development Topical Group has also been set-up at the end 2007 under EFDA. Among other objectives, this Topical Group shall promote and coordinate materials sciences and modelling in support to the development of structural and functional materials for fusion.

European HPC

Secondly, the acquisition of major High Performance Computer (HPC) is essential to progress in the development of the theory of plasma phenomena and material physics as mentioned in section 5.1. State-of-art codes in all these areas require large resources (CPU hours, storage capability) on massively parallel HPC systems. Two EU ad-hoc groups, headed by J. Connor[24] and F. Jenko[25], respectively, have concluded that the provision of the equivalent of 100 Tflops dedicated HPC power, for collective and coordinated use, would be the right next step to satisfy the scientific needs and also to take account of the fact that advanced modelling codes are becoming mature tools, which should be used not only by the code developers, but also a much larger community of scientists. The provision of this computer power has been deemed necessary to provide the EU fusion community with the tools to effectively plan and exploit ITER in a competitive environment, and to build up the integrated modelling capability needed for the design of DEMO and future power plants. Furthermore the availability of such a computer will facilitate collaboration in the EU fusion community on developing the necessary numerical models.
A proposal to satisfy these needs has been made in March/April 2008 to the EFDA-SC and CCE-FU[26]. This new computing facility (HPC for Fusion) is proposed to be installed at Jülich Supercomputing Center (JSC) and be operational at the beginning of 2009. It would be closely connected to the PRACE Project (Partnership for Advanced Computing in Europe) and DEISA initiative (consortium of 11 leading supercomputer centres in Europe forming the Distributed European Infrastructure for Supercomputing Applications). This would be facilitated by the leading role of JSC in these European undertakings. This project is presently under discussion.

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[24] EFDA(06)-32/4.13 *Report to the EFDA Steering Committee by* the Ad Hoc Group on 'Support measures for theory and modelling activities under EFDA'

[25] EFDA(07)-34/6.4 *Report to the EFDA Steering Committee by* the EFDA Expert Group on High Performance Computing

[26] EFDA(08)-36/4.3 Proposal "HPC for Fusion": a dedicated European High Performance Computer for Fusion Applications

High Performance Computing in the frame of the Broader Approach

Beyond the European fusion HPC, the IFERC facility will allow to extend computation to the next stage of model size, which, e.g. in the case of turbulence simulations, should include full size ITER simulations. The procurement of this supercomputer (shared between Europe and Japan in the framework of the "Broader Approach") is expected to start early in 2012.
Strong links should develop between the European HPC and IFERC. First of all, the European HPC will bridge the time gap and allow the European fusion community to prepare for the age of Pflop/s computing on IFERC; international collaborations should also develop during this stage. Secondly, the hyperscaling of fusion-codes has to make further significant progress before implementation on a Pflops computer, and this should be facilitated by the European HPC system.

Cost of High Performance Computing Facilities

A proposal is presently under discussion for a European Fusion HPC, 100 Tflops. The resources required for this project are as follows:

- investment: 9 Meuro, for the hardware, including 4 years maintenance (investment proposed to be made in 2008)
- operation: overall 1.6 Meuro and 12 ppy over the following 4 years (2009-2012)
- high level support team: overall 36 ppy over the same 4 year period.

A "state of art" high performance computer (expected to be around 1 Pflops) will be made available in 2012 in the frame of IFERC and operate during the 5 last years of the BA period. The costs, covered under the BA agreement, are as follows:

- Europe (France) provides an overall contribution of about 81 Meuro, of which about 55 Meuro for computer and 11 Meuro for peripherals;
- Japan provides the buildings, services and technical support about 127 Meuro.

Chapter 6
International collaborations

6.1 Overview of the mechanisms for International collaborations

Fusion research and development (R&D) has been part of the Community research programme since the inception of the European Atomic Energy Community Treaty (EURATOM treaty 1957). Fusion research has been included in all six EU Research and Technological Development Framework Programmes.

The development of International collaborations in the field of fusion research has been / is conducted in the frame of various international agreements and institutions:

- The IAEA's activities in fusion research with its three bodies, the International Fusion Research Council (IFRC), The Board of Editors of the journal Nuclear Fusion and the IFRC Subcommittee on Atomic & Molecular (A&M) Data for Fusion. The IAEA Physics section is responsible for Coordinated Research Projects among which the "Fusion research using small tokamaks" programme for the period 2004-2008.

- The International Energy Agency (IEA) & Fusion Power Coordinating Committee (FPCC). The IEA provides a framework for nine, major international collaborative programs (Implementing Agreements). The IEA Implementing Agreements carry out R&D activities which are relevant to either the ITER project or the "beyond-ITER" program. The FPCC co-ordinates and supports the activities of the IEA Fusion Implementing Agreements.

- The EURATOM fusion bilateral agreements which establish the framework of the EU international collaboration for fusion with Japan, Kazakhstan, Korea, Russia, Ukraine, USA. Bilateral agreements with EURATOM for research further include Argentina, Canada. Negotiations for fusion activities are ongoing with China and India. The International Tokamak Physics Activities (ITPA) are activated through these EURATOM bilateral agreements.

- The EU – ISTC & STCU which are two non-proliferation programmes for Co-operation on fusion energy with CIS institutions.

6.2 Magnetic confinement devices

The EU facilities, Tokamaks, Stellarators and Reversed field pinch, and their role in the fusion research programme were analysed in chapter 4. International facilities and non-EU facilities are here considered from the point of view of opportunities to address the 7 R&D Missions for the realization of energy production through magnetically controlled fusion.

Past and existing collaborations

The tokamak is the most advanced configuration being pursued by the magnetic fusion energy sciences program. The main tokamaks worldwide (operation or under construction) have been listed under Chapter 4; they provide a wide variety of designs and capabilities. The largest facility outside Europe is JT-60U in Japan, which has a number of performance capabilities comparable to JET but without tritium. The other largest tokamaks, foreseen or existing, are JT-60SA derived from JT-60U, KSTAR in Korea, EAST in China and T-15M in Russia, for superconducting devices and D-IIID and ALCATOR

C-MOD in the US for devices using conventional magnets. Besides those, several smaller tokamaks operate worldwide.

Smaller devices provide support to the bigger tokamaks, and help improving the operation as well as the scientific understanding; they can also provide useful tests of new diagnostics techniques and other technologies before application on larger less flexible devices; these are also useful tools in education and training. The collaborative exploitation of tokamaks maintains a lively network of collaborations between major contributors to fusion research and a broader plasma physics community around the world.

Collaborations between the bigger tokamaks around the world are developed either under ITPA (International Tokamak Physics Activities) or more directly through the bilateral cooperation agreements. For instance, JET carries out significant collaborative activities with the US, the Russian Federation and, to a lesser extent, with Japan, China and Korea. In the case of the Russian Federation, the collaborations with JET concentrate on diagnostic systems but also on electron transport, pellet injection, diagnostics software, and plasma modelling. In the case of the US, the collaborations have concentrated on diagnostics, on the ITER-like ICRH antenna, on plasma physics, but also on ITPA high priority coordinated experiments on confinement, MHD and edge physics, energetic particles, pellet injection, DT operation, Advanced Tokamak modes of operation. Joint/coordinated experiments are also regularly conducted between JET and JT60U. These collaborations amount to ~1ppy with the Russian Federation and ~3ppy with the US. Additional information on JET collaborations is given in appendix.

Significant collaborations exist also on ASDEX Upgrade (AUG), TEXTOR, MAST, Tore Supra, etc., as illustrated in the Fiches shown in the Annex 8.

Conversely European physicists are actively involved in the scientific programme of tokamaks in the US (DIII-D, C-Mod, NSTX) and in Japan. For instance, through the IEA Implementing Agreement on large tokamaks, EU scientists participate, in the US, to ICRH modelling and analysis, edge code benchmarking, ELM suppression studies. The European contribution to Alcator C-mod (MIT) relate to diagnostics (X-Ray imaging, CXRS etc.), modelling and theory (lower hybrid, MHD, ICRF, transport etc.) and experimental studies (high Z first wall, dimensionless similarity, SOL transport, etc.). The EU collaboration with NSTX (Princeton) concern diagnostics, ITPA coordinated experiments and collaboration with MAST on spherical tokamak. In Japan, EU physicists are involved in ITPA coordinated experiments on pedestal and on fuelling in JT-60U, and contribute to atomic data for impurity studies.

Opportunities to address the seven R&D Missions

Table 6.1 shows the opportunities for addressing the 7 R&D missions on the main International and non-EU tokamaks. The table also includes two Stellarators (LHD and NCSX) and a Spherical tokamak (NSTX) which are important as they explore physics regimes inaccessible to conventional tokamaks. The main characteristics of these different fusion devices are given in Annex 6.
The analysis of this table and of the annexes shows a number of facts. All new machines of significant size being built or planned to be built outside Europe have superconducting magnets, which permit the exploration of long plasma duration or quasi-stationary plasma states. Obviously superconducting devices open the path to long pulse scenarios, while copper-coil machines, in operation for many years and therefore very well equipped in terms of control systems, heating and current drive and diagnostics, are performing well in terms of detailed physics studies and reliable operation (Mission 2 and 5). Mission 4, exploration of long pulse and steady state, is the one that is best addressed by the various fusion devices either because they are able to sustain long pulse operation (superconducting magnets) or because they are inherently non disruptive (Stellarators). As pointed out in Chapter 4, this represents opportunities for fruitful collaborations between Europe and the parties involved in the ITER project.

Weaknesses are in the ability to develop plasma scenarios with relevant plasma facing components (Mission 3) and the capacity to experimentally simulate the fast particle dynamics (Mission 1 burning plasma physics, developing methods and tools for burn control).

	Tokamaks								**Stellarators**	
	15 MA	**4-5 MA**	**1-2 MA**							
7-M	ITER	JT-60SA	T-15M	EAST	KSTAR	D-IIID	C-Mod	NSTX	NCSX	LHD
	Int.	JA-EU	RF	CH	KO	US	US	US	US	JA
1			?	?	?					
2										
3										
4										
5										
6										
7										

Table 6.1: The major international and non-EU tokamaks, two non-EU stellarators (LHD and NCSX) and 1 non-EU spherical tokamak (NSTX) have been analysed in relation to the 7 R&D missions[27] (assuming potential capabilities are realised i.e. complete and adequate set of diagnostics, heating & current drive and other auxiliary systems becomes ultimately operational). It is uncertain which of the present US devices (D-IIID, C-Mod and NSTX) will still operate in parallel to ITER. The colour code is meant to highlight capability to address the Missions as follows:

Strong capabilities (in its parameter class)

Medium capabilities (in its parameter class)

6.3 Technology developments through international collaborations

The success of the international collaborations is due to a culture that has developed in the fusion community over a few decades. Communication and exchange of technical information have helped to test multiple technology paths that a single research program could not have afforded. It is essential that the on-going collaborations continue and develop further in a number of areas. As mentioned in Annex III of the “Positioning and Strategic outlook” document, the seven ITER large R&D projects constitute a good example of successful international collaboration. These projects carried out between 1992 and 2001 involved the then 4 ITER Parties (Europe, Japan, Russian Federation and USA). Further recent examples of successful international collaborations in the area of technology are provided below as well as opportunities for further developments.

R&D Mission 3:

Plasma Wall Interaction constitutes one of the highest priority areas, in relation to Mission 3, which can benefit significantly from plasma simulators, high heat flux test beds and other smaller facilities. In this area the recent collaborative activities with the Russian Federation on Edge Plasma Energy and Particle

[27] Disclaimer: this analysis was done without feed-back from the related laboratories, who might have a different judgment

Fluxes on Divertor materials has proved very fruitful. These include the experiments performed to characterize the W and CFC targets exposed to ITER-relevant loads in Triniti (Troitsk near Moscow) and experiments to further assess material damage on Be-clad and Be-coated plasma facing components in Kurchatov and Triniti, in Russia. These collaborations are complemented by important on going experiments in PISCES-B (University of San Diego, California, United States), for example with the studies of mixed materials deposits (Be – C). Collaborative research on bulk metallic plasma facing components and coatings is done on TEXTOR with Japanese partners (TEXTOR–IEA). New activities are expected to support ITER in the measurement and removal of tritium and dust, which could be subject of fruitful collaborations. With respect to plasma facing materials, the EFDA Plasma Wall Interaction Task Force has been a driving force in a number of collaborations, which are expected to be reinforced, with an increased participation of material science departments.

Fusion developments also need devices for thermal stress analysis including thermal shocks simulating the ELM dynamics (R&D Mission 3). Besides the EU facilities listed in Chapter 3, there are several High heat flux test or pulsed plasma power deposition stand (PPPDS) facilities worldwide. Of the former type are CEBTF (electron beam) and HTHEL (High Temperature He loop) at Southwestern Institute of Physics in China, JEBIS (electron beam) at JAEA in Japan which is a typical facility for high heat flux tests. Two more fusion relevant electron beam facilities are TSEFEY (electron beam) at Efremov in the Russian Federation and EB-1200 (electron beam) in the US; the former is being upgraded and formally part of the qualification program for the ITER divertor target while the later facility is being exploited for the qualification of the beryllium first wall mock-ups and is part of the ITER program. Amongst the PPDS are the QSPA and MK-200 (plasma gun) at Troitsk RF and the QSPA (plasma gun) at Kharkov Ukraine. These installations are used for ELM and Disruption simulations.

R&D Mission 4:

Because of the problems of reliability and performance, further developments of plasma heating and current drive systems are necessary (R&D Mission 4). These require the use of test beds.
In the case of neutral beams, test facilities have been built worldwide: MeV Test Facility (MTF) in Japan, Neutral beam test stand in India, Neutral beam test facility in Korea, IREK in Russia.
Gyrotron test beds have been developed in Japan, Russian Federation and US.
Ion cyclotron test stands are available in India, Korea, US (collaboration on the JET ITER-like antenna).
Pellet injectors test facilities exist at ORNL, US, on which collaborations have already been successfully conducted.
Long pulse operations rely heavily on superconductor magnet technology. Technology developments are performed worldwide (not only for the fusion program) in particular in Japan, China, Russia, and the US.

R&D Mission 6:

As mentioned several times, fusion power plants are to be built with suitable well-qualified and structural materials, with clearly defined functionalities and able to sustain extreme conditions, in particular for those materials close to the plasma. Therefore the materials development programme is among the top priorities of the fusion programme and constitutes the heart of R&D Mission 6.
As mentioned in previous chapters, the EVEDA phase of the IFMIF project is conducted in the frame of the Broader Approach agreement between EU and Japan. It is assumed that the construction of IFMIF should be achieved in the frame of an international collaboration, thereby preparing a key programme of characterization and qualification of materials for DEMO. Extending even beyond the IFMIF project, international effort should be encouraged that could help achieve the ambitious goals of the materials development programme. An implementing agreement under IEA is in place and provides a framework for a number of collaborative activities.

A few recent examples of fruitful contacts and international collaborations in the field of fusion materials are quoted here. These involve the Ultra-High Voltage Electron Microscopy Group (direct visualization of dynamics of radiation defects) of Osaka University, Japan, where extensive interaction has developed both in the fields of microscopy and modelling, the Shubnikov Institute of Crystallography and A.A. Bochvar Institute of Inorganic Materials (both Moscow, Russia), the Institute of Metal Physics (Ekaterinburg, Russia) where potentially significant line of work is emerging involving the investigation of irradiated FeCr alloys, the Institute of Physics of Strength of Materials (Troitsk, Russia), and the Hong-Kong Polytechnic University (Hong-Kong, China), where there have been significant new developments in the area of atomistic modelling of magnetic materials.
Concerning irradiation facilities, fruitful collaborations took place in the field of (i) neutron irradiation within the fast reactor BOR60 (RIAR Dimitrovgrad, Russia) to test materials up to high dose under displacement cascade regime, and, (ii) alpha particle (~60MeV) implantation via cyclotron at the Kurchatov Institute (Moscow, Russia) followed by microstructure examination and tensile tests.

Some of the non-EU neutron sources, on which collaboration on materials irradiation exist or could possibly develop, are listed here (this list is not exhaustive):

North America

Los Alamos Neutron Science Centre (LANSCE) [target/blanket materials]
Caveat: Abundant production of spallation elements changing the chemical composition of the materials. The situation is comparable to the one of irradiation in spallation target such as SINQ. This type of facility cannot be used for high dose fusion material qualification.

High Flux Isotope Reactor and Spallation Neutron Source, Oak Ridge [ceramic, structural material]
Can be used like the sources in the EU (BR2, Osiris and HFR) with a very high fast neutron flux. This reactor is multi-purpose: (i) isotope production, (ii) neutron science and (iii) material irradiation, thus rather busy. Collaborate with Japan in a program called JUPITER. Caveat: the transmutation production is not fusion relevant, only atomic displacement damage is produced.
The spallation source shows the same drawback as LANSCE, with in addition limited available volume.

Advanced test reactor (ATR Idaho National Lab.) breeder materials
Dpa/rate as high as 15-20 dpa/year achievable, volumes of irradiation rather large compared to MTR of fast reactors. Caveat: transmutation is not fusion relevant. Does not appear to be open to civil activities.

Kazakhstan

EWG 1 (IVG-1M) [studies: ITER structural material such as Be, graphite, steels]

Russian Federation

BOR-60, BN-600, BN-800 [low activation structural materials]
Have high fast neutron flux particularly well adapted for high dose (~20 dpa/year). Caveat: transmutation production not fusion relevant. BOR-60 open to external users.

SM [Be material; 4 to 74 dpa]

IGRIK [Multi pulse operation; structural materials] (Another pulsed reactor is **YaGUAR**)

Asia

Japanese material test reactor (JMTR, JAERI) [breeder material] and **JOYO**.
In both cases an important effort has been done to provide irradiation service with very well controlled parameters, especially the irradiation temperature with a special emphasis on high temperatures.

Hi-Flux Advanced Neutron Application Reactor, Korea [Blanket structure materials]
A ~5 dpa/year can be anticipated. Caveat: transmutation not fusion relevant.

Existing irradiation facilities only partly fulfil the needs for materials development and characterization for a DEMO reactor (about 150 dpa): Fission reactors have large irradiation volumes, appropriate neutron-flux, but neutron-spectra are not adequate (and in addition most are limited in useable temperature window and in-situ testing) while accelerators (e.g. self-ions, protons and/or He) have appropriate dpa to gas production rates, favourable conditions for in-situ tests, do not activate materials, but have small test volumes. It is to be noted that a large number of facilities for ion implantation combining Tandetron, Ion Implanter and/or Van de Graff, are in use worldwide but not necessarily for fusion material studies: dual or triple MeV ion beams in India and Japan, mono or dual beams (>100 keV) coupled to a TEM in Japan and in the US; dual keV ion beams coupled to a TEM in Japan.

In relation to DEMO relevant components, it is worth mentioning here that the Efremov Institute (Saint Petersburg, Russia) has successfully contributed to the fabrication and testing of He-cooled divertor mock-ups under heat-flux ~10 MW/m2.

Finally, the development of an adequate Tritium breeding capability for a fusion reactor requires a very significant effort. This subject is considered with great care by all the ITER parties, with the Test Blanket Module programme. Fruitful collaborations already exist and complementary technologies/facilities are used such as CATS, YAYOI, Tritium-Flibe permeation in Japan, RITM-F in Russia or STAR in the US, all for Tritium breeding, release/extraction and safety studies. Gas cooling loops are used in China while Liquid metal cooling loops (IFMIF related) are in use in China, Japan and Russia.

Chapter 7
Summary: proposals to fulfil the European Fusion development strategy

The document "Positioning and Strategic outlook" and the analysis presented under Chapters 2 to 6 of the present document identify a number of elements which are considered as top priority to achieve the objectives of the fusion development strategy, namely constructing ITER, making an effective use of this device and conducting the necessary complementary programme to allow a decision on DEMO construction to be made in about two decades from now. These top priority elements are summarised in the present chapter.

7.1 Proposal for the Core Programme

According to the approach proposed in Chapter 2, the Core Programme shall include:

- among all the present components the fusion programme, those that address scientific and technical issues that must be successfully resolved to fulfil the "fast track" requirements, and
- additional elements necessary to address gaps and risks identified in the "Positioning and Strategic outlook" document.

Furthermore the analysis made in chapters 3 to 6 identified gaps in facilities and other resources which need to be addressed to fulfil the programme objectives.
The Core Programme presented here includes all these elements. They are presented under two lists, ***(mainly) ITER related*** and ***(mainly) DEMO related***, while these two parts of the programme are strongly coupled and no such clear cut distinction can be made. For example, a number of elements in the first list contribute to DEMO, not just for the fact that ITER is meant to provide a significant contribution to DEMO, but key devices, such as satellite tokamaks, are meant to support ITER as well as complement it in preparing the physics basis for DEMO.

I- ITER related programme:

In addition to ITER construction and the related Broader Approach Projects (JT60SA and ***IFERC super computer), the Core Programme requires the following elements:***

- ***upgrades and additional technology facilities to support the European contribution to the ITER construction***

In relation to ITER construction, the main needs identified under Chapter 2.2 (Core Programme) and Chapter 3 is for a set of facilities supporting the European contribution to the ITER construction, including the Test Blanket Modules. New facilities / upgrades of existing Associations; facilities and their cost are detailed in tables 3.2a and 3.2b respectively.
The overall cost to be spent over the first ~five years of ITER construction is:

- *about 175 Meuro for new facilities (among which two main facilities: the Neutral Beam Test Facility, which is planned to be built in Padua, for about 100 Meuro, and the TF-PF windings cold tests (at 4K with low levels of current) for about 50 Meuro)*
- *about 12-15 Meuro for upgrades of existing Associations' facilities.*

This could be partially offset by the redundancies (although limited) in technology facilities identified in Chapter 3.

- ***in parallel to ITER construction: a strong Tokamak programme to support ITER and prepare its exploitation, comprising:***
  - ***an extension of JET,***
  - ***the continuation of key European tokamaks,***
  - ***some high priority upgrades on existing tokamaks to address identified risks (Steady State, Reliable Operation) by advancing tokamak physics,***
  - ***the further development of collaborations, in particular with new superconducting tokamaks outside EU.***

Table 7.1: Present capabilities of the main EU tokamaks operating today to address the seven R&D missions (proposed upgrades are not considered in this analysis). The colour code is meant to highlight capability to address the Missions as follows:

Strong capabilities (<u>in its parameter class</u>[28])

Medium capabilities (<u>in its parameter class</u>)

	3-5MA tokamaks	**1-2 MA tokamaks**					
Mission	JET	ASDEX Upgrade	Tore Supra	FTU	TEXTOR	TCV	MAST
1 Burning Plasmas							
2 Reliable Tokamak							
3 Wall and Plasma	(after Be/W wall installed)						
4 Steady State							
5 Predictive Capability							
6 Nuclear materials & components							
7 DEMO Integrated Design							

The European tokamak programme on the short and mid-term includes JET and a number of mid-size devices. Their capabilities to address the seven R&D Missions are summarised in Table 7.1 (from Chapter 4, Table 4.5).

It is proposed to extend JET beyond 2010 to fully exploit the ITER-like wall experiment and possibly conduct another DT experiment.

Besides JET, a sufficiently broad tokamak programme should be kept to address the objectives described in section 2.2.

To address the programme described in chapter 2, some upgrades on existing tokamaks should be conducted; the top priority items relate to risks identified in the "Positioning and Strategic outlook" document, in particular Steady State and Reliable Operation. Tokamak enhancements addressing these risks relate to ELM control (installation of magnetic perturbation coils on divertor tokamak(s)) or to the test of an ITER-relevant LHCD coupler on divertor tokamak(s) in ELMy plasma conditions. Other important upgrades could be made, in particular to further address steady state operation and control (additional heating and current drive power). It is also advisable that more technical tests are carried out on tokamaks in support of ITER.

[28] a dark green on a 1-2 MA tokamak is not equivalent to a dark green on JET; in particular parameter plots in Annex 6 show the achievable ranges of parameters on various classes of tokamaks.

The tokamak programme already benefits from significant international collaborations. These collaborations should further develop, with more pro-active, top-down steering from EFDA. They would provide mutual benefit to EU and other Parties. For example, increased participation of collaborators on JET, in particular from ITER Parties, would be fruitful from the scientific point of view as well as for training international teams to jointly prepare experimentation on ITER. Europe would also benefit from developing further its collaborations with the new superconducting tokamaks which start operation in Asia and will offer new experimental capabilities, while these new devices would benefit from the expertise of European scientists.

Resources:
- The cost and resources for the operation of JET and other European tokamaks are provided in the Fiches in Annex 8 and summarised in Table 4.3.
- The high priority upgrades on tokamaks could cost around 40-50 Meuros[29] over the coming 4-5 years, although up to about 96 Meuros of upgrades have been proposed by Associations as shown in table 4.6 (including the MAST upgrade which is also mentioned below in relation to the CTF).

- ***in preparation of a satellite tokamak programme to operate in parallel to ITER exploitation and complementing JT-60SA:***
  - preparing the future operation of at least one Europan tokamak in the 1-2 MA Class (among existing device(s) with refurbishements/upgrades),
  - in order to reduce risk and fill programmatic gaps, launching European studies of a high current tokamak (in the range 5MA or higher) and considering the FAST proposal as a possible option for such a device,

Further joint use of JT60SA beyond the end of Broader Approach is also highly desirable and should be discussed with Japan at an appropriate time[30].

A set of tokamaks will be required in parallel to ITER operation, to support ITER physics, keep a broad physics parameter range to extrapolate to DEMO, test rapidly new concepts, complement ITER experimentation in addressing specific DEMO needs (see Table 2.1) and participate to training.
As pointed out in Chapter 4 and Annex 5, it is unlikely that a single JET-class (>3MA) tokamak worldwide will be sufficient to support ITER and prepare DEMO. The JT60SA project has been launched in the frame of the Broader Approach agreement between EU and Japan for the purpose of being one such device. All efforts shall be made by Europe in collaboration with Japan for a successful construction and joint exploitation of JT60SA. As mentioned in Chapter 4, it is highly desirable to discuss with our Japanese partners a second phase of JT60SA with tungsten plasma facing materials in order to address more appropriately the development of plasma scenarios and tokamak operation in DEMO relevant conditions.

As shown in chapter 4 and Annexes 5 & 6, there is a strong case for strengthening the Satellite Tokamak programme, with a second high current (> 3 MA) device, possibly in Europe:

- Firstly, JT60SA being the only high current device presently foreseen, the worldwide tokamak programme shows a weakness in the range of physics parameters which can be achieved to support ITER and prepare DEMO physics and plasma operation.
- Secondly, the present set of planned devices worldwide will not address all programmatic requirements. In particular there are weaknesses in the ability to address Missions 1 and 3 (see Table 7.2). With respect to Mission 1, JT60SA could be usefully complemented by a high current device providing further data in a range of fast particle physics parameters closest to ITER and DEMO. Addressing the development of plasma scenarios compatible with DEMO relevant wall

[29] Euratom contribution assumed to be up to 40%, the rest being financed by national governments

[30] Joint use of JT-60SA is currently agreed until 5 years after full performance is achieved

materials (Mission 3) requires that one or more divertor tokamaks of sufficiently large size (possibly a few MA) equipped with tungsten plasma facing materials are available worldwide; such device(s) should also be able to operate in the same range of Power/Radius as ITER/DEMO to develop relevant divertor operation. These elements should be taken into consideration if another high current tokamak was to be built.
- Finally, plasma scenario development close to operational boundaries, as required for a high efficiency compact DEMO device, is not a prime mission for ITER and should be done in 'satellite' devices.

A device of this category would be a central element of the programme, allowing Europe to continue playing a key role in the development of tokamaks.
Besides these larger devices, it is also necessary that Europe operates in parallel to ITER at least one tokamak in the 1-2 MA class.

Resources:
- *The FAST project proposed by ENEA is a possible option for a European high current satellite tokamak. It addresses a number of the above mentioned requirements[31]. ENEA provided a cost estimate of about 280 Meuro[32].*
- *The cost of operation of present 1-2 MA class tokamaks is given in table 4.3.*

Table 7.2: Potential capabilities (assuming a complete and adequate set of diagnostics, heating & current drive and other auxiliary systems becomes ultimately operational) of ITER and the new non-EU tokamaks foreseen to operate in parallel to ITER. It is uncertain which of the present US devices (presented under Chapter 6) will still operate in parallel to ITER; therefore no US device appears in this table. The colour code is meant to highlight capability to address the Missions as follows[33]:

Strong capabilities (<u>in its parameter class</u>[34])
Medium capabilities (<u>in its parameter class</u>)

	ITER (15 MA)	**Satellite tokamak(s) (3-5MA)**	**Medium size tokamaks (1-2 MA)**		
7-R&D Missions		JT-60SA	T15M	EAST	KSTAR
1 Burning Plasma			?	?	?
2 Reliable Tokamak					
3 Wall and Plasma					
4 Steady-State					
5 Predictive capability					
6 Nuclear materials & components					
7 DEMO integrated design					

[31] *In view of optimising the satellite tokamak programme, it is understood that the proponents are ready to discuss both the objectives and parameters of the device*

[32] *Euratom contribution assumed to be up to 40% of that amount, the rest being financed by national governments of the involved Associations*

[33] Disclaimer: this analysis was done without feed-back from the related laboratories, who might have a different judgment

[34] a dark green on a 1-2 MA tokamak is not equivalent to a dark green on JT60SA and even less on ITER.

- ***in the frame of a long term High Performance Computing (HPC) policy for theory and modelling of fusion plasmas, in addition to the IFERC computer provided under Broader Approach, and in preparation for its use, a European HPC should be built in the short term.***

The first steps in developing a long term policy in the area of High Performance Computing for fusion were recently made with specialised ad hoc groups; a long term policy deserves to be further developed. However, and as mentioned in Chapter 5, two set of actions should satisfy the needs over the next 8 to 10 years:

- an Integrated Tokamak Modelling Task Force and a Fusion Materials Development Topical Group were set-up and
- significant HPC facilities, are being prepared with a 100 Tflops European HPC, proposed to be financed in the near future, and the IFERC computer (possibly 1 Pflops) foreseen on the longer term in the frame of the Broader Approach with Japan.

While the IFERC computer will be financed under the Broader Approach, the European Fusion 100 Tflops HPC remains to be decided and financed. The resources required for this project are as follows:

- *investment: 9 Meuro, for the hardware, including 4 years maintenance (investment proposed to be made in 2008)*
- *operation: overall 1.6 Meuro and 12 ppy over the following 4 years (2009-2012)*
- *high level support team: overall 36 ppy over the same 4 year period.*

II- DEMO related programme:

In addition to the successful exploitation of ITER and accompanying devices, to the on-going DEMO technology programme (materials and blanket development and R&D for helium-cooled divertors), to the IFMIF EVEDA (Broader Approach) and to the concept improvement programme (stellarator including W7X construction and exploitation), the preparation of DEMO requires:

- ***additional R&D programme to reduce risks in the IFMIF-EVEDA***

It is proposed to minimise risks on IFMIF by considering additional tests & qualification in preparation for construction. Details are provided in Annex 2. Four actions have been identified

- D beam-Li interaction R&D and other liquid Li R&D,
- Rectangular beam shaping demonstration,
- Development of radiation resistant diagnostics for the beam, target and test cell, and
- Development and test of alternative superconducting structures.

Total cost about 25 Meuro to be spent over the coming 4-5 years.

It is also recommended to start without delay the preparation of the decision to site and build IFMIF. *Given that IFMIF is on the critical path for the preparation of DEMO (see “Positioning and strategic outlook” document), the preparation of the IFMIF decision process should aim at allowing a decision to be made immediately after the end of the IFMIF EVEDA phase. This preparatory work should include discussions to possibly enlarge the number of international partners, as well as the definition of site specifications, the pre-selection process of candidate site(s) and any other technical action required to support the decision making.*

- ***European DEMO conceptual studies***

In order to meet the objective of being ready to decide on DEMO construction by the end of the first experimental phase of ITER (i.e. in about 20 years from now) a DEMO design group should be formed as soon as resources permit (human and financial). This group should involve substantial industrial expertise. The objectives of this work should be:

- In a first stage, narrowing down technical options for Fusion Power Plants and DEMO; defining first stage of R&D programme.
- In the following stages: conducting DEMO conceptual design studies and supporting R&D.

The R&D will, inter alia, deal with heating and current drive systems, in-vessel components, reliability and maintainability, ***thereby addressing technical risks identified for the development of fusion power (steady state technologies; overall reliability/availability).***

In support of this programme, a number of upgrades of important facilities in the Associations, are recommended to minimise risks (details are shown in Chapter 3, section 3.2)

The DEMO conceptual studies should also provide essential further guidelines to the tokamak programme, including experimentation on ITER.

The resources needed are as follows:

- *For the DEMO conceptual study and supporting R&D are estimated as follows (see details in Annex 7):*
    - *~380 ppy over 8 years (plus support: Drawing Office etc.) for the central team and external expert support; cost ~60 Meuro (assuming 150keuro/ppy)*
    - *180-200 Meuro R&D over 8 years.*

  After this 8 year period information required to start the Engineering Design Phase should be available.

- *Cost of upgrades of Associations' facilities (Chapter 3, Table 3.2b) : 70-80 Meuros (some of these upgrades might reduce the cost of the DEMO R&D programme mentioned above).*

- *A significant R&D programme will be needed in parallel to the DEMO Engineering Design (in broad terms, running in parallel to ITER experimentation), which can only be defined and costed once the conceptual studies are sufficiently advanced.*

- ***strengthening the Fusion Materials Programme, in order to address one of the main risks of the fusion programme***

To address another technical risk identified for the development of fusion power (Materials), it is proposed to strengthen significantly the Fusion Materials Programme.

As indicated in section 2.3 and Annex 3, a reinforced materials science programme, focussed on the development of experimentally validated predictive modelling tools, should be launched, with the objective of contributing to the optimisation of the IFMIF programme and to reliable extrapolation from the IFMIF data to the broader operating conditions of DEMO. Validation using dual and triple ion beam facilities to create displacement damage, plus helium implantation using, e.g., cyclotrons should forcefully address the critical issue of the effects of the higher (than fission) rates of helium accumulation peculiar to fusion. Novel guidelines are also expected to arise from modelling and validation programmes for new lines of heat and radiation resistant steels, to mitigate the present risk of only having oxide-dispersion-strengthened steel for high temperature applications. The effort should cover structural as well as functional materials (like breeding ceramics, permeation barriers or insulators).

Material development also benefits from international collaborations, in particular under the umbrella of IEA agreements, which could further develop.

The additional resources needed correspond to about:
- *1250 ppy over 20 years (on top of the less than 100 ppy /year in the programme today)*
- *80 Meuro for testing (on top of the level of the present programme)*

Which corresponds to a total additional cost of about 250-270 Meuro over 20 years.

- ***preparing for a possible Component test facility (CTF) with***
  - ***CTF feasibility studies in the frame of the DEMO conceptual design studies,***
  - ***a CTF physics programme (the upgrade of MAST would address this issue);***

To further address the risks and gaps identified for the development of fusion power, it is proposed to include among the objectives of the above mentioned DEMO design group the following tasks in preparation for a possible CTF:

- in a first stage Tokamak based Component Test Facility (CTF) feasibility assessment;
- if the outcome of the first stage is positive, Conceptual design of a CTF and supporting R&D.

Details are provided in Annex 7.

Furthermore R&D on Spherical Tokamaks should be completed in preparation of decision making on CTF. An Upgrade of MAST would allow addressing the issues related to the physics and plasma operation of a spherical tokamak based CTF.

Resources:

- *Overall the resources needed for the CTF and Stellarator Power Plant studies (mentioned below) and supporting R&D are estimated as follows (see details in Annex 7): 10 Meuro over 8 years (mostly manpower).*
- *the MAST upgrade (see Table 4.6 and Fiches in Annex 8) is estimated to cost about 37 Meuro. The investment is proposed by UKAEA to be made over the period 2010-14.*

Moreover, the stellarator programme, as part of the concept improvement programme, must be vigorously pursued. This requires:

- ***the completion and exploitation of Wendelstein 7-X***

The stellarator constitutes the main alternative to the tokamak addressing identified risks (steady-state operation and overall reliability). This includes intrinsic steady-state properties, an extended operational space towards higher density (no Greenwald limit) and a more benign plasma behaviour at the operational boundaries (no current-driven instabilities, no disruptions), resulting in the absence of strong external current drive and less demanding control requirements. As a consequence stellarators promise more reliable operation and lower re-circulating power.

However, stellarators have to face additional engineering challenges which relate to the complex structure and geometry of the coils and vacuum vessel, as well as to a more limited access to in-vessel components for remote maintenance.

The main elements of the further development of the stellarator as an alternative concept are:

- the development of an integrated steady state plasma scenario for W7-X and comparison with other stellarator devices to converge towards an optimum magnetic field configuration,
- the development of numerical tools (numerical stellarator) for an improved extrapolation to a power plant, incorporating also results from ITER (and here in particular alpha heating physics).

- ***to further reduce longer term risks, launching Stellarator Power Plant conceptual studies in the frame of the DEMO conceptual design studies.***

The engineering feasibility of stellarator power plants should be studied on an appropriate time scale. This study is proposed to be added to the objectives of the DEMO conceptual study group, although as part of longer term objectives.

7.2 Desirable complementary programme elements that support the Core Programme

As mentioned in Chapter 2.4, there are complementary elements which are highly desirable not only to support the Core programme, but also because they provide a capability to i) address critical issues in the Core programme as they arise, ii) strengthen the scientific and technical understanding important for the progressive development toward a viable and economic fusion reactor, iii) generate and gather new ideas outside the mainstream thinking.
For this purpose, a portfolio of basic smaller scale activities should be kept in the programme.

7.3 Conclusions

Having analysed how facilities can address the objectives of this Core Programme, the following proposal is made:

I- Programme mainly focused towards ITER:

The Core Programme requires, in addition to ITER construction and the related Broader Approach Projects (JT60-SA and IFERC super computer):

- *upgrades and additional technology facilities to support the European contribution to the ITER construction;*
- *in parallel to ITER construction: a strong Tokamak programme to support ITER and prepare its exploitation, comprising:*
  - *an extension of JET,*
  - *the continuation of key European tokamaks,*
  - *upgrades on existing tokamaks to address high priority risks (Steady State, Reliable Operation) by advancing tokamak physics,*
  - *the further development of collaborations, in particular with new superconducting tokamaks outside EU;*
- *in preparation of a satellite tokamak programme to operate in parallel to ITER exploitation and complementing JT-60SA:*
  - *preparing the future operation of at least one European tokamak in the 1-2 MA class (upgraded existing device(s)),*
  - *in order to reduce risks and fill programmatic gaps, launching European studies of a high current tokamak (in the range 5MA) and considering the FAST proposal as a possible option for such a device,*

- considering further joint use with Japan of JT60-SA beyond the end of Broader Approach;

- *in the frame of a long term High Performance Computing (HPC) policy*[35] *for theory and modelling of fusion plasmas and materials:*
  - in addition to the IFERC computer provided under Broader Approach, and in preparation for its use, approving as soon as possible the proposed European HPC.

II- Programme mainly focused towards DEMO:

In addition to the successful exploitation of ITER and accompanying devices, the on-going DEMO technology programme (materials and blanket development and R&D for helium-cooled divertors) and the IFMIF EVEDA (Broader Approach), the preparation of DEMO requires:

- *considering an additional R&D programme to reduce risks in the IFMIF-EVEDA and thus prepare for a successful fusion materials development programme;*
- *starting without delay the preparation of the decision to site and build IFMIF;*
- *launching European DEMO Conceptual Studies with supporting R&D;*
- *increasing the funding for facilities needed for DEMO oriented technology R&D, in particular heating and current drive systems, in-vessel components, reliability and maintainability (remote handling and design of components), thereby addressing identified technical risks;*
- *preparing for a DEMO engineering Design Study that should follow the Conceptual Studies after about 8 years*[36]*;*
- *increasing the resources for the Fusion Materials Science and Technology Programme, thereby addressing one of the main risks of the fusion programme (Materials);*
- *preparing for a possible Component Test Facility (CTF), and thus addressing identified risks (materials; in-vessel components for tritium-breeding; reliability/availability), with*
  - CTF feasibility studies in the frame of the DEMO conceptual design studies,
  - a CTF physics and technology programme (the upgrade of MAST would address the physics issues).

Moreover, the stellarator programme, as part of the concept improvement programme, must be vigorously pursued. This requires

- *completing the construction of, and exploiting, WENDELSTEIN 7-X;*
- *launching at a later stage, Stellarator Power Plant conceptual studies in the frame of the DEMO conceptual design studies.*

A resource loaded planning showing the cost and time scales for this set of proposals is provided below..

In addition to this Core programme, it is highly desirable to keep a range of facilities i) to address critical issues in the base programme, ii) to strengthen the scientific and technical understanding important for the progressive development toward a viable and economic fusion reactor, iii) to generate and gather new ideas outside the mainstream thinking, which stimulate and foster the essential elements of innovation and creativity.

The programme outlined in this paper would provide a sound basis for the timely development of fusion power, with Europe in a leading position.

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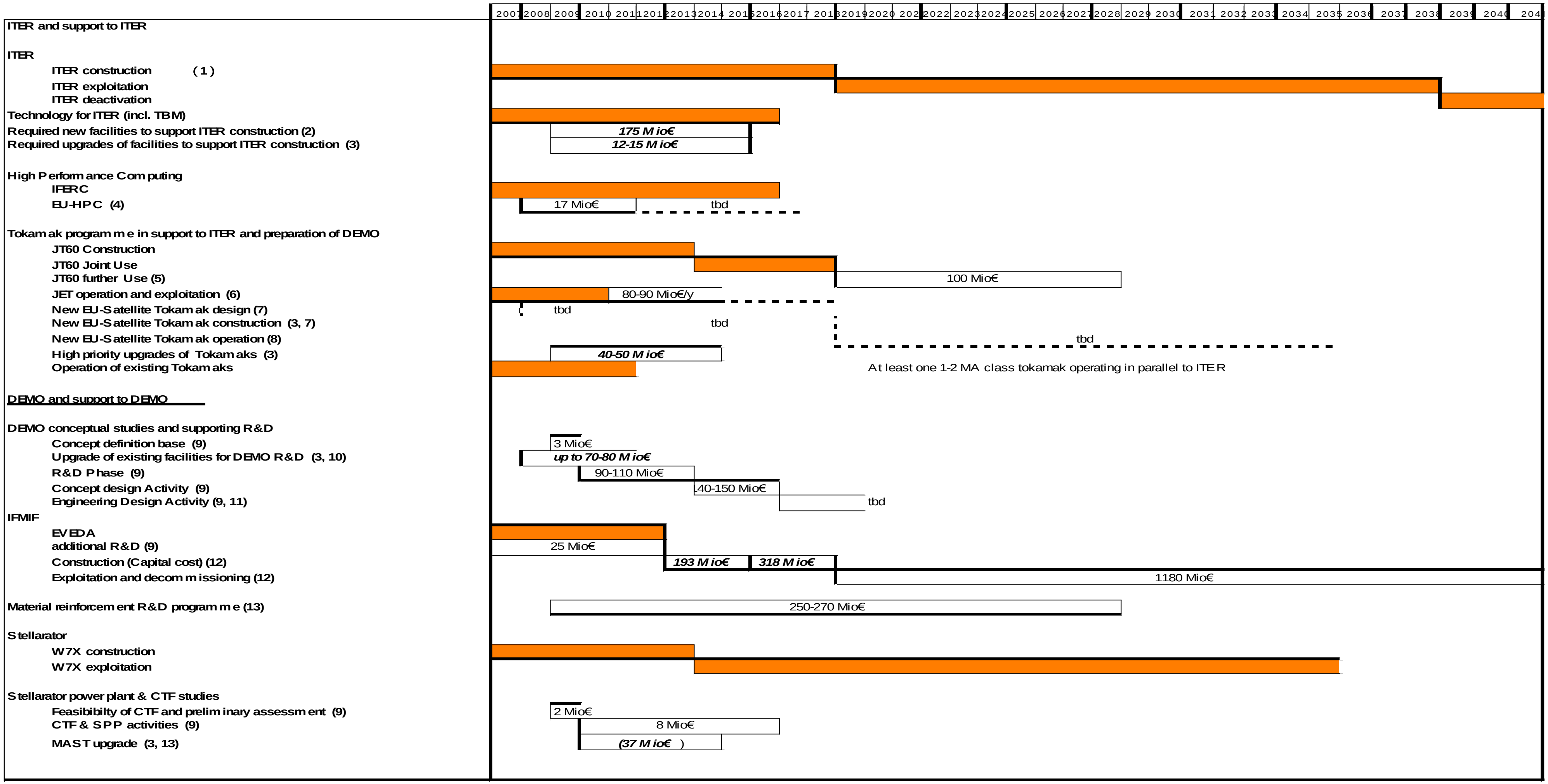
[35] *The first steps in developing this policy were recently made with specialised ad hoc groups; a long term policy deserves to be further developed.*

[36] *this will comprise a significant R&D programme that will require additional facilities and resources; however, this programme will only be costed once the Conceptual Study will be sufficiently advanced.*

RESOURCE LOADED PLANNING PROPOSED FOR THE EUROPEAN FUSION PROGRAMME



The red bars correspond to elements of the programme already decided

The white Bars are the elements proposed to complement the Core Programme

1 Planning under elaboration by the ITER Organisation
2 including the neutral beam test facility in Padova and the coil tests / EC contribution tbd
3 EC contribution up to 40% tbc
4 Cost of investment, operation and high level support; EC contribution 40% to investment and 50% to operation
5 Possible European contribution extrapolated from previous operation period (tbd between EU & JA)
6 Overall EC contribution 60-70 M (75% of operation; 40-100% of other costs)
7 Costs tbd; the capital cost of FAST, which is a possible option, is quoted by ENEA to be about 280 Mio€

Figures in ***Bold Italics*** are for Investment cost only (2008 Euros)

Figures in normal font are for total cost (manpower, investment, operation or R&D when applicable) (2008 euros)

8 Operating cost tbd, EC contribution up to 20% tbc
9 EC contribution tbd
10 Some of these upgrades could reduce the cost of the DEMO R&D programme mentioned under the next line
11 Cost of Engineering Design and related R&D will be an outcome of the Conceptual Design
12 EC contribution assumed to be 50% of total cost (F4E), construction under international collaboration
13 EC contribution 20-40% tbc
14 could be partially overlapping with high priority upgrades of existing tokamaks (above)

ANNEX 1
Addressing the key milestones proposed in the Positioning and Strategic Outlook paper

The set of milestones proposed in the Annex VII of the Positioning and Strategic Outlook document are repeated for convenience in this Annex. To a large extent they address the "gaps" indicated in Annex IV of that earlier paper, as can be seen from the table below. Most gaps will need input from more than one milestone. At a later stage, e.g. in the frame of the DEMO design team, further analysis will be needed to determine whether they are adequately covered.

In the tables below, as and if appropriate, we indicate the type of facilities required to meet the milestone: (a) magnetic confinement facilities (as described in section 4 and Annex 6 of the present paper) that will be used to address these, including ITER; (b) technology facilities (**for details see** Annex4, Table A4.1), (c) High Performance Computing facilities (*HPC*), and/or (4) other resources (e.g. DEMO study group).

The right-hand column indicates if the facilities are sufficient to make major progress towards the milestone but this will need further elaboration e.g. when the DEMO design team is set up and when the detailed plans to meet each milestone are developed.

Modest enhancements (such as diagnostics) are generally not mentioned below – they will be needed but are assumed to be covered within the ongoing programme.

Short Term Milestone related to DEMO

- Launch DEMO conceptual study (Mission 7)
- Conduct a CTF feasibility study and supporting R&D (Mission 7)

10 Years Milestones feeding into first ITER experimental campaign

Milestones	"Gap" addressed	Facilities and other resources making major contributions	Further resource requirements
• diagnostic techniques for erosion, deposition, dust formation etc. (Mission 3)	Plasma facing surface; Tritium inventory control and processing	JET, other tokamaks and plasma simulators	Significant R&D programme needed (under preparation)
• dust and tritium removal techniques (Mission 3)	Tritium inventory control and processing	Tokamaks and plasma simulators	Significant R&D programme needed (under preparation)
• Predictive capability for all aspects of plasma surface interaction	Plasma facing surface; Tritium	Plasma simulators, HPC,	

(erosion, migration, redeposition, T retention, mixed materials, sheath physics etc.) benchmarked on experiments (Mission 3).	inventory control and processing	various tokamaks	
• Fully developed operation strategies (Mission 2)	Plasma Performance (all 6 areas)	tokamaks, mostly larger devices with ITER configuration	
• plasma scenarios compatible with high-Z and mixed materials, and suitable for long-pulse and Steady-State Scenarios (Missions 3 & 4)	Plasma Performance (all 6 areas)	Mostly ASDEX-Upgrade & JET	
• burning plasma physics predictive capability (Mission 1)	Burning plasma (Q>10)	Tokamaks with fast particle simulation capability	
• real-time MHD control, ELM mitigation, disruption avoidance (missions 2 and 4)	Disruption avoidance; Steady State Operation	Specially equipped tokamaks	
• LHCD R&D completed (Mission 4)	Steady State Operation; Heating, Current Drive and fuelling		Not fully addressed with present means; unique compe-tences in Europe; requires ITER-relevant LHCD coupler on a Divertor tokamak
• solution to the coupling of ICCD and LHCD (Mission 4)	Heating, Current Drive and fuelling	JET with some contributions from other devices	LHCD coupling final demonstration requires action as above
• predictive capability for the H-mode pedestal (Mission 5)	Burning plasma (Q>10)	Divertor tokamaks	
• availability of a ‘numerical tokamak’ for planning and analysing experiments on ITER (Mission 5)	Plasma performance (all except disruption avoidance and start-up)	Integrated Tokamak Model-ling Task Force plus EU and broader approach HPC together with tokamak data to test models are sufficient to meet milestone	
• Qualification of reference EUROFER for the Test Blanket	Materials Characterisation	Irradiation and material charac-	Assumes continued access

Modules (Mission 6)		terisation facilities	to international test reactors
• alternative divertor armour materials available (W tbc) (mission 2 & 6)	Plasma-facing surface	High heat flux and other technology facilities (He cooling etc.)	

10 Years Milestones feeding into DEMO Engineering Design

Milestones	"Gap" addressed	Facilities and other resources making major contributions	Further resource requirements
• DEMO physics basis, including preliminary selection of long-pulse and steady-state plasma scenarios (Missions 1 to 4)	Steady state operation; Power plant performance	Tokamaks	Some weaknesses on steady state; opportunities under international collaborations
• specific H&CD and other steady state requirements defined and R&D launched (mission 4)	Heating, current drive and fuelling	Existing test-beds and the intended NBTF will contribute	Additional facilities likely needed. Specifications to come from DEMO conceptual studies.
• confirmation of the optimised stellarator configuration (Mission 4)		W7-X	
• preselection of DEMO Divertor and Blanket concepts (Mission 6)	FW/Blanket/divertor materials; FW/Blanket/divertor components;	Technology facilities	Requires significant DEMO conceptual studies in a first stage. Some specific new technology or materials facilities may be needed
• Completion of a DEMO conceptual study (Mission 7)	Licensing for power plant		Requires significant DEMO conceptual study group to be set-up
• completion of R&D on Spherical Tokamaks in preparation of decision making on CTF (Mission 7)	FW/Blanket/divertor components;	MAST if Upgraded and probably high-heat flux facilities	MAST-Upgrade should be sufficient for physics, requires technology developments in rest of fusion programme

15 years Milestones, before start of ITER DT operation

Milestones	"Gap" addressed	Facilities and other resources making major contributions	Further resource requirements
• hydrogen/deuterium inventory data to establish the basis for DT operation and divertor materials optimisation (Mission 3)	Tritium inventory control and processing; T self-sufficiency	ITER hydrogen and Deuterium Operation with Support from other tokamaks	Possibly additional ion beam analysis facilities

15-20 Years Milestones feeding into DEMO Engineering Design

- within 15 years:

Milestones	"Gap" addressed	Facilities and other resources making major contributions	Further resource requirements
• feasibility of the proposed maintenance procedure for DEMO confirmed by R&D (Mission 7)	Remote handling; Licensing for power plant; Electricity generation at high availability	ITER design and use experience	Requires significant DEMO conceptual study group to be set-up And adequate facilities to be built/upgraded once R&D requirements are defined

- within 20 years: to have completed the DEMO engineering design activity and supporting R&D and be ready for licensing:

Milestones	"Gap" addressed	Facilities and other resources making major contributions	Further resource requirements
• confirmation of DEMO physics basis (Missions 1 to 4)	Plasma Performance (all areas)	ITER and Satellite tokamak programme	Requires an adequate Satellite Tokamak programme and adequate DEMO design team
• full feedback control of plasma (Missions 1 to 4)	Power plant diagnostics and control	idem	idem

• selection of appropriate diagnostics (Missions 1 to 4),	Power plant diagnostics and control	idem	idem
• Fully developed operation strategies (Mission 2) and burn control capability demonstrated on ITER (Mission 1)	Burning plasma (Q>10); Power plant diagnostics and control	Mostly ITER, but would benefit from satellite tokamak	idem
• Availability of a 'numerical tokamak' (Mission 5)	Plasma Performance; Licensing for power plant	ITER, Satellite tokamak pro-gramme and HPC	idem
• final selection of Divertor and Blanket concepts (results from IFMIF, TBM etc. and validated modelling) (Mission 3 & 6)	FW/Blanket/diverto r materials; FW/Blanket/diverto r components;	ITER Test Blanket Module programme, IFMIF, irradiation facilities, materials R&D programme	IFMIF construction to be decided
• Selection of dedicated structural, functional and plasma facing material(s) (Missions 3 & 6)	Plasma-facing surface; materials characterisation	IFMIF, irradiation facilities, high heat flux test beds, materials R&D programme	IFMIF construction to be decided
• selection of no more than 2 H&CD systems (mission 4)	Heating, current-drive and fuelling	ITER, Satellite Tokamak programme	Requires an adequate Satellite Tokamak programme
• R&D on H&CD, PFCs and other steady state components completed (Mission 4)	Heating, current-drive and fuelling; Plasma-facing surface; FW/blanket/diverto r components		new/upgraded technology test beds
• Confirmed strategy for clearance and recycling of DEMO activated materials (Mission 6)	Licensing for power plant		Specific R&D programme tbd on the basis of DEMO conceptual studies
• Availability of DEMO design ready for construction (Mission 7)	Licensing for power plant		Requires DEMO conceptual studies and design studies with supporting R&D

20 years, alternative concept developments:

Milestones	“Gap” addressed	Facilities making major contributions	Further resource requirements
• confirmation of the steady state features of the stellarator configuration (Mission 4)	Steady state operation (back-up)	W7-X	
• Availability of a ‘numerical stellarator’, benchmarked against a variety of experimental devices (Mission 5)		W7-X and HPC	
• assessment of the engineering feasibility of a stellarator power plant (Mission 7)			DEMO conceptual studies expanded to cover stellarator-specific issues

ANNEX 2
Complementary Validation Experiments for IFMIF

Executive Summary

In the preparatory work for DEMO design, IFMIF is becoming a strategic equipment to characterise the structural materials and some key technologies required for the blanket modules. In June 2005 were started officially its Engineering Validation and Engineering Design Activities (EVEDA) in the framework of the Broader Approach (BA) agreement.
In addition to the EVEDA programme, agreed between Japan and EU, and endorsed at the first meeting of the BA Steering Committee, this note lists several proposals to secure the IFMIF project and speeds up its operation.

- The interaction between the deuteron beam and the lithium flow could be experimentally tackled by means of an electron beam at an energy of a few MeV that deposits its energy in the proper lithium flow. Use of existing rodhotrons or renting electron sources and of the lithium loop built in the framework of EVEDA could minimize the cost of this experiment to about 3 M€.
- With respect to the accelerator, validation of the difficult shaping of the beam from the circular cross section to the rectangular footprint could also be performed at a cost of about 4 M€.
- The use of the cutting edge Cross-bar H-mode accelerating structure for the high energy part (from 5 to 40 MeV) could result, if demonstrated, in a further reduction of the accelerator length, and thus the building. This proposal, initiated by University of Frankfurt, is estimated to cost about 15 M€, using the existing infrastructure at Rokkasho.
- Beam Diagnostics at high energy and in CW, as well as diagnostics in the Li target area and neutron monitors in the Test Facilities are considered as a rather challenging task. There are some gaps in the foreseen R&D. Some dedicated development would thus secure IFMIF start. Their development would cost about 3 M€.

IFMIF/EVEDA Validation Activities

The Engineering Validation and Engineering Design Activities (EVEDA) of the International Fusion Materials Irradiation Facility (IFMIF) are one of the three projects of the Broader Approach Agreement, signed between Euratom and Japan on 5 February 2007 and started in June 2007.
The main goal of the project is to deliver in a 6 year framework the detailed engineering file of IFMIF, enabling its rapid construction once a site is decided. This engineering file will rely in particular on the validation of the main systems of the facility:

Test Facilities

- The validation activities of the Test Facilities are mainly focused on the High Flux Test Module and the fabrication, then irradiation of full-size HFTM for the vertical set-up. This task involves the use of heavy experimental structures: nuclear irradiation reactor(s), helium loop, etc. The purposes of such tests are in particular:
    - Check of the irradiation thermal conditions, these ones being ensured by a combination of helium flow, and heating of the samples by means of a heater located in a groove around the capsule. Tests in the helium loop will check the thermo-mechanical and hydraulic behaviour.
    - The technological demonstration (welding, brazing, assembly, etc.) by the construction of a full set-up.

    - Irradiation programme should check, at a less stringent flux than IFMIF, the behaviour of the scale 1:1 capsule: electrical isolation, thermal quality of the NaK surrounding the samples, analysis of the behaviour of one of the most important instruments proposed to monitor the irradiation (gamma microchamber), etc.
    - Post irradiation analysis of all these elements (samples, capsule, heaters, microchamber, etc.)
- A full-size HFTM for the horizontal set-up will be designed, and a heater-integrated (H-I) plate and capsule will be fabricated and intensively tested.
- With respect to the Medium Flux Test Modules, special attention will be devoted to the in situ creep test fatigue, and in particular the actuators providing the mechanical efforts to the samples.

Lithium Target Facility

Several experimental facilities will contribute to better understand lithium loop characteristics and provide a sound basis for IFMIF's Target Facility construction.
The main one is the EVEDA test loop designed and built by JAEA at Oarai, with contribution from ENEA and several Japanese Universities. This loop will be rather representative of IFMIF's one, being constituted of all elements of the latter. The target itself will have a height at a scale 1:1, and only its width is reduced by a factor 3. Nevertheless all side effects should be affordable, only the central part being actually reduced. This loop will enable to tackle the following issues, recalled in the introduction above:

- Hydraulics and thermo-hydraulics of the lithium flow (laminarity, sensitivity to defaults, erosion of the nozzle, etc.);
- Purification system, by providing all hot and cold traps to maintain the lithium impurity level below the 10 wppm or so threshold;
- Possibility to exchange the backplate, both concepts (cut & weld and "bayonet") being accessible in the EVEDA loop;
- Operation of specific diagnostics in a non-irradiated environment but other real conditions (vacuum and vapour pressure, temperature, geometry, etc.).

Two other lithium loops will help, in particular in the preparatory work during the first half of the project: the loop already in operation at Osaka University since a few years, which has already provided very important experimental results and the more recent Lifus 3 loop of ENEA, started during the summer 2007 at Brasimone, and enabling in particular parametric studies of erosion and corrosion with several materials.
Adimensional (i.e. based on constant Reynolds or Froude numbers for example, similarly to wind-tunnel tests for airplanes) studies will also be performed by using water loops (e.g. at Nagoya University).
The last set of experimental work is dedicated to the two options for the removal and exchange of the backplate:

- The **lip-seal solution**, based on the cutting and welding of lips by means of a YAG laser, but requiring the removal of the whole target assembly, will lead to technological tests, as early as 2008, its implementation in the EVEDA loop being planned at the start of its operation.
- The more ambitious **"bayonet"** concept based on the lateral sliding of only the backplate, itself being bolted on the target assembly, for which all relevant technologies must be demonstrated: remote handling tool compatibility with the severe environment, swelling of bolts, sliding capacity with special lubricants, etc.

Accelerator Facility

The IFMIF accelerator, whose low energy section (up to about 9 MeV) will be tested at full current at Rokkasho, is classically composed of four subsystems:

- **The injector (ECR driven source) and Low Energy Beam Transport (LEBT) line** (140 mA – 100 keV): a similar source has been already successfully and reliably tested by CEA in CW with H^+ and pulsed regime with D^+ ions. Its optimisation (electrodes shape, emittance) does not pose *a priori* specific difficulties.
- **The RadioFrequency Quadrupole (RFQ)**, bunching and accelerating the beam up to an output energy of 5 MeV. The work is shared between INFN and JAEA, which provides the RF input couplers. A classical four vane structure is proposed, its main challenges being a very high accuracy (a few 10 µm) all along its about 10 m and minimisation of beam losses.
- **The Drift Tube Linac (DTL) and Matching Section**: the reference solution is today a classical Alvarez room temperature DTL. Two more attractive superconducting solutions, based on (i) CH structure and (ii) Half Wavelength Resonator (HWR) structure have been assessed by a dedicated Ad Hoc Group and the IFMIF/EVEDA Project Committee. Generally, the superconducting solutions are technically preferred. Among both SC options, the HWR solution is preferred, owing to the current European budget profiles within the Broader Approach. *The impact on the project of this solution will be reported to the Steering Committee for decision in May 2008*.
- **The High Energy Beam Transport (HEBT) line and the Beam Dump**: only the line with a circular cross section at an energy of 9 MeV will be built during the EVEDA phase, as well as the challenging beam dump (1.2 MW – CW).

Proposal of new Validation Tasks

Several challenging aspects of IFMIF which are not covered by the current EVEDA are proposed to be addressed as follows.

Deuteron Beam – Lithium Interaction

Recent neutron source experiments in the world show contrasted difficulties in the interaction of the ion beam and the liquid metal of the target:

- At MegaPIE, Paul Scherrer Institute, Villingen in Switzerland, no specific difficulty was observed with the led-bismuth flow; operating conditions provide a rather slow growth of the beam intensity, enabling its gradual thermalisation.
- At SNS, Oak Ridge National Laboratory, USA, some acoustic effects were observed because of the pulsed nature of the beam. These deleterious bursts were treated by means of helium bubble injection.

In IFMIF, the 2 deuteron beams of 40 MeV – 125 mA each will reach the target with a cross section of about 5 x 20 cm^2, with thus a power density of 1 GW/m^2 in CW. These beams interact with the rapid flow (nominally 15 m/s) in a very thin layer, increasing the temperature of lithium by more than 160 °C in the bulk, the surface temperature increasing at 290°C, 54 °C below the boiling temperature at 10^{-3} Pa. Risks could be the creation of differentiated layers in the flow, increasing the risk of turbulence, which could have deleterious effects. Some cavitation could also occur. The windowless characteristic of the flow is also a potential source of risks.

In order to better understand the consequence of the energy absorption, an experiment using electron beam could rather well represent the actual situation[37]. The electron beam energy may be chosen in a range where nuclear reactions do not take place, with a significant simplification in the experiment

[37] The author wishes to express his great thanks to Marco Ciotti, ENEA, who has drafted this experiment in a note, called: “Proposal for a Lithium Heating Experiment (Lithex)”, dated 20 July 2007.

managing. The best choice would be to place the electron beam in the range 1 to 6 MeV. The lower limit is determined by the need to have some penetration inside the Lithium, the upper one is chosen to obtain a complete electron absorption in a lithium thickness of the order of the IFMIF one and to avoid the possible onset of nuclear reactions, leading to a more complicated experiment managing.

A detailed cost evaluation analysis has still not been performed. From a first conceptual analysis based on the experiment described above, and supposing that an existing electron beam source can be rented, the cost would probably reach 3 M€.

Rectangular shaping of the Deuteron Beam

The transition from the circular cross section at the end of the Drift Tube Linac to the required foot print of 5 x 20 cm requires the use of non-linear optics, which could, depending on the actual beam profile, energy spread, halo current, etc. lead to a degradation of the footprint at the target input. Test of these optics could be beneficial to the characterisation of the beam.

Because of the space available in the EVEDA building, this experiment is probably difficult to install and would require, either a strong modification of the building (which seems difficult), or a U-turn after the DTL, thus slightly modifying the beam characteristics. Only pulsed operation would be possible in such a case, because of the activation of the beam dump. The budget of such an experiment would be probably between 3.5 and 4 M€, including manpower.

Alternative CH superconducting DTL

As mentioned above, two structures are competing for the DTL. Another structure, called cross bar H mode has been proposed by the University of Frankfurt[38]. Suffering of a lack of experimental feedback, this proposal, in spite of its potential advantages, was not retained in an early phase. Very compact (the gain on the length of the accelerator would be 10 m more than the half wave resonator, itself being 10 m shorter than the room temperature Alvarez solution), this structure nevertheless requires a very high accuracy in the phase positioning of the beam with respect to the wave. This accuracy could be difficult to maintain, in particular because of space charge effects.
It could be interesting to test this kind of structure, which would result in real estate savings for IFMIF. The extra cost has been estimated to 15.2 M€ by University of Frankfurt.

Diagnostics for IFMIF

The development of diagnostics for IFMIF is concentrated during the EVEDA phase on characterisation of properties to better understand the behaviour of the systems:

- In the accelerator line, they are located in the Low Energy Beam Transport Line (characterisation of the injector), between the RFQ and the Matching Section, and on a specific plate in the “High” (9 MeV, instead of 40 MeV for IFMIF) Energy Beam Transport line;
- In the lithium target facility, most of them could not be operational because of the huge irradiation during the operation of IFMIF. A selection will be made at the end of EVEDA for those compatible with IFMIF operating conditions.

In complement to the development of the upper diagnostics, high energy beam characterisation, specifically devoted to the operation, would be extremely useful. The work could be concentrated on:

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[38] The author is grateful to Pr. Horst Klein for valuable discussions about the CH proposal.

- Relevant operating conditions: high irradiation, appropriate energy beam characteristics
- Very fast time response, to be compatible with the interlock system of IFMIF
- Link with the lithium target ones, to protect the overall facility

In a first approximation, such development could reach an envelope of about 3 M€.

Materials characterisation

Intensive programmes of irradiation are conducted in particular under the responsibility of EFDA. All valuable information on doses similar to those of IFMIF, in particular for the target's backplate and the test facilities will secure the design of IFMIF. Programmes have been already conducted on Eurofer with doses up to a few tens of dpa. The construction of a database with all mechanical properties will be particularly beneficial to IFMIF engineering activities. It should include:

- Conventional steels (316, 304…)
- Ferritic Martensitic Steels (Eurofer, F82H…)
- Other materials, in particular nickel, aluminium…
- Temperature range: between 150 and 400 or even 450 °C
- Doses: a few dpa up to 50 dpa (in particular for martensitic steels)

ANNEX 3
Strengthening the Materials R&D Programme

I Introduction – Requirements on Material Development

An ambitious and successful fusion materials development is one of the key conditions for the development of Fusion Reactors. The main purpose is the development, testing and qualification of structural and functional materials suitable to design and construct breeder blankets and divertors for DEMO and fusion power reactors. The overall objective is to develop radiation resistant materials that nfavo a specified set of physical and mechanical properties through the whole lifetime. The First Wall of a Breeder Blanket, for example, should survive 3-5 full power years or respectively in terms of irradiation damage typically 50-70 dpa for DEMO and 100-150 dpa for a power plant. The aim is to have the materials and key fabrication technologies needed for the DEMO reactor fully developed and validated within the next two decades.
The materials foreseen for DEMO should, therefore, be based on present technologies and knowledge with some reasonable extrapolation as two decades is a short period to completely develop new materials that comply with the requirements set above. EUROFER steel (details below) is the primary EU candidate with appropriate properties in a temperature window of ~300-550C.
In order to increase the thermal efficiency of blankets, the temperature window of the structural materials needs to be increased. Various ODS (Oxide Dispersion Strengthened) Cr-steels and SiC/SiC composite material are candidates for higher temperature application.
For higher temperature gas cooled divertor concepts refractory alloys are currently considered as the most promising candidates to meet the specific requirement of high heat flux, high temperature and structural strength. Excluding radiologically unfavourable elements (like Molybdenum) narrows down to choice of W-alloys, which have the potential to serve for structural application only above 700C, because of their inherent brittleness at lower temperatures. Several concepts of gas cooled divertors are based on a steel support structure and, therefore constitute another driving force for the development of (ferritc) ODS steels, as other candidate materials do not offer a significant overlap in operating temperature with tungsten materials.
High 14 MeV neutrons flux produce large amounts of He and H in association with atomic displacements inducing intensive hardening and embrittlement. This combination is responsible for the degradation of properties of materials, as well as for the possible loss of dimension stability associated with swelling and irradiation and thermal creep. The neutron spectrum, in particular the large fraction of high energy neutrons, results in gaseous transmutations being more than one order of magnitude higher than in fission. Therefore fission based material test reactors can not provide sufficient data for fusion materials qualification and a successful licensing process. For this reason, the construction and use of a facility called IFMIF, designed for simulating as closely a possible the fusion neutron spectrum, is mandatory.
In addition to IFMIF, the Fusion Materials Research programme must focus on running a well-balanced set of activities that combine neutron irradiation in fission reactors, ion-beam sources and physical modelling in order to understand and predict, on quantitative level, radiation effects occurring under fusion reactor relevant conditions. First of all, the integrated use of all these activities are necessary for a bundle of reasons (i) to increase the insight in radiation damage, (ii) to get as much information and data as possible as early as possible, (iii) to prepare for the exploitation of IFMIF, (iv) to assist in optimising the test matrix for irradiation on IFMIF facility, (v) to accelerate the DEMO licensing process.

Another family of materials to be considered in the Materials R&D Programme is ceramic insulators materials. They are key elements of many diagnostics, Heating and current Drive as well as Remote Handling systems. For ITER, radiation effects and associated degradation of the insulator physical properties has imposed severe limitations on their use, fortunately due to the restricted dose, dose rate, and operating conditions, it has been possible to assimilate these limitations into the design. However for DEMO and beyond with far higher total doses and more extreme operating conditions insulators will still form part of essential components. To solve or at least mitigate the problems associated with insulator degradation, a significant coordinated efford must be launched well in advance.

A science based and well-focused modelling programme should help to reduce the uncertainties in transferring data gained from various irradiation experiments in facilities with different neutron spectra to the fusion typical environment. It should also contribute providing guidelines for improved heat and radiation resistant materials and limit the risk of unique development line with critical issues identified as especially difficult to overcome.

The requirements for this programme as well as the resources required for the successful execution of the proposed tasks are presented below.

II- Overview of programme – Status requirements

The current status and some directions of future development are presented for the development of structural materials, i.e. EUROFER, ODS-EUROFER, ODS ferritic steels, W and W-alloys as well as SiC/SiC. In addition, functional materials, i.e. coatings, barriers and breeder materials will be briefly mentioned.

EUROFER – the Core development

The EU reference material for DEMO, EUROFER 9%Cr steel, will be first used with Tritium Breeding Test Blanket Modules (TBM) in ITER and is foreseen for different helium cooled blanket options, namely the Helium Cooled Pebble Bed (HCPB), the Helium Cooled Lithium Lead (HCLL) and the Dual Coolant concepts. The material is the most advanced material with respect to early use. Nevertheless continuous characterisation (e.g irradiation campaigns, development of fabrication and welding processes) is needed. It also requires further optimisation for achieving improved performance for instance improved resistance to He – embrittlement, and better radiological properties towards its successful use in DEMO. The metric to qualify “improvement” or “optimisation” is multidimensional. In short: It needs to guarantee the full set of design related properties within a as large as possible design space of temperature window, performance and life time.
An intensive modelling and materials science programme has to support and complement this base line development. Here, modelling has to be understood in a broader sense including besides the investigations on radiation effects also constitutive and/or damage laws as well as chemistry (corrosion) or technology realed issues such as welding. In summary, modelling here as to provide a link to engineering.

Materials for breeder blankets of the first generation

R&D and qualification of functional materials for DEMO breeder blanket applications is needed. Among the considered functional materials are:

- Anti-permeation and anti-corrosion coatings,
- Tritium breeder materials (both ceramic and liquid breeders),
- Neutron multiplier materials.

Anti-corrosion coatings and in particular anti-permeation barriers (to prevent the tritium permeation from the Pb-Li liquid metal alloy, used as breeder and multiplier material, into the He coolant) are amongst the critical issues in the development of HCLL breeder blankets. The schedule foresees mainly R&D and preliminary qualification of Al or W or Er based materials or compounds during the next decade as well as the full qualification and development of technologies for fabrication at industrial scale during the second decade from now with the final aim for testing in ITER TBM. For tritium breeder materials (ceramic pebbles) and neutron multiplier materials, the strategy is very similar:

development (i.e. improvement of existing materials and exploration of new materials like Be-Titanite), production and basic characterization during the next decade and demonstration of performance thereafter.
The list of issues is long, for example for ceramic breeder pebble beds: stability and performance under irradiation, tritium release and retention, compatibility with structural material, thermo-mechanical behaviour, activation and recycling.
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ODS Steels

Beyond the reference ferritic/martensitic steel EUROFER, Oxide Dispersion Strengthened (ODS) EUROFER and the ODS ferritic steels are being developed with the objective of increasing the temperature window up to the maximum operating temperatures of 650^0C and 750^0C respectively. EUROFER ODS increases the design window for breeder blankets and opens new doors. These materials are currently not foreseen to completely replace the existing material rather than complement the materials systems. One example, given above is the use of ferritic ODS in combination with W-alloys in gas cooled divertor concepts by an overlap of operational windows.
The dual coolant blanket using a material system of ODS EUROFER (at the "hot spots") plated on a conventional EUROFER structure plus SiC/SiC composites for thermal and electrical insulation of flowing lithium lead at high temperature against surrounding structure provides currently the best thermal efficiency achieved with reasonable extrapolation of technology. This system makes efficient use of the high temperature capability and avoids the drawback (low fracture toughness) as well as the main technological issue associated with the use of ODS steels, the welding. The dual coolant concept provides a good example on a future trend, i.e. to separate functions and requirements (e.g. heat load capacity, corrosion resistance and good weldability are needed in blanket but rarely at the same parts) by using different types of materials.
The EUROFER ODS development started around 2001/2002 and had its first breakthrough in 2005, with a first production of 50 kg exhibiting (compared to EUROFER) higher creep strength, however less fracture toughness. Improvement of properties, definition of the best fabrication route and more than one decade of qualification is still ahead before application in a TBM or similar component. The development of nanostructured ODS ferritic steels for DEMO started in 2005 and will need another six years to proceed towards the specification of an "optimized ferritic steels" through several optimisation steps facing thereafter the cycle of evaluation modification and qualification.
The challenge and the risk of the ODS development in general is the fabrication, route, the powder metallurgy, as there is presently no industrial capacity in the EU for manufacturing this type of steel.
The development of technology for joining of parts made of this type of material is also an important issue as any melting processes are excluded, because they destroy specific oxide dispersion microstructure resulting in the loss of high temperature creep strength. Other techniques e.g. diffusion bonding or Friction Stir Welding, have to be developed and/or qualified. Properties of fabricated diffusion bonds are satisfactory. They need to be qualified under irradiation. Friction Stir Welding developed for Al-alloys for Aeronautics application needs development and qualification to be applied to steels.

A strategy that focuses solely on the ODS steel development for a broader application at higher temperature operating conditions implies some risk. Therefore, the programme should not miss the opportunity to explore and develop steels with new alloy compositions based on ideas emerging from the modelling programme. The material is fabricated by conventional metallurgy, with the objective to be used in a higher temperature window satisfying the multiple requirements of fusion environment.

In view of making the best possible use of resources, some milestone, which could be set in approximately 10 years from now, should be defined. ODS steels and improved "conventional steels" should be assessed in the view of their potential use and value for existing or advanced blanket concepts Thereafter, most of the effort should be put on the more promising option.

Refractory Materials

Refractory alloys, in particular W-alloys are used in fusion devices for different purpose, i.e. heat sink and protection material for the divertor, where the performance of the material is strongly affected by both high heat loading and neutron irradiation or FW coatings.
Another area, the use of refractory alloys as structural material (briefly introduced above) in a small pressurized parts of a gas cooled divertor is associated with potential high risk. Typically the from design requirements includes: (i) T-window of 700-1200/1200^{0}C, high thermal conductivity, good ductility and strength. The main issue for the W-alloys under development is their intrinsic brittleness, the origin of which has to systematically studied and has to be understood to enable the development of a strategy for its mitigation. Fracture toughness of tungsten alloys, as a kind of measure to distinguish between brittle and ductile behaviour is very sensitive to fabrication route, heat treatment, grain size and orientation. In addition W-metals and alloys, which have been irradiated in their original metallurgical condition and chemical composition, are characterised by the unacceptable level of brittleness that rules them out as materials for structural applications.
It should be noted that new design concepts imply also the need to solve the issue of joining dissimilar materials, e.g. tungsten to steel.
Whereas in other areas (ODS steels, Silicon Carbide composites) similar materials development programmes are run in Japan and the US fusion and science programmes as well in other areas (fission Generation IV), the European development of W alloys for structural application is currently unique world-wide and needs, therefore, special emphasis and resources.

SiC/SiC and other high temperature materials

EU reference SiC_f/SiC composites hardly satisfied the design requirement for structural application for advanced high temperature tritium breeding blankets. In addition the rapid deterioration of their thermal conductivity under irradiation or its unusual plastic properties or the critical issue of joining are potential critical drawbacks. Finally the possible improvements of this situation using optimised fabrication routes will be explored.

Ceramic insulator materials

The long term objective of this area is the development of radiation-hard components for diagnostics and H&CD systems on DEMO. For DEMO and fusion power plants, a programme on basic material research at higher doses and dose rates needs to be developed to assess potentially suitable materials and to gain understanding of the degradation and radiation effects that occur at such levels. The materials and components include insulators, optical materials such as windows, sensors and potentially also electronics. In addition to the analysis of radiation effects on structural properties of those materials, the focus is on the study of radiation effects on functional properties such as electrical and thermal conductivity, optical transparency and diffusion and accumulation of tritium. Both dose and dose-rate effects on functional properties are important.

III Preparation and exploitation of IFMIF

Preparation for the use of IFMIF (next decade)

First, the capacity of IFMIF for testing and qualifying materials on a short time scale compatible with licensing any early DEMO is limited and hence a rigorous materials pre-selection process should be anticipated. Consequently, prior to the beginning of IFMIF operation a broad programme of qualification of EU reference materials such as improved and optimized EUROFER of a second generation, including also the fabrication and joining procedures, is mandatory during the next decade to be prepared for a selection process.

Exploitation of IFMIF – Material qualification using IFMIF facility (beyond 2019)

The qualification of materials for the IFMIF facility has to be thoroughly prepared. Besides the preparation for the irradiation campaigns, which should be a part of the project, Post-Irradiation Examination (PIE) will require a large effort, too. In IFMIF ~600-900 specimens are foreseen to be irradiated. Two years after start of operation (>2019), these specimens will have to be analysed. Mechanical characterisation will be performed preferably (but not exclusively) at the IFMIF site because of activation. Microstructural analyses can be performed elsewhere in the EU as the activated volume is very small.
In addition, the efficient use of the IFMIF facility and its data requires significant theoretical work. Therefore, besides a basic modelling programme focused on broadening the knowledge and understanding of irradiated materials with the final aim of predicting quantitatively the radiation effects in materials under fusion reactor relevant conditions, dedicated modelling related to IFMIF is necessary in several stages: in the interpretation and understanding of the results as well as in the assistance to secure extrapolation to the operating conditions of DEMO and a Fusion Reactor, in the frame of a modern and reliable licensing process, requiring an additional need resurces.

IV Materials Modelling and experimental validation

The need for correlating the large data-base on radiation effects obtained under various spectra and extrapolating them to Fusion Reactor conditions was the incentive to launch a Radiation Effects Modelling Materials programme. The first objective was to understand and model radiation effects in EUROFER under fusion reactor relevant conditions i. e. up to high dose and contents of transmutation products such as He and H. This modelling is multi-scale in nature, and establishment of a close connection with experimental validation is essential to develop verified physical modelling suitable for quantitative correlation of the results under various spectra and providing guidelines for the development of radiation resistant materials.

Started in 2003 this EU coordinated modelling programme is unique worldwide. Its promising development shows that the objectives for this Modelling and Validation programme, in relation to the design, construction and operation of DEMO can certainly be broadened and more precisely defined as:

- Development of **comprehensive predictive capabilities** for modelling micro-structural evolution and mechanical properties of **EUROFER-type ferritic-martensitic steels under fusion reactor relevant conditions**.
- Assessment of **the effect of atomic displacement and helium & hydrogen generation** through nuclear transmutations by fusion neutrons on the phase stability, microstructure, mechanical, thermal, and functional properties of **tungsten-based materials, and ODS steel**s, developed for high temperature applications.

- Implementation of an **integrated strategy involving modelling and model-oriented experimental validation** as means for the accelerated development and testing of candidate fusion materials at the pre-IFMIF stage.
- **Application** of the newly developed modelling and experimental validation methods **to the innovative development of materials** for the use in irradiation and thermal environment of DEMO and fusion power plants.

The following paragraphs give examples of successful achievements. Quantitative prediction of behaviour of the Fe-Cr-C system, in fact a model steel, under fusion relevant reactor conditions is a realistic medium term objective, provided that (i) the adequate physically based methods are developed and parameterised using presently available high-accuracy *ab-initio* calculations, and, (ii) the experimental validation is adequately linked to the theoretical and numerical predictions. Modelling based guidelines for improving presently selected materials or for innovative development are also underlined.

Radiation Effects Modelling in EUROFER

Phase stability, and, dpa & H accumulation are the main phenomena triggering the radiation induced microstructure and mechanical property evolution.

A Fe-Cr Phase diagram: ab initio based statistical methods:

For the first time ab initio calculations have predicted the correct negative sign of the formation enthalpy of the Fe-Cr, in agreement with the solubility of Cr in Fe-rich domain of the phase diagram. These calculations also showed that the cohesion energy and the α/γ phase transformation at high temperature are triggered by the magnetism of the Fe-Cr system. Magnetic Cluster Expansion (MCE), with an explicit treatment of the magnetism have been developed and fitted to the ab initio mixing enthalpies of various Fe-Cr configurations. As for experimental validation, the MCE method results in excellent prediction of magnetic properties versus temperature, although based on ab initio, i.e, 0K, data. The phase diagram modelling is being carried out.

B He & displacement accumulation: ab initio based kinetic Monte Carlo

The issues linked with He and dpa accumulation is twofold: (i) hardening due to the formation of a dense population of clusters formed of He and vacancies and (ii) intergranular embrittlement even at low temperature ~300 ^{0}C most likely to be triggered by He –atom segregation.

Concerning the point defect and He accumulation an important work has been carried out consisting of Monte Carlo kinetic method development in the Fe-C system in presence of point defects and He, the energetics of which has been calculated ab initio. Such models are able to reproduce accurately the experimental He-desorption of pre-implanted Fe-C samples.

Concerning the He-embrittlement ab initio calculations show that there is an important loss of cohesive energy of grain-boundaries in pure Fe, -2.95 eV/He atom, to be compared with – 0.54 eV/P atom, phosphorus being known as a strong embrittling element for Fe and ferritic steels. Therefore He atoms have a very strong embrittlement effect. It is not the first time that metallurgy is confronted to such grain-boundary embrittlement. The objective of the mitigation measures is always to trap the impurity in the grain by increasing the density of sinks (improvement of the present EUROFER) or introducing new type of possible sinks (the nano- clusters of the ODS steels)

Kinetic Monte Carlo modelling is being developed to include grain boundaries and intra-granular He trapping sites to treat point defect & He accumulation in Fe-Cr-C alloys. Theoretical prediction will be compared to (i) dedicated He-desorption to validate the rate-limiting stages of He migration and desorption, and, (ii) dual beam irradiation to evaluate the partition of He within the grain and to grain-boundaries in order to mitigate the inter-granular embrittlement due to He atom segregation and provide guidelines for optimizing EUROFER.

C Methods for large scale modelling

The dynamic simulation of extended defects such as displacement cascades and dislocations requires calculating systems of millions of atoms, which are out of reach for the present *ab-initio* calculations. Empirical potentials are required, which should transfer the information obtained at the level of electronic structure, including the electron correlation effects responsible of the magnetism, to the extended atomic scale. The "magnetic" one introduces explicitly the spin of the Fe atoms and potentially allows reproducing the correct sign and sign change of the mixing enthalpy in the Fe rich domain of the Fe-Cr alloy. It will be further develop to take better into account the observed changes of elastic constants, the anharmonicity effects and the magnetic properties of Fe. Experimental investigation of magnetic properties and linking to MCE work, magnetic and elastic properties, elastic constants near the Curie temperature, thermal expansion, phase transitions will be essential

d Dynamical properties of dislocations

An extensive experimental and modelling work showed that the ductile-brittle behaviour of pure bcc metals and alloys is controlled by the mobility of screw dislocations, which in turn is determined by the energy of double kinks formed on a migrating screw dislocation under applied stress. *Ab-initio* calculations of formation energies of double kinks on screw dislocations in Fe are being carried out. The various possible methodologies have been assessed and the capability of calculating this important double kink formation energy opened, giving access to an important property governing the value of the DBTT of Fe and in a near future of bcc transition metals.

The range of high-temperature applications of conventional and ODS ferritic/martensitic steels is strongly impeded by the rapid softening of these materials for temperatures exceeding 500°C. The origin of this striking and crucial behaviour for high temperature applications phenomenon has been proven to be associated with the dramatic reduction of one of the elastic shear moduli of the material. The work within the programme will now focus on the elucidation of the physical mechanisms responsible for the softening, which will have to be introduced in the development of Discrete Dislocation Dynamics (DDD) models describing plasticity of iron and iron-based alloys at high temperatures. These investigations will also open the road to innovative solutions to reduce the effect of softening in the range of temperatures exceeding 500-550 ^{0}C.

The interaction of dislocations with He-Vacancy clusters, coupled with in-situ TEM observations, will represent another important topics associated with Molecular Dynamics (MD) and DDD. The interaction of dislocations with radiation defects due to fission neutron irradiation will be further studied using MD and empirical potential within the Fe-Cr system.

E Modelling of the present and the optimised version of EUROFER

There is no doubt that all the effort sketched above will results, in the forthcoming years, in a physically based and validated modelling of the Fe-Cr-C system, in fact a model steel, under fusion reactor relevant conditions.

The next step will be the modelling of the present and optimised versions of EUROFER. This will require specific experiments to determine the critical parameters that characterise EUROFER and have to be incorporated in the previous models of the Fe-Cr-C alloys to have a realistic modelling of the real structural materials. Such process should require a significant effort and will require (i) an iterative process of optimisation based on a good mastering of the modelling accumulated so far, (ii) microstructure characterisation under dual-beam and (ii) dedicated experiments on He-desorption to determine the effect of the increasing complexity of microstructure and chemical composition on the rate-limiting mechanisms of He-desorption. Such step is the only opportunity of the whole fusion materials programme to consider the detrimental He effects and associated mitigation measures in the preparation of the IFMIF programme. In addition modelling industrial materials will be mandatory in the exploitation of the results of IFMIF

Modelling for ODS ferritic steels

The optimised creep strength of ODS ferritic steels is obtained via a high density of Y-Ti-O nano-metric clusters. The significantly better creep resistance of ODS ferritic steels than the one of ODS ferritic/martensitic steels is in agreement with the absence of α/γ phase transition at high temperature. In addition, ODS ferritic steels with a high density of nano-clusters have been proven to harden less after neutron irradiation than EUROFER and conventional ferritic/martensitic steels, because the nano-clusters may act as point defect sinks. As well the nano-clusters have been shown to be effective intra-granular traps for He.

The above developed modelling tools will apply to any question triggered by the Fe-Cr composition of the matrix of ODS steels. Presently the activity is devoted to understand how (i) the nano-clusters are formed during the Mechanical Alloying and subsequent thermal-mechanical treatment during the fabrication process and (ii) their main characteristics (density number, size, chemical composition, crystal structure) vary with the fabrication parameters. On this basis the cohesion energy of these clusters and their interaction with point defects and He will be quantified using ab initio calculation. Kinetic Monte Carlo models developed for the Fe-Cr-C will be further developed to take into account the main characteristics of nano-clusters and provide microstructure evolution under irradiation to be compared with experimental data.

W and W-alloys

Tungsten exhibits intrinsic high DBTT which is triggered by the high activation energy for the glide of screw dislocation as compared to Fe or Ta. This effect can be partially mitigated by severe plastic deformation or by alloying elements such as Re that could decrease the double kink formation energy. The *ab-initio* methodology developed for the screw dislocation in Fe will be applied here, taking into account the specific electronic structure of tungsten and rhenium, and, validated against in-situ TEM observations of dislocation dynamics.

Here also the assessment of effects of dpa and He accumulation will be essential for the quantification of the effect of irradiation on properties The kinetic Monte Carlo modelling tools developed for Fe-Cr-C will have to be parameterised using ab-initio data for W, W-Re in presence most probably of impurities like C, N or O.

Ceramic insulators

The development of modelling and simulation tools for ceramic insulators and their qualification against experiments, still in a low development stage compared with the structural materials modelling activities, is expected to play a major role in the programme, in particular because of the complications and expenses of irradiation at the required level and difficulties in irradiating with the appropriate fusion neutron energy spectrum. Simulation includes the use of complementary irradiation facilities, such as accelerators and gamma sources and using pre-doping of materials.

Link with IFMIF

The modelling programme has already produced exceptional results that revealed, in quantitative terms, the fundamental underlying processes driving radiation effects and dynamic properties of dislocations, which are essential ingredients of the in-service behaviour of materials In addition, modelling should provide, and already did, guidelines for improvement of the various structural materials and for selecting the best possible microstructure and chemical compositions, in effect leading to fine tuning of properties of alloys, hence optimising the IFMIF irradiation matrix in the long term.

The modelling activities focused on real industrial materials are an essential step in the preparation to IFMIF and in the exploitation of the resulting data. Data coming from IFMIF will be of prime importance not only in their own right but also as a basis for the validation of modelling tools. In addition the entire range of operating conditions of a DEMO reactor most probably will not be simulated in IFMIF, and hence modelling methods should contribute to the safe extrapolation of data to the larger DEMO irradiation conditions.

For such objectives to be met, the development process described above including the industrial material for every category is essential. It is a long term effort, which requires keeping on with strong coordination on a firm treaty-based foundation independent of possible political turbulence or delays linked to possible fluctuating political decisions.

Link with Engineering

Design studies and integrity assessments are based on criteria that are often based on good practice mostly accumulated on non-irradiated materials. Constitutive laws, damage accumulation rules for irradiated materials are often difficult to validate experimentally. Physically based modelling should provide guidance to improve the realism of these constitutive laws and damage rules used in the present codes, and, a close interaction between the materials community and design engineers is needed.

The facilities

The modelling programme requires a close link with experiments for the experimental validation of tools and for the demonstration of its predictive capabilities. The key experimental facilities are described below. In addition, the development of numerical methods and algorithms require intensive computation capabilities and access to massive computational resources.

The key pillars of a successful programme here are:

(i) International dimension

One of the necessary conditions for successful, effective and timely development of materials for fusion is the extensive use of the international pool of expertise in the area. The international dimension of the fusion project, highlighted by the very structure of the ITER project, calls for the materials development programme to adopt a similar approach. There are specific areas of expertise where partner countries could make valuable contributions to the development and selection of materials. Japan has particularly strong expertise in the area of steels and alloys, as well as in methods for electron microscope examination of irradiated and structural materials. China is rapidly developing capabilities in the material field, with the University of Hong-Kong and Hong-Kong Polytechnic University being among the world leading centres in the field of mechanical properties and modelling radiation damage. Researchers in Korea have recently performed pioneering investigations of radiation defects, and the country runs extensive programmes on advanced steels and nuclear materials. Russia has strong and extensive track record in materials science, with several major institutions, including the Bochvar Scientific Research Institute for Inorganic Materials, Moscow, the Efremov Research Institute for Electrophysical Devices, St Petersburg, the Institute for the Strength of Materials, Tomsk, leading the development in the field. Needless to say that there are extensive and useful transatlantic links with many national laboratories and universities in the US that are involved in the development, modelling and testing of materials for fusion and nuclear applications.

A reinforced materials programme would strongly benefit from the extension of contacts between the EU laboratories and research centres and universities in the countries noted above. This may take the form of either a network of bilateral contacts supported by the appropriate staff mobility programmes, and be complemented by a programme of extended hands-on workshops on, for example, materials modelling or advanced methods for examination of materials.

(ii) Intensive computational facilities

Recently, a group of experts in their "Report of an ad-hoc Group chaired by J. Connor on Support Measures for Theory and Modelling Activities under EFDA" have identified the allocation of significant supercomputing resources to both plasma physics and to materials modelling as one of the major practical steps in the implementation of a fast track programme of materials development. The deployment of massive computational resources will likely help addressing, in quantitative terms, the issue of brittle behaviour of metals, which will be achieved through application of large-scale density functional calculations to dislocations in the way similar to how calculations recently resolved the problem of structure of radiation defects.
The equally significant revolutionary developments are expected in the field of Monte Carlo calculations, where new accelerated algorithms make it possible to span realistic timescales similar to the lifetime of materials in a reactor, and where massively parallel dislocation dynamics simulations are expected to unravel the microscopic origin of the loss of strength of ferritic steels at elevated temperatures and suggest ways that could help mitigating the effect.

(iii) Fast neutron irradiation

Irradiation with fast neutrons is to be devoted (i) to studying the effects of displacement cascades on post-irradiation (PI) mechanical properties (tensile, Charpy, fracture toughness, fatigue) and (ii) to assessing the combined effect of neutron flux and monotonic or cyclic plasticity, which both exist in service, conversely to PI testing, where there is no neutron flux.

Microstructure examinations are often scarce in this type programme. A link has to be established with the modelling programme to complement microstructure examination with the objective of obtaining microstructure after high atomic displacement doses for comparison with modelling predictions and correlation with mechanical properties.

High dose typical of fusion reactor (at least 50 dpa) has to be achieved in a reasonable time. In the EU, only the Jules Horowitz reactor (JHR), which will start operation in 2013, with 15 dpa/year in the central hole, meets this requirement. The PALLAS reactor is planned in the Netherlands, however missing a political decision and with presently a design maximum dose of 10 dpa/year. For PI testing, well equipped hot labs exist in CEA, SCK.CEN, NRG, FZK and at Rez. Collaboration should be continued with the fast reactor BOR60 in Russia, or could be envisaged under IEA with the US (HFIR) and or Japan (JOYO), where high temperature accelerated testing with temperature control could be performed. Concerning in-pile testing, experience exists in BR2 at SCK.CEN/Mol and Rez, and, will be transferred from OSIRIS to JHR within CEA.

(iv) Multiple ion-beam irradiation

Based on accelerators of a few MV, this type of facility is able to simultaneously and homogeneously produce either atomic displacements and He implantation (dual beam) or atomic displacements and He & H implantation (triple beam) in thickness of a few micro-meters. These techniques are well established, reliable, versatile in terms of irradiation temperature and He/dpa ratios, and have rapid feedback. Since the ion-irradiated samples are non-activated, all the modern chemical-physical characterisation techniques available in cold labs can be applied. Mechanical properties behaviour can

be analysed using nano-indentation or tensile tests on micro-samples machined by Focused Ion-Beam (FIB) techniques.
JANNUS co-operated by CEA, CNRS and the University of Orsay has been optimised towards high doses and He & H contents typical of Fusion Reactors. It includes in-situ TEM irradiation in dual-beam configuration. Dual beam irradiation can be also carried out at FZ-Rossendorf. A dual/triple beam facility is also foreseen at CIEMAT in the Spanish project Tecnofusion with increased energy and implementation depth. Similar facilities certainly exist in Russia and collaboration could be set-up.

(i) He implantation via cyclotrons

This type of technique using α-particles of energies in the range 20-100 MeV allows implanting He in samples of thickness ~100 micrometers suitable for tensile or creep tests, and, post-mortem microstructure characterisation. Helium induced hardening & embrittlement was documented and understood using this type of approach. The He/dpa ratio is high and fixed conversely to the multiple beam techniques. Nevertheless given the present progress of modelling techniques, it is no longer a serious drawback, as it was before.
In the EU, the cyclotrons of FZ-Jülich-and-Karlsruhe have been shut down. CNRS and CEA are setting up the required irradiation and mechanical testing devices at the cyclotron of CERI-Orleans. The KZJ cyclotron has been moved to Ukraine and set-up again there. Collaboration can certainly be envisaged. A cyclotron is foreseen at the Spanish project Technofusion, which would also allow these types of experiments to be conducted.

Link with Engineering

Design studies of in-vessel components and of a complete fusion reactor is in addition mandatory, especially for the Plasma Facing Components and the possible optimisation of the heat flux loading: this design shall guide the Fusion Materials Development Programme by providing precise technical specifications. Furthermore, the inherently different behaviour of the materials developed for high temperature application, as for example their inherent room temperature brittleness, needs new concepts and approaches in the design. Therefore a close interaction between the materials community and design engineers is needed.
This area is missing today modelling capabilities and simulation. Modelling here has to be understood is a very broad sense, including constitutive and damage laws, finite element models, “virtual experiments”, development of new verification tests with the final goal to provide methodologies and physical basis to evaluate relations between material properties and analyses of failure paths and structural integrity. This link between materials science, materials development, engineering design and analysis which in essence is a “virtual structural test assembly” or a “virtual component test facility” strongly requires computational and human resources that are not available by today. In the long term, a programme similar to the large simulation and computation programme of plasma physics (IFERC under the Broader Approach) will be mandatory to bridge the gap between material science and engineering, where material systems and multifunctional systems provide a specific challenge.

V- Human Resources

The human resources of the present Fusion Materials Programme conducted under EFDA and F4E is ~100 Ppy/a[39]. This effort is substantial, but nevertheless significantly less than the one dedicated for example to similar nuclear material programmes, for instance the French programme of radiation and

[39] About 20 (W) + 10 (ODS ferritic) + 20 (Radiation Effects) + ~10 (SiC/SiC) + 30 to 40 for F4E

heat resistant core materials for the Fast Reactors[40](*). In addition, there are more materials to be developed for fusion (structure, heat sink, breeder, protection & support), than for fissile cores. For example, the blanket and divertor components are typically made of 5-8 different types of materials.

To address the programme requirements listed above, the following overall resources are needed.

- Further development of EUROFER: approximately 700 ppy during the next two decades.
- Materials needed for first generation breeder blankets (others than EUROFER): (i) Functional materials (breeder materials, neutron multipliers as well as corrosion and permeation barriers) and (ii) Materials for Dual Coolant option (EUROFER ODS and SiC-composites), both need a total of 200-250 ppy during the next 20 years for each of them.
- Nanostructured ODS ferritic steels: about 10 ppy/a.
- Innovation: for example steels with high creep strength and radiation resistance requires involving broader capabilities from Universities and Industry, in close collaboration with the Associations, with an additional effort of about 10 ppy/a.
- Refractory metals and alloys: an important additional effort of ~20 ppy per year is required for the development of refractory materials since no similar expertise is available worldwide.
- SiC_f/SiC composites and other composites: 10 ppy/a.
- Radiation Effects Modelling: tool development and experimental validation should increase to 30 PPY/year and encompass in addition the capability of 10 PhD and 10 post doc/a
- Preaparation of IFMIF use:
  - Radiation Effects modelling: of industrial materials. Achievable in view of the present status of the modelling programme and mandatory for IFMIF preparation & data exploitation: 10ppy/a.
  - Additional effort for a selection procedure, fabrication of specimens and qualifying new test standards for small samples used within IFMIF: 20 ppy/a
- IFMIF Post Irradiation Examination (>10 years from now): assuming IFMIF is used in the frame of an international collaboration with about 50% European participation, this would require about 50 ppy/a, where 30 ppy/a are “site specific”and only needed if IFMIF is built in Europe.
- Materials sciences: human resources are critical in the fusion material science programme, which is scattered within a relatively large number of small groups. This is a source of possible synergy but certainly of entropy. This situation requires stronger coordination, which is being put in place under EFDA, but this does not compensate for the lack of human resources: indeed the overall capability of the Associations is fully-booked by the present programme. An increase of the human resources from 20ppy/a presently up to 30 ppy/a is mandatory

Finally, teaching Materials Science and Technology in the field of radiation resistant materials for high temperature applications is recognised to be at a low level in the various countries of the EU. An important momentum has to be given to the Associations and Universities. A strong link has to be established between EFDA and the Universities of every country via the Associations. Various periodic Schools exist on fusion, where materials development is taught. This effort should continue and be complemented by a reactive system of grants to be attributed to PhD and Post-Doc on subjects published on a centralised basis. A permanent number of 10 PhDs and 10 Post-doc appointments per year should be envisaged.

In conclusion, human resources is the main bottle neck for the near future.

[40] For comparison the human resources of the French National Fast Breeder programme was ~550 scientists, engineers and technicians around 1980. Within this effort ~100 people were devoted to Basic Research on Radiation Effects. In the 2008/09 EFDA work programme ~20 PPY per year are foreseen.

The additional human resources for the research areas presented above are summarised in the table 1 below. It must be stressed that these realistic figures require an important derivative versus time, which can be achieved only if an important teaching and training effort is targeted to fusion materials science and technology. In addition the priority is not proportional to the human resources requested

The figures given for years 5 to 10 are above the present capacity. The aim should be to arrive within a five years time span at the upper limit (indicated by an →). This seems to be achievable with a focused education and training programme of 20 ppy/y and including additional capacities that have not been used in fusion before. In particular the contribution from Universities in the areas of modelling and development of W alloys could be strengthened.

VI- Other Expenditure

The expenditures are estimated as follows for the 20 forthcoming years:

- 20 M€ for ion beam irradiation and advanced characterisation (TEM, TAP, PAS…).
- 25 M€ for in-pile testing and post-irradiation testing.

In addition, the expenditure for the base programme qualification under fission neutron spectra is estimated to be 35 M€ for the next decade [41] and the insulator materials irradiation testing a few hundred k€/year.

Table 1: Overall resources required for the Fusion Materials Programme

	Year 0-5	Year 5-10	Year 10-15	Year 15-20
Base line programme on structural materials development and characterization towards ITER TBM and DEMO ("EUROFER family")	30→40 ppy/a	30 ppy/a	30 ppy/a	40 ppy/a
Base line programme of materials for DEMO (EUROFER ODS, breeder, neutron multiplier, barriers)	15→20 ppy/a	25 ppy/a	25 ppy/a	20 ppy/a
Innovation for high temperature radiation resistant steels (ODS and/or conventional)	15→20 ppy/a (both lines)	20 ppy/a (both lines)	20 ppy/a (after selection of the most promising line)	20 ppy/a
Innovation for refractory materials for high heat flux PFC	20 ppy/a	20 ppy/a	20 ppy/a	20 ppy/a

[41] *This number gives the order of magnitude for neutron irradiation campaigns. Other costs (e.g. procurement of materials and qualification) over the full two decades will be worked out more precisely later.*

Dedicated programme in preparation to IFMIF (Modelling, in-depth characterisation, fabrication of specimens) (*including 10 ppy/a related to radiation modelling)	10→20 ppy/a (*)	30 ppy/a (*)		
“IFMIF” Programme (exploitation, i.e PI, testing for qualification, feedback to modelling and improvements in R&D)		10 ppy/a	Modelling 20 ppy/ a PI 20 ppy/a *(Site-specific 30 ppy/a)*	Modelling 20 ppy/ a PI 20 ppy/a *(Site-specific 30 ppy/a)*
Innovative high temperature materials (SiC/SiC)	10 ppy/a	10 ppy/a	10 ppy/a	10 ppy/a
Basic modelling development and its experimental validation	20 ppy/a	20 ppy/a	30 ppy/a	30 ppy/a
Insulating Materials	5 ppy/a	5 ppy/a	5 ppy/a	5 ppy/a
Materials Education and training	10 PhD 10 post-docs/y	10 PhD 10 post-docs /y	10 PhD 10 post-docs /y	10 PhD 10 post-docs /y
TOTAL (not including PhDs and post-docs) to be compared to <100ppy/a today	120→150 ppy/a	165 ppy/a	175 ppy/a (IFMIF-site-specific additional 30 ppy/y	180 ppy/a (Site-specific 30 ppy/y)

ANNEX 4
Existing and required facilities for technology R&D
Detailed Mapping Tables

Table A4.1
Technology Facilities: Mapping of Missions and Milestones to Required Generic Means of Execution

Missions	Aims	Required Generic Means of Execution
Construction of ITER (for all Missions)	Testing and qualification of components	IR1: Strand and cable structural materials testing. IR2: Conductor testing. IR3: Magnet testing. IR4: Low & high heat flux component testing. IR5: ECRH testing. IR6: ICRH testing IR7: NBI testing IR8: Cryopump systems testing IR9: Port plug testing IR10: Tritium (fuel cycle) system testing
	Remote handling	IR11: Divertor RH IR12: NBI RH IR13: Transfer casks
	Licensing	IR14: Safety – related testing IR15: Dust and tritium measurement and removal techniques.
	Other	IR16: the ITER Test Blanket Modules and programme.
	Milestones	
Mission 1	Not Applicable	NA
Mission 2	NA	NA
Mission 3	Predictive capability for all aspects of plasma wall interactions (erosion, migration, redeposition, tritium retention, mixed materials, sheath physics, etc.) benchmarked on experiments.	MR1: Plasma wall interaction simulators. MR2: Plasma facing component testing. MR3: High performance computers for modelling of materials.
	Selection of dedicated material(s) tested under neutron irradiation and optimised for plasma-wall interaction processes (tritium retention, embrittlement, erosion). Due to long lead-time this programme needs to be initiated today. (See Mission 6)	MR1 – MR3, plus: MR4: Facilities for non-irradiated materials characterisation. MR5: Plasma facing materials irradiation. MR6: Facilities for post-irradiation examination.
Mission 4	Specific H&CD (as needed: neutral beam energy, ECRH, LH etc. tubes, launchers, power	MR7:Test Beds for H&CD systems (improvement of long pulse capabilities)

	supplies etc.) and other steady state requirements defined and R&D launched).	
	R&D on H&CD, PFCs and other steady state components completed.	See under Mission 3.
Mission 5	NA	The high performance computing required for this Mission could also satisfy the analogous requirements for materials modelling
Mission 6	ITER: Qualification, including validated modelling, of reference EUROFER for the Test Blanket Modules.	MR2 – MR5, plus: MR8: Neutron irradiations of EUROFERs (and other structural and functional, DEMO materials) MR9: Charged particle beam irradiations of model alloys and EUROFERs.
	ITER: Alternative divertor armour materials available (W tbc)	MR1 – MR4.
	DEMO: preselection of divertor and blanket concepts, including coolant and materials.	MR1 – MR6, MR8, MR9, plus: MR10: *Not used*. MR11: Facilities for further blanket developments (beyond ITER TBMs) MR12: Tritium (fuel cycle) facilities. MR13: Helium loops with relevant parameters. MR14: Test facilities for the qualification of mock-ups and prototype components.
	DEMO: development and qualification of structural and functional materials.	MR2 – mR5, MR8 – MR10 plus: R22: IFMIF construction and test programme.
	DEMO: final selection of divertor and blanket concepts, including coolant and materials on the basis of R&D (including results from IFMIF, TBM and validated modelling).	R8 – R13, R15 – R22, plus: MR16: Facilities for the qualification of in-vessel maintenance procedures.
	DEMO: Confirmed strategy for clearance and recycling of DEMO's activated materials.	tbd.
Mission 7	Completion of DEMO conceptual studies, taking account of all requirements, in particular for maintenance (remote handling), availability, tritium generation, waste management and electricity generation, taking into account plasma scenarios developed under other missions.	All the above, plus: MR17: Facilities for the further development and qualification of H&CD systems. MR18: Facilities for the qualification of Balance of Plant components. MR19: Facilities for further strand and conductor development, test facilities for advanced model coils.
	Confirm the feasibility of the proposed DEMO maintenance procedure by R&D.	See MR16, above.

	Completion of the DEMO engineering design activity and supporting R&D (including remote handling) and be ready for licensing, including a strategy for clearance and recycling of activated materials.	All the above.
	Assessment of the engineering feasibility of a stellarator power plant.	None. Covered by elements of the above.

Table A4.2
Technology Facilities: mapping from the required generic means of execution to requirements for key facilities, with note of existing facilities

Required means	Surplus capacity?	Key gaps in capacity?	Key facilities and comments	Existing Facilities
IR1: Strand testing.	Yes	No		Durham Walter Spring (UKAEA), Pacman (FOM), Tarsis (FOM)).
IR2: Conductor testing.	No	No	Satisfactory situation	Twente press (FOM), Sultan (CRPP), or Dipole (CRPP))
IR3: Magnet testing.	No	Yes	TF-PF windings cold tests (at 4K with low levels of current)	TOSKA
IR4a: Low-heat flux Be-compatible component testing	No	Yes	Beryllium-compatible HHFT for series production and acceptance tests of ITER FW panels	BESTH (IPP-CR) ETA-Beta (ENEA) + JUDITH I and JUDITH II (FZJ)
IR4b: High heat flux component testing	No	No		FE 200 (AREVA-CEA), GLADIS (IPP-Garching), JUDITH I and JUDITH II (FZJ).
IR5: ECRH system testing.	Yes, in part	Yes	Sufficiency/surplus: low power gyrotron test beds. Gap: A facility to test ECRH subsystems, such as transmission lines and launchers.	ECRH test beds in CRRP, FZK, IPP-Greifswald.
IR6: ICRH system testing	No	tbc	It remains to be checked if a specific /upgrade of existing test bed is needed for the ITER ICRH antenna	Several relevant facilities are available in Europe (e.g, JET, CEA, IPP-Garching) + tokamak experience.
IR7: NBI system testing	No	No	Neutral beam Test facility (NBTF) to be built in Padua	Facilities at JET, Garching and Cadarache for source development and NB physics
IR8: Cryopump system testing	No	No	Satisfactory situation	TIMO (FZK)
IR9: Port plug testing	No	Yes (tbc)	In addition to ITER on-site facilities, need for EU facility is being assessed in collaboration with other ITER partners.	

			In particular Diagnostics integrated into port plugs will require a significant test programme.	
IR 10: Tritium fuel cycle components testing	No	No	ITER testing requirements (isotope separation system, water detritiation system, exhaust from tokamak operation) can be fulfilled	TLK JET Active Gas Handling System
IR11:Divertor RH	No	Yes	Divertor Test Platform. Allows simulation of divertor in-vessel maintenance operations using prototype divertor RH equipment (Movers, end-effectors and tooling) and in a full scale mock-up of ITER divertor region.	DTP2, VTT, Finland
IR12: NBI RH	No	No	NBTF in Padua will provides a test bed for the RH tools	
IR13: Transfer casks.	No	Yes	Transfer cask transport and docking operation and in-cask operation.	
IR14: Safety – related testing	No	No	Dust explosion, flame propagation for gas-dust mixture, Hydrogen-air and hydrogen-dust combustion, Arc simulation behaviour (busbar…), H_2 and inert gas mixing and distribution, Corrosion product release and transport (high Temp., high pressure and high velocities water).	DUSTEX (FZK), PROFLAM I (FZK), PROFLAM II (FZK), HYDEX (FZK), VACARC (FZK), LONGARC (FZK), MISTRA (CEA), CORELE (CEA)
IR15: Dust and tritium measurement and removal techniques	No	Yes (tbc)	An assessment is under way to evaluate the need for an integrated facility representative of the complex geometry of the ITER Vacuum Vessel for studies of dust mobilization, dust measurement and removal. In situ and dust removal techniques neec to be	

			developed and qualified (most tests foreseen in tokamaks).	
IR16: The ITER Test Blanket Modules and programme.	No	No	Essential, highest priority.	MEKKA (FZK), TRIEX (ENEA), DIADEMO (CEA), HEBLO (FZK), PICOLO (FZK), IPUL-MHD (Latvia), HeFUS3 (ENEA) HELOKA-TBM (FZK), PbLi EBBTF loop (ENEA)

MR1: Plasma wall interaction simulators.	No	Yes	ELM/disruption simulation facilities are desirable, and not fully/reliably covered by existing international collaborations.	MAGNUM-PSI (FOM), Integrated PWI facility (IPP Garching), plasmatron Vision (SCK.CEN-Be), International collaboration PISCES-B (Be compatible) UCSD; Russian plasma guns (QSPA, MK-200 UG)
MR2: Plasma facing component testing.	No	No	This includes for example facilities for acceptance of joints, non destructive examinations, etc.	SATIR (CEA)
MR3: High performance computers.	No	Yes	Needed for multi-scale modelling of materials bombarded by plasmas and/or neutrons.	
MR4: Facilities for non-irradiated materials characterisation.	Yes, in part	tbc	A careful investigation is needed to determine redundancies. There is redundancy with respect of conventional mechanical testing, while for high temperature tests and modern micro structural analyses (TEM, ATP) additional facilities are needed	
MR5: Plasma facing materials irradiation.	No	No	Prior to the availability of IFMIF, fission reactors must be used in addition to additional irradiation facilities as for MR15	

			below	
MR6: Facilities for post-irradiation examination.	tbc	tbc	A careful investigation is needed to determine redundancies and gaps.	
MR7:Test Beds for H&CD systems (improvement of long pulse capabilities and of efficiency)	No	tbc	NBTF, which is planned to be build in Padua, could play a key role in developments towards DEMO. Facilities might be needed for other heating systems.	
MR8: Neutron irradiations of structural and functional DEMO materials.	tbc	No	Prior to, and in complement to, the availability of IFMIF, irradiations must be undertaken in fission reactors, spallation sources and (exclusively for the benchmarking of neutronic and activation modelling) low flux 14 MeV sources. At a first glance there is a surplus capacity in these areas. but only few reactors are available and currently fully booked which can achieve the proper testing requirements (e.g., instrumentation or temperature control, temperatures in the range of 250-600C, reasonable fluence). Successful irradiations to high dpa depends on continued access to reactors in Russia.	
MR9: Charged particle beam irradiations of model alloys and EUROFERs.	No	tbc	Priority should be given to facilities needed to validate modelling activities and simulate the typical fusion He/dpa ratio in materials (e.g. dual/triple beam facilities). Increase in implantation depth desireable.	
MR11: Facilities for further blanket developments.	Yes	No	Existing facilities are more than sufficient, apart from tritium and	

			helium facilities (below), and irradiation facilities for functional materials (as for structural and armour materials).	
MR12: Tritium (fuel cycle) facilities.	No	Yes	Tritium facilities for relevant in-vessel component cooling media, e.g., helium facilities for single integrated tests of processes and components. If water cooling were re-instated in the blanket programme, analogous facilities would be required. The continued availability of tritium laboratories (though currently provision is adequate) is essential.	
MR13: Helium loops with relevant parameters.	No	Yes	See MR19. Also HELOKA TDM for helium-cooled divertor development.	
MR14: Test facilities for the qualification of mock-ups and prototype components.	Yes	No	There is surplus capacity, apart from the items mentioned above.	
MR15: IFMIF construction and test programme.	No	Yes	Essential, urgent and the highest possible priority. In particular, there is a need to accelerate development of lithium technologies (e.g., fluid dynamics modelling and validation, impurities monitoring and purification techniques, corrosion).A medium-size Li-loop is needed to test the effects of high power deposition on Li fluid dynamics.	
MR16: Facilities for the qualification of in-vessel maintenance procedures	No	Yes	A major remote maintenance laboratory.	
MR17: Facilities for	tbc	tbc	A study is needed at a	

the further development and qualification of H&CD systems.			later date.	
MR18: Facilities for the qualification of Balance of Plant components.	tbc	tbc	A study is needed at a later date	
MR19: Facilities for further strand and conductor development, test facilities for advanced model coils.	tbc	tbc	A study is needed at a later date.	

Annex 5
Analysis of the experimental needs of the satellite tokamak programme

1. Introduction

The detailed goals of the mid-term and long-term scientific programme in fusion in Europe and their relevance in view of ITER and DEMO are described in the core document and summarised under the following specific Missions:

1. Burning Plasmas
2. Reliable Tokamak Operation
3. First wall materials and compatibility with ITER/DEMO relevant plasmas
4. Technology and Physics of Long Pulse and Steady State
5. Predicting fusion performance
6. Materials and Components for Nuclear Operation
7. DEMO Integrated Design: towards high availability and efficient electricity production

The general mission of the satellite programme, which is assumed to be in parallel with ITER construction and operation, is:

- to optimise the concepts used in the ITER exploitation in conditions relevant for a reactor and to contribute to the consolidation of the ITER design choices;
- to contribute to the advancement of the physics understanding by extensive plasma diagnostics and modelling tools;
- to complement ITER in the testing of innovative technologies that are not yet foreseen to be tested on ITER itself;
- to contribute to filling the gap between ITER and DEMO in the development of robust regimes of operation characterised by more advanced plasma parameters, especially the investigation of regime sustainment compatible with high fusion gain operation with a minimum number of actuators and sensors, which would have beneficial effects on e.g. the capital cost of and the cost of electricity from a reactor.

The aim of the present document is to elaborate a scientific and technical view on the overall experimental needs of the Satellite Programme. This requires the range of parameters and technological areas that the satellite programme must explore to be defined. Specifically, the experimental needs to fulfil Missions 1-4 are discussed. Although Mission 5 is not addressed, it should be noted that since it is impossible to reproduce the entire range of parameters of ITER and DEMO within the devices that will contribute to the satellite programme, the adequacy of the satellite programme to satisfy Mission 5 should not be judged only on the basis of the **capability of directly simulating ITER/DEMO relevant conditions** but also on **the capability of investigating a range of physics parameters relevant for an extrapolation based on theoretical understanding and advanced numerical simulation tools.**

The document is organised as follows. The criteria for the analysis are established in Section 2. Section 3 and Tables I-IV provide an assessment of Missions 1-4 of the satellite programme on the basis of the criteria established in Section 2. Concluding remarks are given in Section 4.

2. Criteria for the analysis

In order to fulfil Missions 1-4 above, the overall experimental needs for the satellite programme have been analysed on the basis of the following criteria:

- *Focus on issues that cannot adequately be addressed on ITER*
- *Coverage of ITER/DEMO relevant dimensionless parameters* with respect to thermal plasma, fast particles and edge/divertor plasma conditions. This criterion determines the scientific area that can be investigated and constrains the main engineering parameters (plasma current, magnetic field and dimension) of a facility and the characteristics of its auxiliary heating systems;
- *Pulse length requirements*. Different physics/technology issues have different characteristic time scales and to address them properly sets a minimum pulse length capability for a facility;
- *ITER/DEMO relevant technologies*. Capability of addressing the main ITER/DEMO issues in different technology areas (Plasma Facing Components and Heating and Current Drive systems).

In addition, flexibility in operations and enhancement is crucial for investigating different operating conditions, testing new components and improving the diagnostic and control capability.

A short introduction to some of the most relevant dimensionless parameters, their meaning and significance associated with each of these criteria can be found in Annexes 5.A and 5.B.

Whether these Missions can be fulfilled by one device or by a set of devices is analysed in another part of this document. Nevertheless, the assumption is made that JET, being the only device that can at present use tritium, will cover tritium and alpha particle related needs of the programme up to the DT phase of ITER. However, the satellite programme as a whole, including JET, will also address burning plasma physics studies by the generation of fast ions by auxiliary heating systems.

3. Conditions needed to fulfil Missions 1-4

The conditions needed to address Missions 1-4 (Burning Plasmas, Reliable Tokamak Operation, First wall materials and compatibility with ITER/DEMO relevant plasmas, and Technology and Physics of Long Pulse and Steady State) are specified in Tables I-IV respectively on the basis of the criteria established in Section 2.

4. Conclusions

Tables I-IV clearly show that tokamaks in operation in parallel with ITER have to include also one or more devices with the magnetic field strength, size and heating power characteristics of JET/JT60SA (3-5 MA class). They will contribute, together with ITER to the most reactor relevant data points. These tokamaks do not make obsolete, however, smaller (i.e. present “mid-size” 1-2 MA class) devices, which can first test many novel ideas faster, with less effort and reduced hardware risk. The latter will also play a major role in the continuing development and test of theoretical models and, of course, in the education of young scientists.

TABLE I: Conditions needed to fulfil Mission 1: Burning Plasmas

The conditions needed to address the above issues are specified on the basis of the criteria established in Section 2.

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Coverage of ITER/DEMO relevant dimensionless parameters			
Thermal plasma ρ^*	Below specified upper limit	Requirement that fast ions slow down mainly by collisions with electrons yields condition on ratio between fast ion energy and electron temperature that can be cast (for a given value of the fast-ion ρ^*) as condition on maximum value of thermal plasma ρ^*	Satisfied only in JET-class device if simultaneous with high β_N
Thermal plasma β	In the same range as ITER/ DEMO values	Requirement that fast-ion instabilities and effect of thermal plasma stability on fast-ions is investigated in relevant conditions	
Fast-ion ρ^*	Appropriate choice of fast-ion energy	Sufficiently close to ITER/DEMO values to investigate fast-ion dynamics in the relevant range of finite-orbit effects	≈500keV-1MeV for JET-class device
Fast-ion collisionality	Regimes with not too large electron temperature	Requirement to have short electron-ion equipartition times and short slowing-down times compared with the energy confinement time	Simultaneous match of fast-ion and thermal plasma collisionalities not possible in general at fixed thermal plasma β; resulting thermal plasma ν^* is higher than ITER/ DEMO values

TABLE I (continued): Conditions needed to fulfil Mission 1: Burning Plasmas

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Coverage of ITER/DEMO relevant dimensionless parameters			
Fast-ion β	In the same range as ITER/DEMO values	Requirement to reproduce the fast-ion drive	Automatically satisfied provided fast-ion collisionality and thermal plasma β matched (if plasma heating occurs mainly through fast particles)
Pulse length requirements			
t/τ_{res} τ_{res} is current redistribution time	~1	Needed to form target q profile	The pulse length needed to investigate steady-state operation is anyway much longer than this time scale, therefore this condition is fulfilled on a device able to address steady-state physics
ITER/DEMO relevant technologies			
	Highly reliable ICRH systems (e.g. matching systems resilient to changes in edge conditions) Negative ion based neutral beam technologies	Fast particle energy in range 500keV-1MeV	Although technologies are available, these requirements could stimulate further development (e.g. development of high current density negative ion source). Both developments are of direct interest for DEMO and will offer opportunities for progressing in synergy with ITER H&CD systems in more flexible operating conditions

TABLE II: Conditions needed to fulfil Mission 2: Reliable Tokamak Operation

The conditions needed to address the above issues are specified on the basis of the criteria established in Section 2.

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Coverage of ITER/DEMO relevant dimensionless parameters			
Thermal plasma β and ν^*	In same range as ITER/DEMO values	Requirement to reproduce thermal plasma stability properties	
Thermal plasma ρ^*	Sufficiently close to ITER/DEMO	Some aspects of ELM/NTM physics depend on ρ^* (e.g. seed island size, pedestal width)	
Pulse length requirements			
t/τ_{res}	>1	Sufficiently long for plasma control studies	
ITER/DEMO relevant technologies			
	• Disruption mitigation tools • Reliable H&CD systems for NTM control (ECCD, ICCD, LHCD) • Different methods for ELM control (Resonant Magnetic Perturbation, pellets, etc.)		H&CD systems for reliable NTM control would benefit strongly from development of step-tunable gyrotrons that might avoid the need for steerable ECRH antennae and increase the spatial coverage of the ECCD system

TABLE III: Conditions needed to fulfil Mission 3: First wall materials and compatibility with ITER/DEMO relevant plasmas

The conditions needed to address the above issues are specified on the basis of the criteria established in Section 2.

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Coverage of ITER/DEMO relevant dimensionless parameters			
Thermal plasma β and ν*	Sufficiently close to ITER/DEMO values	Interplay between different MHD modes (NTM, sawteeth, ELMs) calls for thermal plasma β and ν* to be matched, both in core and pedestal regions	Necessary condition but not sufficient since also the ratio f_{GR} must be sufficiently close to ITER/DEMO
Divertor plasmas conditions	P/R and n_sR (with P the heating power conducted in SOL, n_s the SOL density and R the major radius) sufficiently close to ITER/DEMO values	P/R and n_sR determine atomic physics which should be reproduced	
$f_{GR}=n/n_{GR}$ Greenwald fraction f_{GR} is ratio between plasma density n and Greenwald density n_{GR}	~1	ITER operates close to Greenwald limit (ratio f_{GR}=0.85) and Power Plants are foeseen to work 20%-50% above Greenwald with confinement above the ITER98 H-mode scaling	To achieve simultaneously low ν* and high f_{GR} values, ρ* must be sufficiently low. This requires JET-class devices
$F_{rad}=P_{rad}/P_{heat}$ Ratio of radiated power to total heating power	Sufficiently close to ITER/ DEMO values	Reasonable values of power load on the divertor plates require regimes with high radiated power fraction	

TABLE III (continued): Conditions needed to fulfil Mission 3: First wall materials and compatibility with ITER/DEMO relevant plasmas

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Pulse length requirements			
t/τ_{res}	>1	High radiative power fraction regimes may have to be prepared from 'normal' H-modes. This may involve significant pressure-profile changes and to establish stability will require some current redistribution times	
t/τ_{equip}	A pulse length of 30s seems adequate	PFCs should work at constant surface temperature for long enough so that this phase dominates the increase and decrease time of the surface temperature before and after the heating phase: this ensures that erosion/redeposition is studied properly in steady conditions High heat load actively-cooled PFCs have characteristic times of about 1-2s (water has to be close to surface to ensure sufficient power handling)	Concerning the wall thermal equilibration time, the argument is still valid if part of the mission is to study highly radiating scenarios: power loads on the wall are still important for actively-cooled components to be needed. However the heat load being smaller than on the divertor, technologies with longer characteristic cooling times (about 10s) could be used
t/τ_p, τ_p being characteristic time for particle content		Characteristic time for particle content can be much longer, especially when using material such as carbon with strong affinity for deuterium (no wall saturation on Tore Supra after 6 minutes plasma operation)	This study should be performed within Mission 4

TABLE III (continued): Conditions needed to fulfil Mission 3: First wall materials and compatibility with ITER/DEMO relevant plasmas

Criterion	**Conditions/ requirements**	**Rationale**	**Comment**

	for matching /approaching ITER/DEMO values		
ITER/DEMO relevant technologies			
	Test PFC materials of interest for ITER and DEMO Actively cooled PFCs at relevant heat loads	ITER relevant materials include: • Carbon-Tungsten-Beryllium • Full-metal (Tungsten-Beryllium) DEMO-relevant materials include Tungsten as a material for the entire wall Joining technologies for tungsten onto water-cooled assemblies would have to be developed and, perhaps, tested in a plasma environment, as a pre-selection for testing with 14MeV neutron irradiation A steady-state heat load of $5MW/m^2$ on ITER and up to $20MW/m^2$ on DEMO Model D, unless a higher fraction of power is lost by radiation ($5\text{-}15MW/m^2$ is assumed in Power Plant Conceptual Study) An effective test of PFCs with transient loads associated with ELMs and disruptions requires plasmas with sufficiently high stored energy	This requires JET-class devices capable of achieving low pedestal collisionality at high density

TABLE IV: Conditions needed to fulfil Mission 4: Technology and Physics of Long Pulse and Steady State

The conditions needed to address the above issues are specified on the basis of the criteria established in Section 2.

Criterion	**Conditions/ requirements for matching /approaching ITER/DEMO values**	**Rationale**	**Comment**
Coverage of ITER/DEMO relevant dimensionless parameters			
Thermal plasma β and ν^*	Close to β-limits foreseen on a reactor at sufficiently low ν^*	Power plant must operate in steady state at $\beta_N \approx$ 3.5-4.5 High bootstrap fraction requires high β and low ν^*	More easily met in small, low magnetic field devices, but extrapolation to a reactor requires JET-class devices that can investigate steady-state scenarios as ρ^* approaches ITER/DEMO values
$f_{GR}=n/n_{GR}$	$\approx$ 1	Steady state regimes require high density to achieve compatible divertor conditions	Requires JET-class devices that can approach simultaneously high f_{GR} and low ν^*
Pulse length requirements			
t/τ_{res}	~several	To demonstrate true steady-state requires several current redistribution times	
ITER/DEMO relevant technologies			
		Test tools to control and optimise plasma scenarios (in addition to those considered under Mission 2), in particular to control RWMs by magnetic coils There is an overriding need for efficient off-axis current drive. There is also a need for a complete suite of all the different H&CD methods to be installed to enable: • current drive efficiencies to be tested at high power densities (including collective instability effects) • scenarios to be developed in which the substitution of one current drive method by another, and the consequent reduction in the number of separate actuating systems used to control an advanced scenario	For all the H&CD systems, the drive to optimise steady-state can be used to bring in technology developments which would improve reliability and availability for DEMO systems Optimising and improving the reliability and power efficiency in various scenario control methods could also drive developments of relevance to DEMO (e.g. the development of step tunable gyrotrons)

		can be studied H&CD methods which are not presently envisaged as day-one systems on ITER should be tested, in particular a Lower Hybrid PAM launcher in reactor-relevant conditions	

Annex 5.A

Extrapolability to ITER and DEMO
(K. Lackner)

The ITER satellite tokamak(s) will contribute to the evolution of ITER systems, to the preparation of ITER scenarios, and will generally broaden the experimental basis for our theoretical modelling capability. They have, however, also to extend the experimental basis beyond ITER in those areas where the requirements of DEMO significantly exceed the design requirements of the former. This is particularly the case for the achievable values of β and for the divertor heat loads, which are both intimately linked to the cost efficiency of fusion power production. To be relevant for DEMO, ITER satellite(s) will have, of course, to satisfy certain minimum requirements on dimensionless parameters, even though issues directly linked to thermonuclear heating will be left to ITER, and satellites will not need to approach reactor-like values of $nT\tau_E$.

The reactor relevance of devices is usually measured in terms of the dimensionless plasma physics parameters: $\rho^* = \rho_i / a \propto \sqrt{T} / (aB_t)$, $\nu^* = Rq/\lambda_{mfp} \propto an_e / T^2$, $\beta_t \propto n_e T / B_t^2$ together with the trivially dimensionless ones fixing the plasma geometry and the poloidal/ toroidal field ratio ($\kappa, \delta, R/a, q..$). [42] The full set of these parameters, however, is actually only known once the device is operating, and it is convenient to use instead dimensionless "engineering" variables: $B^* \propto B_t a^{5/4}, P^* \propto P_{heat} a^{3/4}, n^* \propto na^{3/4} / B_t$, [43] which are fixed by the design or can - within limits - be directly imposed. This allows to plot operational points of geometrically similar devices in a 3-d space, which reduces to 2-d if one assumes $n^* \approx const$, comparing operation only at approximately similar Greenwald density. In this universal map of design parameters the achievable plasma physics conditions, represented by the physics parameters ρ^*, ν^*, β_t can be plotted as contours, but depend now, of course, on confinement law assumptions.

	Bt	Pheat	R
AUGmin	1	2,5	1,63
AUGmax	2,5	25	1,63
JETmin	1	5	3
JETmax	3,5	40	3
ITERmin	3,5	50	6
ITERmax	5,2	150	6
DEMO	6,00	500,00	7,50

Table 1

[42] Thermonuclear performance issues are to be explicitly excluded from this similarity, as nuclear reactions are not directly related to the plasma physics dimensionless parameters - as are atomic physics aspects entering, e.g. impurity behaviour, edge physics, NBI penetration.

[43] Their choice is not unique: the present selection is linear in one main engineering parameter and involves, in addition only plasma dimension. An exception is density, where the present choice of n^* is much closer to the canonic Greenwald/Hugill/Murakami density than the choice na^2 following from the above rule.

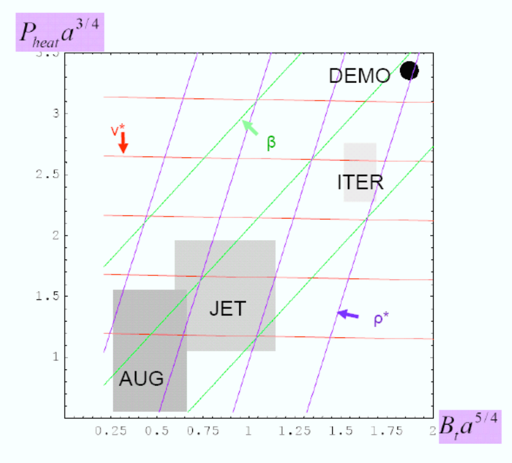

Fig.1

Fig.1. shows such contours, based on the ITER-98$_{y2}$ scaling-assumption in a log-log presentation, with contours spaced apart by factors of 2, together with a reference DEMO design point , and the territory covered by the assumed useful operating ranges (see table 1) of ASDEX Upgrade, JET and ITER. (The parameter range of JT60SA, a designated ITER Satellite tokamak, closely coincides with that of JET.) The arrows in this figure – like in all following ones - point in the direction of increasing value of the indicated parameter.

Fig.1 confirms the general knowledge that today's devices can attain - at least from confinement, field and power constraints - the β_t values of DEMO and ITER, but do so at significantly lower values of $\rho *$. Even larger, however, are the difference in $\nu *$, as - at given Greenwald densities - present devices are much more collisional than ITER&DEMO.[44] Fig.2 shows, in the same coordinate space, that ITER&DEMO will have also a somewhat stronger coupling between electrons and ions (measured by $\nu_{ei}\tau_E$), that DEMO will be more comfortably above the L/H power transition threshold (measured by P_{LH}/P_{heat}) than ITER and that current equilibration on DEMO will take many more energy confinement times than either on ITER or present devices (measured by τ_{skin}/τ_E).

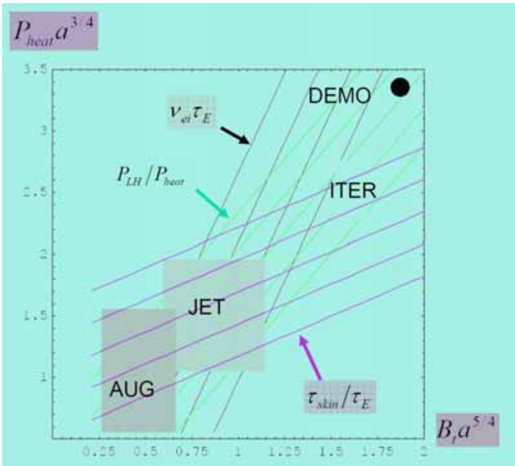

Fig.2.

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[44] This observation holds even stronger, if operation at the exact Greenwald density - rather than its proxy $n*$ - is considered. The parameter $\nu *$ used here is the one appearing in neoclassical theory; drift wave turbulence or NTM physics depend on $\nu_{drift} \propto \nu * / \rho *$, which varies somewhat less.

From these two landscapes it appears clear that a direct simulation of the DEMO parameter regime in a satellite-class device (which will presumably be in the B_t, a, P_{heat} region of JET or JT60SU) will not be possible, in particular due to the incompatibility of the Greenwald density and the collisionality requirements. The extrapolation of results of satellite devices will therefore strongly rely on improved theoretical understanding and modelling, and the separate satisfaction of (e.g.) the $\nu^*, \nu_{ei}\tau_E, \tau_{skin}/\tau_E$ requirements on one, and the Greenwald density similarity on the other hand. We will have to rely not only on integrated, but also "integrating" modelling!

A fusion reactor like DEMO will operate at higher values of β_t (or β_n) than ITER, and over much longer time scales (as measured in either τ_E or τ_{skin}) than present devices, which have partly - but only transiently – already achieved the desirable β-values. The steady-state attainment of this regime necessitates instead control on three quite different time scales, which involve also quite different physics challenges. The stable β-range of tokamak operation depends on pressure and current ($=q$) – profiles and stationarity on the longest time scale is primarily a challenge to current profile control[45]. On an intermediate time-scale slowly growing NTMs appear, if the q-profile allows for their existence. They have to be controlled by direct feedback on the local current distribution within the island region, and ECCD appears to be the method of choice for this. To improve ultimately also the ideal-MHD mode limit, resistive wall mode control, through either magnetic field feedback or toroidal momentum input is required. The need for this is quite pronounced for "advanced" plasma modes of operation with flat or hollow q-profiles, which have low no-wall limits for low-n mode numbers: fortunately in their case also the potential gain from RWM control is large.[46]

By and large, the request for β-improvement will be one main task for the ITER satellite program. While the pioneering work of developing sensors, control strategies and new actuators can be carried out to a large extend on smaller devices with shorter pulse lengths, the ultimate integration of all techniques and the coverage of all time scales requires a superconducting device, with powerful and highly flexible heating and current drive systems.

Divertor power loads are the second key area, in which the presumed DEMO requirements will significantly exceed ITER design values. Table 2 compares some relevant parameters for ITER in two operating scenarios, for two reactor designs considered in the EU power plant study, for one advanced (= aggressive) US power plant concept, and for JET and ASDEX Upgrade. For this purpose a reference heat flux $q_{div,norm}$ [47] is defined, based on the assumption of 80% radiation losses, and a scaling of the scrape-off layer width with device dimension $\sim\sqrt{R}$, normalized to reproduce for ITER the 5 MW/m^2 resulting from more elaborate

[45] Pressure profiles depend on heat source and transport, and it appears rather futile to compete, at reactor Q-values, with α-particle heating. Control of transport is probably also most effective through current profile manipulation.

[46] Improved H-mode type profiles, with higher l_i have higher no-wall limits, and a small difference between wall and no-wall limits even for low-n modes

[47] $q_{div,norm} = (P_{heat} - P_{rad})/(4\pi R_o \lambda F)$, with mid-plane scrape-off layer decay length $\lambda = 0.003\sqrt{R_o}$ and a mid-plane to divertor expansion factor F = 10.

models[48] . This number can be compared to the tolerable fluxes (q_{design}) assumed in the respective engineering studies. Another figure of merit is the ratio P/R of power to linear scale that can be derived assuming atomic physics and collisionality to play the dominant role in the divertor. ITER and present day experiments are within a factor of 2 in both $q_{div,norm}$ and P/R, whereas reactor designs exceed even the ITER values by factors of 4-6.

	P_{fus} [GW]	R_o [m]	P_{rad}/P_{heat}	Q	$q_{div,nom}$ [MW/m²]	q_{design} [MW/m²]	P/R
ITER-ref.	0.5	6.2	0.8	10	5	10	24
ITER-SS	0.36	6.2	0.8	5	5	10	23
PPCS-B	3.4	7.5	0.8	15	22	10	120
PPCS-D	2.5	6.1	0.8	35	20	5	94
ARIES-AT	2.2	5.5	0.8	51	10-20	5	92
ASDEX-Up	**0.025**	**1.65**	**0.8**	**0**	**6.2**		**15**
JET	**0.04**	**3**	**0.8**	**0**	**4.1**		**13**

Table 2

At the same time, the tolerable power fluxes in reactors are likely to be lower than those on ITER, due to the required longer intervals between replacements and the move away from water as divertor coolant. The engineering problems of divertor heat load tolerance and removal will therefore increase significantly even beyond the already severe ones of ITER, and novel physics concepts will be required to help in their solution. The only established method consists in controlled radiation enhancement, which, however, will have to be increased to unprecedented fractions of the heating power (up to the 95% range). As this is a highly nonlinear problem, and atomic physics implies severe additional constraints, such results also cannot simply be extrapolated from lower to higher heat fluxes.

Testing this or other solutions to the power load question at DEMO relevant power fluxes appears, however, impossible in conventional tokamaks like a mainstream “satellite”, as a straightforward large increase in heating power would primarily result in a violation of β -limits[49]. Two escape routes from this dilemma appear possible: either a very low aspect ratio device, capable of reaching very high values of β, but being geometrically dissimilar to ITER&DEMO, or the exploitation of the plasma-physics invariance into the range of compact, high-field devices. Plasma physics-wise identical experiments (ρ^*, ν^*, β = const) require for the attainment of a given regime a heating power scaling like $P_{heat} \propto R^{-3/4}$, corresponding to very strong increases in P_{heat}/R $\left(\propto R^{-7/4}\right)$ and $q_{div,norm}$ $\left(\propto R^{-9/4}\right)$. A device at ½ the linear dimensions of JET, with 2.4 times its magnetic field strength, would require a P_{heat}/R about 3.4 times and a $q_{div,norm}$ about 4.7 times larger than JET to reach (at equal ρ^*, ν^*) the same value of β . It could therefore presumably attain DEMO divertor load conditions in plasma regimes already demonstrated on JET.

It appears therefore unlikely that all DEMO requirements not met by ITER research could be covered by a single, complementary device. A mainstream ITER satellite, supported itself by exploratory research on other tokamaks, would concentrate on improving

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[48] The values given for Aries-AT depend on whether credit is taken for double null operation

[49] the alternative of deliberately deteriorating confinement (by some as yet not identified procedure) would also introduce physics atypical for the desired operating regime of a reactor.

plasma control over all relevant time scales up to a sufficiently large multiple of the skin time, and the extension of the stationary operating range to reactor β_t -values (and bootstrap current fractions) with sufficiently good confinement. A tokamak in the JET range of $B^* \propto B_t a^{5/4}$, $P^* \propto P_{heat} a^{3/4}$, with a flexible heating and current drive system and the pulse length capabilities of a superconducting device should satisfy these basic needs. DEMO-critical plasma-wall interaction issues could only partly be addressed by this – or any presently existing – device in a direct manner, and would probably have to be covered by different experiments addressing diverse aspects. Any tokamak below the DEMO-scale will have difficulties in studying the very long pulse (largely technical) issues of plasma wall interaction at high plasma densities, as external current drive gets very inefficient in this regime. These aspects might actually best be delegated to the superconducting stellarators, which do not suffer from the Greenwald density constraint and require no current drive.

Annex 5.B

The mission of a satellite tokamak in support of ITER (focused on the burning plasma physics needs) (F. Zonca)

Introduction

The seven strategic "missions" for the fusion program, identified by Working Group 1 in support of ITER and in the perspective of DEMO, put "Burning Plasmas" among the issues of high priority and urgency. A detailed discussion of the physics issues that are peculiar to burning plasmas can be found in the report by A. Fasoli "How to be best prepared for Burning Plasma physics on ITER?" According to this report, the peculiar physics characterizing burning plasmas are:

- The physics of fusion produced alphas (fast ions), in particular their interaction with plasma instabilities and turbulence;
- The strong nonlinear coupling that will take place between fusion reactivity profiles, pressure driven currents, MHD stability, transport and plasma boundary interactions, mediated by the a particle population;
- The stability and control of the fusion burn.

The self-organized system behaviour of burning plasmas makes it possible to investigate the complex interplay of these aspects only in ITER-class devices. However, a satellite tokamak can address these issues separately, even if with different level of depth and extrapolability, and provide the necessary new physics insights that are necessary for the successful exploitation of a burning plasma experiment.

In the following, the key plasma parameters for a satellite tokamak to fulfil its mission of "Burning Plasma Physics and Control" are identified and discussed, with emphasis on the novel physics peculiar to burning plasmas that can be addressed in the satellite scientific program. The use of tritium is excluded a priori, since a satellite tokamak operating in sub-ignited regimes cannot provide a significant power density contribution from fusion alphas with respect to the fast ion power density delivered by additional heating sources (ICRH and/or NBI).

Burning plasmas peculiar physics issues and choice of dimensionless parameters

A unique characteristic of burning plasmas is that the energy density of fast ions (MeV energies) and charged fusion products is a significant fraction of the total plasma energy density. Consequently, one can address two major issues of practical concerns in such plasmas: (i) whether fast ions and charged fusion products are sufficiently well confined that they transfer their energy and/or momentum to the thermal plasma without appreciable degradation due to collective modes (and plasma turbulence); and (ii) whether, on longer time scales, mutual interactions between collective modes and energetic ion dynamics on the one side and drift wave turbulence and turbulent transport on the other side may decrease the overall thermonuclear efficiency of the considered system. In terms of consequences, these two issues have different practical implications: the first one has direct impact on the operation scenarios and boundaries, since energy and momentum fluxes due to collective losses may lead to significant wall loading and damaging of plasma facing materials; while the second one poses soft limits in the operation space. In the framework of plasma theory, meanwhile, the first issue is connected with the identification of burning plasma stability boundaries with respect to collective mode excitations by fast ions and charged fusion products as well as with the non-linear dynamics above the stability thresholds; and the second issue is associated with long time-scale non-linear behaviour typical of self-organized complex systems.

The choice of key dimensionless plasma parameters for a satellite tokamak, including those to fulfil its mission of "Burning Plasma Physics and Control", assume similarity of equilibrium profiles of the relevant quantities (density, temperature, current, etc.).

1. Fluctuation induced energy and particle transport reflect the fluctuation spectrum in frequency and wavelength. Given a similarity scaling of the thermal plasma parameters, discussed in Annex A), similarity arguments for the fast ion induced fluctuation spectrum require that the energetic particle characteristic dimensionless size is preserved. This condition provides a criterion for the choice of $\rho^*_{fast} \sim \rho^*_{fast,ITER}$.
2. Resonant excitations of collective modes by fast ions as well as the important role played by resonant particles in transport processes impose that the ratio of characteristic fast ions frequencies to the Alfvén transit frequency ω_A should be maintained. Characteristic fast ion frequencies are transit, bounce and precession frequency, depending on the particle orbit, all indicated here generically as ω_{fast}. The invariance $(\omega_{fast}/\omega_A) \sim (\omega_{fast}/\omega_A)_{ITER}$ reduces to preserving the Alfvén Mach number for circulating fast ions.
3. The strength of wave-particle interactions involving energetic ions provides a criterion for the choice of fast particle power density input, which can be controlled externally, i.e. β_{fast} or equivalently the ratio of the fast ion slowing down time to the energy confinement time: $\beta_{fast} \sim \beta_{fast,\,ITER} \Leftrightarrow (\tau_{SD}/\tau_E) \sim (\tau_{SD}/\tau_E)_{ITER}$.
4. A final but not less important condition is related with the dominant electron heating due to fusion alphas and fast ions in burning plasmas. This peculiar aspect implies a different weighting of the free energy channels readily available for turbulence drive with respect to present day experiments and is of crucial important for investigating the cross scale couplings of fast ion induced collective effects with micro-turbulence and the related transport processes. As an example, fusion alphas in ITER will deliver ~70% of their energy to electrons. This latter condition implies that fast ion populations in a satellite tokamak should have an energy sufficiently larger than the critical energy, i.e. $E_{fast}>E_{crit}$. Obviously this condition reflects the nature of the fast particle distribution function, e.g. it is different for perpendicular energetic ion tails generated by ICRH and for nearly tangential fast particle tails due to (N)NBI.

3. The unique role of burning plasma experiments.

The proximity of a fusion reactor to the ultimate target of *self-sustained burn* can be quantitatively expressed via the parameter $f_\alpha = P_\alpha/(P_L + P_b)$: the ignition condition, thus, becomes $f_\alpha = 1$. For a fusion reactor based on D-T fuel, the *fusion gain* Q, i.e., the ratio between the total fusion power and the net power input needed to maintain the plasma burn condition, can be expressed as $Q = 5 f_\alpha/(1 - f_\alpha)$. Thus, ITER operations at $Q = 10$ correspond to $f_\alpha = 2/3$, while DEMO at $Q > 50$ will have $f_\alpha > 0.9$. Since the self-organized behaviour of burning plasmas as complex systems can be investigated only in the presence of a substantial nuclear self-heating by charged fusion products (alpha particles in the case of D-T fuel), it is completely unknown how close one should go to the ignition condition, $f_\alpha = 1$, in order to observe this behaviour in reactor relevant conditions. Qualitatively, we reasonably expect that a transition is likely to occur at $f_\alpha = 1/2$ (corresponding to $Q = 5$), where the 50% of the local power balance is due to nuclear self-heating and, thus, the system is still significantly driven from external power sources.

The value of f_α also provides a measure of burn control problematic issues. At $f_\alpha < 3/4$ ($Q < 15$), burn control is essentially an optimization problem of plasma profiles (both thermal and fast ions) for fusion performance optimization, as discussed in the report by A. Fasoli (op.cit.), while thermal runaway is possible at $f_\alpha > 3/4$.

Studying burning plasma self-organization and burn control is strictly possible only in ITER-class devices. ITER itself may not be capable to provide a final answer for DEMO in this respect. Nonetheless, partial aspects of these problems, with the exception of self-organization of the radial profiles, could be investigated in a satellite tokamak, provided that the power density delivered by fast ions is a significant fraction of the total power density, $f_{\text{fast}} = P_{\text{fast}}/(P_L + P_b) > 1/2$. The possible role of a satellite tokamak in investigations of burning plasma self-organization and burn control is shortly discussed in the next section.

4. The possible role of plasma experiments in the sub-ignited regime.

The scientific programme of a satellite tokamak to fulfil its mission of “Burning Plasma Physics and Control” does not require the use of tritium or long pulse operations. In fact, additional ion heating is more efficient than fusion reactivity in producing fast ion populations at moderate $Q = \mathrm{O}(1)$; meanwhile, the intrinsic times of the physics processes discussed in Section 2 are shorter or of the order of τ_E. In this respect, plasma pulses of a few τ_E are sufficient provided that enough additional power is available for active current profile control.

Excluded the possibility of investigating the self-organized behaviour of burning plasmas as complex systems, which is *unique* to plasmas close to the ignition condition (Section 3), what can be studied in a satellite tokamak is whether fast ions and charged fusion products are sufficiently well confined that they transfer their energy and/or momentum to the thermal plasma without appreciable degradation due to collective modes and plasma turbulence (Section 2). Complex mutual interplays between micro-scale plasma turbulence and meso-scale collective modes or macroscopic MHD modes can be also addressed, at least partly. One of the scientific scopes of such device(s) would also be providing the experimental database for validation and verification of numerical simulations and fundamental theories that are necessary to extrapolate with confidence numerical modelling to burning plasma operations in future experiments. In this sense, diagnostics capability development is an essential element, which becomes a prerequisite itself. In fact, it is necessary to measure and characterize in “real time” both the space-time structures of fluctuations and the fast ion distribution functions with adequate resolution. The satellite tokamak could offer the opportunity of exploring new ideas and methods for the already mentioned diagnostics capability development.

Precious information about the fundamental dynamic behaviour of fast ions in burning plasmas can be obtained by experimental studies of the *fast ion tail* produced by ICRH and/or (N)NBI. The choice of dimensionless plasma parameters for a satellite tokamak to fulfil its mission of "Burning Plasma Physics and Control" has been discussed in Section 2. Electron temperature T_e cannot be too high, in order to avoid a too stringent requirement on the fast ion energy by $E_{fast}>E_{crit} \propto T_e$. On the other hand, T_e cannot be too low either in order to avoid excessive degradation of the plasma conditions in terms of equivalent D-T fusion performance. A reasonable choice for the performance of present day machines is an equivalent $f_\alpha = 1/6$, corresponding to an equivalent $Q = 1$. With a similarity scaling to ITER parameters in the $Q = 10$ reference scenario, this would imply $T_e \sim 10$ keV.

Once $T_e \sim 10$ keV is fixed, the $E_{fast}>E_{crit}$ condition corresponding to a 70% electron heating by the fast ions translates into $T_{fast} \sim 560$ keV for the effective perpendicular temperature of ^{3}He minority ions accelerated by ICRH in a D plasma. Meanwhile, $\rho^*_{fast} \sim 1.7\ \rho^*_{fast,ITER}$ is reached at $I_p \sim 6$ MA, with the $(\omega_{fast}/\omega_A)\sim(\omega_{fast}/\omega_A)_{ITER}$ criterion resulting into $(n/n_{GW}) \sim 0.61\ (n/n_{GW})_{ITER}$.

At $T_e \sim 10$ keV, 70% of electron heating would be obtained with a D beam injected in a D plasma at $E_{fast} \sim 900$ keV. Meanwhile, for a nearly tangential beam, $\rho^*_{||fast} \sim 3.1\ \rho^*_{fast,ITER}$ is reached at $I_p \sim 6$ MA, with the $(\omega_{fast}/\omega_A)\sim(\omega_{fast}/\omega_A)_{ITER}$ criterion resulting into $(n/n_{GW}) \sim 0.33\ (n/n_{GW})_{ITER}$.

The main qualitative and quantitative difference of using ICRH or (N)NBI is not due to the different anisotropy of energetic particle distribution functions in velocity space but rather to the different radial localization of the fast ion sources. In fact, (N)NBI tend to excite collective fluctuations in a more external region of the plasma cross section with respect to ICRH in typical conditions. The joint use of both ICRH and (N)NBI adds flexibility for burn control investigations in a satellite tokamak (Section 3), although it is not essential.

In the framework of burn control, only a few aspects can be addressed in a satellite tokamak, as discussed in the report by A. Fasoli (op.cit.). In brief, these can be summarized as follows:

1. The role of fast ions in sawtooth control/stabilization (related with NTM physics)
2. The consistency of current and pressure profiles assumed in advanced tokamak operations with fast ion (alpha particle) dynamics
3. The control of fast ion profiles and the use of these techniques to manipulate non-linear structures (slowly evolving non-linear equilibria) in order to “guide” the plasma in favorable operation regimes

In particular the last point, which has significant intersections with investigations of the complex mutual interplays between micro-scale plasma turbulence and meso-scale collective modes or macroscopic MHD modes, has potentially important impact on the re-circulating power needs for burn control.

Annex 6
Analysis of tokamaks and their ability to address the seven R&D Missions

Introduction

Around 30 experimental devices are in operation around the world today. These experiments have significantly advanced the development of the physics and technologies required for the use of fusion as a power source. In particular, the JET (1991 & 1997) and TFTR (1994) experiments achieved significant fusion power in Deuterium and Tritium plasmas, in conditions closest to those required by the Lawson Criteria, but still with a negative plasma power balance (below the break even condition) and these plasmas were sustained only for a few seconds. Sustained fusion power production with a positive overall power balance requires further research and development and is the aim of the ITER project. The research needs towards fusion power can be summarized in terms of the seven research and development missions[50] which cover among other issues; burning plasmas, reliable Tokamak operation, first wall materials & compatibility with the relevant plasma conditions, technology and physics of long pulse & steady state, predicting fusion performance; and materials and components for nuclear operation. In order to achieve these objectives, aiming at establishing the technical and scientific basis for the construction and operation of a fusion power plant (DEMO), an extensive programme is in place. In this section, the capability of the different Tokamak experiments to address the programme needs is discussed in the framework of the high priority R&D missions mentioned above.

Burning Plasmas

The plasma can be considered burning if the heating is dominated by the charged fusion products, which in case of a Deuterium-Tritium fuel is provided by the slowing down alpha particles that transfer their energy predominantly to electrons, thus providing different weighting of free energy source channels driving plasma turbulence. Burning plasmas differ qualitatively from plasmas dominated by auxiliary heating; due to the strong coupling between the plasma parameters and the plasma self-heating. This loss of direct external control of the plasma heating requires the development of adequate Burning Plasma scenarios with adequate real time control techniques, in order to maintain the plasma parameters within the constraints given by the required fusion yield and stability limits. The other important element regarding a large fraction of the heating being provided by the fusion alphas is the possibility of collective instabilities generated by the large fraction of fast ions present in the plasma; in particular, instabilities resulting from the resonant interaction between the alpha particles and specific plasma waves. These instabilities can compromise the confinement of the fusion alphas, reducing the plasma self heating; and more importantly alpha particle losses can damage the first wall of the device.

Alpha heating has been demonstrated experimentally by JET (1997) and TFTR (1994), but in these occasions it represented only a small fraction of the total heating <20%. At present, JET

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[50] Chapter 4: Seven R&D Missions for the European Fusion Programme, The European Fusion Research Programme: Positioning, Strategic outlook and need for infrastructure towards DEMO February 2008

is the only experiment currently in operation that is capable of operation with a Tritium-Deuterium fuel mixture, but alpha heating is limited to <20%. For these reasons, the development of burning plasma scenarios is one of the main objectives of ITER; therefore, ITER has been conceived with the objective of 50%-70% (Q=5-10) of the plasma heating to be provided by the fusion alphas in Deuterium-Tritium operation.

In order to prepare the operation of ITER, it is important to develop heating strategies in conditions where a significant fraction of the plasma heating is linked to the plasma parameters. This can be achieved by performing burning plasma simulation experiments in present devices equipped with versatile heating systems, making use of real time control of the auxiliary heating power, where one of the heating systems simulates the plasma self-heating. This allows the development of the right control strategies, together with adequate theoretical modelling. JET is the device best equipped to perform such experiments, due to its auxiliary heating systems and capability of developing plasma scenarios under the conditions closest to ITER.

Regarding alpha particle collective instabilities studies, present devices can achieve pressures of fast ions comparable with the expected fast ion pressures in ITER and in DEMO, when normalised to the magnetic pressure (β_{fast}). This is achieved, mainly using Ion Cyclotron Resonant Heating (ICRH) of minority ions, therefore, creating significant fast ion populations with high energies > 1 MeV. Therefore, a large number of experiments have been performed, providing an extensive knowledge base in this area. However, ITER and DEMO will operate in a parameter regime where the alpha particle Larmor radius is smaller when compared with the plasma minor radius ρ^*_{fast}. An illustration of the current Tokamaks operation domain and the approximate parameter space expected in ITER and DEMO in terms of two key quantities for the study of Burning Plasmas, ρ^*_{fast} and β_{fast}, is shown in Figure 1. This difference in plasma parameters is important in determining the toroidal spectrum of the modes that can become unstable, the non-linear behaviour of these instabilities and subsequent effect on the particle confinement. In addition, Ion Cyclotron Resonant Heating (ICRH) generates fast ions that are mainly trapped in the low field side of the Torus.

In order to prepare ITER operation and consolidate the physics knowledge, it is important to carry out experiments at low $\rho^*_{fast,}$ with significant fraction of fast ions. Experiments to be carried out in JT60-SA with high power Negative Neutral Beam Ion (NNBI) heating will provide information on the plasma behaviour in the presence of high energy circulating ions at moderate values of ρ^*_{fast}; therefore, mimicking the presence of circulating alpha particles in fusion plasmas. These experiments can be complemented by experiments carried out at JET using ICRH acceleration of He^4; or alpha particles in Deuterium-Tritium experiments; and lower values of $\rho^*_{fast.}$ The construction and operation of a high magnetic field and high current device, would provide further data at lower $\rho^*_{fast,}$ therefore, providing the possibility to perform experiments with parameters closest to ITER. Extrapolation to ITER in preparation of ITER operation and from ITER to DEMO will rely on the knowledge from the experiments performed on JET, JT60-SA and other devices; and on theoretical and modelling activities. Experiments in JET and MAST with high toroidal mode number Alfvén Wave Active Excitation Antennae will provide more accurate estimates on the damping of specific waves, improving the confidence of the predicted behaviour of these instabilities in ITER and DEMO. Further experiments will be carried out in devices capable of generating either fast ion populations or fast electrons such as in the case of FTU. Together with further developments in the fast ions/electrons diagnostics, these experiments will play an important role in improving the knowledge on the complex nonlinear fast particle behaviour in the presence of collective instabilities. Therefore, it is also important to maintain, install and upgrade fast ion and instability diagnostics systems in present experiments. The capabilities of European Tokamaks in addressing the research needs in the area of Burning plasmas is summarised in Table 1.

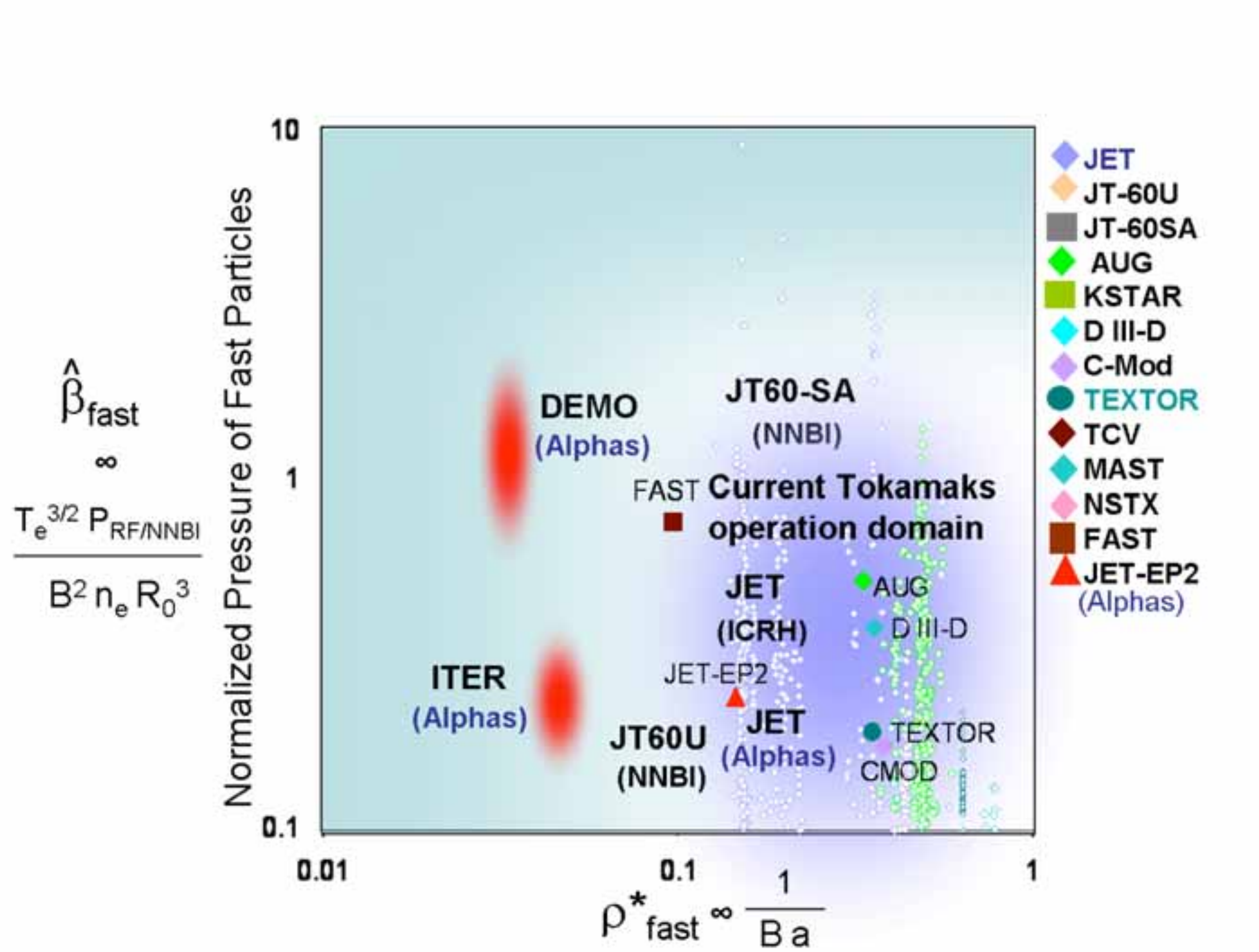

Figure 1 Illustration of the current Tokamaks operation domain and the approximate parameter space expected in ITER and DEMO in terms of two key quantities for the study of Burning Plasmas, the normalised fast ions (E> 1 MeV) Larmor radius ρ^*_{fast} (proportional to 1/(B a) where B is the toroidal magnetic field and a the device minor radius) and normalised pressure of the fast ions β_{fast} (proportional to $T_e^{3/2}$ P_{aux}/ B^2 n_e R_0^3, where T_e is the election temperature, n_e the electron density, P_{aux} auxiliary heating power that generates the fast ions, R_0 device major radius).

Table 1: The capabilities of European Tokamaks in addressing the research needs in the area of Burning plasmas

Device	Burning Plasmas
JET	**DT operation ; Ip>3MA for α confinement; ICRH fast ions; He Neutral Beam Injection; n and γ diagnostics; Active MHD Spectroscopy**
AUG	ICRH fast ions, NBI, dedicated diagnostics
Tore Supra	ICRH fast ions
FTU	Fast Electrons
TEXTOR	ICRH fast ions;
TCV	
MAST	Active MHD Spectroscopy, super-Alfvénic beam ions
Compass	

First wall materials & compatibility with ITER/DEMO relevant plasmas

The materials that can be used in the Tokamak first wall are relatively limited. Carbon (C) has the good properties of high melting point and low Z, but it suffers from significant plasma erosion and forms co-deposits with Tritium; therefore, leading to high fuel retention. ITER and DEMO foresee the use of metal based materials in the first wall, such as the combination of surfaces covered by: Beryllium (Be) a low Z material but with a low melting point; and Tungsten (W) with a high melting point but a high Z material. The use of Carbon surfaces will be reduced to the minimum, if used at all in ITER. Nevertheless, using present technology, the first wall can only withstand power fluxes of the order of 10 to 20 MW/m^2, retaining acceptable plasma purity. Therefore, plasma scenarios need to be developed compatible with the choice of first wall materials. These scenarios need to have a significant fraction of radiated power in order to minimise the power channelled to the surfaces directly exposed to the plasma and minimise the plasma edge temperature; therefore, minimising the erosion of the areas exposed to the plasma. In addition to acceptable steady state power losses, these scenarios have to be developed taking into account the energy lost during plasma transients. The mode of plasma operation that gives the best confinement properties, known as the H-mode, is characterised by frequent instabilities that lead to the loss of the edge plasma confinement and regular transient loss of plasma energy to the exposed surfaces.

Smaller tokamak devices have relatively small power losses, when compared with the surface area directly exposed to the plasma. As the size of the device increases the increase in power loss is not compensated by the smaller increase in the area of the exposed surfaces. Therefore, smaller devices can handle without difficulty the plasma power exhaust; while larger devices are much more limited in the power fluxes that can be handled by the first wall. This poses strong constraints in the operation of ITER and requires the development of scenarios capable of achieving the ITER scientific objective of producing up to 500 MW of fusion power, while maintaining acceptable levels of first wall erosion. Figure 2 illustrates the current Tokamaks operation domain in terms of the loss power divided by the major radius, which quantifies the steady state power density to the plasma facing components, versus the stored energy divided by the major radius which quantifies the energy losses during transient events; loss of a fraction/total of the plasma energy in a timescale shorter than the other relevant timescales.

The development of these scenarios relies on extensive studies to be carried out in JET following the ITER-like wall upgrade; installation of a (Be,W) first wall and in ASDEX-Upgrade presently equipped with a full metal (W) wall. The complementary nature of JET and ASDEX in terms of their sizes and capability to operate with ITER shape and versatile heating systems, will allow the development and extrapolation of these scenarios for an efficient exploitation of ITER. One of the key aspects of these studies is the development of strategies for the mitigation of transient heat losses, since, ITER will operate in a parameter regime where unmitigated transients heat losses will limit the life time of plasma facing components to unacceptable levels. Current upgrades of the JET and ASDEX-upgrade facilities are crucial for the development of such mitigation strategies, i.e., pellet injection, control coils systems.

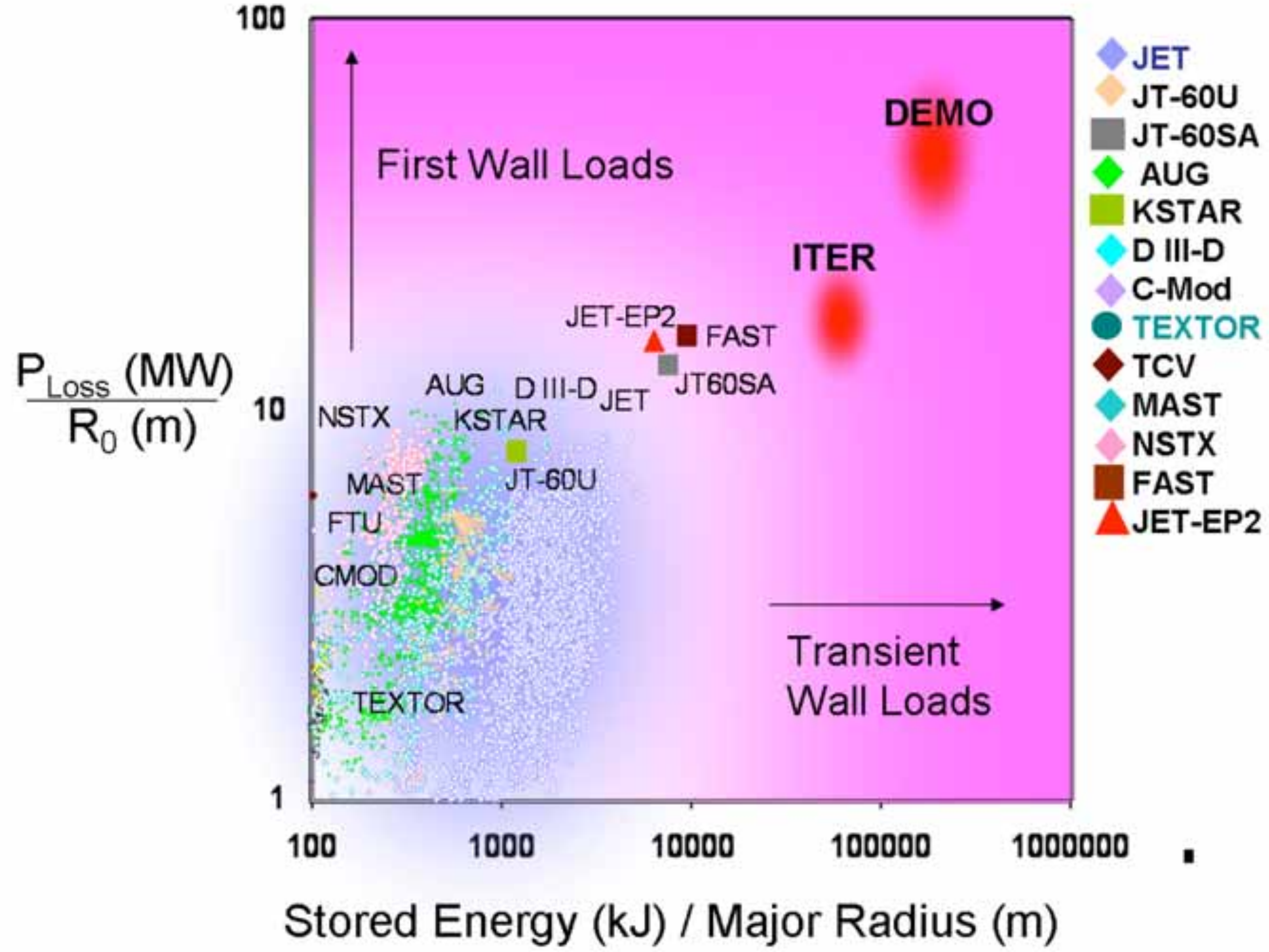

Figure 2 Illustration of the current Tokamaks operation domain in terms of the loss power divided by the major radius, which quantifies the steady state power density to the plasma facing components, versus the stored energy divided by the major radius which quantifies the energy losses during transient events, loss of a fraction/total of the plasma energy in a timescale shorter than the other relevant timescales.

Operation with a full metal wall and acceptable fuel (tritium) retention should be demonstrated in ASDEX-Upgrade; and in JET in Deuterium-Tritium (DT) experiments, ahead of the DT phase of the ITER scientific programme. In addition, these studies must be complemented by the development of Tritium and dust diagnostics and removal techniques in Tokamaks equipped with Remote handling tools. It is also important to point out that the preparation of DEMO operation will also require devices with plasma facing operating at reactor relevant first wall temperatures (400-500ºC); in order to obtained good thermodynamic efficiency. These set of conditions will not be available on ITER and the programme should also start addressing this point.

Alternative, more advanced plasma facing materials and plasma limiting concepts are also being explored, such as the development of liquid limiters in FTU and ISTTOK; and other concepts in the divertor test modules available in the MAST and TEXTOR tokamaks. These studies aim at expanding the first wall heat load capability, therefore, relaxing constraints on the plasma scenarios. The capabilities of European Tokamaks in addressing the research needs in the area of First Wall Materials and compatibility with Plasma scenarios is summarised in Table 2.

Table 2: The capabilities of European Tokamaks in addressing the research needs in the area of First Wall Materials and compatibility with Plasma scenarios.

Device	Remote Handling	First Wall Materials
JET	**Full remote handling capability**	**Unique Beryllium capability; ITER-like wall (2010) with ITER-relevant plasma configuration**
AUG	No	**Full Tungsten Wall with ITER-relevant plasma configuration**
Tore Supra	Remote handling under development	Carbon Limiter with Actively cooled components
FTU	(Ex. in limiter mounting)	Molybdenum and Liquid Lithium Wall;
MAST	No	Divertor science facility for materials testing including erosion, dust studies etc

Reliable Tokamak Operation

A fusion power plant will need to have very high levels of availability in order to be economical and to be able to contribute with an uninterrupted power source to the electricity grid. However, it is also required to operate at the highest pressures possible, making the best use of the available volume and magnetic field for optimal plasma performance. This leads inevitably to operation close to the stability boundaries, which could then cause unrecoverable degradation of the plasma performance or even the termination of the plasma. For example, elongated plasmas have better confinement properties, but require active control of the plasma position, since they are unstable to vertical displacements. Similarly, there are other instabilities that could require active control in order to allow operation at higher plasma pressures. Therefore, it is required to develop control and mitigation strategies for the most relevant instabilities in order to optimise operation at highest possible plasma pressure. Some Tokamak devices can operate at the ITER and DEMO relevant plasma pressures normalised to the magnetic pressure β_N, therefore, important experiments can be carried out in the area of research devoted to control and mitigation of instabilities. However, other important parameters, such as the dimensionless parameter ρ_*, which represent the ion Larmor radius normalised to the plasma minor radius, can not be matched in present experiments. Figure 3 illustrates the current Tokamaks operation domain in terms of ρ_* and β_N. Experiments approaching ITER relevant ρ_* can be carried out at JET following the ongoing auxiliary heating upgrade, because JET is the largest device currently in operation and in the future on JT60-SA. However, JET lacks the Electron Cyclotron Resonant Heating (ECRH) capability, important for controlling plasma instabilities using localised current drive, and also lacks

RWM control capability.. Therefore, these studies need to be complemented by experiments in ASDEX-Upgrade using the versatile ECRH capability and following the installation of in-vessel plasma control coils.

Another important aspect of reliable Tokamak operation is the ability to recover good vessel wall conditioning for plasma operation after planned and unplanned venting/intervention on the machine. The development of ITER relevant strategies of first wall conditioning is required in preparation of ITER operation, including, wall condition in permanent magnetic field and with a metal wall. Therefore, new techniques need to be developed, including experiments in devices with metal first wall such in JET (Be,W) and ASDEX-Upgrade (W); and alternative wall conditioning strategies such as using ICRH. The capabilities of European Tokamaks in addressing the research needs in the area of Reliable Tokamak Operation is summarised in Table 3.

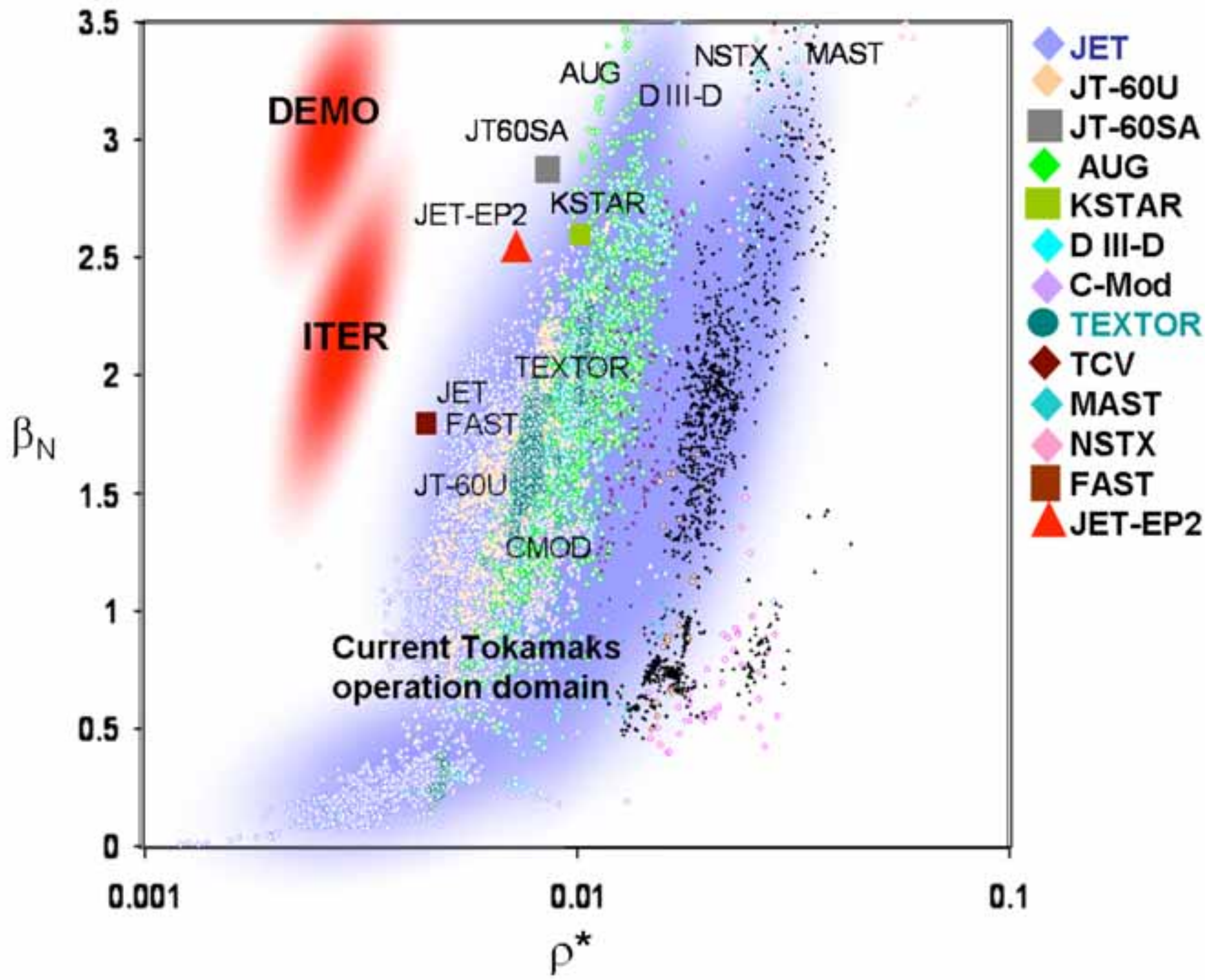

Figure 3 Illustration of the current Tokamaks operation domain in terms of the dimensionless parameter ρ_*, which represents the ion Larmor radius normalised to the plasma minor radius and plasma pressure normalised to the magnetic pressure β_N , important parameters in MHD plasma stability. Most Tokamak devices can operate at the ITER and DEMO relevant $\beta_{N.}$ However, ρ_* and β_N can not be simultaneously matched in present experiments. MAST and NSTX are shown but can comfortably exceed the 3.5 limit of the y-axis.

Table 3: The capabilities of European Tokamaks in addressing the research needs in the area of Reliable Tokamak Operation.

Device	Wall Conditioning	ELM Control	NTM Control	Disruption Mitigation
JET	**320^{o}C operation ICRH**	**Pellets, Coils,**	**ICCD,** LHCD	Large current disruptions, Fast Valve
AUG	**ICRH**	**Pellets, (coils from 2009 onwards)**	**ECRH**	Fast Valve
Tore Supra		No ELMs	ECRH	Fast Valve
FTU		No ELMs	ECRH	ECRH
TEXTOR		**Coils,**	ECRH	Fast Valve
TCV		**Vertical Kicks,**	**ECRH**	Fast Valve
MAST		**Coils,**	ECRH	Fast Valve
Compass		**Coils,**		

Technology and physics of Long Pulse & Steady State

One of the key aspects in fusion research is the development of plasmas in steady state conditions and operation of very long Tokamak pulses. For this purpose, a large fraction; or all the toroidal plasma current needs to be driven non-inductively, and maintained in steady state. Therefore, an efficient current drive capability is a crucial technology for a Tokamak based demonstration power plant (DEMO), and the plasma current redistribution time τ_R is one of the most important timescales. τ_R is significantly longer than the energy confinement time in most cases. Thus, it is important to study in detail plasmas with a long pulse duration $\tau_{pulse}>>\tau_R$, which requires operation of Tokamaks with super conducting magnets. Other longer timescales, such as timescales related to the particle balance, plasma wall interaction and temperature equilibration of the first wall components are also important. Important studies have been carried out in Tore Supra, including long pulse operation >100s. However, the Tore Supra device has a circular plasma cross section and it is not able to operate in the H-mode regime, which is the mode of operation foreseen in ITER. A number of devices equipped with super conducting magnets are planned in the near future, but with limited current capability (I_p <3 MA) such as KSTAR (South Korea), EAST (China), SST-1 (India). In a longer time scale, JT60-SA (I_p >3 MA) could make a significant contribution to the field, in the very long pulse exploitation phase of the device. The work in Tokamaks, will be complemented by experiments in Stellarators, which are inherently steady state devices, therefore, capable of long pulse operation. In particular, Wendelstein 7-X Stellarator (W7-X), will start operation with inertially cooled Carbon Fiber Composites (CFC) plasma facing components in 2014 and with the full steady-state capability after about 4 years.

Optimisation of the current drive capability requires further development of the plasma auxiliary current drive systems, such as the further development of Lower Hybrid Current

Drive (LHCD) systems and Negative Neutral Beam Injection systems. But more importantly, is the development of advanced plasma modes of operation (advanced scenarios) in order to minimise the need for current drive, by optimising the current profile for optimal confinement and stability properties. In particular, adequate coupling of the LHCD waves in the various plasma conditions requires extensive research in Tokamaks, since the wave propagation depends on the specific scenarios in question. Experiments at JET using the installed LHCD system are particularly important and these experiments could be complemented by the installation of LHCD capability in ASDEX-Upgrade. These are important tools in the development of advanced scenarios in preparation of the long pulse ITER operation phase. In addition, advanced plasma scenarios are characterised by large plasma pressures, in which stability limits are approached and in some cases exceeded. It is, therefore, important to complement these studies with an adequate stability control strategy using current profile control and active stabilisation by means of control coils. The development of these control strategies will benefit from experiments in Reversed Field Pinches, such as RFX which is equipped with sophisticated real time control systems for active stabilisation of plasma instabilities and Extrap-2P. The capabilities of European Tokamaks in addressing the research needs in the area of Technology and physics of Long Pulse & Steady State is summarised in Table 5.

Table 5: The capabilities of European Tokamaks in addressing the research needs in the area of Technology and physics of Long Pulse & Steady State.

Device	Long Pulse capability $t_{pulse}>t_{wall}>>t_R$	Off axis Current Drive for advanced scenario development	Resistive Wall Mode study/control capability
JET	No	**LHCD**	EFCC Coils
AUG	No	NBI, ECRH	Internal coils for RWMs (2012 onwards);
Tore Supra	Super Conducting magnets; actively cooled PFCs;	**LHCD**	No
FTU	No	**LHCD**	No
TEXTOR	No	ECRH	RMP Coils
TCV	No	ECRH	No
MAST	No	NBI	No
Compass	No	ECRH	No

Predicting fusion performance

In the limit of vanishing Debye length, plasma behaviour in the different regimes, in particular with regards to energy confinement, can be analysed in terms of the usual dimensionless parameters: β the ratio of plasma pressure to magnetic pressure, ν^* the product of the collision frequency and the thermal transit time and ρ^* the ratio of the Larmor radius to the torus radius. Illustration of current Tokamaks operation domain in terms of the dimensional less parameters ν^* and ρ^* is shown in Figure 4. These physics dimensionless parameters are related to the plasma parameters, temperature (T), density (n), magnetic field (B) and torus major radius (R_0) by $\beta \sim nTB^{-2}$, $\nu^* \sim nT^{-2}R_0$ and $\rho^* \sim T^{1/2}B^{-1}R_0^{-1}$. Low ρ^* values can be obtained at low temperatures, therefore increasing ν^* and vice-versa. The simultaneous operation at low $\rho^*$and low ν^* requires the increase in the size and the magnetic field of the device. Issues that are explicitly excluded from this similarity consideration include nuclear reactions, atomic physics aspects, e.g. impurity behaviour, edge physics and some auxiliary heating aspects such as the NBI penetration. Plasma parameters approaching the conditions required for a Fusion Reactor can only be satisfied simultaneously in ITER. Therefore, it is one of the major objectives of the ITER research programme the study of plasmas under these conditions. JET is the closest device to ITER, due to its size and large plasma current capability, therefore, the results from JET are crucial in preparation for ITER operation. The proposed devices and new devices under construction would not be able to close the gap between the JET operation space and ITER. Therefore, extrapolation to ITER and from ITER to DEMO relies on basic theoretical understanding and modelling activities. It is important to upgrade computation facilities inline to the need for large scale simulations and model integration. In addition, it is also very important to invest significant effort in the development of plasma diagnostics for detailed physics studies, in order to improve basic understanding and increase the confidence levels of the modelling activities. These studies should be complemented by detailed experiments using different auxiliary heating systems with different levels of electron and ion heating and momentum input, in particular using the versatility of the TCV heating systems and proposed upgrades. The capabilities of European Tokamaks in addressing the research needs in the area of Predicting Fusion Performance is summarised in Table 6.

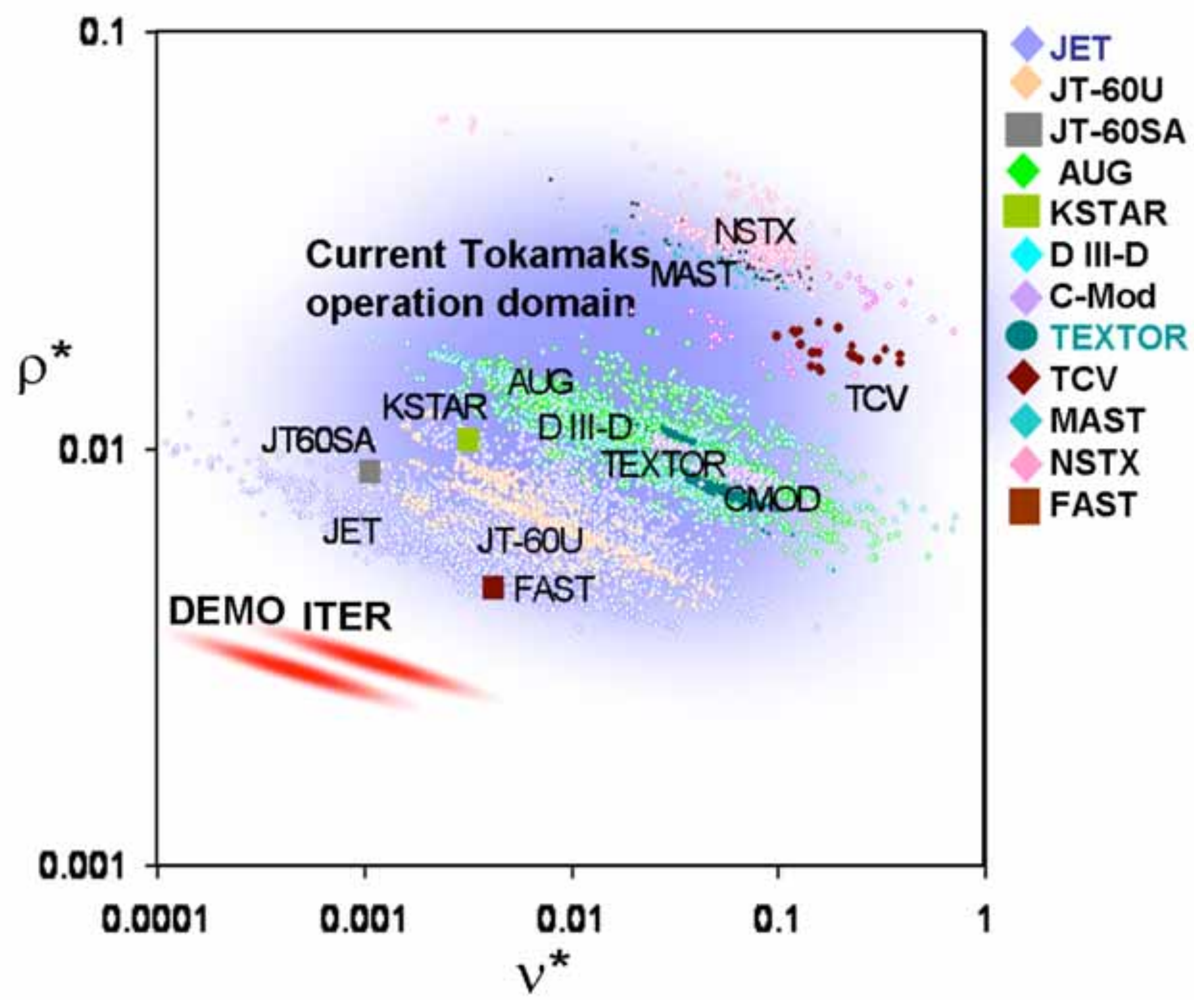

Figure 4 Illustration of current Tokamaks operation domain in terms of the dimensionless parameters ν^*, the product of the collision frequency and the thermal transit time, and ρ^*, the ratio of the Larmor radius to the torus radius. Plasma parameters approaching the conditions required for a Fusion Reactor can only be satisfied simultaneously in ITER.

Table 6: The capabilities of European Tokamaks in addressing the research needs in the area of Predicting Fusion Performance.

Device	Shape	Heating with low momentum input and Central electron heating	Divertor
JET	**ITER-like**	12 out of 40 MW total	**Yes**
AUG	**ITER-like**	14 out of 30 MW total	**Yes**
Tore Supra	Circular	**4 MW**	Limiter
FTU	Circular	**4 MW (ECRH + LH)**	No
TEXTOR	Circular	6/9MW	Ergodic

TCV	**ITER-like**	**4.5 MW**	**Yes**
MAST	ST with ITER-like plasma shapes	0.5 MW out of 5MW total	**Yes**
Compass	**ITER-like**	**2 MW**	**Yes**

Materials and Components for Nuclear Operation.

The operation of Tokamaks in a full nuclear environment will require significant further research and development, in particular, regarding the components and materials to be used in the areas subject to high neutron fluxes. So far, operation with Tritium has been limited to experiments in JET in 1991, 1997 and 2003; and experiments in TFTR in 1994. At present JET is the only device capable of Tritium operation and with a full remote handling capability. ITER will be the first Tokamak to operate under a nuclear license, but still the overall activation of the ITER components is small when compared to the requirements for DEMO. The operation of a Tokamak in a nuclear environment requires established procedures related to the operation of a nuclear device, but also poses specific problems related to the levels of activation of the machine. Diagnostics, in particular, have to be robust to be able to function reliably under the foreseen nuclear fluxes and all components have to be engineered to be able to cope with the neutron fluxes during their life time. Complementary to a fusion irradiation materials test facility, a component test facility will be required in the long term. A Tokamak based component test facility is one possible solution, which is able to provide relevant neutron wall loads with the exact fusion neutron spectrum. The proposed conceptual designs need detailed feasibility studies from both the technical and scientific point of view. In some cases, significant developments are required on the technology and physics side, before a detailed design of a tokamak component test facility is possible. There are a number of challenges associated with the production of the required neutron fluxes (1 MW/m^2) for sufficient long periods with high availability. In order to maximize the neutron wall load, a relatively compact device is required, and a design based on a compact spherical torus has the advantage of providing the required neutron fluxes at moderate tritium consumption.

Annex 7
Cost estimate for DEMO Conceptual Studies

The DEMO Conceptual studies and R&D are outlined in section 1. The target is to be able to start DEMO Engineering Design Activities after 8 years of Conceptual Studies and R&D.
The complementary activities required to conduct an assessment of the Component Test Facility (CTF) concept and a conceptual design study of a stellarator power plant are outlined in section 2.

1. DEMO Studies and R&D

Year 1: DEMO Concept and DEMO R&D Definition Phase
Duration
1 year
Resources
10 ppy for the central team, 10 ppy for expert support
Missions
Narrow down FPP options and, consequently, narrow down DEMO options[51].
Define R&D required to launch DEMO Conceptual Design Activities (CDA).

Year 2-5: DEMO R&D
Duration
4 years (some R&D could/will be completed during the CDA)
Resources
50-60 ppy for the central team, 80-100 M€ for R&D and external expert support
Missions
i) Maintenance.
ii) Layout and architecture of internal components. We recommend restarting the R&D on water cooled internal components pending the final selection for DEMO during the CDA.
iii) Magnets (HTS, LTS with high Jc).
iv) Internal coils.
v) Development and validation of design codes.

Comment: it is assumed that activities carried out under ITER or the BA, in particular physics, materials development (including IFMIF) and TBMs, are funded separately and appropriately. Strong coordination will be required to assess whether, beyond the scope of the ITER/BA work, complementary R&D is required (e.g. on tritium and on heating and current drive systems).

Year 6-8: DEMO Conceptual Design Activity (CDA)
Duration
3 years
Resources
200 ppy for the central team, 100 ppy for external expert support and 100 M€ for R&D.
Missions

[51] for example: should DEMO operate in quasi-continuous mode or in full steady-state.

i) Selection of DEMO parameters (physics and technology).
ii) Define and launch R&D required during DEMO EDA (in particular, “large projects” should start as early as possible so as to allow an optimum integration of their outcome during the EDA).
iii) DEMO conceptual design.

2. Complementary Activities (assessment of the Component Test Facility (CTF) concept and conceptual design study of a stellarator power plant)

Year 1: Component Test Facility (CTF) and Stellarator Power Plant (SPP) Feasibility Assessment

Duration
1 year
Resources
5 ppy for the central team, 5 ppy for expert support
Mission 1: Feasibility of CTF
Assessment of potential CTF showstoppers, in particular the design of the divertor and the replacement procedure for the central column.
Mission 2: SPP Preliminary Assessment
Preliminary feasibility study of a stellarator power plant focusing on engineering issues, in particular: maintenance and segmentation of internal components, neutron shielding of vacuum vessel and magnets.

Years 2-4 CTF and SPP Activities

Duration
3 years
Resources
10-15 ppy for the central team, 15-20 ppy for external expert support,
Mission 1, CTF
Details of mission will depend on the outcome of the feasibility study carried out during year 1. Issues to be assessed include the design of the central column and its experimental feasibility, the feasibility of H&CD systems, the feasibility of remote maintenance and an estimate of the CTF reliability. Some R&D is likely to be required to validate specific design options.
Mission 2, SPP
Details of mission will depend on the outcome of the feasibility study carried out during year 1. A more comprehensive study, including the definition of a physics base, the selection of a preferred configuration and an assessment of all engineering issues could be carried out. No specific R&D is foreseen.

Years 5-8: CTF and SPP Activities

The continuation will depend on the outcome of the first phases.
Duration
4 years
Resources
10 ppy for the central team, 15 ppy for external expert support (assuming one of the two, CTF or Stellarator studies, continues)
Missions
TBD, depending on the outcome of previous studies on the CTF and on the SPP and taking into account the results of W7-X and NSCX.

Annex 8
Fiches: Magnetic Confinement Devices (existing & possible upgrades, projects) and Technology Facilities

INDEX

<table>
<tr><td>FACILITITY</td><td colspan="3">(NAME, ASSOCIATION(S)) JET, EFDA</td></tr>
<tr><td rowspan="4">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: cost in ~~2007~~ euros of the year: 163.9 (1977-1981), 55.98 (1982), 75.48 (1983), 44.56 (1984), 34.47 (1985), 31.36 (1986), 28.31 (1987), 22.15 (1988), 28.29 (1989), 23.69 (1990), 15.06 (1991), 17.79 (1992), 14.18 (1993), 9.87 (1994), 6.67 (1995), 3.21 (1996), 2.53 (1997), 0.55 (1998), 0.0 (1999), 0.007 (2000), 2.50 (2001), 4.68 (2002), 6.90 (2003), 19.36 (2004), 13.01 (2005), 14.59 (2006), 18.09 (2007)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: cost in ~~2007~~ euros of the year:
42.3 (2008), 12.01 (2009), 5.94 (2010) 0.084 (2011)</td></tr>
<tr><td colspan="3">OPERATION: yearly cost of operation in ~~2007~~ euros of the year:
<table>
<tr><th>Year</th><th>2000</th><th>2001</th><th>2002</th><th>2003</th><th>2004</th><th>2005</th><th>2006</th><th>2007**</th><th>Average</th></tr>
<tr><td>Days</td><td>92</td><td>59</td><td>70</td><td>121</td><td>43</td><td>0*</td><td>69</td><td>53</td><td>63/65***</td></tr>
<tr><td>Cost</td><td>51.6</td><td>56.2</td><td>47.1</td><td>54.4</td><td>53.9</td><td>53.9</td><td>52.7</td><td>61</td><td>53.8/52.8***</td></tr>
<tr><td>ppy</td><td>527</td><td>552</td><td>475</td><td>516</td><td>601</td><td>577</td><td>537</td><td>620</td><td>551/541***</td></tr>
</table>
* Shutdown **Provisional ***Excluding 2007</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td rowspan="4">USE OF FACILITY</td><td>Number of facility users:
About 305 users from EFDA Associates; about 100 from international collaborations outside EU.</td><td colspan="2">Yearly integrated equivalent full time facility users: On-site effort of 53ppy and off-site effort of 35ppy contributed by EFDA Associates (2006). Average annual on-site effort contributed by collaborators outside the EU is about 4ppy.</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>(Main) Conferences</td></tr>
<tr><td>2004, 2005, 2006, 2007</td><td>71, 57, 36, 72</td><td>253, 78, 136, 112</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: JET is used collectively by 27 EFDA Associates from 25 European countries, comprising 24 EU member states and Switzerland.
Collaborations outside EU: JET has collaborations with US, Russian Federation, Japan, China and South Korea. During 2003-2007 average annual number of professionals to JET (average annual on-site effort in ppd) from US was 25 (246ppd) and from Russian Federation was 9 (221ppd). 70 professionals from US and 22 from Russian Federation visited JET during these years. During 2004-2007, average number of professionals to JET each year from Japan was 3 (14ppd) and from South Korea was 1 (22ppd). China sent 1 professional to JET in 2004 and 1 in 2005, contributing 95ppd effort in each year for experiments.
Number of experimental contributions to ITPA: During JET Campaigns C15-C19 (25 May 2006 - 4 April 2007), about 30% of JET's experimental time (72 out of 241 sessions) was dedicated to ITPA-coordinated experiments.
Sharing facility with other fields of research: JET (together with CERN, EMBL, ESA, ESO, ESRF, ILL) is a member of EIROforum and in that context, JET grid cluster is shared with other scientific disciplines.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">Plasma current: 6MA (ultimately, with present volume); 5MA (q_{95}=3 requires 4T)
Toroidal magnetic field: Up to 3.45T used routinely; scientific justification required for >3.45T; 3.8T is present internal limit (JET Operating Instruction)
Geometry: R: 2.96m; a/b: 0.96m/1.7m; Elongation: 1.8; Plasma volume: $90m^3$
Triangularity (at full bore): Max upper 0.56 (with lower 0.44); Max lower 0.56
Wall materials: Carbon, beryllium and beryllium evaporation
Ability to operate with tritium
Almost full in-vessel remote handling capability
Cooled systems. NB full and fractional energy ion dumps (achieve steady state within 10s pulse). PF and TF magnets and divertor support structure (all with inter-pulse cooling). Pumped Divertor (PD), NB and LH cryopumps; PD and NB cryopumps can use Argon-frosting to pump, respectively, He ash and He in NB box
NB heating: 24/22MW for 7s. ICRF heating: 32MW (at the generator); 13MW ELM-resilient power to be tested: 6MW (A2 antennas with external conjugate-T system and 3dB couplers) and 7MW (ITER-like ICRH antenna with internal conjugate-T). LH heating: 12MW (at the generator); 4MW/7s L-mode (lower powers coupled up to 20s) and 3.5MW/4s H-mode achieved
Fuelling capabilities Pellet: 10-50Hz (presently limited to 40Hz) 1-$2mm^{-3}$ pellets (~0.6-$1.2x10^{20}$ H or D) 50-$200ms^{-1}$; 0-15Hz 35-$70mm^{-3}$ pellets (2.2-$4.4x10^{21}$ H or D) 100-$500ms^{-1}$
Gas injection: Ar, CD_4, D, H, ^3He, ^4He, N, Ne, T; NB fuelling: D, H, T, ^4He, ^3He
Magnetic field perturbation. ELM control: static n=1 configuration at 2.3kA for 1-2 seconds or n=2 configuration at 1.5kA for 3 seconds. RWM studies: n=1 configuration at 10-30Hz ±500A peak. Error-field correction: static n=1 fields
Massive argon or neon injection to study mitigation of induced disruptions; 0.65 litres reservoir and response tens of ms with Ar injection.
Comprehensive set (about 100) of edge and core diagnostics, 10 Gbytes of data per pulse and 40Tbyte of stored data</td></tr>
</table>

FUTURE TECHNICAL CAPABILITIES	Ability to validate **ITER-like first wall materials** choices (installation of an ITER-like Be/W first wall in 2009/10). NB power upgrade in 2009/10 to **34MW** for **20s** or 17MW for 40s. Enhancement of radial field amplifier for **vertical plasma control** and plasma vertical position control system. About **20 new diagnostics** or upgrade of existing diagnostics.

PROGRAMME: ACHIEVEMENTS

1. Burning plasmas: Fusion Performance

- World's first controlled release of DT fusion power (1.7MW peak producing 2MJ of fusion energy) (1991)
- Record DD power in discharges with optimised magnetic shear (with Internal Transport Barriers, ITBs) (1996)
- World record fusion power of 16.1MW achieved, with 22MJ energy release, in DT ($Q_{DT}=P_{fus}/P_{in}=0.66$) (1997)
- Steady-state ELMy H-mode with 50:50 DT demonstrated, with 4MW fusion power maintained for 4s (1997)
- Up to 8.2MW fusion power in DT discharges with optimised shear (potential for steady-state) and ITBs (1997)

1. Burning plasmas: Fusion Physics

- First demonstration of plasma heating by alpha particles in DT (~3MW) (1997)
- Helium exhaust demonstrated with τ_{He}/τ_E~4 (1998)
- Alpha particle behaviour simulated using central source of ^4He provided by NB injection (1990), showing slower concentration decay in H-mode than in L-mode (with ^4He accumulation in core in some cases) and acceleration to 0.5MeV by application of ICRF to NB injected ^4He ions (2000)
- World's first observation of monster sawteeth (1986), stabilised by fast ions and controlled using ICRH (2002)
- Redistribution/losses of fast particles due to core localised modes (tornados, sawteeth, fishbones) measured; internal saddle coils and dedicated antennae used to probe stability (growth, damping) of Alfvenic instabilities

2. Reliable operation

- **Plasma configuration control** developed to allow first plasma of 19kA (1983), 7MA limiter plasmas (1988), 6MA X-point plasmas (1995; ELMy H-mode demonstrated at 6MA with JET record stored energy of 13.5MJ) and record steady-state (>5s) ELMy H-mode plasmas at 4MA (2003/4); strongly shaped ELMy H-mode plasmas (ITER triangularity), with more precise shape control (XSC); ITER similarity experiments of plasma current ramp-up; use of low electric field for breakdown; quality assurance in pulse management
- **Increased reliability of ICRH and LHCD** auxiliary heating systems by coupling improvements using CD_4 gas injection (plasma/antenna separation 0.1m, 3MW coupled LH power) and D_2 gas injection (plasma/antenna separation 0.14m, ~2.8MW coupled LH power); 8MW coupled ICRH power in ELMy H-modes with 0.14m separatrix/antenna distance; 0.7MW (0.8MW at generator) ICRH power coupled using external conjugate-T system and ~1MW with 3dB couplers
- **Control/destabilisation of sawteeth** for control of alpha particle generated monster sawteeth and for reducing the seed for NTMs demonstrated using ICRH. Threshold in **β for onset of NTMs** found to be low at low q_{95} and higher in absence of large sawteeth and with increased rotation; JET provided data, in particular at low ρ^* and ν^*, to multi-machine NTM scaling studies with ASDEX Upgrade and DIII-D. **Active ELM control** demonstrated with externally applied n=1 and n=2 perturbations using error field correction coils (2007)
- **Thermal energy of disrupting plasmas** just before thermal quench found to be typically 25% of maximum stored energy during normal operation before disruption developed. Improved **halo current measurements** in ITER-like shaped plasmas consistent with earlier data in multi-machine database on toroidal peaking factor vs. halo current fraction, as design input to ITER. **Disruptions** can produce vertical displacements of plasma and large forces (200-450 tonnes, so far) on vacuum vessel (1984), radiation-induced collapse can occur at high density; edge density is critical parameter. **Disruption avoidance techniques for safe approach** to operational limits refined by improved plasma scenario design and better technical capabilities
- **Simultaneous control of plasma pressure and current density profiles in real-time** demonstrated by feedback on temperature gradient scale-length and iota (1/q) using LHCD, ICRH and NBI as actuators
- **Radiative feedback control** used to sustain Type III ELMy/Hybrid discharge close to L-H transition boundary

3. Optimisation of plasma-wall compatibility

- Five different divertor geometries (precision tiles and alignment; edge protection) tested to **improve power handling** (with ELMy H-modes, power deposition profile scaling determined, found to be very narrow on divertor target, dominated by ion energy losses (IR camera, thermocouple, Langmuir probe measurements); record energy accepted by divertor target (180MJ of 326MJ input power); power loading about equal in outer and inner divertor when ion ∇B drift away from X-point, and larger in outer divertor when ion ∇B drift towards X-point (power threshold for L-H transition is also lower then); ELM energy deficit on divertor explained by filament impact on main chamber, quantified by ELM filament modelling; improved density and impurity control confirmed with more closed 'gas-box' divertor (Mark IIGB)); beryllium tested as divertor material (compatible with high-recycling ELMy H-modes and incompatible with low-density operation)
- Carbon **erosion**, predominantly from main chamber walls by physical and chemical sputtering, is **screened** partially by scrape-off-layer (twice as effective at top of vessel than at mid-plane, and ten times as effective in divertor), **flows** in SOL (large flows measured with reciprocating probe) mainly to inner divertor (to outer divertor when toroidal field direction reversed; enhanced by ELMs; measured with quartz microbalances; confirmed with injection of tracer material and post-mortem tile analysis)
- **Type II ELMs** sustained for 9s in quasi-double-null configuration, but only at high collisionality and high triangularity. **ELMs moderated** at 80% of **impurity radiation** (Type III), leading to inter-ELM detachment and pedestal pressure and energy confinement reduction; best results extrapolate to Q=10 on ITER (17MA)

4. Technology and physics of Long Pulse & Advanced Scenarios

- **Non-inductive current drive** studied with **LHCD** (3MA in X-point plasmas); with **NB/LH** current drive and bootstrap currents (full current drive at 1.8MA and 80% at 2MA); 1.7MA L-mode discharge sustained for 1 minute with combined ICRH and LH current drive; 70% bootstrap current in 1MA ELM-free H-mode using ICRH; LHCD used to pre-form current profile in ITB experiments
- **AC plasma current operation** demonstrated as alternative route to quasi-steady tokamak operation
- **Long ($50\tau_E$) steady-state H-modes** demonstrated with controlled density
- **Internal Transport Barriers (ITBs)** with improved central confinement produced by pellet injection (PEP mode: pellet enhanced performance) and by control of current profiles; ITBs sustained with Type I ELMs and with mild ELMs using argon or neon seeding (up to 32MW input power); electron ITB sustained for 11s with ion ITB for 8s ($27\tau_E$); wide ITBs sustained for >2s at high current (up to 3MA) and double ITBs with power input exceeding 22MW; turbulence suppression requires inclusion of magnetic shear in addition to ExB shear
- **High performance hybrid mode (**β_N~2.8; $H_{89}\beta_N/q_{95}^2$=0.4) sustained for one resistive diffusion time. **20s hybrid mode** reached β_N~2.5 with record 186MJ NB power input, with strike point sweeping to reduce divertor heat loading, and sustained for 3 resistive diffusion times; extended towards ITER normalised conditions; stable integrated Hybrid Scenario sustained at Greenwald density limit, β_N ~3.6 reached
- **Resistive Wall Mode** experiments provide favourable scaling towards ITER of critical rotation required to avoid instability; EFCCs used to probe parametric dependencies of no-wall β-limit using Resonant Field Amplification in Advanced scenarios

5. Predicting fusion performance

- Demonstrated influence of toroidal magnetic field (including TF Ripple), plasma density (low density turning point), plasma mass species (H, D, T), divertor geometry (strike point position, proximity to septum) on **L-H transition threshold** and establishment of edge transport barrier
- Experimental measurement of **ITBs** as narrow regions with non-stiff profiles established at very low power in reversed shear plasmas; triggering and sustainment mechanisms involve rational magnetic surfaces and rotation
- First identification of **threshold for onset of ITG transport** in ion heat channel
- JET scaling studies of **energy confinement** in ELMy H-modes, especially for multi-machine scalings in dimensionless parameters show $A^{-0.5}$ mass-dependence, no dependence on neutral gas pressure, no degradation at high current (low q_{95}), degradation with increasing density (with or without impurity seeding), improvement with triangularity (shape dependence), confirmation of ITER scenario requirements with H~1 and n/n_{GW}~1 and β_N~1.9 with an ITER-like plasma shape, weaker dependence than 1/β of ITER98(y2) in similarity experiments between JET and DIII-D; collisionality rather than Greenwald fraction found to be relevant for confinement scaling; two-term confinement scaling separates core and edge transport contributions
- JET has unique capability to vary **TF Ripple amplitude** showing that increased TF Ripple degrades fast ion confinement but indicating fast particle confinement not expected to be affected significantly in ITER with TF ripple amplitude <0.5%; degrades pedestal pressure (above 1% in JET/JT-60U identity experiments) and energy confinement (above 0.5%); produces counter-current torque which lowers co-current toroidal rotation
- **Peaked density profiles** obtained on JET with pellets (densities above Greenwald density, without strong confinement degradation; record peak density of $4 \times 10^{20} m^{-3}$) and with gas puffing over wide range of currents (0.95-2.5MA), correlating with internal inductance in L-mode and with decreasing collisionality in H-mode, extrapolating to $n_e(0)/\langle n_e \rangle$=1.4 on ITER; density control improved with beryllium as plasma-facing material
- **Accumulation of high-Z impurities** observed in long-pulse discharges with peaked density profiles and controlled by central ICRH; Z-dependence of transport determined experimentally and compared with neoclassical and turbulence theory
- JET experimental measurements of **momentum confinement** show that poloidal rotation velocity is anomalous, Prandtl number for core plasma momentum is unexpectedly low with the existence of a momentum inward pinch being demonstrated by measurements obtained by modulating applied torque by NB injection

6. Materials and components for nuclear operation

- JET with carbon first wall materials pioneered studies of **fuel retention** in carbon-based flakes deposited in shadowed region below inner divertor (increasing with recycling flux and ELM energy, and as high as 40% retention during DT operations) and its removal (to 11% after intense cleaning campaign using He, H, D plasmas; venting; GDC; also laser and flash-lamp ablation tested for tritium removal)

PROGRAMME:ADDRESSING THE PROGRAMME NEEDS: Five to ten year perspective:

1. Burning plasmas

JET cannot access deeply burning plasma regimes (Q>5), but with weakly self-heated plasmas (self-organised with fusion alpha particles, bootstrap currents, mhd and transport) is well-suited to study fast-ion and alpha particle physics, fast-ion driven instabilities and redistribution, by virtue of its size, current, heating, diagnostic and analysis capabilities: JET is currently the only DT-capable tokamak worldwide and is used to study fusion-born alpha particles with isotropic energy distribution; JET is the only tokamak capable of confining energetic particles in the MeV range, and can therefore investigate fast particle physics; JET can generate MeV-range alpha particles by ICRF-acceleration of NB ions; JET can generate fast ions with energies of a few hundred keV using ICRH.

JET can achieve, or even exceed fast alpha particle (^4He) normalised pressure in ITER (β_{fast}~3% peak, 0.3% volume-averaged compared with β_{fast}~1.2% peak, 0.3% volume-averaged in ITER), using ICRF-acceleration of NB ions. The fast particle pressure gradient (which drives fast particle instabilities) in JET is also comparable to that in ITER. The fast particle slowing down time on JET (≈1s) is short compared with the ICRF heating pulse, so that a steady-state fast-particle distribution is reached, driving ITER-relevant instabilities. The orbit width of fast particles normalised to the minor radius is similar to that of ITER for energetic ions in the 500keV range and high plasma current.

JET has an array of burning plasma diagnostics including a unique gamma-ray system for simultaneous D and ^4He fast-ion population profiles with a time resolution of ~100ms; a number of neutron flux detectors and spectrometers; Faraday cups to measure energy and poloidal distribution of lost fast particles (principally alpha particles), a scintillator probe measuring with high time resolution; active TAE antennas to study growth/damping of Alfvénic instabilities (modes up to n=16, reaching estimated ITER-relevant range of $15<n<50$); and neutral particle analysers measuring in 40keV-100keV and 500keV-2MeV ranges (being upgraded to measure also in range 100keV-500keV).

Specific experiments on JET could include He ash accumulation, transport and pumping; control of simulated burn (heating, fuelling, exhaust, actuators); fast-ion and alpha particle effects on sawteeth, fishbones and other MHD; effects of MHD on fast-ion and alpha particle distributions/losses (safe operation on ITER requires <5% losses); after-glow experiments to study Alfvén eigenmode growth/damping; isotope scaling of confinement and TAE physics; ion heating by energetic alpha particles with isotropic birth distribution; DT qualification of ITER-relevant scenarios, development and qualification of burning plasma diagnostics and models. A high performance DT experiment with ITER-like Wall could demonstrate for first time alpha particle heating in ITER-relevant core conditions ($T_i=T_e$, low rotation) with Q≈0.5 (thermal) in stationary conditions at high plasma current and with P_{fusion}=20-25MW for ITERH98 scaling.

2. Reliable Tokamak Operation

JET approaches ITER in complexity and reliable operation has always been a priority, with experience gained being of direct relevance to ITER. **Physics studies**: A significant part of JET experimental programme addresses operational limits and avoidance in view of potential damage by disruptions or large ELMs, extrapolation to ITER and avoidance or mitigation using passive and/or active techniques of relevance to ITER. Active techniques on JET include disruption mitigation with massive gas injection; ELM control with magnetic field perturbation, high frequency pellet injection, vertical kicks; impurity seeding to reduce inter-ELM power loading and ELM suppression; improved vertical control during ELMs; avoidance of NTMs and understanding of RWMs. The similarity between JET and ITER (in particular, access to low ρ^* regimes) is key to increase the relevance of the results. High priority is given to a fully developed operational strategy with pulse management for reliable tokamak operation including wall conditioning, break-down, current ramp-up and safe termination for ITER, and the development of a discharge simulator. **Control**: JET is well-placed to build on its comprehensive set of control tools and associated architecture. **Quality assurance for upgrades**: Past and future upgrades to JET result in greater complexity. Quality assurance procedures are essential to maximise reliability, are continuously refined and are directly applicable to ITER. **Maintenance reviews**: As a complex tokamak with 25 years of operation, JET is well-placed to identify potential failure modes for ITER, and to develop procedures to minimise the chance of occurrence of such failures. **Technology tests**: JET is well-suited to test technology for ITER, for example to optimise ITER auxiliaries such as heating systems and diagnostics. An ITER-like ICRH antenna will be tested during 2008 for its ability to couple up to 7MW reliably in the presence of ELMs. Also, the A2 ICRH antennas have been upgraded (3dB couplers/ external conjugate-T systems) to deliver more than 6MW ELM-resilient power to plasma. LH and ICRH coupling is also being optimised with ITER-relevant plasma-wall separation, and ITER-relevant power densities with LH.

3. First-wall materials and compatibility with ITER/DEMO relevant plasmas

During 2009/10 a new first wall, made with ITER first wall materials (beryllium in main chamber and tungsten in divertor), will allow the only possible test of an ITER-like first wall in a tokamak. Furthermore, this test on JET will be under the closest possible plasma conditions to ITER. Key elements of the forward programme include: Characterise **radiating ELMy H-modes**, including detached plasmas with intrinsic and extrinsic impurity radiation. Demonstrate **ITER-relevant integrated scenarios** with full plasma compatibility with carbon and ITER-like wall. Characterise **large ELMs** and their effects on the wall, including **melt-layer studies** in main chamber and divertor; this requires a JET-class device. Disruption studies including **interaction of halo currents with melt layers** formed by ELMs or during disruptions. **Erosion and particle flow characterisation** with carbon and with ITER-like wall materials. **Minimise tritium inventory**, including deposition minimisation and supported by modelling, together with the **testing of tritium removal techniques** (e.g. laser ablation or oxidation). Studies of **alloying effects of the tungsten divertor**. Optimise and qualify scenarios and control strategies as the basis for extrapolation to ITER. Validate spectroscopic diagnostics for ITER-like wall materials. **Minimise impurity sources** due to heating systems (eg. NB shine-through and RF screens). Develop **dust removal technologies** (under consideration)

Following installation and commissioning of the ITER-like wall, the test programme is foreseen to extend from early 2011 to mid-2013, thereby providing timely input for decisions on the ITER detailed design in 2013. A Full Tritium Experiment could aim at quantification of the benefit of the ITER-like wall for the minimisation of tritium inventory.

4. Technology and physics of long pulse and steady state

JET heating and current drive systems (ICRF, NB and LH) provide a test of heating and current drive technologies, improved RF coupling in ITER-relevant scenarios, significant non-inductive current drive for steady-state operation and effective current profile tailoring to produce advanced operating regimes with optimised MHD stability (RWMs and β limits), energy confinement and bootstrap current.

The duration of high power experiments can be in the range of tens of seconds, enabling long-pulse/steady-state behaviour (albeit, not at full heating power) to be studied with respect to important physics, including energy and particle confinement and current diffusion. This capability will be extended following installation of the ITER-like wall and the NB enhancement. Furthermore, the heating pulse duration will be comparable to the current diffusion time, thereby providing access to steady-state physics.

Since the late 1980s, these features have made JET suitable for advanced scenario development, aimed at optimised steady state operation. With its upgraded power during the testing of the ITER-like wall, JET will be able to access both the hybrid and steady-state regimes at low ρ^* with plasma currents above 2.5MA, explore high confinement (H_{98}~1.5), high β_N (=3.4), high density ($n>6x10^{19}m^{-3}$) and high bootstrap current fraction (75%). These studies will provide a major scenario development opportunity at divertor-compatible densities (high Greenwald density and with edge-core compatibility) and sophisticated control of magnetic shear and plasma pressure profiles to obtain optimised profiles for improved MHD stability, energy confinement and non-inductive current drive.

5. Predicting fusion performance

Following installation of the ITER-like ICRH Antenna (2007/8) and the NB Enhancement (2009/10), the upgraded power on JET will approach 45MW. This will allow the first stable operation of ELMy H-modes at 3.4T with densities up to the Greenwald density and $q_{95}<3.5$. With ITER-like wall materials, this will provide the opportunity to extend confinement and MHD scalings, including the H-mode pedestal scaling, simultaneously at high Greenwald density and low collisionality. The present uncertainty in the β-dependence of confinement will be resolved as the uncertainty of the stored energy measurement (~10%) becomes less significant at the higher levels of stored energy achievable with the upgraded power. In addition to the development of empirical scalings, these experiments, together with an extended diagnostic capability, will be essential for validating models being developed at JET and elsewhere within Europe.

In the hybrid regime, ITER-relevant conditions with T_i~T_e and low plasma rotation will be accessible at high density at 2.3-2.8MA and high heating powers with a toroidal magnetic field above 2.6T. JET results in this regime are crucial since confinement with $T_i>T_e$ and high rotation may be optimistic for extrapolating to ITER. The upgraded power will also allow studies of the influence of NTMs on the evolution of the current density profile at high β, to define the requirements for the control of q_0 and to optimise the stability of the scenario by varying q_0.

In AT regimes, 0-D scalings and 1-D transport/theory-based models are not capable of predicting ITB characteristics under ITER conditions with confidence. This requires auxiliary heating powers in excess of 35MW on a JET-class device to determine the confinement scaling at low ρ_*.

6. Materials and components for nuclear operation

A few diagnostics are being tested for their potential application to ITER and DEMO.

7. DEMO integrated design: towards high availability and efficient electricity production

Aspects of the JET programme relevant to high availability and efficiency for a future reactor include: Optimisation of plasma-wall compatibility; Experiments aimed at achieving reliable operation in future devices; Experience in achieving high availability of JET Facilities, in view of comparable complexity with ITER/DEMO; Technology tests aimed at optimising the ITER detailed design; Optimisation of remote handling techniques; Experience gained in quality control during device construction

FORWARD PLANNING:	Present end-date for exploitation of JET is end-2010. Discussion has started on the prolongation to 2014 to fully exploit the ongoing enhancements and perform a DT campaign.

<table>
<tr><td>FACILITITY</td><td colspan="6">Tore Supra, Euratom-CEA Association</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="6">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
• Tore Supra tokamak: 154 M€ (1989)
• CIEL project: upgrade of inner components, 10 M€ (2002)
• CIMES project: upgrade of LHCD and fuelling systems, 13.5 M€ (2010)
Total cost in 2007 euros : 240,5 Mio€</td></tr>
<tr><td colspan="6">COST OF FORESEEN UPGRADES:
Proposed for consideration:
• CIMES 2 project: upgrade of ICRH system to long pulses, 2014, 15 M€
• CIMES 3 project: installation of an ECRH system, 2013, 10 M€</td></tr>
<tr><td colspan="6">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 70
- yearly cost of operation in 2007 euros: 7 M€ (excluding manpower)
- yearly manpower for operation in ppy: 100 (40 professionals, 60 technicians)</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="3">Number of facility users: 250</td><td colspan="3">Yearly integrated equivalent full time facility users: (ppy) 140</td></tr>
<tr><td colspan="6">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 50 (+20 in progress)</td></tr>
<tr><td colspan="6">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td colspan="2">Year</td><td colspan="2">Journals</td><td colspan="2">Conferences</td></tr>
<tr><td colspan="2">2004</td><td colspan="2">33</td><td colspan="2">98</td></tr>
<tr><td colspan="2">2005</td><td colspan="2">67</td><td colspan="2">60</td></tr>
<tr><td colspan="2">2006</td><td colspan="2">23</td><td colspan="2">75</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="6">Collaborations inside EU: 50 collaborations with 22 institutes in 17 countries
Collaborations outside EU: 20 collaborations with 12 institutes in 6 countries
Number of experimental contributions to ITPA (if applicable) : 11
Prospects: increase of collaboration with new superconducting tokamaks in Asian countries; increased involvement in fusion programme of other French research institutes through the newly created “Fédération Nationale de Recherche”</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="6">• I_p up to 2 MA, B_t up to 4.2 T (superconducting toroidal magnets)
• R = 2.4 m, a = 0.75 m, plasma volume = 25 m^3, circular cross section
• Power exhaust: up to 25 MW steady state, all PFC actively cooled, toroïdal pump limiter material = CFC
• ICRH 12 MW/30s, LHCD 8 MW/60s, ECRH 700 kW/10s (transmitter power)
• Particle exhaust: up to 4 Pam^3s^{-1}
• Steady state high reliability pellet injector, supersonic gas injection
• Comprehensive set of 40 diagnostics including an unique set of IR cameras for power load studies and reflectometers for turbulence studies</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="6">Already decided:
• 2010: 11,2 MW LHCD power steady state (transmitter)
• 2010: articulated inspection arm operating under vacuum with process for in situ inspection, deposited layer analysis and detritiation
Proposed for consideration:
• 2012: 6 MW ICRH power steady state, 12 MW/30s (transmitter)
• 2013: 6 MW ECRH power steady state (transmitter)
• 2014: 15 MW ICRH power steady state (transmitter)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="6">Integration of physics and technology constraints for long pulse high power discharges:
• 20 years of reliable operation of superconducting magnets in a tokamak environment
• Long duration discharges up to 6 minutes at 3 MW injected power level and zero loop voltage (1GJ injected/exhausted energy)
• Practical implementation and safe operation of actively cooled PFC up to 10 MW injected power level
• Physics of evanescent loop voltage discharges in a situation where current diffusion has fully taken place (density peaking without Ware Pinch, oscillating regimes with interplay between transport and current profile, steady state ITB..)
• Physics of plasma wall interaction in a situation where PFC works at constant surface temperature (deuterium retention, erosion/redepostion studies in a C environment)</td></tr>
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<tr>
<td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td>
<td>
The main drive of the Tore Supra programme is based on its unique capability to routinely produce long discharges at high heating and current drive injected power. Presently, we do not foresee, worldwide, another device with such capability until the middle of next decade. For several years, Tore Supra will thus be a unique facility to investigate high power, long discharges issues for ITER and DEMO.
Tore Supra will allow to progress towards:

A: scenario development for continuous operation of tokamaks through the study of discharges at evanescent loop voltage in a situation where the current diffusion has fully taken place,
B: long discharge operability and reliability of tokamaks through the integration of all constraints (on physics and technology sides) stemming from working in an environment where all PFC are actively cooled.

Continuous operation of tokamaks
This field of study is dominated by the complex non linear coupling between transport, MHD, pressure and current radial profiles. As such, it is of vital importance, for preparation and extrapolation of scenarios to ITER and DEMO, to be able to experimentally validate scenario modelling on time scale much longer than the current diffusion time. It is worth to note that this physics is largely decorelated from the one of H mode pedestal, the transport properties of the plasma core being similar in L or H mode. Such studies strongly rely on advance in transport and turbulence understanding through experiments (Tore Supra is equipped of a comprehensive set of diagnostics for turbulence measurements), integrated modelling and first principle based theory. On a 5 year perspective, the completion of the 1^{rst} phase of the CIMES project will allow extending current profile capability and the operational domain of zero loop voltage discharges to higher density and higher plasma current. Studies will concentrate on elaboration of steady state scenarios with dominant electron heating and no torque injection. On a 10 year perspective, with the capability of the subsequent phases of the CIMES project, Tore Supra would be in an unique situation to study a key point for development of the steady state tokamak: the demonstration that regime with plasma core characteristics suitable for steady state scenarios, can be sustained with low power heating and current drive actuators. In such studies, the major part of the H/CD power would be used to simulate the α heating and complement the edge bootstrap current that is missing in L mode. This programme would be accompanied by the development of innovative feedbacks of interest for ITER and takes advantage of the C environment of Tore Supra that offer the maximum experimental flexibility (notably at lower density) for scenario elaboration and studies. It would allow gaining considerable insights for the choice of the heating mix for DEMO. This field of study mainly contributes to mission 4 and mission 5 and to a lesser extent to mission 1.
Long discharge operability and reliability of tokamaks
Tore Supra is presently, and for several years the only facility with all PFC actively cooled. As such Tore Supra can be viewed as a test bed that qualifies operational procedure and technological objects as well as investigates maintenance, safety and reliability issues related to long pulse operation. This, inter alia, includes:

studying the ageing of the superconducting toroidal field system
studying the ageing of actively cooled PFC in a real tokamak environment
studying long discharge related issues as erosion, D (i.e. T) retention and dust production
developing know-how, actuators, sensors and feedback loops to optimise plasma performance while maintaining PFC in the safe operation domain,
developing and testing of steady state, ITER relevant RF couplers
studying strategies for simultaneous high power coupling of several H/CD schemes
developing conditioning techniques in the presence of the toroidal field

In a five years perspective, these studies will benefit of the completion of the first phase of the CIMES project, that will extend the operational domain of long discharges on Tore Supra and allow testing the PAM LHCD launcher, a concept relevant to ITER. In a 10 years perspective, with the capability of the subsequent phases of the CIMES project, besides testing ITER relevant, ICRH and ECRH antennas for long discharges in a tokamak environment, these studies would be extended to higher power handling. Specific operational procedures to be qualified for ITER might be studied as well.
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	Of specific interest for next generation devices is the operational study of high heat flux actively cooled PFC. It is worth to note that during ramp-up and ramp down phase, ITER and even more DEMO will operate in limiter configuration for duration much longer that the characteristic cooling time of PFC. Physics of plasma wall interaction in limiter configuration has thus to be studied and optimised with the constraints of long duration discharges. Furthermore, CFC armoured PFC are one of the option contemplated for ITER divertor during the first phase of operation. Outcome of the use of C in a tokamak environment thus deserves being fully investigated. Tore Supra by routinely producing repetitive long discharges can simulate next device very long discharges (up to hours) with the fundamental feature that PFC works at constant surface temperature. It is thus a unique tool to study, in a C environment, erosion phenomena, D (i.e. T) retention and dust production. Tools and diagnostics for in situ measurement of erosion/deposition, characterisation of deposited layers and detritiation are presently being developed. In a 5 years perspective, the articulated inspection arm will allow to perform these measurements and/or operations when desired during an experimental campaign without breaking the vacuum. In a 10 years perspective, it will allow to develop Tritium and dust inventory minimisation strategies and the experimental qualification of adequate surface conditioning. Such programme, pertaining to mission 2, 3 and 4, is of vital importance for developing the necessary know-how for operational integration of all constraints linked to mastering the handling of high H/CD power in an environment of actively cooled PFC. This is a key point for the success of ITER operation. Tore Supra, through EFDA collaborative activity, will allow not only to further develop this know-how but, even more importantly, to train European teams in this field. This is an asset placing Europe in a foremost position for ITER exploitation.
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above)

<table>
<tr><th>FACILITY</th><th>ASDEX UPGRADE (AUG), IPP GARCHING</th></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="2">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: 220 Million €</td></tr>
<tr><td colspan="2">COST OF FORSEEN UPGRADES:
– ECRH-II (4MW/10s): in total 12.7 M€, 6.9 M€ spent
– Internal coils and stabilising shell: 6.9 M€
– LHCD, preliminary estimate: 11 M€</td></tr>
<tr><td colspan="2">OPERATION:
– Yearly cost of operation including manpower: 9.75 Million €
– Average number of operation days / year (over the past 5 years): 60
– Manpower for operation: 85 ppy</td></tr>
<tr><td rowspan="3">USE OF FACILITY</td><td>Number of facility users: 105 IPP + 30 external</td><td>Yearly integrated equivalent full time facility users: 90 ppy</td></tr>
<tr><td colspan="2">Number of PhD/diploma thesis using experimental data from the facility in the last-10 years: 44 / 27</td></tr>
<tr><td colspan="2">Number of yearly publications based on experimental results from facility:
<table>
<tr><th>Year</th><th>Journals</th><th>Conferences</th></tr>
<tr><td>2004</td><td>57</td><td>41</td></tr>
<tr><td>2005</td><td>91</td><td>66</td></tr>
<tr><td>2006</td><td>55</td><td>34</td></tr>
<tr><td>2007</td><td>91</td><td>58</td></tr>
</table></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="2">Collaborations inside EU: JET, CEA, CIEMAT, CRPP, DCU, ENEA, ERM/KMS, FOM, FZ-J, FZ-K, HAS, Hellenic Rep., IPP.CZ, IST, MEdC, ÖAW, RISØ, TEKES, UKAEA, Univ. of Latvia, VR + Universities of Augsburg, Bayreuth, Munich (LMU, TU), Stuttgart, Strathclyde, Tübingen, Vienna

Collaborations outside EU: ITER, ALCATOR-CMOD/MIT, DIII-D/GA, EAST/IPP-CAS, HL-2A/SWIP, IAP/RAS, JT60U/JAEA, KSTAR/NFRI, Kurchatov Inst., NSTX/PPPL, ORNL, SST-1/IPR, Univ. of Wisconsin

Number of experimental contributions to ITPA: 44 (in 2008)

Sharing facility with other fields of research: Atomic Physics, Astrophysics

Prospects: Since 2002 the AUG programme has been opened to all EU Associations (annual call for participation, joint international programme committee with 10 EU Associations defining next year's AUG programme). A further increased co-operation on specific programmatic missions will be welcomed. In order to give EU Associations an even better opportunity to conduct their scientific programme at AUG, the weekly operation hours of the device have to be extended. This idea comes from the Commission Services and the EFDA leadership. IPP has explored the possibility of increasing the number of operational time (6 days of AUG operation instead of 5 per fortnight or increase the number of extended shifts which have a 30% higher pulse rate in comparison with standard shifts). The implementation of such a step will require an increase in the number of staff by10 people (8.5 engineers / technician, 1.5 physicist), which will only be possible by substantial external support from the Associations / EFDA. The integration of engineers and technicians into a complex working environment like AUG is a difficult task and will require a sufficient phase of training in 2008. If all necessary resources can be provided for the coming years, IPP proposes to initiate in 2008 all necessary steps for an extended AUG operation in 2009.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="2">(Based on the optimized use of generator power and repair of the presently damaged flywheel generator, scheduled in mid-2008)
• I_p =1.6MA , B_t=3.1T , R=1.65m, a=0.5m, V= 14 m^3, pulse duration (10s @ 1.2MA) up to 5 current redistribution times
• ITER-like coil configuration & shape, X-point plasma, δ= 0 – 0.6 (at 1MA), κ = 1.4 – 1.8
• Plasma facing materials: tungsten coated carbon tiles (coating thickness: 200μm on LFS divertor tiles, otherwise 4 μm)
• Heating systems: NBI(20MW/10s), ICRH(8MW/10s), ECRH-I(2MW/2s) ECRH-II(1MW/10s/105&140GHz); in total 30 MW -> P/R ratio close to ITER value
• Diagnostics: 58
• Pellet injection systems for fuelling & ELM pacing (centrifuge: 80 Hz, 130 pellets; blowergun: 120 Hz, 120 pellets)
• Routinely used disruption mitigation systems (fast gas puff with up to 10^{23} atoms, valve opening time 1ms)</td></tr>
</table>

FUTURE TECHNICAL CAPABILITIES	➢ Heading systems: • ECRH-II (4MW/10s/4-f gyrotrons 105-140GHz) with fast steerable launchers (poloidal angle variation of 10o in 100ms) for – avoidance of impurity accumulation by application of central heating – suppression of NTMs – sawtooth control – off-axis CD in advanced tokamak regimes – support for Collective Thomson Scattering (CTS) • LH(4MW/10s, 400kA off-axis CD), under consideration, an additional option could be a replacement of ECRH-I by ECRH-III(4MW/10s) for current profile control of improved H-mode as well as ITB scenarios ➢ Diagnostics: • Collective Thomson Scattering • additional Fast Ion Loss Detectors (FILD) for ploidal lost ion distribution • upgrade SXR (128 lines, 2MHz + 80 lines, 500kHz) • MSE for E_r (half energy system) • ELM resolved spectroscopic measurements in the divertor • Li/Na BES for edge fluctuations • simultaneous core/edge Thomson scattering • increased time resolution and RT capabilities of many diagnostics ➢ Pellet injection systems (with increased pellet number) for ELM control ➢ Internal coils & conducting shell (2010-2012) for – ELM suppression – NTM rotation control – RWM stabilisation
PROGRAMME ACHIEVEMENTS	*(in chronologic order)* • Ohmic H-mode • Highly reliable RF ion sources for NBI • **Completely Detached H-Mode (90% radiation)** *(pioneered)* • Operation with W-strike point tiles • Identification of stiff temperature profiles • **Edge operational diagram (H-mode power threshold, ELM regimes, H-mode density limit)** *(pioneered)* • Optimization of divertor geometry • Development of advanced diagnostic methods for plasma core, edge, SOL and divertor • Influence of shaping on good H-mode confinement at high density • **Coupling of 6MW ICRF power in the presence of strong Type-I ELMs (3-dB couplers)** *(pioneered)* • **HFS pellet fuelling** *(pioneered)* • HFS pellet fuelled H-modes close to and above Greenwald density • Ion-ITBs with T_i~30keV • Electron-ITBs with ctr-ECCD (T_e~30keV) • **Discovery of improved H-modes (ITER Hybrid scenario) with enhanced confinement and stability** *(pioneered)* • **Discovery of FIR NTMs** *(pioneered)* • **Complete 3/2 NTM-stabilisation with ECCD** *(pioneered)* • Full non-inductive current drive at 400kA • Thermographic characterisation of ELM and disruption power loads • Standard H-mode with Type-II ELMs • **High β_N=3.5 improved H-modes with Type-II ELMs** *(pioneered)* • Verification of QH-mode regime • **ELM pacing of Type-I ELMs** *(pioneered)* • Extended operational range for improved H-modes including ν^* and ρ^*scaling • Integrated ITER-relevant H-mode with impurity seeding and ELM pace making • **Influence of collisions on deuterium and impurity transport and benchmarking of turbulent transport models** *(pioneered)* • Electron transport from ECRH heat pulse propagation and verification of turbulent transport thresholds • Broadening of NBCD density profiles by anomalous fast ion diffusion • **Fast ion losses caused by TAE** and global MHD instabilities • **Disruption mitigation for heat loads and forces by killer pellets and strong gas puffs (routinely used)** • Development of quantitative tungsten diagnostics for the core plasma • **First demonstration of H-mode operation with a full W-device with tolerable W-concentration by applying central wave heating** *(pioneered)* • Tungsten sputtering by fast NBI ions and ICRF sheath accelerated ions • Material migration path based on spectroscopy and surface analysis • **First demonstration of plasma operation without wall conditioning by boronization with a full W-wall** *(pioneered)* • Demonstration of the reduction of the deuterium inventory in the transition from an all-C to an all-W device

PROGRAMME: ADRESSING THE PROGRAMME NEEDS	*AUG contributes to R & D Missions 1-5 in substantial manner:*

PROGRAMME: ADRESSING THE PROGRAMME NEEDS

AUG contributes to R & D Missions 1-5 in substantial manner:

Mission 1 (Burning plasmas):

- AUG is sufficiently large for fusion relevant studies of fast particle effects. Study of fast ion physics and their interaction with MHD instabilities using new diagnostics like fast ion loss detectors (FILD) and CTS. Strong ICRH can be used for creation of fast ions, ICRH created beatwaves can be used to probe AE stability.
- Further development and understanding of reactor relevant scenarios, e.g. improved H-modes extrapolate to ITER either reaching nearly ignition at full current or to Q=10 at reduced current of 11MA permitting pulse lengths of 3000s. The installed heating power at AUG is sufficient to achieve β_N well above three even for the highest plasma currents. Two different routes will be followed either at ITER-relevant ν^* or at densities close to the Greenwald density for power and particle exhaust. (Both values can only be simultaneously met at devices of ITER size.)

Mission 2 (Reliable tokamak operation):

- Integrated plasma control of β (NBI, ICRH, ECRH), rotation (NBI vs. wave heating, internal coils), density (gas puff and pellets) and current density profile using a large set of RT diagnostics and RT regime recognition.
- Active control of NTMs and sawteeth using ECRF and internal control coils.
- Active Type-I ELM control with pellets.
- Plasma fuelling to densities above Greenwald density relevant for DEMO operation, e.g. HFS pellets.
- ELM mitigation / avoidance by internal coils and QH-mode;
- Development of small ELM regimes (grassy, Type-II).
- Disruption avoidance by tailoring of discharge scenario including neural network prediction of disruptions; routine disruption mitigation by strong gas puffs.

Mission 3 (First wall material & compatibility with ITER/DEMO relevant plasmas):

- Operation of a full W device including compatibility with heating methods, in particular the use of ICRF and corresponding antenna optimisation.
- Assessment of W and C PFCs with emphasis on plasma performance, C-deposition and tritium retention, erosion at high heat loads (ITER-relevant P/R), hydrogen and noble gas balance. This is supported by the application of state-of-the-art surface analysis methods.
- Assessment of compatibility of standard and advanced scenarios with fully W-coated wall.
- Development and testing of advanced plasma facing materials (Ti-C, Ti-N-C) and proof of their plasma compatibility.

Mission 4 (Technology & Physics of long pulse and steady-state):

- Besides new devices with superconducting coils, AUG has the longest pulse duration with full applied heating compared to the resistive current diffusion time in ITER geometry and is therefore well suited to study plasma scenario related steady-state issues.
- Physics of improved H-modes (ITER Hybrid scenario) will be further investigated with the aim to prepare long pulse operation for ITER (Q=10 at I_p=11MA and pulse length > 3000s).
- On AUG fully non-inductive CD at relevant plasma currents and β_{pol} values is considered to be achieved by a LH system (4MW/10s) with a PAM launcher in addition to NBCD and ECCD. The use of a PAM launcher in a divertor tokamak will be a new element in the EU fusion programme. As an alternative CD tool a further extension of the ECRH (additional 4MW for 10s) could be envisaged.
- Active pressure and current profile as well as MHD stability control (ECRF, internal coils and conducting shell for RWM stabilization) is a major element of the future AUG programme.

Mission 5 (Predicting fusion performance):

- Based on the gained experimental results and theoretical understanding of ITER scenarios, their extrapolation to future devices will be improved. This will be supported by the development of first principle based theories for transport and stability and their benchmarking against experimental results in collaboration with the strong theory department at IPP Garching.
- The integration of ab-initio theories for plasmas on both closed and open flux surfaces has been successfully started (ASTRA, SOLPS, GEM) and will be continued towards a 'numerical tokamak'.

5 year perspective:

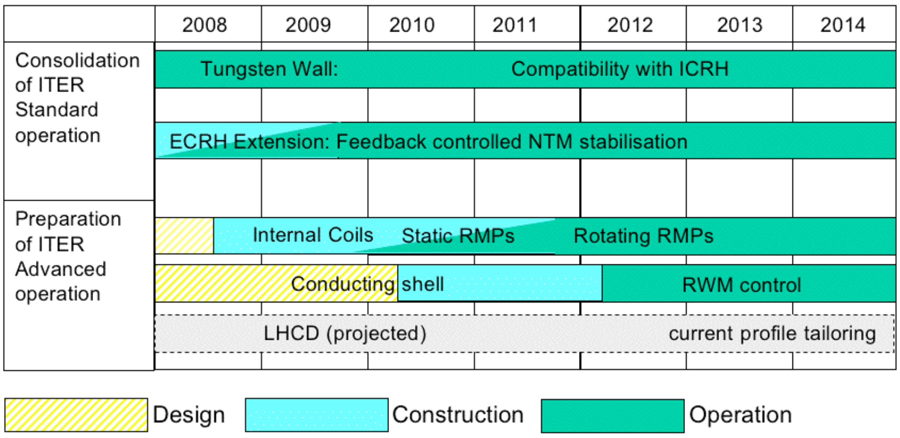
- Full exploitation of all ITER-relevant scenarios in a full W-device and assessment of their performance and compatibility with high-Z walls.
- Verification of NTM control strategies for ITER.
- Use of internal coils for ELM and rotation control. Operation with tolerable ELMs at sufficient high confinement.
- Optimization of disruption avoidance and mitigation schemes for ITER / DEMO.

10 year perspective maintaining the for AUG typical high level of flexibility:

- Development and control of long pulse or even steady-state tokamak operation at reactor relevant performance.
- Resistive wall mode control with internal coils and stabilizing shell.
- In addition, the AUG device has the capability to run large volume plasmas (~25m^3) at currents of up to 2.5MA by using ferritic inserts, however, flexibility in power and particle handling will be reduced. Such a major rebuild would partly overlap in dimensionless operational space with JT60-SA and thus fulfil requirements of an ITER satellite.

FORWARD PLANNING

Timetable and planning for the operating period up to 2015:



<table>
<tr><td>FACILITY</td><td colspan="3">FTU, EURATOM-ENEA</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: ~(125+12.5+1.3)M€
FTU+LH 1983-91 + IBW+ECRH 1993-99 + 2 spares LH gyrotrons 2000</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: 2009-2011, 6 M€
60° Toroidal sector Liquid Lithium Limiter, RT steerable ECRH launcher, Refractometry, GEM, CTS new equatorial test. TBD: 2 extra lines ECRH with 850kW/1s gyrotrons</td></tr>
<tr><td colspan="3">OPERATION
Average number of operation days/years (last 5y) = 70
yearly cost of operation (in 2007 €) = 2.5 M€/y
yearly manpower for operation ~ 35 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of Facility users: ~110</td><td>Yearly integrated equivalent full time facility users: ~55</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~10/y</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility: ~40</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>20</td><td>22</td></tr>
<tr><td>2005</td><td>18</td><td>15</td></tr>
<tr><td>2006</td><td>11</td><td>27</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Inside EU: CEA, CIEMAT, KFJ, IPPLM, CRPP, ULB, UTh, MECT
Outside EU: PPPl, UCSD, UCI, IAP, Triniti, SWIP, HEFEI, Zhejiang Univ. IFTS
Number of experimental contribution to ITPA: Transport, MHD. Diag, SSO
Sharing facility with other field of research: Atomic Physics (Spectroscopy DB)
Prospects: Latvia, IST, Malta</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">I_P=1.6MA, B_T=8T, t=1.5s, 1.5MW/1s-LH(8GHz), 1.6MW/0.5s-ECRH (140GHz), 1MW/1s-IBW (433MHz), 8 pellets/shot-1.5kms, Full Metallic PFC, rail Liquid Lithium Limiter
$n_i T_e \tau_E \sim 10^{21}$, $T_e \geq T_i$, $Z_{Eff} \sim 1$, $\tau_E \leq 120$ms, $\rho^* \geq 1.6 \cdot 10^{-3}$, $\nu^* \geq 0.3$, P/R~5MW/m</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">t=3s @ $B_T \leq 5$T, 3MW/1s-LH(8GHz), Real Time steerable ECRH launcher, allowing: 1)RT MHD control, 2)Toroidal launching angles $-35° \leq \theta \leq +35°$ for ECCD, 3) Electron Bernstein Wave heating above 140GHz cutoff density and 4) CTS in ITER-like configuration.
Toroidal sector Liquid Lithium Limiter
TBD: extension up to 3MW/0.5s of ECRH system (140GHz).</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">FTU routinely works at its design parameters and mostly at ITER values of magnetic field and electron density thus allowing testing scenarii and additional heating and current drive systems in a similar range of parameters.
-Electron transport barriers were firstly achieved on FTU in 1998 with ECRH in the plasma current ramp phase[1] and extended in 2000 to higher density [2]: T_{e0}=14keV, $\bar{n}_e=0.4 \cdot 10^{20}$m^{-3}.
-The combined use of LHCD and ECRH produced wide ITBs r/a~0.5[3] further extended to higher density $n_{e0}=0.9\text{-}1.2 \cdot 10^{20}$m^{-3}, with $H_{97} \sim 1.6$, for a time duration much longer than the energy confinement time, comparable to the current diffusion time[4,5]. The neutron yield was substantially increased, with ion heating occurring through ion-electron collisions[6].
-In the so called PEP modes, $n_{e0} \sim 5 \cdot 10^{20}$m^{-3} and a fusion product $n_i T_i\ t_E = 0.8 \cdot 10^{20}$ m^{-3} keV·s have been obtained at I_P=1.2MA with repetitive pellet injection, limited only by the time duration of the current plateau[7].
-Full I_p=0.5 MA LHCD has been achieved at high density[8] $0.8 \cdot 10^{20}$ m^{-3} while, at the same I_p, up to 75% current drive has been achieved at $n_e=1.2 \cdot 10^{20}$m^{-3} with a 6 times increase of the neutron yield.
-ECRH has been successfully used to study[9], control and automatically stabilise in real time[10] MHD. More recently, ECRH has been successful to control and avoid disruptions[11].
-Synergy between ECRH and LHCD[12] has been achieved both at a much higher field, thus extending operating range, or at a lower field than the cold resonance (ITER condition).
-IBW have produced clear indications of improved core confinement, possibly ITB[13].
- Very fast plasma restart and cleaner plasma operations have been obtained with boronisation[14] and more recently also with a Liquid Lithium Limiter, which handled so far 5MW/m^2. Increased SOL T_e with reduced recycling and quasi quiescent MHD are observed that seems to trigger a new high density regime characterized by peaked profile up to densities close to Greenwald values[15].
-Relevant results for burning plasmas have recently been produced on FTU analyzing MHD activity driven by fast electron accelerated by LH waves[16], the so called "Electron Fishbones". Systematic theoretical studies of these phenomena[17] have produced a sound basis for their modelling and interpretation. These results suggest that fast electron driven MHD activities can be used for investigating nonlinear fluctuation behaviours and fast particle transport in regimes that are relevant for burning plasma experiments.

- The first observations of Beta induced Alfvén Eigenmodes (BAE) nonlinearly excited by</td></tr>
</table>

	large magnetic islands have been made in FTU and interpreted with a theoretical model that gives good qualitative and quantitative agreement with experimental results[18,19]. -A new LHCD launcher, based on the concept of Passive Active Multijunctions (PAM), which is suitable to be used on ITER, has been succesfully tested in 2003. Its performance supports this kind of choice for the ITER LH launcher[20]. 1. F.Alladio et al., Proc. 17th Fusion Energy Conf.,Yokohama, Japan, October 19–24, 1998, IAEA-F1 CN-690OV404, International Atomic Energy Agency (1998) 2. P. Buratti et al., Phys. Rev. Lett., **82**, 560 (1999) 3. F. Alladio et al., Proc. 18th Fusion Energy Conf., Sorrento, Italy, October 4–10, 2000, IAEA-OV01, International Atomic Energy Agency (2000) 4. B. Angelini et al., Proc. 19th Fusion Energy Conf., Lyon, France, October 14–19, 2000, IAEA-OV04-5, International Atomic Energy Agency (2002) 5. V. Pericoli Ridolfini et al., Plasma Phys. Control. Fusion **47**, p. (2005) 6. V. Pericoli Ridolfini et al., Nucl. Fusion, **43**, 469 (2003) 7. D. Frigione et al., Nucl. Fusion, **41**, 1613 (2002) 8. V. Pericoli Ridolfini et al., Phys. Rev. Lett., **82**, 93 (1999) 9. E. Lazzaro et al., Phys. Rev. Lett., **64**, 6038 (2000) 10. S. Cirant et al., Fusion Science and Technology **53** (2008) 11. B. Esposito et al., Phys. Rev. Lett., accepted for publication 12. G. Granucci et al., Proc. 29th Conf. Plasma Physics and Controlled Fusion, Montreux, Switzerland, 17–21 June, 2002, ECA26B, P-4.038, (2002) 13. R. Cesario et al., Phys. Plasmas, 11, 4721 (2001) 14. M. L. Apicella et al., Nucl. Fusion, 45, 685 (2005) 15. G. Mazzitelli et al., 34th EPS Conference on Plasma Phys. and Contr. Fusion, Warsaw, Poland 2 - 6 July 2007 ECA Vol.31F, O-2.001 (2007) 16. V. Pericoli Ridolfini et al., Nucl. Fusion, 47, S608 (2007) 17. F. Zonca et al., Nucl. Fusion **47**, 1588 (2007) 18. P. Buratti et al., Nucl. Fusion **45**, 1446 (2005) 19. S.V. Annibaldi et al., Plasma Phys. Control. Fusion **49**, 475 (2007) 20. V. Pericoli Ridolfini et al, Nucl. Fusion **45**, 1085 (2005)
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	Five years perspective: FTU is a full metallic machine that can work at ITER density and magnetic field. Its plasmas are only heated through RF on electrons that in turn heat ions via collisions. Moreover because of FTU compactness, plasma wall issues and the behaviour of liquid metals (Li) as PFC can been addressed at relevant P/R values (typically P/R~ 5MW/m or more). The presence of LH driven fast electrons also allows studying fast particle driven MHD thus investigating nonlinear fluctuation behaviours and fast particle transport in regimes that are relevant for burning plasma experiments. FTU contribution in the coming years can then be summarized as follows with respect to the missions identified for the future of the fusion programme: **Mission 1)** With full LH and EC power (the latter possibly extended to 3MW) FTU will provide a powerful tool to check and benchmark in a wide range of parameters the newly developed linear and nonlinear modelling of fast particle driven MHD and related transport phenomena. Investigations of nonlinear excitation of Alfvénic modes by MHD fluctuations will continue. The new EC launcher will also provide on FTU a unique environment to check, in ITER conditions, the CTS recently classified among “ITER enabled” diagnostics. **Mission 2)** Combined use of LH and ECRH (with flexible launchers) will allow: a) studying plasma start/ramp-up for operation optimization, flux saving and plasma control; b) Real Time control of MHD modes (tearing, sawtooth); c) disruption avoidance and d) studying of runaways formation, in presence of LH driven fast electrons, at the disruption quench. **Mission 3)** Behaviour of liquid lithium limiter up to loads of $10MW/m^2$ will be investigated and associated improved plasma regimes with higher SOL Te, low recycling and quiescent MHD will be studied. Plasma wall interaction study at relevant P/R will be pursued with insertion of different metals as PFC. Formation and dynamics of metallic dust generated in the SOL will be studied and technique for its diagnostic will be pursued. Testing of CTS as a diagnostic for fast particle in burning Plasmas will be pursued. **Mission 4)** CD at high density will be studied with particular attention to possible efficiency degradation mechanism due to wave interaction at plasma periphery. Study of advanced regimes with only electron heating will be continued. 10years perspective: **Not considered pending decision on FAST**
FORWARD PLANNING	Refurbishment of LH and ECRH system is undergoing, full power availability foreseen in 2009. Refurbishment of poloidal systems PS foreseen late 2008 as well as of the existing MJ LH grill to allow easier current ramp up CD. New Real Time steerable ECRH (able to include a new CTS equatorial line). Possible implementation of two new 140 GHz lines (2x850kW/1s) to be discussed.

FACILITITY	**TEXTOR, FZ Jülich (TEC)**		
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: 132 Mill. Euro (1981 – 2003)		
	COST OF FORESEEN UPGRADES: 2 Mill. Euro 2008-2012		
	OPERATION - average number of operation days/year (over the past 4 to 5 years): 75 - yearly cost of operation in 2007 euros (full cost with overhead): 4.9 Mill. - yearly manpower for operation in ppy: 11		
USE OF FACILITY	Number of facility users: 135	Yearly integrated equivalent full time facility users (ppy): 40	
	Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 70 in 10 years		
	Number of yearly publications based on experimental results from facility:		
	Year	Journals	Conferences
	2004	61	94
	2005	52	129
	2006	54	101
COLLABORATIONS	Collaborations inside EU: 31 Collaborations outside EU: 23 Number of experimental contributions to ITPA : 27 Sharing facility with other fields of research: no Prospects: no significant changes		
PRESENT TECHNICAL CAPABILITIES	Tokamak: R=1.75 m, a= 0.47 m, circular cross section, toroidal belt-limiter (pumped), plasma volume 7 m^3, 16 TF coils, $B_{T,\ max}$= 3.0 T, $I_{P,\ max}$= 0.8 MA, pulse length 12 s; auxiliary heating power: NBI 2x2 MW co/ctr, ICRH 2x2 MW, ECRH 1 MW Dynamic Ergodic Divertor: 16 helical in-vessel RMP coils; base modes: 12/4, 6/2, 3/1, I_{max} = 15 kA, DC and rotating field up to 10 kHz Plasma densities 1-25·10^{18} m^{-3} (last closed flux surface - LCFS), 1-8·10^{19} m^{-3} (core), electron temperatures 20-200eV (LCFS), 0.5-5 keV (core), PWI test facilities: 2 air-lock systems with gas feed, external heating and active cooling, equipped with comprehensive diagnostics, capable of parallel heat and particle flux densities of up to 300 MWm^{-2} and 5·10^{24} $m^{-2}s^{-1}$, respectively.		
FUTURE TECHNICAL CAPABILITIES	Upgrade of the Plasma-Wall Interactions (PWI) test facilities: laser systems for in situ fuel desorption, material ablation and dust detection, new systems to enable in situ W-coating techniques		
PROGRAMME: ACHIEVEMENTS	First precise measurement of the central current density by polarimetry showing **q<1** during the whole sawtooth cycle. First demonstration of efficient **He-exhaust** with the pumped limiter ALT-II. Definition of a figure of merit for He-exhaust: the ratio τ^*_{He}/τ_E. Pioneering work by introducing low-Z wall coatings: **carbonization, boronization** and **siliconization**. First demonstration of feed-back controlled **radiation cooling** (90% of total power) with seeded noble gases. Discovery of an **improved confinement** regime at high density with stationary plasma energy and strong edge radiation by seeded and intrinsic impurities (RI-mode). Investigation of the effect of externally imposed **radial electrical fields** (electrodes) to the plasma edge to study bifurcation phenomena observed during L-H transitions. Pioneering work on **basic plasma-wall interaction** processes and the link between erosion processes, plasma edge transport and plasma core properties: Quantification of **chemical erosion** processes and **radiation enhanced sublimation** of low-Z wall components at high flux densities. Evidence for hot spot formation on carbon PFCs due to thermal electron emission. Evidence for **enhanced carbon transport** due to enhanced erosion of re-deposited carbon: pioneering of **$^{13}CH_4$ tracer techniques** for material migration studies. **Qualification of high-Z PFC**: erosion, melt layer stability, blistering. Development and tests of concepts for bulk W divertor tiles for JET. Identification of **hydrogen recycling processes** (atomic and molecular release, reflection, charge exchange, excitation) with spectroscopy. Basic research on transport and stability with resonant magnetic perturbations generated by the **Dynamic Ergodic Divertor (DED):** Identification of generic transport properties in stochastic magnetic topologies (formation of a helical divertor, application to ELM-mitigation, control of intermittent transport), study of tearing		

	modes excited by external error fields and their suppression by localised heating and current drive methods. Pioneering work in **diagnostics**: active spectroscopy (CXRS, He-/Li-Beam, LIF), high-resolution emission spectroscopy, colorimetry, VUV spectroscopy, high-resolution Thomson Scattering, Collective Thomson Scattering, ECE-imaging, dispersion interferometry. Development and benchmarking of the local erosion deposition code **ERO-TEXTOR** and the Janev-Reiter database for hydrogen and hydrocarbons (HYDKIN). Development and benchmarking of the codes **EIRENE** and **B2-EIRENE** (SOLPS) for neutral particle transport and integrated divertor physics studies, 3D integrated edge transport with **EMC3-EIRENE** in (partially) ergodized B-fields.
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	TEXTOR provides a PWI test facility, tools for RMP studies and a test-bed for diagnostic developments for ITER and W7-X. The following R&D Missions for the fusion roadmap are addressed, most of them embedded in joint experiments (ITPA and TEXTOR-IEA agreement on plasma-wall interaction) with large divertor tokamaks: <u>Mission 1:</u> Development of fast particle detection diagnostics (CTS, CXRS) <u>Mission 2:</u> Impact of transient power and particle fluxes to PFCs (disruptions, runaways, ELMs, blobs) and development of control schemes (disruption mitigation, RMP), control of instabilities with local ECRH/ ECCD, development of wall conditioning applicable under permanent magnetic field, development of start-up scenarios in limiter configurations. <u>Mission 3:</u> Qualification of high-Z PFCs for fusion applications: high temperature behaviour, melt layer studies, material mixing and T retention. Identification of migration behaviour of low and high Z materials (local transport, castellated structures and remote areas). Development of shot resolved diagnostics for erosion/deposition, tritium inventory and dust accumulation, removal methods for tritium. Qualification of diagnostic mirrors in a tokamak environment. Development of in-situ cleaning methods. Benchmarking of erosion-deposition codes. Exploration of high-Z in-situ coating techniques for fusion devices. <u>Mission 4:</u> Investigation of power exhaust in helical divertor structures in preparation of long pulse and steady-state operation in stellarators, benchmark of 3d plasma edge codes (EMC3-Eirene). <u>Mission 5:</u> Code validation for detailed quantification of atomic, molecular and PWI processes on kinetic, gyro-averaged and fully gyro resolved kinetic level (EIRENE, ERO). Contributing to numerical tokamak (and stellarator) by development of kinetic code modules and databases for neutral gas and impurity transport with standardized interfaces: focus on PWI and edge plasma physics related aspects, online A&M database interfaces (HYDKIN) and benchmarking in TEXTOR. <u>Human resource planning</u> – Education and Training programme on TEXTOR for engineers and physicists. Access to fusion facilities for students. Strong role in education in fusion within a net-work of university programmes.
	TEXTOR will address the Satellite Tokamak requirements by combined R&D in TEXTOR and JET: JET ITER-like wall, disruption mitigation valve, ELM mitigation, diagnostic refurbishments and tests, edge modelling 10 year or longer term perspective: subject to national funding
FORWARD PLANNING (subject to national boundary conditions)	Use of TEXTOR as test facility for PWI concepts and diagnostics complementary to the TEC facility Magnum- PSI and material test facilities Judith and Marion in FZ Jülich

<table>
<tr><td>FACILITITY</td><td colspan="3">TCV TOKAMAK, CRPP</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
20 M€ (device) + 20 M€ (gyrotrons and the flywheel upgrade)+ 10 M€ (diagnostics),
manpower over 20 years (incl design): approx 60 M€ (on the basis of 2007)</td></tr>
<tr><td colspan="3">COST OF UPGRADES (proposed, approximate): 1M€ (in-vessel components, see below, 2009-10), 5 M€ (3 MW Neutral beam heating, 2011-13), 4 M€ (3 MW X3 ECH upgrade, 2011-13). Manpower cost not included</td></tr>
<tr><td colspan="3">OPERATION
- average number of operational days: 80 days/year
- yearly cost of operation in 2007 euros: about 1.5 – 2 M€ expenses & 3.5 - 4 M€ manpower
- yearly manpower for operation: 35 ppy scientific (including PHD students) and 10 ppy technical support</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: approx 60</td><td>Yearly integrated equivalent full time facility users: approx ppy</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility completed in the last 10 years: 19 completed. Currently 19 PhD students are working on TCV. 13 diploma students have been awarded a masters degree (or equivalent) for work on TCV</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>8</td><td>17</td></tr>
<tr><td>2005</td><td>14</td><td>20</td></tr>
<tr><td>2006</td><td>9</td><td>30</td></tr>
<tr><td></td><td>2007</td><td>20 (as of Feb.2008)</td><td>38</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations within EU: JET, IST, FZK, HAS, IPP-Garching, IPP-CAS, CEA, CNR Milano, ENEA, Uni Warwick, UKAEA, Gøteborg, Risø,
Collaborations outside EU: Keldish, Triniti, MIT, Kurchatov,
Number of experimental contributions to ITPA : 183 experimental samples to three databases, input to diagnostics database, memberships in 5 ITPA groups
Prospects: the current trend for increasing collaborations is expected to continue</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">$B_T \le 1.52$T, Ip≤1.02MA,
pulse duration 2 s (Ohmic), up to 5s with ECCD, 3-5 pulses per hour
kappa≤2.8, -0.7 ≤delta≤0.9
6×0.5MW ECH/ECCD at 83GHz ($n_e < 4\times10^{19}m^{-3}$)
3×0.5MW ECH at 118GHz ($n_e < 10^{20}m^{-3}$)</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Ongoing upgrades (2008-2009)
<u>Digital control system</u> in collaboration with IST Lisbon to replace the hybrid controller dating back to 1992 (MIT collaboration), providing enhanced operational flexibility. Selected diagnostics will be equipped with real time signal processing capabilities for the control of advanced tokamak experiments and for highly shaped plasmas.
<u>Diagnostic upgrades</u> under implementation reflect the current focus of research: plasma turbulence (phase contrast imaging, reflectometry, correlation ECE), ECH/ECCD physics (2 hard-X diagnostics, suprathermal ECE), transport barrier physics (high resolution Thomson scattering, Faraday rotation diagnostic).
<u>Fast gas injection</u> system for impurity transport experiments and for 'killer gas pulse' experiments aimed at safely disrupting a plasma.

Minor upgrades (design studies underway)
These upgrade all (most likely) involve in-vessel components and are considered together.
<u>Alfvén Wave antennae</u> for Alfvén Eigenmode (AE) physics in shaped plasmas and controlled spatial redistribution of fast ion populations. Fast ions from proposed Neutral Beam Heating project (below) would mimic fusion alphas.
<u>Ergodisation coils</u>. Provide a spectrum of resonant magnetic perturbations at the plasma edge sufficient to control ELMs. Provide an error field correction/generation capability.
<u>Low Field Side power handling tiles</u>. TCV inner wall at LFS currently not suitable for power handling, i.e. not suitable for negative triangularity divertor plasmas. To check if L-mode confinement gain with $\delta<0$ holds for H-mode, LFS carbon tiles need upgrading to CFC. Other materials also considered.</td></tr>
</table>

	Major upgrades (heating upgrades) We are considering increasing X3 ECH (up to 3 additional MW) and acquiring an ion heating capability in the form of Neutral Beam Heating (NBH, 3 MW), or a combination of both. Additional power is needed to approach β limit in H-mode and to heat larger and higher current plasmas at high density ($n_e>4\times10^{19}m^{-3}$). For the choice of extra X3 ECH using the source developed for W7X, the maximum allowable frequency would be about 130 GHz, allowing a 25% increase in the optimum density for X3 and somewhat better equipartition. The acquisition of an ion heating capability would open up the largest domain for physics exploration, by allowing a wide range of temperature ratios Ti/Te to be explored using mixed ECH/NBH. Ti/Te is one of the fundamental parameters governing turbulent transport. Currently, with ECH alone, 0.1<Ti/Te<0.5. NBH units applying opposite torque allow rotation control and investigation of β-limiting modes (Resistive Wall Modes, RMW) at low or no rotation, like in a reactor. A major motivation is fast ion physics. Fast ions have profound effects on MHD, which may be controlled using localised ECCD (e.g. sawteeth) or more speculatively, active AE antennae (TAE's). A feasibility study for NBH is well underway. Suitable beam energies 20-40keV. Tangential injection essential for low current experiments. Near normal suitable for Ip>200kA.
PROGRAMME: ACHIEVEMENTS	• Plasma shaping and control: Unprecedented variety of new plasma shapes, often with radically different properties. • L-mode heat transport in shaped plasmas: Unique experimental data on the dependence of L-mode transport of energy, particles and impurities in a range of plasma configurations, including extreme elongation and triangularity, which is unmatched by any other device in the world. In particular it has been shown that confinement increases by a factor of about 2 as elongation is increased from 1 to 2 or as triangularity is reduced from 0.4 to -0.4. • Particle and impurity transport L-mode density profiles closely follow the predictions of Turbulent Equipartition theory, irrespective of plasma shape. Impurity pinches can be reversed, at least in electron heated L-mode plasmas. In ECH electron heated H-mode plasmas TCV results show that the favourable inverse ν_{eff} scaling of the density peaking factor (which will have a significant impact on fusion performance in ITER and DEMO), observed in AUG, JET and C-MOD, does not hold, prompting further investigations. • Momentum transport: Significant plasma rotation occurs without external torque and different rotation regimes exist with abrupt transitions from one to another. May provide natural stabilisation against RMWs. • MHD stability: Effect of shaping on the stability of the internal kink mode (sawtooth instability) conforms to theoretical predictions based on the Mercier criterion. Experiments and modelling of localised ECH and ECCD in the vicinity of the q=1 surface, show that depending on deposition, sawteeth can be stabilised or destabilised. The latter effect may be applied to avoid degradation of confinement by Neoclassical Tearing Modes (NTM) in ITER and DEMO, by avoiding magnetic seed island formation. TCV has also shown that NTM's can arise in the absence of a seed island (triggerless NTM's). • Electron Cyclotron Current Drive: TCV is the only tokamak in which steady-state discharges have been fully non-inductively driven by electron cyclotron current drive, with a record 210 kA current driven using 2.7 MW of power. • Physics of ECH and ECCD: The extremely high power densities have permitted the first test of quasilinear enhancement of the ECCD efficiency, which is reproduced correctly only when suprathermal electron transport, clearly observed experimentally, is included in the modelling. Second harmonic-accelerated suprathermals have also been shown to strongly enhance the absorption of third harmonic waves. TCV has demonstrated Electron-Bernstein wave heating at densities > second-harmonic cut-off. • Advanced tokamak physics: A unique, non-inductively sustained electron internal transport barrier (eITB) is generated in TCV by reversing the central magnetic shear with off-axis ECCD. The regime has a confinement merit factor of up to 3 with respect to ITER L-mode scaling and 6 with respect to RLW scaling and a bootstrap fraction of up to 80%, β_p up to 2.4 and durations of more than 10 current redistribution times. TCV has shown that these barriers arise exactly where and when the local magnetic shear reverses. Sustained eITB's have also been obtained with no net current drive and no

	inductive current, demonstrating the physical possibility of tokamak plasmas sustained to 100% by bootstrap current. • H-mode physics. Contributions to the ITPA threshold and confinement databases. Using the X3 system with 0.9-1.3MW, stationary type I ELMy H-modes, $\beta_N \cong 2$ and $H_H \cong 1.3$. Although type I ELMy H-modes are by no means a first as such, these TCV results are the first to show that this operating mode also exists in fully electron heated plasmas (as ITER and DEMO will be), with encouraging confinement (all other tokamaks operating in type I ELMy H-mode are NBI heated). In addition, these experiments demonstrated the existence of electron heated ELM-free stationary H-modes with same confinement as type I ELMy H-modes and no impurity accumulation. • Physics of Edge Localised Modes: Using its unique in-vessel, fast vertical stabilisation coils, TCV has been the first to demonstrate that Type III ELM frequencies can be manipulated through the application of axisymmetrical magnetic perturbations. In X3 heated H-modes ELM-free stationary H-modes were obtained. • Edge and divertor physics: The non-diffusive transport in the SOL has been unambiguously identified as driven by interchange instabilities.
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	Advancing the physics underlying the concept(s) of future fusion reactors is an essential requirement for the fusion programme and one fully embraced by TCV. **Mission I: Burning plasmas** Burning plasmas produce alpha particles, which may produce deleterious MHD instabilities. They may also partly stabilise sawteeth, leading to less frequent, but larger crashes, which may trigger Neoclassical Tearing Modes. We propose to conduct experiments to show how sawteeth, which are stabilised by fast particles, can be simultaneously destabilised by local ECCD near the q=1 surface. The fast particles would most likely be produced by the proposed NBH system. We also propose experiments for controlled redistribution of fast particles using AE excitation by external antennae, with the aim of avoiding large MHD instabilities. **Mission II: Reliable tokamak operation** The current implementation of a digital control system is aimed at providing reliable operation over a wider domain, with more flexibility than previously. Various diagnostics advances and the ergodisation coils for ELM control will contribute towards this goal. Part of the proposed work under mission I also pertains to mission II. **Mission IV: Technology and physics for long pulses and steady-state** Our continued effort on advanced tokamak (AT) physics, with new diagnostics and control systems, to understand ECCD and ITB physics is directly relevant to this mission. The gyrotrons on TCV (whether upgraded or not) are directly contributing to building up expertise for DEMO, as has already been shown by our involvement in the ITER ECH/ECCD design and collaborations. **Mission V: Prediction and performance** It has always been our deep belief that in order to test our physics understanding, models and theories must be validated of a wider parameter range than foreseen for a future application (e.g. a particular ITER scenario). TCV is ideally suited for this because of its flexibility and wide operating domain. The wide operational range is also expected to show ways to improve plasma properties using shaping, providing options for concept improvements. The proposed addition of ion heating in the form of NBH vastly augments the accessible operating domain in terms of electron/ion heating mix ratio, encompassing the ITER values and linking up to other, predominantly ion heated tokomaks. Allows studies on fast ion populations and associated instabilities, effects of rotation (and their absence), providing code validation and strategies to use in future burning plasmas. The proposed ergodisation coils address the need to understand and control ELMs and protect plasma facing components from excessive heat loads (also pertains to mission II). The host of new diagnostics is generally aimed at testing our understanding and improving predictive capabilities, especially (for current developments) in the areas of turbulence, fast electron physics and transport barriers.
FORWARD PLANNING	Near term: Experiments to focus on transport, ECH/ECCD, AT physics Ongoing upgrades: 2008-2009 Minor upgrades: 2010-2011 (may require preferential support) Heating upgrades: 2011-2013 (will require preferential support)

<table>
<tr><td>FACILITY</td><td colspan="3">Mega Amp Spherical Tokamak (MAST), UKAEA</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros 1Euro=£0.7
MAST device (Phase II costs, 12/1995): 14.2 MEuro
MAST NBI (Phase II costs 11/1997): 2.1 MEuro
COMPASS systems used for MAST (Phase II 1984): ~20 MEuro
Subsequent investments (estimated, general support) ~10 MEuro</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2007 euros
2010-2014: ~37MEuro. 1Euro=£0.7.
~2/3 of the UK funding has been identified already.</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): ~90
- yearly cost of operation in 2007 euros: ~6.5 MEuro (incl staff and overheads). Will rise to ~8MEuro after the upgrade.
- yearly manpower for operation in ppy: ~45 + 15 technicians</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, where appropriate for technology facilities)</td><td colspan="2">Number of facility users: ~105</td><td>Yearly integrated equivalent full time facility users: (ppy): ~35</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~20 since start of MAST Operations(2000)</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:
Estimated numbers.</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>20</td><td>~40</td></tr>
<tr><td>2005</td><td>21</td><td>~45</td></tr>
<tr><td>2006</td><td>20</td><td>~45</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: CEA, CRPP, DCU, ENEA, FOM, FZJ, HAS, IPP, IPP.CR, IPPLM, IST, VR, Univ Marseilles, 14 UK Universities.
Collaborations outside EU: PPPL, ORNL, MIT, GA, LLNL, Ioffe, SPbSPU, Kurchatov, MSU, U Tokyo, Kyoto U., Kyushu, INPE, IPP-Hefei, ANU Canberra,
Number of experimental contributions to ITPA (if applicable): Involved in 23 joint experiments in 2007, 26-30 in 2008. Spokesperson for 4-5
Sharing facility with other fields of research (if applicable): N/A
Prospects: generally strengthening collaborations, including new equipment loans</td></tr>
<tr><td rowspan="2">PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">(factual set of parameters (engineering as well as physics parameters) and other technical capabilities)</td></tr>
<tr><td colspan="3">I_p:1.5MA; B_T:0.52T; R: 0.85m; a: 0.65m; Volume: ~10m^3. Flat top ~ 0.5 sec.
X-point (limiter, DND, SND) $\kappa \lesssim 2.5$; $\delta \leq 0.5$
Graphite PFCs
NBI(5MW), ECRH/EBW startup system (0.4MW)
Diagnostics: ~35 distinct systems, some very substantial, generally with very high spatial resolution. Unique full plasma views.
Other: 12 ELM coils, TAE drive antennae, pellet injector, 4 error field correction coils, divertor test station (single location), biased divertor, digital control with real-time equilibrium reconstruction
Remote participation facilities.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">(if applicable: new technical capabilities after foreseen upgrade(s); including planning)
Upgrade to give high performance, steady state (~5 sec) regimes:
3 additional 2.5MW 5s neutral beams: permanent off-axis and counter NBI to add to 5MW co-NBI
1MW ECRH/EBW heating and current drive system (proof of principle)
Divertors: upper and lower cryopumped, closed
6 additional PF coil pairs with supplies for shape and divertor control
New solenoid ~double the Vs
New TF centre rod and supply to allow order of magnitude increase in I^2t
External ELM mitigation coils (on equatorial port plugs)
Continuous pellet injector
Divertor test station (full toroidal rings) – ITER-like heat flux > 10 MWm^{-2}
Tungsten coated/bulk tungsten divertor (under consideration)
Planning: could be implemented within 4 years from EURATOM approval.</td></tr>
</table>

PROGRAMME: ACHIEVEMENTS	(a summary of key scientific achievements) **Demonstration of good tokamak performance of ST at 1MA level and contributions to key ITER physics issues and databases** 1. *Major research in ELM filaments (propagation, energy content, nature vs collisionality), and other filaments (L-mode, inter-ELM)* 2. *H-mode threshold scaling - removing degeneracies, establishing sensitivity to geometry (especially near DND)* 3. *TAE and EPM observations (super-Alfvénic NBI) including first evidence of damping of antenna-driven modes, CAE modes, Alfvén cascades, hole-clump pairs etc* 4. *β, aspect ratio dependence of τ_E scaling* 5. *Pellet fuelling efficiency with shallow pellets* 6. *On- and off-axis NBCD and tests for anomalous fast ion diffusion* 7. *Start-up without central solenoid, using outer PF coils or EBW* 8. *Halo currents: comprehensive measurements, and local mitigation by changing path resistance* 9. *Neoclassical tearing modes: confirm drive terms at low A and stabilising effect of curvature* 10. *Error-field correction and locked mode threshold scaling* 11. *Disruption studies (e.g. energy quench timescales, SOL broadening)* 12. *Influence of rotation on confinement (e.g. transport barriers) and stability (e.g. sawteeth)* 13. *SOL transport and flows (mirror effects)*
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(Is the facility an "ITER" facility, paid or to be paid under ITER credit through F4E.): *NO* (how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions) ***The whole programme (and upgrade) will be explicitly designed, in conjunction with EFDA, around the 7 Missions for ITER and DEMO and a gap analysis/roadmap towards an ST CTF to meet the proposed EFDA 10-year milestone ("Completion of R&D on Spherical Tokamaks in preparation of decision making on CTF").*** Five year perspective: ***M1****: Wide range of TAE and EPM studies with $v_{fast}>v_A$: nearly isotropic fast ions, TAE antennae (damping rates), detectors up to 5MHz, strong modelling support (linear and non-linear – MISHKA, HAGIS)* ***M2****: Scenario physics with ELMs, ELM mitigation and interaction with shallow pellet fuelling. SND, DND for H-mode access and hysteresis studies. Disruption mitigation valve with imaging diagnostics.* ***M3****: ELM mitigation coils and diagnostics; ELM impact on PFCs – input to predictive model from well-diagnosed ELMs; SOL power scaling; SOL-pedestal interface (flows, L-H access); test divertor physics with 1-D and 2-D diagnostics and OSM-EIRENE modelling; optimisation of thermographic diagnostics, PFC studies including dust production and transport* ***M4****: NBCD on and off-axis, validation of numerical models* ***M5****: Develop and test ELM models including mitigation; test models of H-mode threshold and edge and internal barrier formation; NTM threshold and island transport with fast cm-resolution Thomson scattering; impurity transport in inductive scenarios with strong flow; use high resolution (ρ_i) T, n, $v_{\theta,\phi}$ diagnostics to benchmark specific features of core transport models (e.g. GS2+rotation, ORB5, TGLF, CENTORI). Modelling specific to CTF.* ***M6****: No major input* ***M7****: ST-CTF relevant non-solenoid start-up studies (EBW, different PF configurations).* ***Training****: flexible training facility for tokamak operators and physicists. Upgrade would provide project, design, engineering and procurement opportunities.* 10 year perspective (if applicable): **Major upgrade completed and exploited. Above activities greatly expanded given the very substantial increase in capability and flexibility in H&CD, pulse length, shape, fuelling, exhaust/divertor. In particular:** ***M1****: Higher beta (beyond that needed for a CTF), wider range of fast particle distributions and wider range of scenarios (H-mode, ITB, hybrid).*

	***M2**: More flexible PF system, higher power, fully-non-inductive scenarios (CTF-relevant). Operation with high frequency pellet injection (pedestal perturbations, fuelling efficiency)* ***M3**: ELM mitigation coils (including external equatorial coils) and diagnostics with modelling support; SOL power scaling at ITER-level heat fluxes at divertor (can raise to melt tungsten); divertor plasma physics including testing radiative divertor physics at increased power density; capability to test materials, novel divertor concepts using the divertor test station in a tokamak divertor with ELMs.* ***M4**: Pulse length >> current redistribution time. Enhanced current profile control. Much more flexibility to test specific ITER/DEMO scenario issues. Demonstrate physics for steady state high performance ST (Mission 7).* ***M5**: Use cw pellets to determine nature of pedestal-core interplay; counter injection to control flow and change edge* E_r *(orbit losses); impurity transport in inductive (variable flow) and fully driven scenarios (EBW core electron heating as a control technique). Test elements of an integrated model for an ST CTF.* ***M6**: No major contribution, but note the intention to have a cyanate-ester insulated central solenoid and TF rod.* ***M7**: Based on a ST-CTF gap analysis, test ST-specific physics and scenario issues for an ST CTF, including operating limits, in some cases in conjunction with other STs and conventional tokamaks. Specific studies to inform DEMO design e.g. comparison of single and double null divertors.* Longer term perspective (if applicable): *As Europe's newest tokamak MAST will be able to make leading studies, beyond the 10 year timeframe, in support of ITER/DEMO physics and ST CTF (and ST Power Plant)*
	(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>) Five year perspective(when relevant): 10 year perspective (when relevant): Longer term perspective (when relevant):
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above) *0-5 years – further exploitation of the present MAST device* *2-5 years: – implementation of a major upgrade (15 month shutdown)* *>5 years: exploitation of the upgraded MAST, implementation of remaining upgrades.*

<table>
<tr><td>FACILITY</td><td colspan="3">COMPASS, IPP.CR</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
1983 (UK), 20 million 2007 euros (?) (cca 6 mio GBP in 1983)
2007, re-installation – upgrade; 7.5 M€ (incl. new building)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
2008-2010, 3.9 M€ for completion of the re-installation/upgrade</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year: (foreseen) 100
- yearly cost of operation in 2007 euros: (foreseen) 850 k€
- yearly manpower for operation in ppy: (foreseen) 20 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY (form magnetic confinement and, when appropriate, for technology facilities)</td><td colspan="2">Number of facility users: (foreseen) 50</td><td>Yearly integrated equivalent full time facility users: (foreseen) 35</td></tr>
<tr><td colspan="3">Number of PhD/diploma theses using experimental data from the facility in the last 5-10 years: N/A
(currently, 12 PhD and 6 undergraduate students are involved in the re-installation)</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>N/A</td><td>N/A</td></tr>
<tr><td>2005</td><td>N/A</td><td>N/A</td></tr>
<tr><td>2006</td><td>3</td><td>5</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: IST, HAS, UKAEA
Collaborations outside EU: RF, Ukraine
Number of experimental contributions to ITPA (if applicable): N/A
Sharing facility with other fields (if applicable): No
Prospects: collaboration with IST, HAS, CEA, ENEA, OAW, IPPLM, etc. envisaged</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">The only tokamak in the new EU countries, with ITER-like magnetic configuration and a unique set of saddle coils;
I < 250 kA, B < 2.1 T, R = 0.56 m, a = 0.18-0.23 m, plasma volume 0.5 m^3,
X-point plasma, $\delta < 1.8$, $\kappa = 0.3–0.7$,
Wall: divertor and central column – graphite, vessel – Inconel, passive cooling
No. of basic diagnostics: 8 systems
Pulse length up to 1 s</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">NBI heating (40 keV, 2x 300 kW), LHCD (1.3 GHz, 400 kW), edge plasma diagnostics with a high spatial-temporal resolution, Resonant Magnetic Perturbation technique for ELMs mitigation</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">N/A (in 2007, new building was constructed, and the COMPASS tokamak transported from Culham, UK)</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">Is the facility an “ITER” facility, paid or to be paid under credit through F4E: No
How does the present (future/upgraded) facility address the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions)
5 year perspective:
- (Mission 2) Stable tokamak operation with emphasis on edge plasma studies
• Application of new control and data acquisition concepts
• H-mode, ELMs mitigation, pedestal physics
• Plasma wall interaction
- (Mission 4) Wave plasma Interaction
• Parasitic LH wave absorption in front of the antenna
• LH wave coupling in detached plasmas
- Practical training following the tradition of the former IPP.CR tokamak CASTOR</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning:
2008 – Installation of new power supplies, new CODAC and basic diagnostics, first plasma expected at the end of 2008
2009 – optimization of OH performance in hydrogen, installation of NBI heating, installation of advanced edge diagnostics
2010 – deuterium plasma, H-mode, edge plasma studies, ELM’s mitigation by RMP technique, installation of LHCD system
2011 - continuation of edge plasma studies, coupling and fast particle generation at LHCD</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td colspan="3">(ISTTOK, ASSOCIATION EURATOM/IST)</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: ORIGINAL: 1.75 M€ (1990/1)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2007 euros: No significant investment planned</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year: 180 days
- yearly cost of operation in 2007 euros: 175 kEuros (of which 125 k€ for manpower)
- yearly manpower for operation in ppy: 4 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: 30 (2007)</td><td>Yearly integrated equivalent full time facility users: 8 ppy (2007)</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: PhD: 6, MSc: 16; BSc: 21</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>7</td><td>8</td></tr>
<tr><td>2005</td><td>4</td><td>7</td></tr>
<tr><td>2006</td><td>4</td><td>7</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: Ciemat, IPP.CR, ENEA, Latvia, HAS, IPP Greifswald, Humboldt-Universitat Berlin, IPPLM (Poland), ÖAW
Collaborations outside EU: Brazil, Russia, Saskatchewan (Canada)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">(factual set of parameters (engineering as well as physics parameters) and other technical capabilities)
• I_p = 4-8 kA, B_T = 0.3-0.6 T, R = 46 cm, a = 8.5 cm, V_{plasma} = 0.065 m^3, circular cross-section, stainless steel vessel; graphite limiters; 15 diagnostic systems</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">• Real-time control of the plasma current and density</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">• Control and characterization of the edge turbulence using electrode bias
• Achievement of long duration alternating current discharges (10 cycles / 250 ms)
• Successful operation with a gallium jet limiter
• Development of several diagnostics (Bolometer Tomography, Heavy Ion Beam, Retarding Field Energy Analyser, Mechanical Force Sensor)
• Real-time control of the plasma position</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">(Is the facility an “ITER” facility, paid or to be paid under ITER credit through F4E.)
Five year perspective:
• Test of new control and data acquisition solution
• Development of new fusion relevant plasma facing materials</td></tr>
<tr><td colspan="3">(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>)
Five year perspective:
• Test of new control and data acquisition solutions</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning (if not already adequately addressed above):
• Cooperative use of the Tokamak ISTTOK in the framework of the IAEA project “Joint Research Using Small Tokamaks”
• Provide necessary facilities for education of students, scientific activities of post-graduate students and for the training of personnel for large tokamaks</td></tr>
</table>

FACILITY	**FAST (Fusion Advanced Studies Torus) is the proposal for a European ITER Satellite Tokamak by the Italian Association, open to contributions of other EU Associations.**
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: ~280 M€
	OPERATION Average number of operation days/years (last 5y) = **NA** yearly cost of operation (in 2007 €) = **NA** yearly manpower for operation **~ 150 ppy**
USE OF FACILITY	The Facility will start operations few years before ITER
COLLABORATIONS	This facility is aimed at the broadest possible involvement of the EU Associations, which is the prerequisite for its realization and exploitation. The exploitation of this facility could be attractive for all the ITER Partners.
PRESENT TECHNICAL CAPABILITIES	NA

FUTURE TECHNICAL CAPABILITIES

	H-mode Ref-a	H-mode Ref-b	H-mode "Perf"	Hybrid	AT
I_p (MA)/q_{95}	6.5/3	6.5/3	7.5/2.8	5/4	3/5
B_T (T)	7.5	7.5	8.0	7.5	6
H_{98}	1	1	1	1.3	1.5
$<n_{20}>$ (m^{-3})	2	4	4	3	1.3
n/n_{GW}	0.4	0.8	0.7	0.8	0.5
$P_{th,LH}$ (MW)	14-18	20-28	22-29	18-23	8.5-12
β_N	1.3	1.7	1.6	2.0	2.0
$t_{flat-top}$ (s)	13	13	6	15	60
τ_{res} (s)	5.5	3	4	2.8	2.5
τ_E (s)	0.43	0.57	0.67	0.52	0.25
T_0 (keV)	13.0	8.5	9.0	8.5	15
f_{rad} (%)	27 (39)	18 (75)	18 (75)	20 (55)	63
Z_{eff}	1.06 (1.55)	1.0 (1.2)	1.0 (1.2)	1.0 (1.3)	1.35
Q	0.65	1.2	1.8	0.9	0.18
$t_{Discharge}$ (s)	20	20	14	20	70
I_{NI}/I_p (%)	15	20	18	30	60

FUTURE PROGRAMME: ACHIEVEMENTS	See the Previous table. FAST is a High Plasma Current and Magnetic Field Tokamak, proposed as Satellite Tokamak meant to operate in parallel to ITER and aimed at the early development of ITER operation scenarios. FAST will operate with D plasmas in a range of parameters close to that of ITER, thus avoiding the complexity of operating and maintaining a D-T facility. Consequently, its flexibility will give major contributions to most of the problems that ITER will meet in a time and cost effective way. In particular, to the understanding of Fast Particle and charged Fusion Product behaviors, of very large Wall Loading and of Plasma Operations in ITER relevant conditions. Moreover, FAST will be capable of investigating advanced alternative concepts for the divertor plates relevant for DEMO and CTF, based e.g. on liquid Lithium, and of giving contributions to JT-60 SA on any Physics aspects of the Advanced Tokamak Scenarios.
PROGRAMME: ADDRESSING THE PROGRAMME	Five years perspective: Not Applicable

NEEDS	
Contribute to the R&D Facilities on three Different Time Perspective	Fifteen years perspective: It is assumed that the Facility will Start Operations around 2016. During the construction phase and from the Technological point of view, the facility construction will give important information related with general aspects of the W armored Plasma Facing Components fabrication as well as specific Remote Handling issues of actively cooled components (Missions 2-3). During the first years of operation, it will be able to address several different aspects of the strategic Missions. Fast Particle physics issues will be addressed and responses will be available for anticipating and predicting ITER burning plasma operations. In FAST, ICRH will generate well confined energetic ions above 0.5 MeV, producing dominant electron heating in the 40-90% range of the total, depending on plasma parameters (Mission 1). General aspects of ITER Relevant Plasma Operations (Plasma Control, ELM handling, Toroidal Ripple issues, Plasma Break-Down assisted by ECRH, Coupling of large Power in Presence of strong ELMs…) will be investigated and useful input for the initial phase of ITER operation will be produced (Mission 2). ITER relevant P/R wall loads (~ 22 MW/m) will be reached, developing the capability of handling large ratios between the heating power and the device dimensions at high density and low collisionality, thus allowing the investigation of the physics of large heat loads on divertor plates as well as the production and control of ITER and DEMO relevant ELMs (Mission 3); Advanced Tokamak scenarios will be investigated with pulse lengths of about 30 times the current resistive diffusion time, making it possible to compare experimental results compared with those of JT-60 SA. (Mission 4) Mutual positive feedbacks between theory, numerical simulations and experimental results from FAST will promote Verification and Validation of numerical simulation codes, which are an essential ingredient for reliable extrapolations to predicting fusion performance and ITER burning plasma operations. (Mission 5)
	Long term perspective: On the longer time scale, this Facility will complete the Missions described above for the fifteen years perspective and address any other relevant matters, which will eventually arise during its prior operations, thanks to the flexibility of its design both in terms of operation scenarios as well as additional heating power sources. Moreover, the use of liquid Lithium as Divertor material will give important information about the possibility of relying on advanced and effective solutions for DEMO design and construction, which can possibly be extended to CTF that certainly requires divertor plates with ultimate capability.
Role As Satellite Tokamak	FAST has been conceived to play the role of European ITER Satellite Tokamak, addressing most of the physics issues identified by programmatic strategic missions. JT-60 SA has been planned and designed with this same perspective. Naturally and by definition, one single ITER satellite experiment cannot exhaustively fulfil all strategic missions and programmatic needs. FAST has been designed to address crucial issues of burning plasma operations, while remaining fully complementary to the already approved JT-60 SA. In fact FAST has: ITER relevant geometry (same shape of magnetic surfaces and divertor configuration); ratio between energy confinement time and electron-ion equipartition time similar to that of ITER; capability of producing and confining energetic ions in the half-MeV range in order to produce dominant electron heating, in the range 40-90% (taking into account that fusion alphas in ITER will deliver ~70% of their energy to electrons); ITER-like fast ion induced collective effects and cross space/time scale couplings of meso- and micro-scale turbulence; large ratio between the heating power and the device dimensions at high density and low collisionality. The need of having a sufficiently flexible facility, integrating most of the major areas of programmatic interest identified by the strategic missions, motivates the choice of maintaining in FAST the capability of investigating AT scenarios, characterizing JT60-SA operations. Thus, FAST candidates itself as a European ITER Satellite with complementary scientific program with respect to JT60-SA but still with the possibility of investigating long

pulse operations, generating synergies between these two experiments. For this purpose, FAST will be able to study Advanced Tokamak regimes on a time scale about 20÷30 times longer than the plasma resistive time. Moreover, High Beta Scenarios are foreseen, where internal active coils will control the Resistive Wall Modes problem.

From the operational point of view, FAST will operate in a range of parameters close, to those of ITER. Consequently, its flexibility will allow giving a major contribution to general problems that ITER will encounter. Few examples are given hereafter on some problems, which are already foreseen.

The presence of ECRH will allow significant contributions to solving general problems connected with obtaining a reliable and robust ECRH assisted Break-Down.

FAST will operate with large ELMs at high density and low collisionality, which will release to the divertor plates an amount of energy comparable with that foreseen for ITER; consequently, FAST will test the use of different tools to keep ELM activity under control. For instance, active coils are foreseen in FAST with the aim of ergodising the Plasma Edge magnetic field.

Advanced Plasma Control Tools are designed to have an integrated control of arbitrary Plasma shapes in FAST. In particular, the Shaping Control will be integrated with the Kinetic quantities and with MHD activity (ELMs, RWM, NTM…) control systems.

FAST will work with a P/R value comparable to that of ITER (up to ~ 22MW/m); consequently, great attention has been given to Plasma Wall interaction aspects. The possibility to remotely change divertor plates will allow studying the role played by different materials subject to very large power fluxes. In particular, very high Z pure Tungsten solution will be compared with the use of liquid lithium.

The use of ICRH in the range of Plasma Parameters that are peculiar to burning plasma operations will allow investigating FAST particle behaviours in regimes consistent with those producing a noticeable amount of Alpha Particles. Not only the physics of general instabilities excited by fusion Alphas will be studied; various Plasma Scenarios will be investigated as well, with emphasis on their compatibility with the onset of these instabilities and on the spatio-temporal cross-scale couplings of meso- and micro-scale plasma turbulences, affecting the plasma fusion performance on long time scales.

The coupling of significant ICRH power is a very challenging goal. However, the design of FAST ICRH system will integrate from the very beginning possible solutions of decoupling problems (optimization of Faraday shield shape minimizing local electric field, phase and frequency real time control to minimize antennae cross talk, etc.).

FAST is also drawn up for hosting, in a future phase, a substantial amount of NNBI power. Upgrades and/or optimizations of ECRH and LH systems are also possible. All these facts are part of the intrinsic flexibility of the FAST conceptual design, which is open to contributions from other Associations and is aimed at providing significant answers to the broadest possible range of programmatic issues, formulated by the strategic missions.

It is worthwhile emphasizing that FAST has been conceived to investigate a comprehensive set of the strategic missions in a compact and cost effective facility.

FACILITY	WENDELSTEIN 7-X (W7-X), IPP GREIFSWALD
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: W7-X basic device (1997 – 2019): 346 Mio. Euro W7-X diagnostics (1997 – 2019): 26 Mio. Euro
	COST OF FORESEEN UPGRADES: Stage II of W7-X-heating (NBI after 2019) 38 Mio. Euro
	OPERATION - average number of operation days/year (over the past 4 or 5 years): n/a - yearly cost of operation in 2007 euros: n/a - foreseen yearly manpower for operation in ppy: 120
USE OF FACILITY	Foreseen number of facility users: IPP 140 + external users
COLLABORATIONS	Collaborations inside EU: CEA Saclay & Cadarche, CIEMAT, CRPP Lausanne, ENEA Frascati & Milano, FZ-Jülich, FZ-Karlsruhe, IPHT Jena, IPPLM Warsaw, IST/CFN Lisbon, KFKI-RMKI Budapest, PTB Braunschweig, UKAEA Culham Universities of Cork, Ljubljana, Opole, Rostock, Stuttgart, Szczecin (Techn. & Maritime Universities), Vienna Collaborations outside EU: Australia: Australian National University Canberra Japan: Kyoto University, National Institute for Fusion Science (NIFS) Russia: A.F. Ioffe Physico-Technical Institute of the Russian Academy of Sciences, Budker Institute of Nuclear Physics Novosibirsk, Efremov Institute St. Petersburg, Institute of Applied Physics (IAP) Nizhnynovgorod, Kurchatov Institute Moscow, Technical University St. Petersburg Ukraine: Institute for Nuclear Research Kiev, Kharkov Institute of Physics and Technology US: Oak Ridge National Laboratory (ORNL), Princeton Plasma Physics Laboratory (PPPL), University of Wisconsin Madison Number of experimental contributions to ITPA (if applicable): n/a Sharing facility with other fields of research (if applicable): n/a Prospects: The present collaborations are dominated by the technical support for the construction and assembly of W7-X including heating systems and diagnostics. New collaborations with Polish institutions are in being implemented at present: Assembly personnel and error field correction coils (trim coils) from the Institute of Nuclear Physics (INP), Polish Academy of Sciencies in Cracow, design and construction of NBI components from the Soltan Institute for Nuclear Studies (Soltan INJ) in Swierk, and NBI cryo-pumps from the Wroclaw University of Technology. For the operation of W7-X a scheme similar to the one established for the participation of EU Associations in the ASDEX Upgrade programme is envisaged.
PRESENT TECHNICAL CAPABILITIES	n/a
FUTURE TECHNICAL CAPABILITIES	B = 3T, R = 5.5m, <a> = 0.53m, V = $30 m^{-3}$, ι = 5/6 – 5/4, 5-fold symmetry (70 superconducting coils (modular arrangement), 50 non-planar, 20 planar coils Discharge duration: 30 minutes, initially limited to ~5sec at 10 MW Plasma facing materials: carbon Heating systems at start of operation (including foreseen upgrades): 10 MW ECRH (steady state), 10(20) MW NBI (10sec), 50kW(3MW) ICRH (10sec; initially only for vessel conditioning) Diagnostics at start of operation (including foreseen upgrades): 30(50)
PROGRAMME: ACHIEVEMENTS	n/a

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The stellarator is an alternative magnetic confinement concept, with the specific advantages of an intrinsically steady state magnetic field, which is generated predominantly by external currents. For the development of a steady state fusion device complementary to the tokamak, stellarators would not only make external current drive obsolete, thus reducing the recirculating power, but also avoid current driven instabilities and disruptions. Moreover, stellarators have no density limit and may thus be operated at much higher densities than tokamaks, resulting in lower alpha-particle pressure and reduced drive for fast-ion instabilities. However, the 3D magnetic field geometry needs an elaborate optimization to guarantee confinement properties which meet the basic requirements of a fusion reactor plasma. While the tokamak has already shown these basic properties, culminating in the design of ITER, the stellarator still has to demonstrate that in particular fast particle confinement, a feasible divertor concept and impurity control are compatible with a 3D magnetic field configuration.

The optimisation basically led to three lines capable of confinement of trapped-particle orbits: Averaged drift in helical direction (quasi-helical symmetry), averaged drift in axial direction (quasi-axis-symmetry) and averaged drift in poloidal direction (quasi-isodynamicity, generalizing isodynamic systems - characterized by absence of

neoclassical transport - once studied by Donato Palumbo, later director of the EU fusion programme). Wendelstein 7-X is designed to achieve quasi-isodynamicity at finite plasma pressure. The task of W7-X is to demonstrate the basic reactor suitability of this concept. Some of the properties have already been tested by Wendelstein 7-AS (W7-AS), the predecessor of W7-X, which was partially optimised in its magnetic field characteristics.

The optimization criteria underlying the W7-X design aim to combine

- magnetic field equilibrium properties which are largely independent of the normalized plasma pressure (β) and sufficient MHD stability in order to achieve high β ($<\beta> \approx 5\%$).
- good confinement properties (for a stellarator this means low neoclassical fluxes at low collisionality / long mean free path) also for fast particles (this is essential to transfer the energy of the fusion α-particles to the plasma and to avoid undue fluxes of α's to the wall).
- low plasma currents, and here in particular low bootstrap current (and therefore an equilibrium largely independent of β), and a low Shafranov shift for a stiff plasma edge configuration which is a prerequisite to realize plasma exhaust based on the magnetic island divertor.

Compared to other stellarators in operation or under construction W7-X is the most integrated approach combining reactor relevant confinement, equilibrium and stability properties, a feasible (and in W7-AS already tested) exhaust concept, full steady state technology and a plasma volume well within the range of today's larger tokamaks. Other major stellarator devices are the Large Helical Device (LHD), operated since 1998 at NIFS in Japan, and the National Compact Stellarator Experiment (NCSX) under construction in Princeton, US. LHD is a so-called heliotron / torsatron with a plasma volume similar to W7-X and superconducting coils but with the disadvantages of a non-optimized classical concept. NCSX is a quasi-axis-symmetric system which relies on a large (toroidal) bootstrap current and in a way is a hybrid between a tokamak and stellarator. As a short-pulse device technological contributions to steady-state operation are not on its agenda. In any case, these stellarators and their missions complement each other in developing the stellarator to a viable alternative to the tokamak.

Besides the validation of the basic optimization criteria, the main goal of W7-X is to develop integrated steady state plasma scenarios with high plasma pressure and confinement time (i.e. high triple product $nT\tau_E$) which can be extrapolated to a burning fusion plasma. Regarding the seven R&D missions outlined in Chapter 4, W7-X will, in a complementary way, make major contributions to the main line of the research – the further development of the tokamak.

Burning plasmas (mission 1) require first of all α-particle confinement which is one of the optimization criteria of W7-X. Although tritium will not be used in W7-X, as the size and the magnetic field strength of the device would not produce large α populations, with the help of auxiliary heating fast particles can be generated to study the important effect of energetic particle modes in 3D geometry and possibly synergies with 3D-orbit losses.

In W7-X the question of ***first wall materials & compatibility with ITER/DEMO (mission 3)*** is directly related to the possibility to run 30 minutes high heat flux discharges. Since erosion and re-deposition are in general not equilibrium processes such long discharge durations are necessary to develop integrated plasma scenarios which demonstrate the control of heat and particle fluxes, and impurity sources and sinks. Since stellarators have often shown impurity accumulation, W7-X will start operation with high and medium heat flux wall elements covered with the low-Z material carbon. Only at a later stage high-Z wall materials such as tungsten might be considered.

A major contribution of W7-X is in the area of ***technology and physics of long pulse and steady state (mission 4)***. The development of super-conducting tokamaks started in Russia. While in Asia (with LHD, EAST, SST-1, KSTAR and JT60-SA) five super-conducting experiments are either operating or under construction, in Europe W7-X is the only super-conducting device next to Tore Supra (which went into operation in 1988) and ITER (which is an international project). Besides the intrinsic advantage of a steady state, disruption-free magnetic confinement device, W7-X has made and will make significant contributions to the technology and physics of long pulse and steady state:

- Already now the problems with high voltage insulation of the W7-X coils led to the decision to Paschen test also the ITER coils.
- In continuation of the Tore Supra development and similar to the technique of the ITER beryllium wall elements, actively cooled divertor targets which can withstand 10 MW/m^2 are development for W7-X. As it turned out, the technology is extremely complex and constant long-term quality of the targets is difficult to achieve.
- The main initial heating of W7-X is provided by 10 MW stationary electron cyclotron resonance heating. Along with the successful development of 140 GHz (2.5 – 3 T) gyrotrons, the optical transmission line for 10 MW has been developed and already partially tested which would fulfil the ITER-ECRH transfer requirements.
- Diagnostics, developed for short-pulse devices, now have to be operated under steady state conditions. New developments become necessary which range from the handling of low heat flux levels at the diagnostic-plasma interface, which add up to significant energy levels, to the avoidance of small signal drifts, which is not a problem in short-pulse devices.
- For steady state operation new control concepts have to be developed. Although the plasma control requirements are less demanding in stellarators, W7-X still needs major advances in device and plasma control, data acquisition and real time data analysis.
- Compared to present day tokamaks and stellarators, W7-X will be one of the first fusion experiments which will address high density, high power plasmas with discharge durations of up to 30 minutes. In contrast to tokamaks two reasons make high density operation much easier to achieve. Firstly, the Greenwald density limit has not been observed in

stellarators and, secondly, without the need of current drive (the efficiency of which drops with the density squared), steady state operation is not limited to lower densities. Operating well beyond the Greenwald density allows access to density regions which, in fact, are envisaged for DEMO or divertor plasmas in reactor relevant regimes.

- Long discharges will be particularly valuable to study plasma wall interaction issues and impurity behaviour. For W7-X the main question is to find a steady state operating regime which combines good neoclassical confinement, high fusion product and tolerable impurity concentrations. A possible starting point could be the HDH-mode, discovered in W7-AS, which at high density showed improved energy confinement while the impurity confinement was reduced.

When establishing the predictive capability for the fusion performance (***predicting fusion performance, mission 5***) in stellarators the 3D magnetic field has to be taken into account. As a unified confinement scaling for stellarators has not been found yet, an important issue both for first principle theory and experiment is the dependence of the turbulent transport on the 3D magnetic field configuration. For this task W7-X is well positioned, as it is equipped with comprehensive heating system (which would also allow to study current drive) and high configurational flexibility. The long-term aim is to achieve a first principle understanding of the plasma transport in tokamaks and stellarators so that for the development of a fusion power plant a joint knowledge base can be established. In this context IPP is in the unique situation to accommodate both device types (ASDEX Upgrade and W7-X) and theory groups which cover both research lines.

Based on the three main stellarator lines (LHD, NCSX, W7-X) corresponding stellarator reactor studies have been conducted (***DEMO Integrated Design: towards high availability and efficient electricity production, mission 7***). The results from W7-X, as the most integrated and comprehensive approach to realize stellarator reactor relevant plasma scenarios, will be crucial to establish the stellarator as an alternative to the tokamak fusion reactor. This (i) reduces the risk developing a fusion reactor, should e.g. satisfactory solutions for the current drive and stability problems of the tokamak not be found, (ii) offers an alternative to the tokamak fusion reactor with differing and possibly favourable properties, such as steady state operation at lower recirculating power i.e higher efficiency.

FORWARD PLANNING

The present planning foresees a completion of the device assembly in 2014, with plasma operation starting in 2015. During the first two years of operation W7-X will be only equipped with inertially cooled in vessel components and a test divertor, limiting the discharge duration at 8 MW heating power to about 5 sec. Conversely, stationary plasmas will be restricted to about 1 MW.

After verifying the vacuum field configuration and establishing first plasmas, the first part of the experimental programme will start to explore the optimization criteria, such as the neoclassical confinement and the residual bootstrap current, at moderate densities (below the ECR X2 cut-off $n = 1.2 \times 10^{20}\ m^{-3}$). The investigation of the island divertor transport and topology for different magnetic field configurations will also be used, if necessary, to adapt the actively cooled divertor elements, already being built and to be installed at a later stage, to the actual plasma conditions. If the impurity content can be kept low and the heat load on the divertor is acceptable, first long-pulse, low-power experiments will be performed to demonstrate the integral technical capability of all involved systems. The next step in the programme would be the start of the development of plasma regimes for high power, steady state operation. This includes ECR current drive if necessary for configuration control, and divertor operation with high recycling and high edge densities with a controlled island edge structure satisfying the necessary symmetry of the divertor load. For high densities (above the X2 cut-off) this would also mean to establish ECR heating and current drive in O2 polarization, which up to now has not been routinely used. Together with neutral beam injection (NBI), first studies of the fast particle confinement at moderate β could be carried out.

After these first two years of operation, the actively cooled wall components, the actively cooled divertor and the corresponding periphery will be installed in a one-to-two-year shut-down. Additional extensions are the installation of a cryo-pumps, the upgrade of the heating power supplies, more diagnostics, and control and data acquisition for steady state operation, and the installation of 3 MW ion-cyclotron resonance heating. The subsequent experimental programme will comprise the verification of all optimization criteria, including fast particle confinement at high β and the β -limit (at 5% or above), and the development of steady state discharges with relevant fusion product and divertor conditions which can be extrapolated to a reactor scale plasma.

Foreseen upgrades of W7-X at a later stage are the increase of the heating power (10 MW NBI to 20 MW NBI). Other conceivable measures are the upgrade of the NBI from 10 sec pulses to steady state. Depending on the impurity behaviour in steady state a gradual modification of the divertor and first wall elements to more reactor relevant plasma facing materials could be also considered.

<table>
<tr><td>FACILITY</td><td colspan="3">TJ-II CIEMAT</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: 40 M€ (60 M€ IN EUROS 2007)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: 4 M€ in FP7 (Heating, Divertor, diagnostics)</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years):55
- yearly cost of operation in 2007 euros: 2.6 M€
- yearly manpower for operation in ppy: 20</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td>Number of facility users: 70</td><td colspan="2">Yearly integrated equivalent full time facility users: 41</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5 years:10/15</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>37</td><td>50</td></tr>
<tr><td>2005</td><td>41</td><td>76</td></tr>
<tr><td>2006</td><td>32</td><td>41</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">- Collaborations inside EU: JET, IPP-Greifswald/Garching, IPF (Stuttgart), IST, HAS, CULHAM, CEA, IPP-CZ, FOM, ERM/KMS ,OAW. More than 15 Spanish R&D institutions/universities.
- Collaborations outside EU: Russia (IOFAN, Kurchatov Institute, IOFFE), US (ORNL, PPPL,UCSD), Ukraine (KIPT), Japan (Kyoto University, NIFS), Australia (ANU)
- Number of experimental contributions to ITPA (if applicable) :
- Sharing facility with other fields of research: Atomic Physics, Turbulent phenomena, pattern recognition...
- Prospects: increase links to stellarator groups, JET and EU task forces.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">(factual set of parameters (engineering as well as physics parameters) and other technical capabilities) B 1T, R 1,5m, r 0,2m , bean shaped plasma, 4 periods, ECH 2x300kW, NBI 2x1 MW. Diagnostics: ~40 (incl. 180 point TS and HIBP system). Metal wall Li/B cover.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">(if applicable: new technical capabilities after foreseen upgrade(s); including planning)
EBW: proof of principle 2008 (200 kW), upgrade to 1 MW 2010.
Divertor: 2010. NBI under study</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">TJ-II main objective is in the area of concept improvement (mission 4), but in addition, its research programme has exploited synergies with the tokamak wherever meaningful (missions 2 3,5). The main scientific achievements are the summarized below:

Global stellarator confinement studies. TJ-II has contributed significantly to establish the positive dependence of energy confinement on rotational transform of the global energy confinement time scaling for stellarators (ISS04). (M4)

Confinement and magnetic topology. Magnetic configuration scan experiments have highlighted the interplay between magnetic topology (rationals, magnetic shear) transport and electric fields. (M4, also M2: rationals vs transport barriers in tokamaks)

L-H transition physics. TJ-II has shown the possibility of spontaneous and biasing-induced improved confinement regimes in a device designed for high beta operation but not optimized for neoclassical transport (high neoclassical viscosity) (M4, M2).

Rotation physics. Perpendicular core rotation is strongly coupled to plasma density in consistency with neoclassical expectations. Edge sheared flows are strongly coupled to plasma turbulence, consistent with expectations for turbulence-driven flows. First evidence of parallel flows driven by turbulence have been reported (M4, M2)

Turbulence studies. First evidence of predicted tilt of convective cells by shear flows using 2-D imaging techniques has been reported, providing a critical test for the basic prediction of the shear decorrelation model (M4, M2)

Plasma-wall. Hydrocarbon fuelling experiments in configurations with a low order rationals in the edge have shown the impurity screening properties related to the expected divertor effect. The local injection of hydrocarbons has opened the possibility of carbon transport studies relevant to co-deposit formation in fusion devices. (M4, M3)

Techniques for T removal or inhibition were also developed in laboratory experiments. The mechanism of film inhibition by scavengers has been addressed in laboratory experiments in cooperation with other devices. (M3,M6)

Training & education: starting from TJ-II staff and including newly trained experts CIEMAT has established a group of >40 experts to work on JET (TFD leader/dep leader,</td></tr>
</table>

	ECE, Langmuir probes & camera enhancements), IFMIF (accelerator & test system), JT60-SA (cryostat), ITER (diagnostics, TBM, divertor, NBI, CODAC), EFDA-TFs (Transport TG leader) and basic plasma computer simulation (EGEE, EUFORIA projects). **Link to industry**: Spanish companies won Art7 contracts for 30 M€ in FP6 (JET & ITER), 80% was related to companies which started business in Fusion working for TJ-II
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	TJ-II will continue with our long standing tradition to extend our physics studies to different confinement concepts looking for common clues as a fundamental way to investigate basic properties of magnetic confinement beyond any particular concept. TJ-II research programme will focus (with 10 years perspective) in the following areas. **Stellarator physics and concept improvement: (M4)** The development of stellarator working groups (Confinement database and profile Database working groups and divertor physics) will continue as an active area of research. TJ-II related research programme will contribute for the design/development of new stellarators, addressing questions like importance of optimization for ballooning stability, magnetic shear, neoclassical viscosity, divertor in flexible configurations). Exploring regimes of high beta plasmas will keep our research programme to increase heating capabilities (ECRH, EBW, NBI). **Transport studies (M4, M2):** Rationals, magnetic shear and transport. Investigation of the mechanisms linking magnetic topology with the development of transport barriers. Physics of momentum transport. Studies of neoclassical and turbulent driven flows. Investigations of the influence of magnetic topology on zonal flow physics will be addressed. Physics of edge transitions. The flexibility of TJ-II to trigger smooth edge transition makes it a unique plasma physics laboratory to study the interplay between electric fields, profiles and fluctuations and critical test of transition models. Fast particle transport, MHD and turbulence. The flexibility of TJ-II to explore configurations with reduced magnetic well should allow the investigation of the influence of instabilities on fast particle transport. **Plasma Wall Activities (M4, M3, M6):** The main undertaking of plasma-wall issues in TJ-II will be the systematic investigation of plasma-wall conditions (e.g.full lithiumization of the machine) on confinement and plasma control. Respect to ITER-oriented research, further activity will be aimed at the problem of T inventory under carbon scenarios. **Theory and modelling (M4, M5):** Studies on the statistical description of transport processes in fusion plasmas as a complementary approach to the traditional description of transport based on effective transport coefficients. Theoretical studies of plasma heating scenarios will be focussed on EBW studies and modelling of kinetic effects. Computation developments on Grid Computing for fusion will be also addressed (e.g. Porting of VMEC and DKES to the Grid). Moreover, the ab-initio simulations in stellarators using the gyrokinetic code TORB is underway. Stellarator equilibrium studies including rational values of rotational transform will be addressed. **Training and education:** The Association will continue its participation (making used of the TJ-II facility) in educational activities, dissemination of information about fusion to the general public and education of young researchers. In this line it is worth to mention the project: Zivis, a project of volunteer computation for stellarator (TJ-II) physics based on BOINC , with the participation of more than 5000 persons.

<table>
<tr><td>FACILITY</td><td colspan="3">RFX-mod, Consorzio RFX, EURATOM-ENEA Association</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT: 60 M€ in 1985-1991
INVESTMENT FOR UPGRADES: 11 M€ in 2001-2004</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: since the device has been recently upgraded, no further upgrades are foreseen in the next 3 to 5 years; later on, possible upgrades may be proposed on the basis of achievements.</td></tr>
<tr><td colspan="3">OPERATION: Average number of operation days/year (2005-2007) : 155 (this includes runs for commissioning)
yearly cost of operation (in 2007 €) : 2.5 M€ (including manpower for operation)
yearly manpower for operation ~ 25 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of Facility users: ~50</td><td>Yearly integrated equivalent full time facility users: ~35</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~5/y PhD, ~7/y Master, ~10/y Diploma.</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility: 40 to 50</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2005</td><td>23</td><td>21</td></tr>
<tr><td>2006</td><td>15</td><td>31</td></tr>
<tr><td>2007</td><td>23</td><td>21</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">The experimental program is based on a call for proposals open to EU and international laboratories. External scientists are performing experiment on RFX-mod.
Inside EU: RIT Stockholm (MHD control, turbulence), IPP Garching (MHD stability and control experiments), IPP Prague (turbulence), UCC Cork (polarimetry), VR Goeteborg (RWM control).
Outside EU: University of Wisconsin, Madison (RFP Physics), AIST Tsukuba (density control, turbulence), PPPL Princeton (Codes, edge measurements), MIT Boston (edge measurements), GA San Diego-DIII-D (MHD control), Kyoto Institute of Technology (RFP physics)
Sharing facility with other fields: astrophysics, statistical mechanics, control engineering, industrial applications of plasmas.
Prospects: collaboration with tokamak groups has been recently enforced and will be strongly expanded, particularly on: active control of MHD instabilities, RWM stabilization, density limit, transport and turbulence modelling. Collaborations with Stellarators will be enhanced as well, in particular on density limit, transport and MHD in helical structures, turbulence.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">I_P=2.0 MA, B_T=0.7 T, t=0.5s, R=2.0 m, a=0.5 m, plasma volume=10 m^3, plasma-facing material: graphite 100%, heating: ohmic only, available flux swing 12 Vs.
Main diagnostic systems: multichord interferometry, multipoint core and edge Thomson Scattering, high-resolution SXR and bolometric tomography, 6-chord polarimetry, spectroscopy, active spectroscopy with neutral beam injector, reflectometry, 600 in-vessel magnetic, Langmuir and calorimetric probes, movable edge probes, multicolor SXR for Te profile.
Other key features: system of 192 saddle coils independently driven and covering the whole plasma boundary, fed by 192 fast, independent amplifiers for feedback control of the radial field. Full multivariable feedback capabilities with flexible digital controllers.</td></tr>
<tr><td>PROGRAMME ACHIEVEMENTS</td><td colspan="3">During the last 10 years RFP research and RFX in particular has made significant advances in understanding and improving confinement and in controlling plasma stability. The device has been modified in 2001-2004, the main upgrades being the replacement of the thick shell (τ_{shell}=500 ms) with a thinner one, whose time constant (50 ms) is shorter than pulse duration, and a system of 192 feedback-controlled coils. This is allowing strong progress in plasma parameters (also by producing a more axisymmetric magnetic boundary) and in the capability of contributing to the world-wide effort on MHD mode control.
The key scientific achievements of RFX-mod are summarized in the following.
Full complete stabilization of multiple resistive wall modes (RWM) by means of active coils has been demonstrated in RFX-mod for many RWM growth times. Experiments to force RWM rotation, and on error field amplification are being performed.
First demonstration of tearing mode control and PWI mitigation by active coils. A crucial improvement upon the intelligent shell concept came from de-aliasing the fluctuation measurements from the high periodicity sidebands</td></tr>
</table>

<table>
<tr><td></td><td>produced by the control coils. This improvement produces the spin-up of the TMs up to 100Hz and, as a consequence, phase locking is considerably mitigated and wall-locking avoided.
Operation at plasma current > 1 MA with confinement improvement. RFX-mod has set the world record for an RFP both in plasma current, with 1.5 MA, and in pulse duration with 0.5 s. Electron temperature has been raised to the 1 keV range and is observed to linearly increase with plasma current with no degradation of beta in the explored current range.
Single Helicity regimes with global improvement of plasma confinement. RFX was the first experiment to recognize the importance of Single Helicity (SH) regimes, which establish a clear and promising route toward chaos-free RFP plasmas. RFX-mod has also demonstrated the existence of SH states where the growth of the helical structure is accompanied by a synergic decrease of magnetic turbulence throughout the plasma and a global reduction of transport is found, in agreement with theoretical predictions
Stimulated SH enhanced confinement scenario by Oscillating Poloidal Current Drive (OPCD). The possibility of extending current profile control to a quasi-stationary regime has been proven with OPCD, where the best values of confinement have been achieved.
m=0 magnetic islands predicted by numerical simulations were found to be a key ingredient to the transport barrier existing in RFPs close to the reversal surface.
Exploration of density limit. The RFP has the same Greenwald density limit as the tokamak. Different from the tokamak, it is not disruptive. Toroidally localized strong radiation belts appear at high density, associated to m=0 magnetic islands. Smaller particle diffusion is found in the region with safety factor q<0 when these m=0 radiative structures are present, which hints for a relationship between them and the density limit. These RFP results set stringent conditions on the explanation of the Greenwald limit, and may lead to discard those that would be only tokamak relevant.
Understanding turbulence and transport. Key contributions on the interplay between sheared E × B mean flows and turbulence have been provided and common features are observed in tokamak, stellarator and RFP. A universal relationship between the small scales of plasma turbulence and the large scales of the plasma mean flow is observed.
Momentum transport studies. Electrostatic fluctuations rule the momentum balance equation representing the main driving term for sheared flows counterbalancing anomalous viscous damping. Energy transfer from turbulence to mean flow is also addressed
Multi-coils feedback control and innovative models for fusion devices. A full electromagnetic model of the active control system in the presence of passive conducting structures has been developed and mode controllers based on the physics of Tearing Modes have been designed.
Many innovative diagnostic systems have been developed for RFX-mod. This experience has allowed to give key contributions to European fusion experiments, and in particular to JET (where RFX has led the development of the High Resolution Thomson Scattering, magnetic probes and halo current sensors).</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td>On a 5 years perspective, RFX-mod can give the following contributions to the missions.
<u>MISSION 2 (RELIABLE TOKAMAK OPERATION)</u>
Contribution to the integrated development of plasma control tools in preparation to ITER operation, in particular for real-time control of active coils. Specific issues are:
1. Coil magnetic field penetration delay and asymmetries.
2. Effect of coil distance from the plasma boundary.
3. Coupling between coils. Analysis of the effects (e.g. limit in the maximum gains) and development of decoupling techniques, like stationary decoupling matrix and model-based MIMO (multiple input-multiple output) approach.
4. Real time correction of sideband aliasing and effect of sidebands.
5. Design of mode controllers based on the physics of Tearing Modes.
6. Comparison between different saddle coil geometries and number (being complete, the RFX coil system is easily “downgradable”).

7. Tearing Mode torque balance and saturated amplitude dependence on resonant and non resonant perturbations.
8. Tearing mode locking avoidance (as a tool to prevent disruptions in the tokamak)</td></tr>
</table>

	MISSION 4 (TECHN. AND PHYSICS OF LONG PULSE AND STEADY STATE) RFX-mod can test critical techniques or physics that transfer fairly directly to the tokamak, like the development of integrated tools for tokamak plasma control, in particular for the Advanced Tokamak (AT) Scenarios, which operate close to operational stability limits, and require active control of MHD instabilities like Resistive Wall Modes (RWM). Specific issues, which can be addressed in RFX-mod are: 1. issues 1-4 and 6 mentioned for Mission 2, applicable also to Advanced Tokamak scenarios and RWMs. 2. Numerical code benchmarking: experiments in RFX-mod can be designed to measure very precisely growth rates. These experiments are an optimal basis for clean code benchmarking in simple reference cases (i.e. without additional complications): circular cross-section, no (partial) stabilization by plasma rotation, etc. 3. Interaction between RWMs and Tearing Modes (TM). This is a hot topic for tokamak RWM control (see DIII-D experience). RFX-mod can contribute to answer the following questions: (a) are these TMs triggered/driven by stable RWM with finite amplitude? (b) Which is the physics behind the interaction? (c) What is the effect of island formation on RWM stabilization? (d) Is this going to happen to ITER? 4. Comparison and benchmarking with RWM experiments performed in ohmic tokamaks (potential collaboration with DIII-D), in particular as far as the issues of different saddle coil geometries, number and location are concerned. 5. General understanding of RWM physics (codes and experiments). 6. Control engineering: advanced feedback models applicable to generic control systems. ***MISSION 5 (PREDICTING FUSION PERFORMANCE)*** Full or complete predictive capability would be able to say what happens when particular choices are made of the major toroidal confinement variables: B_T, I_P, R/a, shape, external vs. internal transform via 3D field, density, input power, etc. To truly predict toroidal confinement, one should ideally know what happens at literally every value of these major variables. Some combinations are naturally going to be more interesting, but the present degree of predictive capability cannot define an obvious optimum (or set of nearly equal optima), and the chances of building a reliable numerical tokamak increase if more physics and a broader parameter space are included. RFX-mod can then *explore RFP confinement at currents similar to those of large tokamaks,* but with 10 times smaller B_T. Moreover, it can contribute to the mission with critical and unique tests of tokamak physics stretched to the extreme of low field. Examples are: 1. *density limit*: RFX-mod can help in understanding the origin of the density limit, which is found very similar to that of tokamaks, but it is not disruptive. In particular the links with radiative instabilities and MHD can be explored. 2. *Turbulent transport in the core*: when magnetic transport is suppressed, there are indications from MST that the RFP is dominated by electrostatic transport also in the core. This corresponds to a low field tokamak with electrostatic transport, which helps to understand how electrostatic turbulence and transport behaves at strong magnetic shear and relatively large gyro-orbit. It will be a key tool in the validation of tokamak transport codes such as gyrokinetic codes which can run for the RFP. *3. Edge turbulence and transport*: with its well-recognized work on the link between sheared $E \times B$ mean flows and turbulence and on turbulent velocity structures, RFX-mod can contribute to first principle understanding of pedestal transport properties. 4. *Beta studies:* RFPs are exceeding beta limits due to interchange (Mercier) and tearing modes, without plasma disruption or serious degradation. The stellarator also exceeds Mercier without apparent effect. The physics of what happens when different MHD stability limits are exceeded (a nonlinear problem) is very general. 5. *Momentum transport*: A significant issue in tokamaks. The RFP both spins up spontaneously and transports momentum anomalously. Magnetic instability explanations are rather advanced in the RFP. 6. *Non-linear MHD,* thanks to the rich portfolio of diagnostics and numerical codes. 7. *Sawtooth oscillations*: the RFP sawtooth has similarities (and some

	differences) to the tokamak and has been studied in a way that is ripe to transfer to the tokamak. 8. *Role of tearing modes and magnetic islands in current transport*. Might contribute to understanding q profile dynamics in hybrid H-mode (see for example DIII-D phenomenology with small NTMs) 9. *Effect of stochastic magnetic fields*, now central to several main subjects of tokamak physics, like ELM control. They can be controlled and tuned in an RFP.

<table>
<tr><td>FACILITITY</td><td colspan="6">EXTRAP T2R, VR</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="6">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: Investment period 1991- 2000. Original investment (Pref Support): 3 500 000 Euro. . (4,1 Mio€ in Euros 2007)</td></tr>
<tr><td colspan="6">COST OF FORESEEN UPGRADES: No major cost for upgrades foreseen. Small upgrades to control system within operation budget.</td></tr>
<tr><td colspan="6">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 60 to120
- yearly cost of operation in 2007 euros: 550 000 Euro
- yearly manpower for operation in ppy: 4 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="3">Number of facility users:
10-12</td><td colspan="3">Yearly integrated equivalent full time facility users: 4-6 (ppy)</td></tr>
<tr><td colspan="6">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 17 PhD theses (last 10 years).</td></tr>
<tr><td colspan="6">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td colspan="2">Year</td><td colspan="2">Journals</td><td colspan="2">Conferences</td></tr>
<tr><td colspan="2">2004</td><td colspan="2">8</td><td colspan="2">9</td></tr>
<tr><td colspan="2">2005</td><td colspan="2">10</td><td colspan="2">5</td></tr>
<tr><td colspan="2">2006</td><td colspan="2">9</td><td colspan="2">4</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="6">Collaborations inside EU: RFX-Padua; AsdexU-Garching; MAST-UKAEA; JET
Collaborations outside EU: Univ of Wisconsin, USA ;
Number of experimental contributions to ITPA (if applicable) :
Sharing facility with other fields of research (if applicable):
Prospects: COMPASS, Prague (RWM active control).</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="6">Ip = 150 kA; <B> = 0.1 T; R/a = 1.24 m/ 0.18 m; Vol = 0.8 m^3
PFM: Stainless steel with Molybdenum armour (8% coverage).
Ohmic heating only.
Spectroscopy (visible and VUV). TS (single point), Interf (single chord), SXR camera (10 chord), ToF (Ti). Insertable edge probes (Te, n, Vf, B-components, surface collector).
Comprehensive B diagnostics, array 64 (toroidal) x 4 (poloidal) of three B components-Br, Bphi, Btheta.
Active saddle coils for feedback control of MHD, full-coverage array 32 (toroidal) x 4 (poloidal).
Real time digital feedback controller and a set of operational algorithms for implementing various relevant feedback laws and for feed forward tests of mode control.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="6">Improved real time controller and new algorithms to test advanced feedback models.
Arrays of edge probes to measure statistical properties of turbulence and coherent structures in the edge region.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="6">First demonstration of full active control of multiple MHD resistive wall mode (RWM) instabilities in an RFP (P R Brunsell et al, Phys Rev Lett 93 (2004) Art No 215003).
Quantitative measurements of resonant field error amplification (RFA) and successful comparison with MHD theory (D Gregoratto, J R Drake, et all, Physics of Plasmas, 12 (2005) Art No 092510).
Measurements of statistical properties of turbulence in the edge region including the Reynolds and Maxwell stress tensors (N Vianello et al, Nuc Fusion 45 (2005) p 761).</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	EXTRAP T2R is not an ITER facility under ITER credit through F4E. EXTRAP T2R addresses the R&D Missions and the fusion roadmap as follows: Five year perspective: Mission 4: Technology and Physics of Long Pulse and Steady-State. EXTRAP T2R will be used for generic studies of active control of MHD (RWMs and RFA) using advanced control algorithms (Multivariable input and output (MIMO), Kalman-filtering with associated estimated state-feedback in the form of LQG-design, and other state-model control synthesis. EXTRAP T2R is the platform for collaboration with AsdexU in the area of RWM control. Mission 3: First wall materials. The Alfven Lab is the platform for technical support and collaborative studies with JET in area of Spectroscopy for ITER-like Wall Experiments. Mission 5: Predicting fusion Performance. Generic studies of turbulence and related transport in the edge region. The statistical properties of the edge turbulence and the relationship to transport of particles and momentum will be examined.
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above) The EXTRAP T2R device addresses generic issues for control of MHD instabilities and for understanding of confinement. The device also provides a national platform for persons engaged in collaborations with JET and other devices in the European programme. The device is operated in a university environment and education has always been a strong feature of the programme. We feel that the device is important in the Human Resource Planning-Education and Training Programme for the fusion community. PhD education and training is a major part of the activity on EXTRAP T2R. There are PhD students currently doing research on the experiment. Furthermore, the EXTRAP T2R device is used for experimental projects in Masters of Science programmes. These include courses and MSc thesis projects within the Erasmus Mundus European Master in Nuclear Fusion and Engineering Physics Masters and within the School of Electrical Engineering Masters Programme. The operation of the EXTRAP T2R device beyond 2010 is dependent on the funding level for fusion research within KTH and from Government research funding agencies.

<table>
<tr><td>FACILITITY</td><td colspan="3">Divertor Test Platform DTP2, Tekes - VTT</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
2007: DTP2 Basic structure, divertor cassette mock-up 1,000 k€
2008: CMM / CEE mover and manipulator 2,500 k€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
2009: DTP2 toroidal extension and CTM (toroidal) mover 2,500 k€
investment, cost in 2007 euros</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): experiments not started
- yearly cost of operation in 2007 euros: 1,500 k€
- yearly manpower for operation in ppy: 10 ppy</td></tr>
<tr><td rowspan="8">USE OF FACILITY</td><td colspan="2">Number of facility users: 5
Associations expressed interest (TEKES, CEA, UKAEA/OT, IST, CIEMAT)</td><td>Yearly integrated equivalent full time facility users: 15 ppy</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis during design and construction of the facility in the last 5-10 years (experiments not started yet): 1 PhD and 7 Diploma Thesis</td></tr>
<tr><td colspan="3">Number of yearly publications based on the design and construction of DTP2 (experiments not started yet):</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2005</td><td>2</td><td>4</td></tr>
<tr><td>2006</td><td>-</td><td>5</td></tr>
<tr><td>2007</td><td>3</td><td>7</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: TEKES, CEA, UKAEA/OT, IST, CIEMAT
PREFIT engineering training programme in collaboration of CEA and UKAEA/OT
Prospects: Demonstrating and training ITER divertor cassette RH operations</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">1:1 size ITER divertor sector, lower port, 1:1 cassette (9 tn), cassette movers CMM/CEE and CTM plus manipulators and tools.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Future RH-operations: radial removal of full size divertor cassettes, second cassette removal, toroidal transfer of next cassettes to the port, various RH operations, cutting/welding pipes, locking and unlocking, rescue operations….</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Basic stell structure and 9 tn divertor cassette ready since May 2007, CMM (mover) from EU industry late over half year</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">DTP2 is an "ITER" facility, paid or to be paid under ITER credit through F4E.
DTP2 addresses the Missions 6 being a necessary demonstration of remote handling maintenance after the nuclear operation is started in ITER.
Five year perspective: to demonstrate ITER divertor remote handling operations
10 year perspective (if applicable): to test the real ITER movers and cassettes before delivered to ITER site and to train ITER RH handling engineers for divertor maintenance procedures</td></tr>
<tr><td colspan="3">Remote handling is needed in long-pulse Satellite Tokamaks like JT-60SA resulting in activation by DD-neutrons.</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td colspan="3">FE 200, Euratom-CEA Association and AREVA-NP</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
• Estimated 1.1 M€ in 1991 (updated 1,4 M€ in 2007)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Proposed for consideration:
• Coupling FE200 facility with an He loop : will depend on specifications</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 130
- yearly cost of operation in 2007 euros: 0.45 M€ (including manpower)
- yearly manpower for operation in ppy: 3 (1 professional, 2 technicians)</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: 3-10</td><td>Yearly integrated equivalent full time facility users: 1</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 5</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>2</td><td>3</td></tr>
<tr><td>2005</td><td>5</td><td>3</td></tr>
<tr><td>2006</td><td>1 (6 in 2007)</td><td>4</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: 4
Collaborations outside EU: 3 (Russia, Japan, USA)
Number of experimental contributions to ITPA (if applicable) : No
Prospects: Possible increase with Asian laboratories developing actively cooled PFCs</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">• Power 200 kW (200 kV – 1A)
• Heat Fluxes : 1- 50 MW/m² for fatigue tests
• up to 1 GJ/m² for disruptions simulations
• Vacuum chamber 8 m^3, mock-up length up to 2 m
• Pressurized water loop up to 35 bar, 50 to 230°C
• Digital infrared camera 3-5 µm, 1 and 2 colours pyrometers, pyro-reflectometer</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">To be considered

Coupling FE200 facility with an Helium cooling loop.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">• More than 135 000 cycles (of ~10 sec.) performed
for thermal fatigue tests (on SS316L, Cu, CFC, W but no Be)
• Availability up to 1000h/ year (only operating without data processing – total 1300h)
• In about 15 years : Development Tore supra, NET, ITER, W7-X
• 400 critical heat fluxes
• Some specific tests : - Disruptions
• - Tests with glancing incidence
• - LOFA (Loss of Flow Accident)
• Operational diagnostics tests : CHF acoustic monitoring, pyro-reflectometry</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	• This facility can still address many issues and contribute significantly to ITER high heat flux plasma facing components industrialisation. High heat flux testing of small and medium-size plasma facing components for ITER : - Completion of R&D for 1st Divertor - R&D for the 2nd Divertor - Support (to industrials, to F4E, to laboratories) for industrialisation * Tests of EU Vertical Target before final acceptance (in Russia) * ITER operation preparation and support - Specific tests : safety (LOFA), qualification of operational instrumentation • Other PFCs qualification for other projects involving actively cooled plasma facing components may appear. • A coupling to an He loop may be considered for ITER TBM or DEMO He cooled PFCs development. Implementation of an He loop for ITER TBM and DEMO Helium-cooled-Divertor could be considered. Implementation to be defined
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above) In 2008 : completion on the experimental work on the topic "Acceptance criteria for the ITER Divertor" Future planning will depend on the programme needs

<table>
<tr><td>FACILITITY</td><td colspan="3">(NAME, ASSOCIATION(S))
Facility for testing laser-induced co-deposit removal and dust characterisation; IPPLM, Warsaw</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros (details when appropriate): First investment in 2005, costs in 2007 ~30 kEuro.</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2007 euros (details when appropriate): Foreseen investment in 2008 ~50 kEuro.</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 60 days/year in 2005, 2005 and 2007,
- yearly cost of operation in 2007 euros: 30 kEuros,
- yearly manpower for operation in ppy: 2ppy.</td></tr>
<tr><td rowspan="8">USE OF FACILITY</td><td colspan="2">Number of facility users: 3</td><td>Yearly integrated equivalent full time facility users: 4 ppy.</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 1 PhD student (2006-2007).</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td>1</td><td>3</td></tr>
<tr><td>2006</td><td>2</td><td>4</td></tr>
<tr><td>2007</td><td>1</td><td>4</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: FZJ Juelich, FZJ-IPP, Juelich, Alfven Laboratory, KTH, Stockholm, IPP ASCR, Prague, Inst. of Fundamental Technological Problems, Warsaw, the Soltan Institute for Nuclear Studies (SNIS), Swierk, n. Warsaw.
Collaborations outside EU: none.
Number of experimental contributions to ITPA (if applicable):
Sharing facility with other fields of research (if applicable): The facility may be used for study of laser-matter interaction, laser-induced technologies and development of diagnostic methods,
Prospects: modernisation of facility including new laser system and advanced diagnostics.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">(factual set of parameters (engineering as well as physics parameters) and other technical capabilities): Nd:YAG pulse laser system, repetition of 10 Hz, 3.5 ns pulses of energy up to 0.8 J at 1060 nm wavelength. Peak power density up to 10^{10} W/cm^2, for co-deposit removal optimized peak power in range of 10^8 - 10^9 W/cm^2, average power is about 5-8 W
Ion collectors and electrostatic ion energy analysers with good both time and spectral resolution which allows to precisely measure energetic parameters of the ions and identify elements present in the samples being subject of research.
Spectrometer systems were available due to courtesy of the collaborators. The systems allow to identify and measure intensities of carbon and deuterium lines which is crucial for in-situ co-deposit removal and for dust characterisation.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">(if applicable: new technical capabilities after foreseen upgrade(s); including planning):
The spectroscopic system is put out to tender and will be bought in 2008.
A new (fiber or YAG) laser is put under consideration and will be bought in 2008</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">(a summary of key scientific achievements): At IPPLM in Warsaw there have been prepared experiments on removal of deuterium from tokamak in-vessel components using laser-induced ablation methods. Original graphite tiles taken from the TEXTOR poloidal limiter were received from FZJ Juelich. The investigations were carried out with the use of Nd:YAG repetitive system. Determination of the characteristics of ions emitted from the laser-illuminated targets was performed with the use of ion collectors and an electrostatic ion-energy analyzer as well as with optical spectrometers. The characterisation of tile after laser treatment was performed in FZJ in cooperation with teams from IPPLM and Alfven Lab. The research performed during last 3 years confirmed that the ion diagnostics and optical spectroscopy could be convenient methods for future real-time observation of the co-deposited layer removal by the means of laser ablation. It can be considered reasonable to use data being collected by this diagnostics method as the feedback signal for the automated system dedicated to remove the co-deposit from components of tokamak devices.</td></tr>
</table>

	The results of preliminary investigation performed in IPPLM show the possibility of simulation of dust production with the use of laser-ablation of the deuterium/tritium containing co-deposited layers from the surface of tokamak tiles (eg. taken from a limiter). Analysing the spectra recorded by the spectrometer one can estimate the elemental content of the ablated dust and evolution of this content during subsequent laser shots, therefore, this method enables to estimate the depth distribution of different elements in the (co-)deposited layer of dust. Simultaneously with optical measurements the laser-ablated material was collected on the surface of glass and metallic substrates located at different distances from laser illuminated graphite tile. The collected dust was roughly characterised with the use of optical microscopy as well as SEM and NRA in collaborating laboratories (FZJ, Juelich, IPP ASCR in Prague and in Alfven Lab.).
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(Is the facility an "ITER" facility, paid or to be paid under ITER credit through F4E.) (how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions) Five year perspective: **The facility can be used for future development and testing of laser-induced techniques prepared for in-situ co-deposit removal and for dust characterisation. This apparatus will be used also for development and testing the diagnostic methods for co-deposit and dust characterisation.** 10 year perspective (if applicable): **Not estimated.** Longer term perspective (if applicable): **Not estimated.**
	(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>) Five year perspective (when relevant): **The facility will be used for development and testing the laser techniques prepared for in-situ characterisation of dust in JET (JET Task: Fusion Technology JW8-FT-2.31).** 10 year perspective (when relevant): **Not estimated.** Longer term perspective (when relevant): **Not estimated.**
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above)

FACILITITY	14 MeV PULSED NEUTRON GENERATOR (IPPLM)		
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: **1966, 1995, 50 000 euros**		
	COST OF FORESEEN UPGRADES: in **2009, 25 000 euros**		
	OPERATION - average number of operation days/year (over the past 4 or 5 years): **200** - yearly cost of operation in 2007 euros: **34 500** - yearly for operation in ppy: **3**		
USE OF FACILITY	Number of facility users: **4 institutions**	Yearly integrated equivalent full time facility users: (ppy)	
	Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: **4/1**		
	Number of yearly publications based on experimental results from facility:		
	Year	Journals	Conferences
	2004	**1**	**4**
	2005	**4**	**2**
	2006	**5**	**4**
	2007	**1**	**1**
COLLABORATIONS	Collaborations inside EU: Collaborations outside EU: Number of experimental contributions to ITPA (if applicable) : Sharing facility with other fields of research (if applicable): Prospects:		
PRESENT TECHNICAL CAPABILITIES	Neutron yield during pulse**: $5{\cdot}10^8$ n/s** Pulse duration**: 25 – 1000 µs** Pulse repetition**: 0.3 – 100 ms**		
FUTURE TECHNICAL CAPABILITIES	(if applicable: new technical capabilities after foreseen upgrade(s); including planning) as above		
PROGRAMME: ACHIEVEMENTS	(a summary of key scientific achievements)		
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(Is the facility an "ITER" facility, paid or to be paid under ITER credit through F4E.) (how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions) Five year perspective: 10 year perspective (if applicable): Longer term perspective (if applicable):		
	(How the facility addresses/will address the Satellite Tokamak requirements: for JET, JT60SA and satellite tokamak proposal(s) only) Five year perspective(when relevant): **tests for neutron activation diagnostics** 10 year perspective (when relevant): Longer term perspective (when relevant):		
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above)		

<table>
<tr><td>FACILITY</td><td colspan="3">BESTH, Nuclear Research Institute Rez, plc, Czech Republic</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007: 376 000 euros
(details when appropriate)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2008: 27 000 euros
(details when appropriate)</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over past 4 or 5 years): 150 days
- yearly cost of operation in 2007: 600 000 euros (4 x 4 mock-ups)
- yearly manpower for operation in ppy: 2</td></tr>
<tr><td rowspan="7">USE OF FACILITY (form magnetic confinement and, when appropriate, for technology facilities)</td><td colspan="2">Number of facility users: 5</td><td>Yearly integrated equivalent full time facility users: 2</td></tr>
<tr><td colspan="3">Number of PhD/diploma theses using experimental data from the facility in the last 5-10 years:</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td>2</td><td>4</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: Yes
Collaborations outside EU: Yes
Number of experimental contributions to ITPA (if applicable): -
Sharing facility with other fields (if applicable): -
Prospects: -</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">(factual set of parameters (engineering as well as physics parameters) and other technical capabilities)
Material mock-ups thermal cycling by graphite heating panel. Power input 40 kW, heat flux 70 W/cm^2, water cooling 6 bar, Beryllium working place, spec.ventilation. Two parallel experimental boxes for 2+2 mock-ups.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">(if applicable: new technical capabilities after foreseen upgrade(s); including planning)
Divertor testing, other PFW materials.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">(a summary of key technical achievements)
Fusion reactors - first wall mock-ups thermal cycling test Be – Cu joint connection.</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">(Is the facility an “ITER” facility, paid or to be paid under credit through F4E.)
(how does the present (future/upgraded) facility address the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions)
5 year perspective: ITER facility
10 year perspective (if applicable):
Longer term perspective (if applicable): DEMO R&D</td></tr>
<tr><td colspan="3">(How the facility addresses/will address the Satellite Tokamak requirements: for JET, JT60SA and satellite tokamak proposals only)
Five year perspective (when relevant): -
10 year perspective (when relevant): -
Longer term perspective (when relevant): -</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning (if not already adequately addressed above) -</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td>(ITM EFDA GATEWAY, Associazione Euratom/ENEA sulla fusione)</td></tr>
<tr><td>RESOURCES INVOLVED</td><td>ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment,
2007 : 344.3 k€ (hardware : cluster HPC + 32 TB Storage Data)

COST OF FORESEEN UPGRADES:
2008 : 32.5 k€ (hardware : +32 TB Storage Data)
2009 : 32.5 k€ (hardware : +32 TB Storage Data)

OPERATION
- average number of operation days/year (over the past 4 or 5 years): NA
- yearly cost of operation in 2007 euros: NA
- yearly manpower for operation in y:1.5 ppy</td></tr>
<tr><td>USE OF FACILITY</td><td><table>
<tr><td colspan="2">Number of facility users:
more than 100 are foreseen for 2008</td><td>Yearly integrated equivalent full time facility users: (ppy)</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td></td><td></td></tr>
</table></td></tr>
<tr><td>COLLABORATIONS</td><td>Collaborations inside EU: NA
Collaborations outside EU: NA
Number of experimental contributions to ITPA (if applicable) : NA
Sharing facility with other fields of research (if applicable): NA
Prospects:
ITM Groups</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td>ITM EFDA Gateway is the group's first joined computing facility; it was projected to allow the task force ITM members to work together on a common platform and share their codes, development tools and technologies. Technically Gateway is a rather small (1 TF, 100 TB) installation which is not meant for massive production; it is however more than sufficient for development activities. it is hosted at ENEA/CRESCO premises in Portici (Naples).</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td>Upgrading of data storage area to archive large amounts of data generated by simulation codes and originating from experiments of various fusion devices.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td>(a summary of key scientific achievements): Integrated Tokamak Modelling (ITM) , High Performance Computing (HPC)</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td>(Is the facility an "ITER" facility, paid or to be paid under ITER credit through F4E.) NO
(how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions)
It is essentially to code development for mission 5 – Predicting Fusion Performance
Five year perspective: This facilities will be available until 2011
10 year perspective (if applicable):
Longer term perspective (if applicable):</td></tr>
<tr><td>FORWARD PLANNING</td><td>Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td colspan="2">“GYM”, IFP-CNR, EUR-ENEA-CNR, Milan, Italy</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="2">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros

The device is under construction. The initial investment in 2007 has been:
Infrastructures 200 k€
Gyrotron and power supply 300 k€</td></tr>
<tr><td colspan="2">COST OF FORESEEN UPGRADES:year of foreseen investment, cost in 2007 €

Gaussian mode converter for the gyrotron output radiation (2008) 15 k€
3 new power supplies for the upgraded version of the machine (2008/09) 100 k€
2 extra magnetic field coils for the high density plasma source (2008/09) 100 k€
Power supply for the 2 coils for the gyrotron plasma source (2008/09) 30 k€
Vac. chamber and vac. system for the high density plasma source (‘08/09) 40 k€</td></tr>
<tr><td colspan="2">OPERATION
the device is not yet operating</td></tr>
<tr><td rowspan="3">USE OF FACILITY</td><td>Number of facility users:
10</td><td>Yearly integrated equivalent full time facility users: 4</td></tr>
<tr><td colspan="2">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years:
the device is not yet operating</td></tr>
<tr><td colspan="2">Number of yearly publications based on experimental results from facility:
the device is not yet operating</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="2">Collaborations inside EU: University of Milano-Bicocca; ENEA, Frascati; INFN LNS, Catania; EPFL-CRPP, Lausanne.
Collaborations outside EU: IAP-RAS, N.Novgorod (Russia).
Sharing facility with other fields of research: possible use as ion source for particle accelerators;
Prospects: establishing collaborations with Columbia Un. (USA), Auburn Un. (USA), New Delhi Un. (India). In perspective, collaborations with EU laboratories where linear devices are operating (IPP-Greifswald, KIWI-Kiel, etc.)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="2"><u>Magnetic field system and main vacuum chamber</u>:
10 water-cooled copper coils, internal/external Ø = 0.52/0.83m, 36 turns each, B_{max} = 1.3kG @ I = 1kA, B ripple < 2%.
Total length: 2 m; inner radius: 25 cm, material: stainless steel AISI 304L, pumping capability: 600 l/s by cryopumps.
<u>Expected main plasma parameter</u>:
Ø = 5 – 10 cm, L ≈ 100 cm; n = 10^9–10^{11} cm^{-3}, $T_e \approx T_i \approx$ 10 – 20 eV, ionization degree > 90 % with the gyrotron-based source.
<u>Filament plasma source</u>: cathode: W wire cloth.
<u>RF source</u>: f=2.45 GHz, CW, P_{max}=3 kW, equipped with an auto-tuning matching system, a circulator, 2 power meters, and a user-friendly software interface.
<u>Gyrotron</u>: GyCom, f=28 GHZ, pulsed (1-100 ms) and CW, P_{max} = 15 kW, ripple ≤ 1%, output mode TE_{01}. Fully equipped with a dummy load capable of 15kW CW, an arc detector, a mw power meter, and a user-friendly software interface.
<u>Diagnostics</u>: Langmuir probes with radial scan for measuring density and temperature profiles, in two axial positions are beeing implemented.</td></tr>
</table>

FUTURE TECHNICAL CAPABILITIES	DAS: the DAS based on a flexible PXI National Instrument technology and process control on "Compact Field Point" platform will be installed. Both controlled by LabView environment. Planned for 2008. Diagnostics: New system of Langmuir probes; electrostatic probe arrays for plasma fluctuation measurements in poloidal direction, Mach probe for plasma flux measurements, Optical spectroscopy VIS-NUV. Planned for 2008-09. Gyrotron-based high-density plasma source: magnetic cusp configuration with B_{max} = 1.7 T. Planned for 2008-09.
PROGRAMME: ACHIEVEMENTS	The GyM machine is a linear CW magnetized plasma device where a low density fully ionized collisionless plasma, with $T_i \approx T_e = T$, will be produced and contained in a cylindrical vacuum chamber by means of a magnetic field, produced with a coaxial set of 10 copper coils. The device will be used for experiments in the field of fundamental plasma physics, mainly oriented to wave-plasma interaction studies. The expected plasma parameters are n = 10^9-10^{11}cm^{-3} (depending on the plasma source), T = 10-20 eV. The underlying idea is to perform experiments following the physical similarity principle: a few relevant dimensionless parameters will be made of the same order as those occurring in large tokamaks, despite the difference in absolute dimensional values, in order to perform scalable experiments. The scientific activity will consist of experimental studies on drift wave instabilities and turbulence, sheath-driven instabilities relevant to the SOL and to the divertor region, use of ponderomotive forces to produce a plasma current, gas breakdown induced by strong microwaves, production of multi-species ions for ion implantation experiments. The general approach will be of establishing action-response patterns by controlling separate plasma parameters. In order to carry out the experiments under the required physical conditions, the device will be equipped with ion heating to get $T_i \approx T_e$, and with control systems for the radial distributions of plasma density, temperature, azimuthal rotation and axial flows. It will allow one to investigate drift instabilities induced by density, temperature (ITG) and flow (KH) gradients. These investigations require the use of high-resolution diagnostics for the plasma parameters, the wave electric fields, and the electrostatic fluctuations associated with the excited instabilities. The planned long pulse operation shall enable accurate study of plasma fluctuation spectra. Moreover, the gyrotron-based plasma source will be used for the production of highly-charged ions, that will be accelerated and made to interact with materials, whose structural modifications will be subject to investigations in GyM.
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(how does the present (future/upgraded) facility addresses the 7 R&D Missions) The foreseen experimental activities in GyM will increase the understanding of basic aspects of drift-wave induced density and momentum transport across the magnetic field, in conditions of $T_i \approx T_e$, $\rho_{Li}/a < 1$, $\nu_c/\omega_{bounce} < 1$ (Mission 5). Such a device will be the ideal tool to investigate sheath-driven instabilities relevant to the SOL and to the divertor region, and specifically the role of sheath boundary conditions. The use of the gyrotron will allow one to investigate the ionization process driven by microwaves (M2). New concepts of non-resonant current drive based on the use ponderomotive forces will be explored, in view of applications to start-up problems in tokamaks (M5). The gyrotron-based ion source will make possible experiments of ion implantation (M3). In addition, GyM will be an ideal plasma device for training of young scientists in the physics of magnetized collisionless plasmas. Five-year perspective: upgrade from Phase I to Phase II (see below); extend collaborations; Improve the accessible experimental regions, develop more sophisticated diagnostics and interpretation tools.
FORWARD PLANNING	The planning of operations consists of two phases. Phase I (2008-09): After an initial characterization of the plasma parameters which can be achieved with the W filament or the RF (2.45 GHz) plasma sources, the experimental activity will be focused on mode excitation/instabilities due to azimuthal rotation; sheath driven instabilities excited by a radial T_e gradient, varying the boundary conditions at the sheath; ponderomotive forces induced by a low frequency (150 Mhz) antenna placed inside the vacuum chamber. In this phase, the plasma will be only partially ionized and ions will be cold. Phase II (2009-11): Operation with the gyrotron-based plasma source and RF ion heating. This will allow one to get a fully ionized plasma and to achieve $T_e \approx T_i$. The activity will be focussed mostly on the excitation of ITG waves, in gases with different atomic masses, and their interaction with plasma rotation and axial flows; plasma breakdown; production of multi-charged ions and relevant studies.

<table>
<tr><td>FACILITY</td><td colspan="3">DRP (Divertor Refurbishment Platform)
Site ENEA CR Brasimone</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">- Capital investment: 1.6 M€
- Cost of foreseen upgrades: 0.5 M€, to be concluded in 2009
- average number of operation days per year: 120 on average in the last four years
- yearly cost of operation cost in 2007: 10 K€ (energy+maintenance), 22 k€ with manpower
- Yearly manpower for operation, ppy: 2</td></tr>
<tr><td colspan="3"></td></tr>
<tr><td colspan="3"></td></tr>
<tr><td rowspan="8">USE OF FACILITY</td><td colspan="2">Number of facility users:</td><td>Yearly integrated equivalent full time Facility users (ppy)</td></tr>
<tr><td colspan="3">Number of PhD/Diploma thesis using experimental data from facility in the last 5-10 years:
0</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>2</td><td></td></tr>
<tr><td>2005</td><td>1</td><td></td></tr>
<tr><td>2006</td><td>6</td><td></td></tr>
<tr><td colspan="3"></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: IHA Tampere; Gradel (L);
- DRP is used in the frame of ITER (divertor PFC assembly) and IFMIF (target assembly)
Collaborations outside EU :
Number of experimental contributions to ITPA (if applicable): NA
Sharing Facility with other fields of research (if applicable): NA
<u>Prospects</u>:
collaboration with CIEMAT in the frame of IFMIF</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">DRP is able to operate in a hot cell relevant environment with the possibility to:
- handle heavy components up to 5 tonn
- assure a good positional accuracy, ± 0.25 mm</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">It is ongoing the upgrading design of the SW control system for the PFCT (Plasma Facing Component Transporter) and for the heavy manipulator.
New tools are under design for the refurbishment of the last ITER divertor cassette design.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">- Development and optimization of the refurbishment process of the ITER divertor cassette;
- Test of the refurbishment process
- Design of the EU concept for the IFMIF lithium target replaceable back-plate (bayonet concept)
- Design and test of the RH procedures for the IFMIF back-plate replacement</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS
FORWARD PLANNING</td><td colspan="3">- DRP has been conceived and is operated to fulfill one of the main missions of ITER design development, which is to optimize and qualify the RH operation.
- Five year perspectives: to continue experimental activities in the frame of ITER and IFMIF Projects</td></tr>
<tr><td colspan="3">- ITER divertor cassette assembly: 2008
- Possibility to perform trials of the tools envisaged for the ITER Hot cell operation 2008-2011
- Procurement of the IFMIF target assembly: 2009
- Development and procurement of tools for the IFMIF target RH: 2009
- Tests on IFMIF target assembly: 2011</td></tr>
</table>

<table>
<tr><td>FACILITY</td><td colspan="4">EBBTF (European Breeding Blanket Test Facility): coupling of He-Fus 3 (He loop) and IELLLO (Pb-16Li loop). IELLLO is presently in commissioning phase

Site: ENEA CR Brasimone</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="4">- Capital investment: 1.8 M€
- Cost of foreseen upgrades (He-Fus 3): 0.20 M€, to be concluded in 2008 + 1.0 M€ for the new compressor (end of 2009)
- average number of operation days per year: EBBTF not active so far, only He-Fus 3 operative (average 90 days/y in the last years)
- yearly cost of operation (supposing 70% of loading factor): 530 k€
- yearly manpower for operation, ppy: 5</td></tr>
<tr><td colspan="4"></td></tr>
<tr><td colspan="4"></td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users:</td><td colspan="2">Yearly integrated equivalent full time Facility users (ppy)</td></tr>
<tr><td colspan="4">Number of PhD/Diploma thesis using experimental data from facility in the last 5-10 years:
3 (from He-Fus3)</td></tr>
<tr><td colspan="4">Number of yearly publications based on experimental results from facility:
2 per year, from 2004 to 2006, Fusion Engineering and Design + Fusion Technology, presented at ISFNT, SOFT, CBBI</td></tr>
<tr><td>Year</td><td colspan="2">Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td colspan="2">2</td><td></td></tr>
<tr><td>2005</td><td colspan="2">2</td><td></td></tr>
<tr><td>2006</td><td colspan="2">2</td><td></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="4">Collaborations inside EU: with University of Palermo (I) , with FZK under the supervision of EFDA- He-Fus 3 is being to be used for the experimental characterization of a compact heat exchanger (HEATRIC) in the frame of Raphael Project (VHTR fission reactor, Gen. IV)

Collaborations outside EU :
Number of experimental contributions to ITPA (if applicable)
Sharing Facility with other fields of research (if applicable):
Prospects:</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="4">EBBTF is able to test HCLL and HCPB TBM mock-ups up to scale 1:3</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="4">EBBTF will be:
- able to test HCPB and HCLL-TBM mock-ups up to scale 1:1
- able to test the main TBM auxiliary systems (TES, CPS)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="4">He-Fus 3, as a part of EBBTF, provided the blanket development community with important experimental results, mainly in the field of thermo-mechanics of HCPB blanket concept.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(is the facility an “ITER” facility, paid or to be paid under ITER credit through F4E)**: NO** EBBTF has been designed to fulfill one of the main missions of ITER operation, which is to test and develop the EU Test Blanket Modules in view of their use in ITER. Moreover, some technologies tested through EBBTF are DEMO and Reactor relevant.
FORWARD PLANNING	- Tests of HCPB small scale mock-ups: **2008** - He-Fus 3 upgrading: **2008** - Further tests on HCLL and HCPB-TBM mock-ups, included auxiliary systems: **2009-2012**

<table>
<tr><td>FACILITY</td><td colspan="3">FNG (Frascati Neutron Generator)

Site ENEA Frascati</td></tr>
<tr><td>RESOURCES INVOLVED</td><td colspan="3">- Capital investment: 4 M€
- Cost of foreseen upgrades: 7 M€, (acquisition of a 30MeV cyclotron to produce IFMIF like neutrons)
- average number of operation days per year: 60
- yearly cost of operation cost in 2007: 1,5 M€ (including manpower)
- Yearly manpower for operation, ppy: 1,6</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: 6</td><td>Yearly integrated equivalent full time Facility users (ppy) 1</td></tr>
<tr><td colspan="3">Number of PhD/Diploma thesis using experimental data from facility in the last 5-10 years:
8</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>4</td><td>3</td></tr>
<tr><td>2005</td><td>5</td><td>4</td></tr>
<tr><td>2006</td><td>6</td><td>4</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU:CEA – FZK – UKAEA – CIEMAT – TUD + several universities

Collaborations outside EU : JAERI - TRINITY
Number of experimental contributions to ITPA (if applicable)
Sharing Facility with other fields of research (if applicable):
<u>Prospects</u>:
Continue the collaborations with the aforesaid Institute also in the frame of TBM activities</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">The Frascati Neutron Generator (FNG) is an unique 14 MeV neutron source important for material activation, cross sections measurements and fusion nuclear calculation tools and data, benchmarking - validation. FNG consists of a steady-state or pulse 300 KeV 1 mA deuteron beam impinging on a tritiated target. A very detailed model of the target support and surrounding structure is used to obtain fluxes and spectra around the neutron target. It is possible to produced clean D-T fusion neutron spectrum as well as FW fusion reactor like neutron spectrum using an “ad hoc” cavity. Time dependent neutron output is measured with an accuracy better than 3%. FNG is located in a large (2000 m3) heavily shielded bunker. Room return background is very low.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">For IFMIF program there is an urgent need to improve the nuclear databases at higher neutron energy. FNG upgrade is a facility to perform integral cross section testing and activation cross-section validation up to about 33 MeV. This facility, taking advantages of the existing FNG structures, systems and services, will cost about 7 M€, a very small fraction of the IFMIF project.
High-energy databases validations are important also for other nuclear programs..</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Several benchmark experiments to validate neutron cross section databases have been performed with FNG.
Development of innovative neutron detectors (diamond detectors).
- Design of the EU concept for the IFMIF lithium target replaceable back-plate (bayonet concept)
- Design and test of the RH procedures for the IFMIF back-plate replacement</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">FNG is the ideal facility do project and develop neutron diagnostics for ITER</td></tr>
</table>

<table>
<tr><td>FACILITY</td><td colspan="3">PROTO-SPHERA (Euratom-ENEA)
Spherical torus formed and sustained from a screw pinch plasma central column (fed by ring electrodes), which replaces central metal column of toroidal magnet and solenoid of ohmic transformer.</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT: 2004-2009,
1.25 M€ for Phase 1 (central plasma column only)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: 2010-2011, 1.5 M€ for Phase 2 (spherical torus around central plasma column)</td></tr>
<tr><td colspan="3">OPERATION
not yet ready (foreseen at the end of 2009)
yearly cost of operation 50k€
yearly manpower for construction 4 ppy
yearly manpower for operation 7 ppy</td></tr>
<tr><td rowspan="6">USE OF FACILITY</td><td colspan="3">Number of yearly publications (design and construction): 2-3</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td>Innovative Confinement Conference (1)</td></tr>
<tr><td>2005</td><td>Phys. Plasmas (1)</td><td>ST Workshop (1)</td></tr>
<tr><td>2006</td><td>Phys. Plasmas (1)
Nuclear Fusion (1)</td><td>ST Workshop (1)</td></tr>
<tr><td>2007</td><td>Phys. Plasmas (1)</td><td>ST Workshop (2)</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Inside EU: Euratom-UKAEA, implemented since 2004
Outside EU: Tokyo University, to be implemented in 2008 through IEA Implementing Agreement on ST</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">PROTO-PINCH (central plasma column current 1kA)
Phase 1(foreseen 2009, central plasma column 10kA)</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Phase 2(foreseen 2011, central plasma column 60kA, -
spherical torus plasma current 120-240 kA)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Heated cathode and hollow gas puffed anode module able to withstand a plasma current density of 1 MA/m^2 and
a power load of 20-30 MW/m^2 for a few seconds, fully tested on PROTO-PINCH testbench</td></tr>
<tr><td rowspan="3">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">Five years perspective:
Obtain results from Phase 1: with suitable multi-modules ring electrodes avoid anode arc anchoring of the central plasma column; upgrade to Phase 2 and obtain results from Phase 2: form spherical torus around central plasma column at highest possible current, investigate instabilities and sustain spherical torus for at least one resistive time, investigate energy confinement.</td></tr>
<tr><td colspan="3">Ten years perspective:
Form configuration with electrodes, but then remove it from electrodes and try sustaining it (CKF configuration)</td></tr>
<tr><td colspan="3">Long term perspective:
CKF configurations (if scalable to a burner) could easy direct energy conversion and the use of a burner as a space thruster</td></tr>
</table>

FACILITY	**NBTF -Neutral Beam Test Facility (ENEA)**
RESOURCES INVOLVED	Total investment: 101,4 M€
	Cost of foreseen upgrade: *not applicable*
	Manpower: Design, follow-up procurements, R&D 10.5 M€/y (2008-2013), Operation 2,7 M€/y (2011-2013), Operation 7.5 M€/y after 2013 (start of operation of the full power injector)
USE OF FACILITY	*not applicable since not yet in operation*
COLLABORATIONS	Collaborations inside EU: 3 Associations, CEA, IPP, UKAEA Collaborations outside EU: ITER, Japan, India
PRESENT TECHNICAL CAPABILITIES	*not applicable*

FUTURE TECHNICAL CAPABILITIES	The nominal performance of a single ITER injector should be:
	• Power injected into the plasma P = 16.5 MW
	• Equivalent negative ion current I = 40 A
	• Acceleration voltage V = 1 MV
	• Pulse duration t_{oper} = 3600 s
	The facility will support ITER with the objectives to:
	• Realize and test the neutral beam injector (NBI)
	• Achieve the nominal performance
	• Assist the NBI operation on ITER
	• Optimize NBI operation at ITER under different scenarios, e.g.:
	    • Injected power modulation
	    • Low energy operation for low-performance plasmas
	• Maximize the ***reliability*** of the injector
	• Develop ***new technologies*** for the injector
	• Test ***Remote Handling*** tools and procedures
	The facility will allow to separately test the negative ion source (Ion Source Test Facility)
	Extended experimental exploitation will support the design of NBIs for DEMO in the following areas:
	o Further development of the NBI technology
	o Improvement of ***availability***
	o Test of new concepts in order to improve the NBI ***efficiency***, e.g.:
	    ▪ Higher neutralisation efficiency
	    ▪ Maximization of D- production yield
	    ▪ Minimization of Electron production yield and their soft suppression

PROGRAMME ACHIEVEMENTS	*not applicable*
PROGRAMME: ADRESSING THE PROGRAMME NEEDS	***Mission 1: Burning Plasmas*** To provide a Neutral Beam Injector for ITER, with output power and pulse duration adequate to achieve the fusion target goal; to assist ITER testing operations (like power modulation) aimed to aid burn control. ***Mission 2: Reliable tokamak operation*** To provide flexible and reliable burning plasma heating and current drive by neutral beam injection for a range of plasma parameters and related profile control. ***Mission 4: Physics and Technology of Long Pulse and Steady-State*** To test all technologies related to long pulse and possibly steady state operation, with the aim of ensuring reliability, availability, limited maintenance, full compatibility with Remote Handling requirements. ***Mission 7. DEMO Integrated Design: towards high availability and efficient electricity production*** To improve efficiency and availability by developing new components (as, for example, a more efficient neutralizer and an Ion Source without Caesium) and improving the design of all components according to the previous experimental results at the Test facility.
FORWARD PLANNING	Start of experimental activities on the Ion Source Test Facility: 01-2011 Start of experimental activities on the full Neutral Beam Test Facility: 01-2013

<table>
<tr><td>FACILITITY</td><td colspan="3">JUDITH / HML (Hot Materials Laboratory) - FZJ</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT:
Judith 1: 0.75 M€ (1990), Judith 2: 1.5 M€ (2005), HML: 13 M€ (1966)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Judith 1: 0.5 M€ (new beam control unit), Judith 2: 0.25 M€ (extension of vacuum chamber, new Judith 3: 1.5 M€, hot coolant loop, 2007/2008), new hot cell building for fusion and fission research: 6 – 8 M€ pro-rata share devoted to fusion.</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 to 5 years): 200 days/ year
- yearly cost of operation in 2007: Judith 1: 0.2 M€, Judith 2: 0.3 M€, HML: 3 M€
- yearly manpower for operation in ppy: Judith 1, 2 and HML: 16 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users:
23 research institute</td><td>Yearly integrated equivalent full time facility users: 5 ppy</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 10 years: average 1 PhD- and 1 diploma-thesis per year</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>16</td><td>8</td></tr>
<tr><td>2005</td><td>20</td><td>9</td></tr>
<tr><td>2006</td><td>32</td><td>9</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: 8 association
Collaborations outside EU: 11 research institutes / universities
Sharing facility with other fields of research: industrial applications, ExtreMat-project (with emphasis on aerospace, electronics, fission etc.)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">High Heat Flux Test Facility: | JUDITH 1 | JUDITH 2
electron energy | 120 – 150 keV | 30 - 60 keV
beam power: | 60 kW | 200 kW
irradiation area | 10 x 10 cm^2 | 50 x 50 cm^2
pulse length: | 1.0 ms ... ∞ | 1.5 µs ... ∞
beam scanning mode: | ≤ 100 kHz | digital mode
Facilities capable for quasi-stationary <u>and</u> transient thermal loads;
testing of neutron irradiated and toxic materials (Be, T-implanted samples)</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">new technical capabilities after foreseen upgrade(s); including planning)
- upgrading of Judith 1 (installation of a more flexible beam control unit)
- installation of a new electron beam facility JUDITH 3 (to expand test capacity)
- construction of a new hot-cell building (to be operational in 2012)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Systematic investigation of divertor and first wall solutions under ITER-relevant thermal loads including neutron irradiation effects; investigation of material degradation (brittle destruction) under transient loads, thermo-physical characterization of n-irradiated PFC-materials.
Participation in the EU fusion training scheme; supervision of PhD- and diploma theses; lectures at universities and summer schools.</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">Five year perspective: qualification testing of FW-modules, improvement of W-armoured PFCs for the 2nd ITER divertor, simulation of ELM-loads with high repetition rate (on W, CFC and Be), .[in agreement with mission 3]
10 year perspective: Investigation of steady and transient power fluxes to PFCs for later ITER applications and for DEMO (incl. neutron degradation effects with DEMO relevant fluences), [in agreement with missions 3 and 6]
Longer term perspective: neutron irradiation studies on improved PFMs for DEMO, [in agreement with mission 6]</td></tr>
<tr><td colspan="3">(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>)
Five year perspective: procurement testing of W-armoured divertor tiles for JET, [in agreement with mission 3]</td></tr>
<tr><td></td><td colspan="3"></td></tr>
</table>

<table>
<tr><td>FACILITY</td><td>GLADIS, IPP GARCHING</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td>ORIGINAL INVESTMENT : 2 M€ (start of operation: 2005)</td></tr>
<tr><td>COST OF FORESEEN UPGRADES:
Upgrading for full size ITER target prototype testing: Σ 0.85 M€
- Simultaneous operation of 2nd beam line: 0.2 , (spent 0.1)
- Enlargement of test chamber and target translation mechanism: 0.3
- Hot water cooling loop (if requested): 0.25
- Improvement diagnostics, data acquisition: 0.1</td></tr>
<tr><td>OPERATION
- average number of operation days/year (over the past 4 or 5 years): 200 days/year
- yearly cost of operation in 2007 euros: 0.7 M€
- yearly manpower for operation in ppy: 5</td></tr>
<tr><td>USE OF FACILITY</td><td>Number of facility users: 6 research institutes
Number of yearly publications based on experimental results from facility:
<table>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>Start of operation 2005</td><td>2</td><td>3</td></tr>
<tr><td>2006</td><td>1</td><td>14</td></tr>
<tr><td>2007</td><td>17</td><td>8</td></tr>
</table></td></tr>
<tr><td>COLLABORATIONS</td><td>Collaborations inside EU: 6 association, JET, EFDA, Univ. Marseille
Collaborations outside EU: 2 research institutes / universities
Sharing facility with other fields of research: industrial applications, ExtreMat-project (with emphasis on aerospace, electronics, fission etc.)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td>High Heat Flux Test Facility, Type: H ion beam, 2 beam lines
Ion beam energy 15-55 keV
Beam power: 2 x 1.1 MW
Heat flux: 3 – 45 MW/m²
Loaded area up to 300 cm²
pulse length: 1 ms ... 45 s
Target dimension: up to 2m
Target cooling (high purity water): 8.5 l/s
Target pressure: 2.5 MPa</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td>new technical capabilities after foreseen upgrade; including planning:
- Installation of vacuum lock to increase efficiency of testing (2008)
- GLADIS is capable for operation with He or mixed H/He beam
- Simultaneous and independent operation of two beam lines up to 90 MW/m²
- Tests of ITER full size target prototypes (see above)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td>The aim of this facility is to provide testing capabilities for full size HHF loaded divertor components which have both active water cooling and large dimensions.
- W7-X: Evaluation of CFC/ Cu bondings in the frame of development for the actively cooled divertor. Qualification tests during manufacturing of W7-X divertor targets.
- ASDEX Upgrade: contribution to the W programme, HHF tests of components
- Participation in the EU fusion training programme.</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td>Contribution to Mission 3: (First wall materials & compatibility with ITER/DEMO relevant plasmas):
- HHF testing and evaluation of W plasma facing components in interaction with hydrogen (He possible) for heat loading conditions similar to ITER/DEMO.
- Numerical modelling of components and materials
- Investigation of the thermo-mechanical behaviour of PFCs under thermal load
Comprehensive characterization of tested PFCs within IPP.
5 year perspective:
Acceptance tests of industrial manufacturing of W7-X divertor targets.
Qualification testing for the JET ILW project. Tests of tungsten PFCs with improved joining technique (MMCs).
Support for ITER divertor target fabrication by EU industry:
- HHF tests during prototype, pre-series and fabrication phase.
- Continuous feedback prior to final tests performed in RF
10 year perspective: Evaluation of the thermo-mechanical behaviour of PFCs for future ITER applications and for DEMO (in agreement with mission 3).</td></tr>
<tr><td>(How the facility addresses/will address the Satellite Tokamak requirements: for JET, JT60SA and satellite tokamak proposal(s) only)
Five year perspective: Quality assessment for procurement of W coated CFC tiles and other PFCs for JET and ASDEX upgrade (in agreement with mission 3).</td></tr>
</table>

<table>
<tr><th>FACILITY</th><th>INTEGRATED PWI FACILITY, IPP GARCHING</th></tr>
<tr><td>RESOURCES INVOLVED</td><td>ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: 7 Million €
COST OF FORSEEN UPGRADES:
– Beamline dedicated uniquely to PFC analysis 0.5 Million €
OPERATION
- yearly cost of operation in 2007 euros: 1.0 M€
- yearly manpower for operation in ppy: 7.0</td></tr>
<tr><td>USE OF FACILITY</td><td>- Simulation of PWI processes using in-situ ion beam experiment
- Analysis of probes and PFCs from tokamak exposure
- High energy irradiation for n-damage simulation
<table>
<tr><td>Number of facility users: 35</td><td>Yearly integrated equivalent full time facility users: 24 (ppy)</td></tr>
</table>
Number of PhD thesis using experimental data from the facility in the last-10 years: 10
Number of yearly publications based on experimental results from facility:
<table>
<tr><td>Year</td><td>Journals</td><td>Not-refereed conference contribution</td></tr>
<tr><td>2004</td><td>28</td><td>1</td></tr>
<tr><td>2005</td><td>42</td><td>8</td></tr>
<tr><td>2006</td><td>18</td><td>5</td></tr>
<tr><td>2007</td><td>66</td><td>6</td></tr>
</table></td></tr>
<tr><td>COLLABORATIONS</td><td>Collaborations inside EU:
Close cooperation with EFDA PWI Task Force
- Leadership of EFDA PWI TF since 2006, - Completion of 8 EFDA Tasks since 2000
Main bilateral collaborations:
- CEA (DITS project), JET (ILW Project), TEXTOR (^{13}C material transport)
- Univ. Marseille: Ab-initio calculations of phase formation in binary mixtures of Be/C/W
Collaborations outside EU:
- PISCES-B, UCSD: Influence of Be on erosion and hydrogen retention of CFC and W
- Inst. Phys. Chem, Moscow: Analysis of D in CFC and W at large depths
- MEPHI, Moscow: Hydrogen retention in and permeation through metals</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td>Quantitative hydrogen analysis by NRA: 15 μm into C, 7 μm into W
Surface composition by backscattering analysis: up to 20 μm
Analysis of boundary and divertor probes and large tiles from ASDEX Upgrade and JET
Dedicated analysis chamber with glove box for Be and tritium (<1GBq) use
Simulation experiment of PWI processes:
- Erosion and deposition studies in dual beam experiment
- Chemical analysis of binary and ternary mixing of Be/C/W
- Synergistic chemical erosion of carbon with H^o and ions in triple beam experiment
Implantation experiments for damage simulation:
- Energies up to 15 MeV C, 24 MeV Si, W
- Beam currents: up to 50 μA,
- Scanned area: 20 x 20 mm^2
- Dose homogeneity: Lateral non-uniformity < 1 %</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td>• Addition of dedicated beam-line uniquely for analysis of PFCs
• Completion of Be and T compatible beam-line for JET ILW experiment
• Addition of sputter-ion source for highly sensitive tritium depth profiling using AMS</td></tr>
<tr><td>PROGRAMME ACHIEVEMENTS</td><td><table><tr><td>• Quantitative determination of the material migration and deuterium inventory in the all-C and all-W ASDEX Upgrade
• Hydrogen inventory build-up in Be, CFC and W
• Determination of compound phases in the interaction of Be/C/W</td><td>• Development of metal-doped graphites with low chemical erosion
• Development of Si-doped W with low oxidation rate
• Development of hydrogen diffusion barriers for W coatings</td></tr></table></td></tr>
<tr><td>PROGRAMME: ADRESSING THE PROGRAMME NEEDS</td><td>Mission 3: First wall materials & compatibility with ITER/DEMO relevant plasmas:
• State-of-the-art surface analysis for the assessment of W and C PFCs with emphasis on plasma performance, erosion/re-deposition balance and tritium retention. This is achieved from dedicated laboratory experiments and material comparison in fusion devices.
• Material compatibility with respect to mixed material phase formation and thermo-mechanical stability.
• Extensive modelling of solid-state processes in the interaction of a hot plasma with PFC surfaces, as well as erosion, transport and re-deposition in tokamaks.
5 year perspective:
• Assessment of the expected tritium inventory for different wall choices in ITER
Longer term perspective:
• n-irradiation simulations for its influence of tritium retention in ITER and DEMO</td></tr>
<tr><td>FORWARD PLANNING</td><td>2009/2010: Dedicated beam-line and analysis chamber for large tile PFC analysis</td></tr>
</table>

<table>
<tr><td>FACILITY</td><td>ECRH test facility, IPP GREIFSWALD</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td>ORIGINAL INVESTMENT : 27 M€ (start of test operation: 2004)</td></tr>
<tr><td>COST OF FORESEEN UPGRADES:
Upgrading required for ITER ECRH-component testing: Σ 0.8 M€
- cw-dummy loads with higher performance 0.3
- Improvement of rf-diagnostics (t-f analyser + rf-equipment) 0.25
- Upgrade of the IR-diagnostics + DAQ 0.1
- Vacuum window and pumping system 0.15</td></tr>
<tr><td>OPERATION
- average number of operation days/year: 80 days/year
- yearly cost of operation in 2007 euros: 1.2 M€
- yearly manpower for operation in ppy: 7</td></tr>
<tr><td>USE OF FACILITY</td><td>Number of facility users: 5 research institutes
Number of yearly publications based on experimental results from facility:
<table><tr><td>Year</td><td>Journals</td><td>Conferences</td></tr><tr><td>Start of operation 2004</td><td>2</td><td>9</td></tr><tr><td>2005</td><td>5</td><td>7</td></tr><tr><td>2006</td><td>4</td><td>6</td></tr><tr><td>2007</td><td>5</td><td>16</td></tr></table></td></tr>
<tr><td>COLLABORATIONS</td><td>Collaborations inside EU: FZK, CRPP, ENEA-Milano, EFDA, Univ.Stuttgart, Univ. Greifswald
Collaborations outside EU: Russian Academy of Science, Institute of Applied Physics, Nizhny Novgorod, Russia, Nizhny Novgorod State University, Nizhny Novgorod, Russia
Sharing facility with other fields of research: dusty plasmas, gyrotron development, microwave technology.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td>High Power, cw Microvave Facility
Frequency 140 GHz
Total power: 10 x 1 MW
RF-performance adjustable rf- beam parameters, arbitrary polarization
Transmission quasi-optical at normal pressure
pulse length: 10 μsec – 30 min
Power modulation < 10 kHz</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td>Power up to 1.7 MW with Beam Combination of two Gyrotrons
- Simultaneous and independent operation of 2 RF wave-beams
- Tests of ITER vacuum-based components
- Tests of ITER full size RF-components
- Two frequency operation at 140 (full power) and 104 GHz (half power)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td>This facility enables testing of μ-wave components under high-power, cw conditions with an rf-frequency close to the ITER ECRH-frequency. So far:
- High Power Tests (HPT) of a 2 MW prototype load for ITER (with CNR, EFDA).
- development and HPT of a remote steering launcher mock-up for ITER (EFDA)
- HPT of a Fast Directional Switch for ‘Advanced ECRH for ITER’ (5 Laboratories)
- development and test of improved Gyrotron Collector Power dissipation for next step ‘2 MW Gyrotrons for ITER’
- HPT of two frequency operation</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td>Contributions to Mission 2 (reliable tokamak operation), Mission 4 (technology and physics of long pulse and steady state):
- HPT and evaluation of a many ITER-ECRH components (e.g. electrodynamic systems of gyrotrons, dummy loads, mirrors,..), excl. narrow band rf-components, which operate only at 170 GHz.
- HPT and evaluation of mock-up and prototype versions of ‘scaled to 140 GHz’ components for ITER
5 year perspective: Contribute to viable/robust component design for ITER-ECRH
10 year perspective: Two-year time slot available for further testing during W7-X major reinforcement shut-down</td></tr>
<tr><td>(How the facility addresses/will address the Satellite Tokamak requirements: for JET, JT60SA and satellite tokamak proposal(s) only)
JT60SA will install 140 GHz, W7-X EU-Gyrotrons, the test facility is perfectly matched to do HPT’s for JT60SA.
ASDEX upgrade: close collaboration, uses same frequency, applies R&D components in plasma experiments (e.g. FADIS)</td></tr>
</table>

FACILITY	**Microwave Stray Radiation Launch Facility MISTRAL, IPP GREIFSWALD**
RESOURCES INVOLVED	ORIGINAL INVESTMENT : 120 K€ (start of routine operation scheduled : 4/2008)
	COST OF FORESEEN UPGRADES: - system currently commissioned, no upgrades yet
	OPERATION - average number of operation days/year (planned) : 40 days/year - yearly cost of operation expected: 165 k€ - yearly manpower for operation in ppy: 1.5
USE OF FACILITY	Facility users: IPP only microwave homogenity: characterization completed (DA)
COLLABORATIONS	Support by Univ.Stuttgart. Facility also used for training of engineers in the frame of EFTS program "Microwave Diagnostic Engineering for ITER" (MDEI) in cooperation with IST (Lisboa), CIEMAT (Madrid), CEA (Cadarache).
PRESENT TECHNICAL CAPABILITIES	**Impact of cw microwave stray radiation from ECRH on in-vessel components** Input Frequency 140 GHz Total cw power input: MISTRAL designed for 30 kW cw 100kW for 1 minute obtained from a modulated cw-gyrotron RF-performance isotropic and homogenous microwave field, Power impact on Device under Test 30 kW/m^2 - $100kW/m^2$ Pulse length: 30 min Size of Device Under Test max flange 40cm*100cm, max length 270 cm (corresponding to maximum in-vessel component of W7-X, i.e. the ECRH launcher) Standard Diagnostics: RF sniffer detectors, thermocouples, IR- and VIS cameras
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	**Contribution to Mission 2** (reliable tokamak operation)**, Mission 4** (technology and physics of long pulse and steady state): - High-power and vacuum tests of in-vessel components in an homogenous 140 GHz stray radiation environment as it will occur with ECRH at densities close to cut-off and during advanced heating scenarios beyond X2 ECRH, namely O2 heating, Electron Bernstein Wave (OXB) heating and during plasma start-up. - Tests of windows, sealings, gaskets, moveable parts like shutters and plasma facing optical components. Test of divertor- and diagnostic components and integrated in-vessel systems.

<table>
<tr><td>FACILITITY</td><td colspan="4">SULTAN, CRPP</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="4">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: most capital investment was done in 1980-1996. Approximate value ≈ 40 MEuro, incl. cryo-plant</td></tr>
<tr><td colspan="4">OPERATION
- SULTAN operates all year, except 2 weeks/year for maintenance of the cryo-plant
- including manpower and electricity : 1.5 MEuro/year
- yearly manpower for operation in ppy: 3 physicists and 5 technicians/engineers. In case of extensive use, with double shift and weekend operation, the yearly operation cost is estimated 2 MEuro/year</td></tr>
<tr><td colspan="4">Number of facility users: see “collaborations”</td></tr>
<tr><td rowspan="7">USE OF FACILITY (technology facilities)</td><td colspan="2">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: > 2</td><td colspan="2">Yearly integrated equivalent full time facility users: on site 1 ppy, off-site >5</td></tr>
<tr><td colspan="4">Number of yearly publications based on experimental results from facility: > 10 (we do not keep track of publications on SULTAN results made by the users)</td></tr>
<tr><td>Year</td><td colspan="2">Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td colspan="2">10</td><td>1</td></tr>
<tr><td>2005</td><td colspan="2">10</td><td></td></tr>
<tr><td>2006</td><td colspan="2">9</td><td>2</td></tr>
<tr><td>2007</td><td colspan="2">9</td><td></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="4">Collaborations inside EU: CEA, ENEA, Univ. Twente, EFDA, W7-X
Collaborations outside EU: ITER, JAEA, NFRC, ASIPP, MIT, Bochvar
Prospects: All ITER DAs and industries involved in magnet procurement for ITER</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="4">SULTAN is the only facility able to apply full operating conditions to large size superconductors: background dc field up to 11 T, background ac field up to ± 0.43 T, 0.1 – 10 Hz, DC current source for sample up to 100 kA, supercritical helium flow with temperature range 4.3 K to 10 K, mass flow rate up to 20 g/s per sample, high accuracy instrumentation and data acquisition</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="4">• Qualification of conductors and joints for ITER, W7-X and EAST (2002–to date). Before starting production, the prototypes are tested for design verification
• Transverse load and cyclic load degradation in Nb3Sn Cable-in-conduit (CICC) conductors. The performance loss from the strand to the CICC and its progression with cyclic load was first observed in SULTAN in 2000
• Qualification of developmental Nb3Sn conductors. A non-CICC conductor was compared with standard CICC in 2003. A newly developed react&wind conductor was developed and tested with excellent results in 2006
• Qualification of very high current density, Nb3Sn CICC. For the use in the EDIPO magnets, several medium size CICC with parametric variations were tested in 05-07, showing the inadequacy of the original design
• Self-field instability in large NbTi Cable-in-conduit conductors. The high field gradient across the CICC cross section, combined with high n-index and high dJ/dB of NbTi conductors, drives a runaway of the local electric field and a premature quench before any current re-distribution among the strands takes place, 2002-2004
• Current re-distribution on large NbTi CICC. The current distribution is artificially unbalance at the electrical termination and the re-distribution in high field is studied at different extent of unbalance, as a function of the interstrand resistance in the cable (strand coating), 2003</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	SULTAN is the facility selected by ITER for qualification, acceptance and quality control of the whole production of superconductors (CICC). The funding of the facility operation during the construction of ITER, either directly by ITER or through F4E and the other DAs, is being discussed between F4E and ITER **Five year perspective**: Full load by the ITER conductor qualification plan, and the initial phase of quality control. Qualification and acceptance for the superconductors of JT60SU (broader approach). Quality control for the conductors of the EDIPO facility (in construction). Possible, seldom, occasional use by projects in other fields. **10 year perspective**: Together with the EDIPO facility (in construction), full load by quality control and acceptance tests for the series production of ITER conductors. Possible use for tests of HTS or LTS prototype and developmental conductors for DEMO. Possible, seldom, occasional use by projects in other fields **Longer term perspective**: SULTAN and EDIPO remain in the long term the only facilities for qualification, quality control and acceptance test of large size, force flow superconductors. DEMO and other fusion devices will need to use SULTAN for their conductor tests. Other fields (energy storage, hybrid high field magnets) may also need to use SULTAN to test prototypes and qualify the conductor production.
FORWARD PLANNING	The flow of samples to be tested in the facility is the key for the planning of operation. Basically SULTAN is kept cold, i.e. ready for operation, with the operators team available, all the year around, with minimum discontinuity for the maintenance of the cryo-plant and replacement of components. The coordination of the operation and flow of samples is done by CRPP accounting for the priority given by EFDA (in future F4E) in agreement with ITER. The activity of sample assembly (quite costly and demanding, mostly in case of Nb3Sn conductors) is also part of the planning. The intention of ITER is to delegate the sample assembly exclusively to the SULTAN team.

FACILITITY	**EDIPO, CRPP**
RESOURCES INVOLVED	ON GOING INVESTMENT: the overall expenditure (2005-2010), excluding the cryo-plant, is estimated ≈ 10 MEuro, incl. design, development, procurement and assembly. To large extent, the expenditure is already committed
	OPERATION - EDIPO will operate in parallel with SULTAN, using the same cryo-plant. On top of 1.5 MEuro/year of SULTAN operating cost, 0.8 MEuro/year must be added for the parallel operation of EDIPO, due to the increased power consumption and team upgrade (2 more physicists and 2 more technicians/engineers) - in case EDIPO would operate without SULTAN (i.e. shutting down SULTAN) the yearly operation cost would be the same as SULTAN, 1.5 MEuro/year.
	Number of facility users: see “collaborations”
USE OF FACILITY (technology facilities)	EDIPO is planned to complement SULTAN during the construction phase of ITER and beyond. Due to the strategic importance of a facility for conductor qualification, EDIPO is also meant to provide a back-up in case of failure of SULTAN, i.e. the guarantee that at least one facility world-wide is operating
COLLABORATIONS	Same collaborations / users as for SULTAN facility
PRESENT TECHNICAL CAPABILITIES	EDIPO will have similar technical capabilities as SULTAN. The same samples can be measured both in EDIPO and in SULTAN. Compared to SULTAN, the peak background field is increased from 11 T to 12.5 T and the conductor length exposed to high field is increased from 0.45 m to 1.1 m. Background dc field up to 12.5 T, background ac field up to ± 0.3 T, 0.1 – 10 Hz, DC current source for sample up to 100 kA, supercritical helium flow with temperature range 4.3 K to 10 K, mass flow rate up to 20 g/s per sample, high accuracy instrumentation and data acquisition
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	EDIPO will share the load of work of SULTAN, addressing, both in medium and long term, the same programme
FORWARD PLANNING	EDIPO is scheduled to be commissioned in the end of 2009 and start normal operation in 2010. The achievement of the schedule targets depends on the delivery of the superconducting winding, procured by EFDA (now F4E) at the industry.

<table>
<tr><td>FACILITITY</td><td colspan="6">2MW ITER Gyrotron Test Facility, CRPP</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="6">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
2.519M€ + 2.498M€ (gyrotron) + 0.6M€ (magnet) + 0.3M€ (BPS) + 0.11M€ (crowbar) + 0.112m€ (Transmission line)</td></tr>
<tr><td colspan="6">COST OF UPGRADES: 1.3M€ MHVPS installation in 2008:</td></tr>
<tr><td colspan="6">OPERATION
- average number of operation days/year: 40/200 days (started 11/07 with all equip.)
- yearly cost of operation in 2007 euros: about 700k€ budget
- yearly manpower for operation in ppy): 4.15ppy +1ppy scientific coordination</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="3">Number of facility users: 5</td><td colspan="3">Yearly integrated equivalent full time facility users: ppy</td></tr>
<tr><td colspan="6">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 1 PhD in progress /1 Marie Curie Fellowship Trainee</td></tr>
<tr><td colspan="6">Number of yearly publications based on experimental results from facility: 1 FED 2003, 1 FED 2007, 1 IRMMW conference 2007, during the construction phase</td></tr>
<tr><td colspan="2">Year</td><td colspan="2">Journals</td><td colspan="2">Conferences</td></tr>
<tr><td colspan="2">2004</td><td colspan="2">1</td><td colspan="2">1</td></tr>
<tr><td colspan="2">2005</td><td colspan="2">1</td><td colspan="2">1</td></tr>
<tr><td colspan="2">2006</td><td colspan="2"></td><td colspan="2"></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="6">Collaborations within EU: FZK, CNR/ENEA, NTUA, TEKES ; industries
Collaborations outside EU: GA (future component testing)
Number of experimental contributions to ITPA (if applicable) : NA
Sharing facility with other fields of research (if applicable): NA
Prospects: EU launcher project (testing)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="6">10s, 85kV, 80A Regulated High Voltage Power Supply (RHVPS); 40kV, 150mA Body PS; 4.5MW cooling plant; integrated control and data acquisition for CW operation; 12m ITER transmission line available; 2MW short pulse (0.1s) calorimetric, RF load; 5000ltr Liquid Helium refrigeration plant at test stand.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="6">CW 2MW RF load to be delivered (CNR/ENEA) in early spring 2008.
CW, 60kV, 80A Main High Voltage Power Supply (MHVPS) by end of 2008.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="6">Presently testing 1st prototype 2MW ITER gytroron. 1s RF pulse testing and CW collector testing planned for 2008.</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="6">The facility is optimally geared towards establishing the Heating and Current Drive reliability in CW operation. Space is allocated to allow transmission line/ launcher testing as well as gyrotron development along the fusion roadmap.
Five year perspective: Development/testing of gyrotron prototypes to series tubes
10 year perspective: ‘Factory’ testing of series ITER gyrotrons
Longer term perspective: Repair/upgrade/reliability/burn-in testing of ITER/DEMO gyrotrons. Potential launcher test bed with upgrade</td></tr>
<tr><td colspan="6">JT60SA gyrotron+magnet systems will be tested at this facility
Five year perspective: mechanical supports and water cooling connections will be adapted (2010). The system is dimensioned to handle 2MW tubes whereas JT60SA requires 1MW tube testing (2010-2012).
10 year perspective: Available when ITER gyrotrons not being tested.
Longer term perspective: NA.</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="6">Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td colspan="3">Neutronics Laboratory, TU Dresden</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 Euros: 4 M€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euro:</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 24/12*
- yearly cost of operation in 2007 euros: 210 k€ / 70 k€*
- yearly manpower for operation in ppy: 1.8
* The first number refers to the DT neutron generator, the second to the photo-neutron source (PNS)</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
12</td><td>Yearly integrated equivalent full time facility users: (ppy) 3</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 0/2 completed, 1/3 in progress since commencement of operation 2005</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td colspan="2">commencement of operation</td></tr>
<tr><td>2006</td><td>2</td><td>3 (refereed papers)</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: FZK, PTB, ENEA, UKAEA, Univ. Vienna
Collaborations outside EU: Osaka University, JAEA/FNS
Number of experimental contributions to ITPA: n.a.
Sharing facility with other fields of research: Dosimetry
Prospects: collaborations to be continued…</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">DT neutron generator: 10 mA, 300 keV, pulsed / cw, 1*10^12 n/s
Photo-neutron source: tungsten radiator driven by electron beam, currently test operation at 33 MeV and 10 uA</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Photo-neutron source: electron beam at 40 MeV / 1 mA, 8*10^13 n/s</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Fusion neutronics: Material activation and nuclear data validation, blanket experiments (tritium production rate - TPR, shielding, neutron and gamma-ray flux spectra)</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The laboratory at TU Dresden (TUD) has participated in EFDA tasks concerning material activation measurements and various blanket experiments for many years. The new TUD neutronics laboratory constructed at the ELBE facility of Forschungszentrum Dresden-Rossendorf came into operation in 2005. It is designed for high neutron source strength at 14 MeV and pulsed/cw operation of the new neutron generator (DT-NG). In addition, a photo-neutron source (PNS) has been constructed which will be driven by a 40 MeV / 1 mA electron beam bombarding a tungsten radiator.

The intense fast neutron field in a fusion reactor generates radioactive nuclides in all structures surrounding the fusion plasma. In order to address the issues of safety, accident scenarios and decommissioning with acceptable end-of-lifetime activity adequately for engineering design, licensing, and construction, a well-validated database of nuclear parameters such as activation cross sections needs to be maintained. Within the framework of Mission 6 (Materials and Components for Nuclear Operation), the two neutron sources will be used to continue irradiation experiments of material samples and blanket mock-ups as mentioned above in order to understand and extrapolate the behaviour of materials under reactor conditions. The DT-NG will be utilized for irradiation of ITER/DEMO-relevant materials in 14 MeV neutron fields to validate nuclear data libraries such as the European Activation File. The PNS is able to produce significant neutron flux densities above 14 MeV and is therefore expected to be used for validating IFMIF-relevant nuclear data which need to be qualified to a level adequate for the nuclear licensing of IFMIF. The intense neutron flux in the lower MeV range allows for deep-penetration shielding experiments which are of interest especially for protecting the superconducting coils. The PNS can be used also as a source of an intense gamma-ray field with energies up to several MeV for experimental measurements of radiation effects on diagnostics components.

Ensuring a tritium breeding ratio (TBR) greater than 1 is of crucial importance for fusion devices beyond ITER. Although ITER aims at demonstrating the feasibility of tritium blanket concepts, a well-developed database of nuclear parameters relevant to modelling the neutron transport in a blanket is required for a reliable engineering design of the blankets. Blanket experiments for measuring nuclear responses such as the tritium production rate (TPR) from which the TBR can be calculated will be performed with the DT-NG and sufficiently accurate methods of TPR measurement. TPR and shielding experiments will include also measurements of fast and slow neutron as well as gamma ray flux spectra. The equipment for such measurements is available on-site. Recently we have performed experiments with a mock-up of the European Helium-Cooled Pebble Bed Blanket, and an experiment with a mock-up of the Helium-Cooled Lithium-Lead blanket is scheduled for 2008. Both experiments are done in collaboration with FZK and ENEA.

FORWARD PLANNING

Five year perspective:

Activation and blanket neutronics experiments (currently TTMN-002)

10 year perspective:

Extension to activation experiments for IFMIF with photo-neutron source in addition to regular DT neutron generator operation, testing of diagnostics components, experimental support for validation of modelling

Longer term perspective:

Neutronics performance experiments for qualification of structural and functional materials in 14 MeV neutron fields for DEMO, experimental support for validation of modelling.

<table>
<tr><td>FACILITITY</td><td colspan="3">Fusion Materials Laboratory (FML), Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
Construction of the building and setup of both a lead-shielded hot cell and 4 glove boxes for investigation of neutron-irradiated functional materials (1986 – 1990): 17.5 M€$_{2007}$
First upgrade: Installation of 4 more glove boxes (1996 – 2001): 0.6 M€$_{2007}$
Second upgrade (2002 – 2005): Setup of 2 new lead shielded hot cells for materials testing and metallography. Installation of a scanning electron microscope (SEM) for radioactive samples, 7.7 M€$_{2007}$.</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
2008: Replacement of the shielded light-optical microscope (LM) by a new state-of-the-art device, ~0.3 M€
2008: Installation of a new transmission electron microscope (TEM), ~1.25 M€
2009: Installation of a nano-indentation device for very small irradiated samples (glove box operation), 0.5 M€
2011 Installation of a dual beam focussed ion beam system (FIB+SEM) for radioactive samples, ~0.4 M€</td></tr>
<tr><td colspan="3">OPERATION (includes maintenance but excludes upgrades/large refurbishments.)
- average number of operation days/year (over the past 4 or 5 years): 365
- yearly cost of operation in 2007 euros: 3.2 M€ (including personnel)
- yearly manpower for operation in ppy: 14.0</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
So far FML is not a User Facility, but is only used by FZK scientists in the frame of the FZK-EURATOM programme. FML could be made available to external users by introducing an additional shift.</td><td>Yearly integrated equivalent full time facility users: (ppy)
10 (FZK scientists)</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 7 PhD and 2 diploma theses</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>6</td><td>11</td></tr>
<tr><td>2005</td><td>3</td><td>5</td></tr>
<tr><td>2006</td><td>7</td><td>7</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: EFDA Technology Workprogrammes, HFR (EU), NRG (NL), CEA (F), GKSS (D)
Collaborations outside EU: SSC RF RIAR (RUS)
Prospects: Collaboration with UCSB (USA)</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">The Karlsruhe Fusion Materials Laboratory (FML) is a Post-Irradiation-Examination (PIE) facility which specially meets the requirements of research on irradiated fusion materials (structure materials, functional materials, materials for plasma facing components). Unlike conventional universal hot cell plants it is a very compact, low operating cost laboratory. It has three highly flexibly operated lead shielded hot cells: Two materials testing cells are equipped with all necessary devices for mechanical testing of irradiated materials (including small specimens technology), one metallography cell is equipped with a LM. The cells can be kept under a very pure nitrogen atmosphere.
There are eight glove boxes also with connections to the nitrogen atmosphere system. They are equipped with measuring devices for tritium adsorption and desorption, gross and fine structure and gamma spectra of irradiated materials.
The equipment is completed by a scanning electron microscope (SEM) with wavelength-dispersive (WDX) and energy-dispersive (EDX) spectrometers for the analysis of chemical elements in irradiated samples and a transmission electron microscope (TEM) with EDX spectrometer.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">The TEM will be replaced in 2008 by a new instrument covering the resolution range down to 0.25 nm (equipped with devices for EDX, EELS and EFTEM analysis). It will be the only analytical TEM in Europe allowing the observation of atomic scale irradiation defects in highly radioactive specimens in a hot cell environment.
An indenter for instrumented hardness tests of structural materials and thin coatings of divertors up to DEMO-relevant temperatures will be made available in 2009. This very powerful and unique tool allows extracting all relevant mechanical material parameters only from the indentation of very small samples as they will be available e.g. from IFMIF irradiations.
The installation of a dual beam focussed ion beam system (FIB+SEM) for radioactive samples in about 2011 will enable the use of nanotechnology methods for analyzing irradiated fusion material samples of very small volume.</td></tr>
</table>

PROGRAMME: ACHIEVEMENTS	The FML and the associate institutes participate in the EU fusion program since the beginning of the FZK-EURATOM association. The participation included the EFDA-Tasks on structural materials (steels), functional materials (Li-based ceramics and beryllium/alloys), and materials for protecting plasma facing components (carbon, beryllium, tungsten). Materials from EU irradiation experiments were investigated: 1. The PIE of irradiated Reduced Activation Ferritic Martensitic (RAFM) steels qualified EUROFER as a reference material for the construction of DEMO: RAFM steels have been studied after various low, mid, and high dose irradiation programmes (MANITU, HFR irradiations Ia to IIb). The effects of irradiation dose and irradiation temperature on the material properties were studied by tensile, charpy and fatigue testing as well as by microstructural characterizations. It was demonstrated that the substitution of the radiologically unfavorable elements not only reduced the neutron-irradiation induced activation, but also substantially reduced the irradiation induced embrittlement of the EUROFER steel. As an example, helium bubbles were identified in the microstructure of irradiated steel and their influence on the mechanical properties was determined by extensive studies of steels with different contents of the constituent element boron. The results of the above research considerably influenced the development of the actual candidates for structural materials. 2. The second field of research are the materials for the Helium-Cooled-Pebble-Bed (HCPB) blanket. FML has been involved in this topic since the beginning of the development of ceramic breeders. FML did not only participate in post-irradiation investigations of irradiated materials, but was also regularly involved in the quality control or the characterisation of pebbles prepared for European irradiation experiments. The laboratory has also carried out several out-of-pile tritium release experiments on samples from European irradiation programs (e.g. EXOTIC). 3. Protective materials for the first wall (Be, C, CFC) and window materials for the ICR-heating successfully have been investigated. FML participated in the determination of tritium inventories of JET divertor and first wall protection tiles and the development of detritiation techniques.

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

FML provides significant input to the materials development for the European Fusion Programme. It contributes to all materials issues identified in the R&D Needs and finally to the licensing of the power plant. The laboratory will address the Missions 3 (*First wall material & compatibility with ITER/DEMO relevant plasmas*) and 6 (*Materials and Components for Nuclear Operation*) by testing and characterizing irradiated and thus highly radioactive materials. These materials are foreseen as structural, plasma facing or functional materials for DEMO and for the TBMs inserted in ITER. The capability of handling and testing of radioactive samples is a key technology for materials development. Especially the small scale specimen testing technology for PIE, available at FML, meets the requirements of limited space in future reactor- and IFMIF-irradiations. For all future purposes, PIE includes mechanical testing and microstructural investigations by the installations and methods described above, and providing data for the modelling of irradiation damage.

For structural materials for the First Wall (DEMO and TBM), FML's main contribution will be the mechanical, microstructural, and radiological characterisation of irradiated steel and ODS-alloys, supplemented by compatibility tests of different materials.

For the blanket, BeTi and Ceramic Breeder Materials (Li_4SiO_4, Li_2TiO_3) will be analyzed mechanically and in their microstructure in irradiated and unirradiated condition. FML contributes to the tritium removal development programme by tritium release tests after loading or co-deposition on material foreseen for use in blanket and divertor.

Divertor materials for ITER and DEMO as W and C are characterized concerning the erosion behaviour; and material coatings can be mechanically characterized by instrumented indentation. Those materials and their joints to other materials will be characterized at high temperatures in irradiated condition.

Five year perspective:

In the frame of the materials development programme the following activities will be pursued:

- PIE of high-dose fast breeder irradiation (steel) and ongoing low- and mid-dose experiments (e.g. the Petten High Flux Reactor experiment SPICE)
- Mechanical characterisation of irradiated tungsten and coatings
- PIE of ongoing (HICU, HIDOBE) and future irradiation experiments of breeder materials (Li-ceramics and Be-alloys)
- Out-of-pile investigation of the tritium release behaviour of ceramic breeder materials.

10 year perspective:

Emphasis will be put on PIE of future reactor-irradiations (irradiation resistance of improved and novel structural materials and joints (15 dpa between 250 and 650°C):

- EUROFER97-2
- Advanced ODS and advanced W-alloys
- Advanced diffusion-, TIG, and EB welds as TBM-relevant joining technologies.

Additionally, the laboratory will participate in the development of breeder materials (Li_4SiO_4 and Li_2TiO_3) and neutron multipliers (e.g. $Be_{12}Ti$) within the Broader Approach. It will also support the regular quality control of ceramic breeders to be used in the mock-ups for the development of the EU HCPB Test Blanket Module for ITER, as well as the ITER TBM itself. In addition to the database generation, the novel analytical TEM and the instrumented hardness tester are essential tools for the experimental validation of modelling the dependence of irradiation damage on irradiation parameters.

Longer term perspective:

In IFMIF, structural materials for future reactors (DEMO and beyond) are to be irradiated in a fusion-relevant spectrum. For the materials development based on IFMIF-irradiations, FML will play an important role by doing PIE, especially by testing small samples by means of micro-investigations.

Functional and structural materials from the TBMs will be analyzed after use and thus irradiation in ITER.

Damage analysis of irradiated ITER and later on DEMO components will be done.

FORWARD PLANNING

2008: Availability of the new LM and TEM.

2009: Supplying data for licensing of materials used in TBMs (ITER). Availability of the high-temperature micro/nano-indentation device for advanced characterisation of brittle materials.

2011: Procurement of a dual beam focussed ion beam system (FIB+SEM).

2017: ITER Operation Phase 1 (H_2-D_2): investigation of irradiated components.

2017 – 2026: DEMO Engineering Design: contribution to qualification of materials.

2019 – 2031: Operation of IFMIF: investigation of irradiated small scale material specimens.

2021 ITER DT-Operation Phase: in addition, investigation of tritium release and tritium removal possibilities

2026: Supplying input to the database for the DEMO licensing procedure. Damage analysis of irradiated ITER components after first shutdown.

<table>
<tr><td>FACILITITY</td><td colspan="3">Gyrotron Test Facility, Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: Investments of existing and operational gyrotron testbed equipment (dating back up to 20 years and more) to the present: € 6.3 Mio. (excluding buildings).</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: approx. € 4.3 Mio. (in 2007 Euros) investment planned within next 3-5 years</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 150
- yearly cost of operation in 2007 euros: 700 k€ (including manpower)
- yearly manpower for operation in ppy: 3.5 (5 Professionals x 70%)</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="3">Number of facility users: Facility is mainly used for the FZK gyrotron development within EURATOM and only occasionally as a user facility (< 10 %).</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: PhD's: 12, Diploma: 7</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>11</td><td>36</td></tr>
<tr><td>2005</td><td>17</td><td>62</td></tr>
<tr><td>2006</td><td>13</td><td>41</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: IPP (Garching+Greifswald), Thales, EFDA, CRPP, IPF Universität Stuttgart, CEA, CNR (Milano), FOM Rjinhuizen, TEKES (Helsinki), Euratom Helenic Republic (Greece), Univ. Karlsruhe, Univ. Hamburg-Harburg
Collaborations outside EU: IAP (RU), Gycom (RU), St. Peterburg State University (RU), Calabazas Creek Research (USA), BINP Novosibirsk (RU), CEERI India, JAEA Japan.</td></tr>
<tr><td>PRESENT TECHN. CAPABILITIES</td><td colspan="3">Optimisation and testing of gyrotrons and gyrotron components up to 2 MW, up to 170 GHz, up to 10sec (1 MW to 190 sec, up to 500 kW continuous), including all relevant power, frequency and mode purity diagnostic equipment for high and low power.</td></tr>
<tr><td>FUTURE TECHN. CAPABILITIES</td><td colspan="3">Testing of gyrotrons up to 4 MW output power in continuous wave (CW) operation.
High Power Testing of EC H&CD components like transmission lines and antennae.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">• Development of world´s first 1 MW CW 140 GHz gyrotron with high mode purity
• Introduction of the CVD-diamond window technology in collaboration with JAEA and Element Six (formerly DeBeers)
• Construction of 10 MW ECRH Plant for W7-X (ongoing)
• Development of various short pulse (5msec) gyrotrons up to 2.2 MW / 165 GHz
• Development (unique in Europe) of computer codes for key gyrotron components (gun, cavity, quasi optical mode converter, phase correcting mirrors, output window, collector etc.)</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The facility is not an “ITER” facility, although collaboration in the development of a 2 MW CW 170 GHz coaxial cavity gyrotron exists. In case of an upgrade of the ITER EC H&CD system to 40 MW this facility would supplement the CRPP gyrotron test facility.

Electron Cyclotron Heating and Current Drive (EC H&D) are essential tools to generate and maintain a burning Tokamak Fusion Reactor plasma. ECCD is required to stabilize neoclassical tearing modes (NTMs) in order to get reliable long-pulse and steady state operation. This means that the development and testing of highly reliable continuous wave (CW) gyrotrons with high unit power and high efficiency as sources for the EC-waves is addressing the missions I (Burning Plasma, burn control by ECH), II (reliable Tokamak operation, NTM stabilisation) and is directly related to mission IV (R&D Technologies and Physics of Long Pulse & Steady state, ECH and ECCD).

Besides the continuous training for students (PHD, Diploma and Term-papers) of the University of Karlsruhe, the facility is also used for training young scientists within the EURATOM FUSION TRAINING SCHEME EC TECH (Electron Cyclotron System Technology for ITER, Contact Nr. 042636 Fu(06) and the "EFDA Goal Oriented Training Programme" (Network Power Supplies).

Five year perspective:

- Development of a highly reliable continuous wave 2 MW 170 GHz CW coaxial cavity gyrotron. The need for larger gyrotron unit power is even now considered for ITER and is likely to be mandatory for reactor-size machines.
- Design and testing of a high efficiency 4 MW short pulse coaxial cavity gyrotron with two 2MW output beams.
- Development of a multi frequency step tuneable gyrotron for NTM stabilisation with a fixed antenna mirror system (fixed frequency ECCD requires movable antenna mirrors, avoiding those could be advantageous).

10 year perspective:

- Development of a high efficiency, CW 4 MW output power gyrotron
- Development of a 1 MW CW gyrotron at higher frequencies (≈ 200 GHz) for more efficient central ECCD for steady state Tokamak operation.

Longer term perspective:

- Continuation of 4MW gyrotron development with frequency tuneability option.
- General Gyrotron development and maintenance and extension of know-how of the design of key gyrotron components.

FORWARD PLANNING

EC H&CD and therefore gyrotrons will play a major role in plasma heating and plasma control in machines beyond ITER. Even central Electron-Cyclotron Current Drive (ECCD) is being considered for continuous Tokamak operation. In order to reduce cost and complexity of future more powerful ECH plants, the unit output power and efficiency needs to be increased beyond current capabilities. Today's Gyrotrons can provide continuous power up to 1 MW per unit, the design and testing of 2 MW gyrotrons is ongoing. In order to advance Gyrotrons up to 4 MW output power, it is necessary to extend the existing test facilities to accommodate such development. Within the next 3-5 years FZK intends to significantly upgrade the test facility, starting with the procurement of a 10 MW High Voltage DC power supply, which can satisfy all the needs that could occur for the testing of gyrotrons of up to 4 MW output power. Furthermore, the design and development of such gyrotrons shall be commenced with a view to have these available within 10-15 years, also addressing the need for higher efficiencies. Frequency tuneability of Gyrotrons is highly desirable, as it extends the operating range of both, the ECH plant and the Tokamak. This topic is currently being addressed within the EFDA for short pulse gyrotrons (105 GHZ to 170 GHz) and will be further explored for CW gyrotrons.

Within the next 5 years it is also envisaged to extend the test facility in such away as to accommodate the testing of larger ECH&CD -components and -systems.

<table>
<tr><td>FACILITITY</td><td colspan="3">Helium Loop Karlsruhe – High Pressure (HELOKA-HP), Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: TBM Part of HELOKA-HP Loop under construction. Estimated loop investment is 4.6 Mio. Euro (2007: 1.33 Mio €, 2008: 2.53 Mio. €, 2009: 0.774 Mio. € + Compressor + TBM related equipment such as dedicated data acquisition.)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros: HELOKA-HP/TBM see above;
HELOKA-HP/TDM:
• 2009: 3.1 Mio. € (HT Bypass + Preparation TDM-Loop),
• 2010: 2.5 Mio. € (TDM-Loop),
• 2011: 1.9 Mio. € (TDM-Loop),
• 2012: 4.0 Mio. € (TDM-Loop + Surface Heating System Stage I),
• 2013 ff: 4.0 Mio. € (Loop + Heating system finalization).</td></tr>
<tr><td colspan="3">OPERATION- average number of operation days/year (over the past 4 or 5 years): not yet operational
- yearly cost of operation in 2007 euros:
1.4 Mio €. (electricity, media, maintenance, etc. – including personnel).
- yearly manpower for operation in ppy: 6 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
not yet in operational
→ does not apply</td><td>Yearly integrated equivalent full time facility users: (ppy)</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years:</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td></td><td></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: Co-operations with other Euratom associations and institutions will be possible. Research organizations in France have already expressed interest.
Collaborations outside EU: Research organizations in USA have already expressed interest.
Sharing facility with other fields of research/Prospects: e.g. gas cooled fission reactors could be a future research field to share the facility.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">Helium Loop for TBM compound and system testing available in early 2009.
Operating pressure (max.): 10 MPa
Operating temperature (max.): 500°C
Helium Mass flow rate: 1.3 kg/s
Loop heating power: 750 kW
Test section available heating power: ~ 1 MW</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Helium Loop for divertor /TDM advanced blanket development and testing
Operating temperature: more than 700°C
Helium mass flow rate: 3.9 kg/s or more
Surface Heating System (neutral beam source)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Does not apply as the facility is under construction.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

HELOKA is needed to support several R&D Missions for the European Fusion Programme. The Facility is dedicated to allow for developing the plasma facing components (PFC), the breeding modules technology and the dedicated Helium cooling systems for DEMO. An important section in the time line towards DEMO is the ITER testing phase for the PFC and breeding technology needed in DEMO.

Thus, the main focus of the HELOKA facilities is on the breeding blanket technologies and the high temperature Helium cooled divertor technologies including the related test programs in ITER, namely the Test Blanket Module (TBM) program and the envisaged Test Divertor Module (TDM) program. The PFC for DEMO need appropriate, reliable und efficient cooling systems, therefore the development of the Helium Loops (HELOKA) themselves has to be highlighted in the context of the EU "reactor-oriented" approach.

The efficiency of the Fusion Power Plant will highly depend on the heat extraction system and its parameters like outlet temperatures and internal losses like pumping power demand. Also the *reliability of the PFC is mandatory for Fusion reactors. HELOKA will allow to develop and test the efficient and reliable components and cooling systems needed. This includes optimized heat transfer mechanisms, improved high temperature performance (e.g. evolving divertor operational temperatures without exceeding tolerable thermal stress limits) and reliable mass flow balancing in complex Helium cooled components with complicated manifold systems.*

- *Mission 6, "Materials and Components for Nuclear Operation", is the most important purpose of HELOKA. The key in-vessel components needed for the fuel cycle (tritium breeding blankets) and power and particle control (divertor) will be developed using the HELOKA facilities. For many years, FZK as a part of the EU efforts has been developing concepts for advanced components, in particular the blanket modules and the divertor for DEMO that have to be tested in ITER. Full thermal testing of these components is required before they are installed in ITER, not only to validate the computer codes used for the design and material selection but also to ensure they can operate safely and reliably over long periods in ITER and DEMO. Experimental test loops are indispensable also to test the ancillary components required for operation in ITER/DEMO, such as the pumps and the cooling and purge loops. HELOKA will therefore be a complete pilot system for the blanket modules to be installed on ITER. HELOKA has to demonstrate safe and reliable heat extraction systems.*
- *Mission 7, "DEMO Integrated Design: towards high availability and efficient electricity production", is another important area, where the HELOKA facilities will be used intensively in the context of the development of the internal structures, mainly blanket and divertor, and their adjustment to the related maintenance strategies. The ongoing development in regard to availability and efficiency will make necessary the test of special blanket geometries (not always cubic and separated as a single TBM), helium cooled shield elements and manifolds at various temperature and load levels. The related R&D is not part of the TBM and TDM program but needs extensive experimental testing in dedicated facilities. HELOKA will be available for these DEMO related future tasks.*
- *In regard to Mission 4, "Technology and Physics of Long Pulse and Steady-State", HELOKA will be used to test the time dependent features of the PFC like creep effects, fatigue and thermo-cyclic behavior for real geometries. Overload experiments will help to get information on the possible lifetime during a reduced experimental time. The experimental results will allow optimizing the PFC.*
- *In regard to Mission 3, "First wall materials & compatibility with ITER/DEMO relevant plasmas", again HELOKA will play an important role. The surface heat loads and their impact on the PFC will be studied in HELOKA. The graphite radiation heaters for the blanket modules and for advanced TDM a neutral beam source will be used to provide the according high heat fluxes.*
- *In regard to Mission 2, "Reliable tokamak operation", next to the reliability of the plasma operation also the reliability of the highly loaded PFC and the related cooling systems are important aspects, which will be optimized during the HELOKA R&D programs.*

HELOKA is planned to be developed in several consecutive stages. The HELOKA-HP/TBM section is already under construction for the development of the ITER test blanket modules and the corresponding Helium cooling cycles will serve as basis for the necessary upgrades and divertor test sections (mission 6). The mass flow and temperature levels available in HELOKA-HP will be increased according to the needs for the development of the Helium cooled divertor for DEMO and the corresponding test module for ITER. HELOKA-HP/TDM will serve as a pilot project for the Helium circuit required for Divertor tests in ITER.

As it is anticipated that very high temperature helium technologies will play an important role in the future energy production market (mission 7), advanced types of very high temperature divertor mock-ups will be developed. Therefore, the bypass in the blanket loop will be further upgraded for temperatures of up to 900°C. This very high temperature loop section of limited size will allow a continuous improvement and development of the divertor and Helium loop technologies for high temperatures. In addition, this loop section allows for performing essential overload experiments for enhanced scenarios in ITER and DEMO (mission 2, 3, 4, 6).

- Five year perspective: HELOKA-HP is designed to perform full-scale tests of HCPB TBM prior to its installation in ITER. Additionally, the loop is prototypical for the Helium cooling and extraction loops to be installed in ITER thus providing the conditions to develop and qualify the TBM helium cooling system. In parallel with the TBM testing program, the test of a representative section of the TDM target plate made of steel (parallel flow under moderate heat input) is foreseen. First TDM tests at higher temperature are envisaged for the end of this period.
- 10 year perspective: In the ten years perspective, the loop is going to be used intensively for testing TDMs. Divertor target plates made of W/steel can be tested between 2013 and 2016. For this purpose, the loop will be extended by constructing the complete High Temperature test section and installing additional circulators to increase the flow rate. The high temperature loop

will be build in parallel with the TBM test section. In addition, a small very high temperature test section is planned.

- Longer term perspective: On a longer time perspective, the test of a complete divertor cassette and new blanket configurations (see above) is foreseen. When the divertor loop for enhanced scenarios will be available, also breeding blanket tests at enhanced scenarios and with higher mass flow will become feasible in HELOKA. HELOKA gives the possibility to test large divertor mock-ups and DEMO blanket modules.

FORWARD PLANNING

- HELOKA-HP/TBM for breeding blanket tests, TBM qualification, Helium Loop development (starts operation in early 2009)
- In 2009: Start of preparation for divertor/TDM mock-up tests.
- In 2010: Construction of high temperature bypass in HELOKA-HP/TBM for testing of small TDM mock-ups. Also further preparation of the construction of the parallel TDM loop section for higher mass flow.
- 2011-2013ff: Construction of HELOKA-HP/TDM with heating system.
- 2013ff: experimental campaigns in regard to mission 2, 3, 4, 6 and 7 as shown above.

<table>
<tr><td>FACILITITY</td><td colspan="3">Launcher Structural Test Facility , Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: 400k€ (in 2007)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros: 2M€ in 2008-2011</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): Not applicable. Facility is still under construction.
- yearly cost of operation in 2007 euros: estimated: 300 k€ (including personnel)
- yearly manpower for operation in ppy: estimated: 2.5</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
Estimated: 10</td><td>Yearly integrated equivalent full time facility users: (ppy) Estimated: 4</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: Not applicable. Facility is still under construction.</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:
Not applicable. Facility is still under construction.</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td></td><td></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: 4 (CRPP, CNR, IPP, FOM)
Collaborations outside EU: 3 (JA, USA, RF)
Prospects: External assembly and acceptance test site for ITER</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">ECRH launcher modules alignment & fixation; Joining parts (bolting, welding).
ECRH launcher performance tests (thermo-hydraulic, -mechanical and optical).</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">External Launcher Acceptance Tests:
- Pressure and vacuum tests.
- Bake-out in vacuum.
- Cooling system performance tests.
- Geometric tolerances.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Not applicable. Facility is still under construction.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The Launcher Structural Test Facility in its first stage provides launcher components prototype testing capabilities with a water loop covering the ITER relevant cooling water temperatures and pressures (operation and bake out at 240°C, 44bar). The planned extensions to be paid under ITER credit through F4E will cover also the assembly and acceptance tests of manufactured components as well as the development of remote handling tools and procedures. This offers an extensive design check in the development of the ECRH heating and plasma stabilizing system (upper and equatorial launcher, diagnostic port plugs) bringing together the goals in plasma heating and stabilizing physics with the engineering requirements for a successful operation in ITER.

Mission 1: Burning Plasmas:

The ECRH system plays a substantial role in heating the plasma. In order to provide a functional mm-wave optical system, a correct component alignment during the assembly is required. The assembly procedures, remote handling maintenance and different acceptance tests are developed at the facility and will ensure proper beam propagation within the operational tolerances at different temperatures and cooling water pressures.

Mission 2: Reliable Tokamak operation:

ECRH heating systems are vital for stabilizing plasmas. In contrast to ICRH and neutral beams only ECRH can provide localized heating of individual plasma surfaces to suppress plasma instabilities which is essential for tokamak operation. As the heating spot has to be small and well positioned, a very precise steering of the mm-wave beam has to be guaranteed. The assembly procedures and different acceptance tests developed at the facility will ensure proper plasma stabilization and thus provide the vital plasma stability. The remote handling system also developed at the facility will allow a proper maintenance within the ITER hot cells.

Mission 4: Technology and Physics of Long Pulse and Steady-State:

The plasma stabilization by the ECRH system as referred to under “Mission 2” is the key to reach long pulse duration up to near steady state operation in ITER. As mentioned above, the Launcher Structural Test Facility allows the development of suitable assembly and remote handling maintenance procedures, as well as the necessary acceptance test programme before the shipping of the Upper Launcher to ITER.

Five year perspective: Within the next five years the Launcher Structural Test Facility will used for prototype testing, design development (including remote handling tools and procedures), and assembly tests/component alignment. Further it is planned to perform a part of the acceptance tests for components manufactured at industry in the facility which cannot be done effectively at the manufacturer.
10 year perspective: On the ten years time scale, further the launcher delivery to ITER will include a set of acceptance tests at the assembly site which is planned in the Launcher Structural Test Facility.

FORWARD PLANNING

In 2008, the Launcher Structural Test Facility will be extended to provide a platform for remote handling development. Tools and procedures will be developed into a system offering a fast and reliable maintenance scheme. In parallel, acceptance test programmes for the different components of the Upper Launcher will be defined. After an upgrade of the Launcher Structural Test facility in 2010, manufactured components will be tested following the programme as defined. A further upgrade in 2011 will allow the assembly of the complete Upper Launcher before the final acceptance tests and the installation in ITER.

FACILITITY	**Materials Institute, Forschungszentrum Karlsruhe**
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: ………………………. (Over the last 15 – 20 years acquired and depending on necessity upgraded) Experimental units: - 50 instrumented creep testing machines, ~ 3.0 M€ - 6 HT universal testing machines (Instron, MTS, Zwick), ~ 3.6 M€ - 3 univ. testing machines and 2 Charpy test machines (Zwick, Schenk), ~ 2.0 M€ - 1 biaxial test facility, ~ 2.0 M€ - 8 thermo-mechanical testing machines, ~ 3.0 M€ - 1 liquid metal loop (Picolo-Loop), ~ 1.4 M€ - 11 processing units for functional materials (Be, BeTi, Li_4SiO_4), ~ 2.0 M€ Characterisation units: - 2 TEM's (Tecnai, CM 30), ~ 2.5 M€ - 1 SEM with EDX unit (FEI), ~ 0.8 M€ - 1 Dual Beam FIB with EBSD unit (FEI), ~ 1.2 M€ - Metallography Lab with optical microscopes etc., ~ 1.1 M€
	COST OF FORESEEN UPGRADES: Year of foreseen investment, cost in 2007 euros: In 2008 / 2009: - Upgrade of a servo-hydraulic testing machine with a high temperature vacuum furnace (1600°C) and a 3-point bending machine for experiments under higher temperature, acquisition of an ultrasonic crack detector, ~ 0.5 M€ - Upgrade of processing units for functional materials, ~ 0.2 M€ - Acquisition of a HT-creep facility (1300°C), ~ 0.2 M€
	OPERATION - average number of operation days/year (over the past 4 or 5 years): 220 days/year - yearly cost of operation in 2007 euros: ~ 1.8 M€ (including personnel) - yearly manpower for operation in ppy: 8
USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)	Number of facility users: ~ 40 Yearly integrated equivalent full time facility users in ppy: ~ 32
	Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~ 150
	Number of yearly publications based on experimental results from facility:

Year	Journals	Conferences
2004	~ 30	~ 50
2005	~ 30	~ 50
2006	~ 30	~ 50

COLLABORATIONS	Collaborations inside EU: EURATOM / EFDA: all major associations Collaborations outside EU: UCSB (USA), JAEA (Japan), PSI (Switzerland), ITEP (Russia), ISSP (Latvia) etc.
PRESENT TECHNICAL CAPABILITIES	The core capabilities of the Materials Institute are: - Development and qualification of structural and functional materials - Development of fabrication routes and thermomechanical treatments, - Mechanical and micro-structural characterisations, - Modelling including experimental validations - Understanding of structure-property correlations and interfacial engineering of high performance materials For all current activities, the Institute is adequately equipped with a multitude of state of the art machines, technical units (see above) and modelling tools.
FUTURE TECHNICAL CAPABILITIES	In 2008 /2009: Enlargement of some technical devices concerning the development, fabrication and characterisation of high temperature materials (ODS alloys, tungsten based alloys) and functional materials (Be, $Be_{12}Ti$ and Li_4SiO_4 pebbles) .
PROGRAMME: ACHIEVEMENTS	- FZK is leading in the development of structural materials for fusion: The European reference RAFM steel "EUROFER" and, more recently, the RAFM-ODS alloy "EUROFER-ODS" are predominantly based on FZK alloy R&D. - Characterisation of RAFM- steels, ODS alloys and refractory metals / alloys. - Development and modification of models and design rules for

	components. - Fabrication of diffusion welded cooling components at lab and industrial scale. - Fabrication and characterisation of functional materials.

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The Materials Institute addresses **mission 3** (First wall material & compatibility with ITER/DEMO relevant plasmas) and **mission 6** (Materials and Components for Nuclear Operation) of the European Fusion Programme. Presently and in future, the Institute is mainly working on the development and characterisation of structure materials for DEMO (first wall, divertor) and TBM materials – especially on EUROFER, RAF- ODS alloys, tungsten and tungsten alloys and functional materials like Be, BeTi (neutron multiplier) and Li_4SiO_4 (breeder). Additionally a further development of design rules for components of DEMO including experimental validations will be done.

Five year perspective:

Further development of RAF(M)-ODS and tungsten alloys including mechanical and micro-structural characterisations for generating a comprehensive data base.

10 year perspective:

- Enlargement of the activities in direction of materials for high temperature applications for DEMO (RAF- ODS alloys, refractory alloys, SiC_f /SiC).
- Up scaling of materials processing towards industrial fabrications.

Longer term perspective:

Further development of new materials and materials processing for advanced TBM and DEMO reactor applications

FACILITITY	**MEKKA, Forschungszentrum Karlsruhe**
RESOURCES INVOLVED	ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: 1990-1995, 3 M€
	COST OF FORESEEN UPGRADES: Year of foreseen investment, cost in 2007 euros: …………………………… - Installation of a turning mechanism for the magnet (20 tons): 30-40 k€ New measuring equipment suitable for ITER TBM experiments: 30 k€
	OPERATION - average number of operation days/year (over the past 4 or 5 years): 50-60 - yearly cost of operation in 2007 euros: 150 k€ (including personnel) - yearly manpower for operation in ppy: 0.7
USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)	Number of facility users: 5 Yearly integrated equivalent full time facility users: (ppy) 3.5
	Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 7
	Number of yearly publications based on experimental results from facility:

Year	Journals	Conferences
2004		**3**
2005	**1**	**6**
2006	**2**	**6**
2007	**4**	**5**

COLLABORATIONS	Collaborations inside EU: ULB (Brussels), CEA, University of Thessaly and Coventry Collaborations outside EU: UCLA, in the past joint experiments have been conduced with Argonne National Laboratory. Sharing facility with other fields of research: performed MHD research is relevant for any liquid metal device. Prospects: consolidate the collaboration with CEA to support ITER TBM experiments.
PRESENT TECHNICAL CAPABILITIES	MEKKA is a liquid metal magnetohydrodynamic (MHD) laboratory for investigating fundamental MHD phenomena and liquid metal flows in strong magnetic fields. NaK inventory: 200 litres, max flow rate: 28 m^3/h, max pressure head: 9 bar Dipole magnet, max uniform B: 2.1 Tesla in space of 800x480x165 mm Hartmann number, Ha = 500 ÷ 6000 Solenoid magnet, max B: 3.7 Tesla, homogeneous magnetic field dimension: 450 mm Measuring equipment - precise measuring techniques are available for flow rates, electric potentials on test section surfaces and for pressure distribution Up to date control room and monitoring system, improved safety system It is possible to reach experimentally values of the main characteristic non-dimensional parameters of the same order of magnitude as those in liquid-metal fusion blankets.
FUTURE TECHNICAL CAPABILITIES	- The turning mechanism for the magnet will allow analysing ITER safety issues e.g. emergency draining for which a horizontal magnetic field is required. - Up to date techniques for velocity measurement in liquid metal flows under fusion relevant conditions.
PROGRAMME: ACHIEVEMENTS	▪ Fundamental research: - Analysis of MHD flow in a straight electrically conducting circular pipe with transverse **B** to understand entrance and exit effects in a fringing magnetic field.Investigation of stability of laminar MHD flow in rectangular ducts, study of properties of MHD turbulent flows and influence on heat transfer.MHD flow in bends of rectangular ducts and electric flow coupling.MHD flow in sudden expansions of rectangular channels (these are important geometric elements in liquid metal devices as well as in fusion blankets). The results have been used to validate developed numerical codes.Applied research: - Preliminary analysis of pressure drop in a scaled mock-up of HCLL blanket. ▪ Gained experience in measuring techniques for velocity, pressure and electric potential in liquid metal flows. ▪ Development of database of experimental results to be used as benchmark for validating numerical codes. ▪ Successful training and educational programme (PhD/diploma thesis /practice).

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

The Fusion Programme at FZK has been operating successfully for many years the liquid metal magnetohydrodynamic (MHD) laboratory MEKKA with the aim of improving the present understanding of fundamental MHD phenomena and liquid metal flows in strong magnetic fields, as they occur in HCLL and DCLL blankets, and to create a database for validating computational numerical tools.

- As pointed out in the R&D strategy of the European Fusion Research Programme, fusion development requires also basic physics research. Fundamental research in magnetohydrodynamics and thermodynamics has been performed in the MEKKA laboratory, showing the importance of this facility to improve understanding of MHD flows and to create a "broad knowledge base in support of R&D Missions". This experimental activity contributed to raise questions and ideas for future studies.

- At the present time experiments in a MHD mock-up of ITER HCLL TBM are being carried out with the aim of assessing the effects of magnetic field intensity and flow rate on pressure drop and flow distribution in blanket modules. Dedicated experiments are fundamental for predicting TBM performance and selecting a proper blanket design (Mission 6).
- The obtained experimental results are used as benchmark problems against which numerical codes are validated. They have been successfully used to test a code developed at FZK for prediction of liquid metal MHD flows in magnetic fields of various strength and in arbitrary geometries. The database resulting from the ongoing experimental campaign will support the development of a 3D multi physics code for simulation of liquid metal MHD flows, coupled with heat and mass transfer problems, including tritium transport. This numerical tool will be the result of a collaborative research activity (CEA, University of Brussels and other European partners). A predictive code is required for planning and analyzing the experiments performed in ITER (see Mission 5).
- The planned experiments and the development of new measuring techniques will help to design suitable instrumentation and to define tests for ITER TBM testing programme. This latter should set clear objectives for the experimental campaign, such as measured outputs, figure of merit to characterize the performance of ITER TBM and to develop a procedure for extrapolating the results to DEMO reactor.
- The new experiments will consider flow channel inserts in electrically conducting ducts to reduce MHD pressure drop in blanket modules. Tests for qualification of insulating materials will be performed.

During the next 10 years the laboratory will contribute to the improvement of the HCLL TBM design, optimization of flow path, prediction and reduction of MHD pressure drop. It will help to define further R&D issues and experiments to be performed in preparation for ITER TBM operation. Moreover, recent studies for a long-term power plant project showed the high potential of dual coolant blankets. These latter operate in a completely different parameter range compared to HCLL blankets so that for a DEMO reactor additional experimental and theoretical investigations of MHD flows will be required.

Five year perspective: Activities related to ITER TBM experiments, suitable measuring techniques and instrumentation. Inputs to safety concerns related to emergency draining of liquid metal blankets. Supply benchmark experimental data for validating computational design tools.

10 year perspective: Experiments related to DCLL blanket concept. Here the liquid metal flows at higher velocity compared to HCLL blanket and insulating flow channel inserts are necessary.

Longer term perspective: supporting of R&D for DEMO engineering design activity, final selection of blanket concepts.

FORWARD PLANNING

See above.

<table>
<tr><td>FACILITITY</td><td colspan="3">PICOLO, Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: Base investment in 1985: 0.3 M€
subsequent upgrades till today: 0.7 M€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros:
Modification of test section for very low flow velocities: 25 k€ (2008)
Additional control systems (flow rate): 25 k€. (2008)</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 300 days/y
- yearly cost of operation in 2007 euros: 275 k€ (including personnel)
- yearly manpower for operation in ppy: 2 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
6</td><td>Yearly integrated equivalent full time facility users (ppy): 3</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 2</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>2</td><td>4</td></tr>
<tr><td>2005</td><td>1</td><td>3</td></tr>
<tr><td>2006</td><td>2</td><td>2</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: EURATOM Associations (CEA, ENEA, IPP.CR, IPP-Garching)
Collaborations outside EU: JAEA/Japan, KAERI/Korea, FDS/China
Prospects: Reference loop for Pb-17Li corrosion in Europe</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">PICOLO is a forced convection Pb-17Li loop for corrosion testing of bare and coated RAFM-steels at temperatures up to 550°C and flow velocities up to about 0.3 m/s and also used for analyzing transportation effects of corrosion products and their precipitation behaviour. Total test duration since start-up ca. 150,000 hours.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">- Extended testing campaigns at lower flow velocities at around 1-2 cm/s and increase of reliability of modeling tools by testing at different flow regimes (laminar, mixed, turbulent)
- Qualification of coated samples for qualification of barrier systems in Pb-17Li
- Long-term corrosion testing of coated loop components for the first time
- Begin of testing of new ODS steels.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">- Elaboration of a corrosion data base for RAFM steels (e.g. Manet, F82H-mod., Optifer, EUROFER) in flowing Pb-17Li at 480 and 550°C (kinetic analysis)
- Determination of corrosion mechanisms
- Evaluation of corrosion rates vs. temperature
- Code development for compatibility of FM-steels in dynamic Pb-Li systems
- Model development for simulating transport and precipitation phenomena.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

PICOLO is a facility supporting the experimental activities for addressing **mission 6** (Materials and Components for Nuclear Operation).

Reduced activation ferritic-martensitic steels (e.g. EUROFER) or ODS-steels are considered for application in future fusion technology as structural materials for e.g. blanket components, which are in contact with the breeding material Pb-17Li. Beyond their compatibility with the breeding material also the T-permeation behaviour will be essential for a successful application in fusion technology which will therefore require both corrosion and T-permeation barriers. The development of these materials and coatings incl. the coating process technologies require a detailed characterization and qualification.

PICOLO is a forced convection Pb-17Li loop for corrosion testing of bare and coated RAFM-steels at temperatures up to 550°C at the moment and flow velocities up to about 0.3 m/s and is also used for analyzing transportation effects of corrosion products and their precipitation behaviour. PICOLO is the reference loop for Pb-17Li corrosion testing in Europe and thus together with the existing data base an essential tool on the path for qualification of materials and functional scales for DEMO and for validation of modeling tools.

Five year perspective:
The near term perspective will be focused on corrosion testing at low flow velocities of Pb-17Li to extend the existing data base by corrosion values at mixed and laminar flow conditions and to collect more data on the newly started topic examination of transportation and precipitation phenomena. Parallel to these testing of bare materials existing coatings on Al base – deposited by organic electrochemical methods – will be analyzed concerning their compatibility with Pb-17Li. This series will be completed by testing of coated loop segments (corrosion barriers).
10 year perspective:
The medium term perspective is the testing of more advanced coatings for corrosion and T-permeation barriers. The deposition will be based on new electrolyte systems which allow the homogeneous coating of large and complex shaped structures under industrial relevant conditions from ionic liquids on single or multi phase composition e.g. Al, W, refractory alloys.
Longer term perspective:
is the additional characterization of ODS materials in bare and coated form in PICOLO under DEMO relevant conditions.

FORWARD PLANNING

The detailed planning based on the obvious needs in TBM development for testing scenarios in the forced Pb-Li loop PICOLO is well embedded into the material development programme towards a fusion power plant.
The entering of the near term perspective will be possible without any changes of the PICOLO facility.
The proposed facility upgrade will enable the testing at rather small flow rates (like in TBM's) inclusively the testing of coated loop segments for the first time under real conditions. The medium and long term programme will also integrate any feedback from coating and ODS steel development and design for DEMO.

<table>
<tr><td>FACILITITY</td><td colspan="3">TIMO-2 Cryovacuum Test Bed, Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: 1995, 0.2 M€; 1998, 0.9 M€; 2000, 1.5 M€; 2006, 0.2 M€; 2007 0.3 M€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros: 2008/09, 0.4 M€ (Upgrade to decouple facility operation from competing users of the cryosupply facility)
2011, 0.8 M€ (Integration of the ITER sized forepump train)
2013, 0.7 M€ (Instrumentation and test environment for cryomechanical performance testing for DEMO)</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 130 d/a
- yearly cost of operation in 2007 euros: 450 k€ (including personnel)
- yearly manpower for operation in ppy: 2.0</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
TIMO-2 is not a user facility but is used for the execution of the FZK R&D programme on cryopump development for nuclear fusion.</td><td>Yearly integrated equivalent full time facility users: (ppy)
0-5 (depending on the R&D programme)</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 2</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>4</td><td>12</td></tr>
<tr><td>2005</td><td>7</td><td>14</td></tr>
<tr><td>2006</td><td>8</td><td>14</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: IPP Garching, CEA-Grenoble, CEA-Cadarache, UKAEA, ENEA-RFX, Volos Greece
Collaborations outside EU: IPR India, Kurchatov Institute Russia, Efremov Institute Russia, IPC Moscow, LANL US, ORNL US
Prospects: excellent as being a unique test facility</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">Simulating 1:1 ITER cryogenic conditions in terms of mass flows and temperatures (200 g/s supercritical helium at 4.5K@ 4 bar, 160 g/s helium at 80K@ 18 bar), licensed for hydrogen operation (explosion safety), covering a wide pressure range (0.2 MPa to 10^{-9} Pa) in a large test volume (10 m^3), simulating torus vacuum, NBI, cryostat and leak detection conditions.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Cryo-supply at reduced temperatures with supercritical helium at 4.3K, flexible cryo-connection concept for versatility in testing different components, e.g. the pellet injector, full vacuum pump system including tunable mechanical and cryogenic switchover.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">- Demonstration of cryo-sorption as reference pumping concept for ITER
- Complete characterisation of the ITER cryo-sorbent and its vacuum performance under ITER-relevant conditions
- Validation of the torus cryopump design with integrated inlet valve
- Confirmation of the cryopump fast regeneration cycling and staggering pattern
- Providing experimental data for benchmark of Monte Carlo and thermohydraulic codes
- Determination of cryopump behaviour under off-normal events (LOVA)</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

Mission 1 (Burning plasmas): Burn control and efficient helium ash removal via the divertor and its recycle flows is a key aspect for maintaining a burning plasma. Helium pumping is expected to play the key role in burning plasma control. This has to be achieved by advanced pumping systems with tunable pumping which can be developed only in TIMO-2. The second key area for burning plasmas is plasma disruption, its mitigation and especially the associated recovery times which are essential issues for a commercial reactor. They depend significantly on the performance of the vacuum pumping system under the conditions following a massive gas injection event. The TIMO-2 facility will provide benchmark data for viscous operation of cryopumps (fast recovery cannot be achieved by any other pumping system).

Mission 4 (Technology for steady-state): In order to allow long pulse and steady-state operation of a DT magnetic fusion device, it would be beneficial to replace cryogenic pumps which require frequent cyclic operation, especially for DEMO with its large plasma. If higher divertor pressures become acceptable for DEMO, the integration of tritium-compatible mechanical pumps to take over the major pump duty during burn would be extremely beneficial. The interaction between mechanical and cryogenic pumping can be studied in full scope and size at TIMO-2. This progress in pumping technology is a pre-requisite to come to manageable long pulse/steady-state operation without compromising the high burn control capabilities of today´s cryopump system.

Mission 7 (High availability): The vacuum pumping system is a core part of fusion technology without which the machine is not operable. High reliability and availability has to be implemented and will be taken as design driver for the next generation vacuum pumping systems.

Finally, as being the only versatile and full scale cryogenic and mechanical vacuum test bed in the fusion community worldwide, TIMO-2 is expected to take over the leading role to educate and train young researchers.

The future programme of TIMO-2 will be an essential complement to the vacuum pump system experience gained at ITER. The TIMO-2 facility represents an important contribution to meet the goals of three out of the 7 R&D Missions and will demonstrate a sound solution to fusion pumping issues within the next 10 years.

FORWARD PLANNING

Milestones in preparation to ITER assembly (5 years):

- Detailed testing of the 1:1 scale torus prototype cryopump for ITER (under ITER credit through F4E) accomplished
- Tests of a NBI cryopump model (under ITER credit through F4E) finished.

Milestone in preparation to ITER operation (10 years):

- Functional testing of the serial torus cryopumps for ITER performed

Milestone in preparation of DEMO construction (20 years):

- Usage of the TIMO-2 test environment for development of hybrid large tritium compatible mechanical and cryogenic pumping systems to have an option to replace a large portion of the cryogenic pumps for DEMO and, thus, to provide better commercial features to fusion energy. This next generation pumping system adds considerably to achieve better burn control, reduced operational efforts during long-pulse conditions and increased efficiency and RAMI (reliability, availability, maintenance) performance.

<table>
<tr><td>FACILITITY</td><td colspan="3">Tritium Laboratory Karlsruhe (TLK), Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: 40 M€ investment 1995 plus 10 M€ subsequent upgrades from 1995-2007</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros: 2015-2018, 8 M€
Extension of building and infrastructure for a He and tritium test facility (“He and T test facility”) to extract tritium from tritium breeding modules</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 365
- yearly cost of operation in 2007 euros: 4M€*(fusion 3.2M€, including personnel)
- yearly manpower for operation in ppy: 13* (fusion 10.4)

*) The cost for operation and the manpower are the total values; the fusion part is about 80%</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
22 from TLK, 3 trainees, 3 students, 6 guests (scientists e.g. from Japan), total: 34</td><td>Yearly integrated equivalent full time facility users: (ppy)
21 (TLK) + 2 (trainees) + 3 (students) + 0.4 (guests), total: 26.4</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 8 PhD / 5 diploma</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>13</td><td>20</td></tr>
<tr><td>2005</td><td>36</td><td>8</td></tr>
<tr><td>2006</td><td>10</td><td>11</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: EFDA Technology Programme incl. JET (CEA, ENEA, MEC, HAS, UKAEA), U Heidelberg + U Karlsruhe
Collaborations outside EU: NIFS, Japan; AECL, Canada; Kyushu University, Japan; Toyama University, Japan
Number of experimental contributions to ITPA: not applicable
Sharing facility with other fields of research: About 20% of the infrastructure is used to provide the necessary tritium for the international experiment KATRIN to determine the neutrino mass by measuring the ß-decay spectrum of tritium
Prospects: Collaborations are expected to be continued, respectively expanded.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">The TLK is an almost unique technical facility and has many similarities with the proposed ITER Tritium Plant, since it has a closed tritium cycle with a storage system, a plasma exhaust processing system for the recovery of tritium, an isotope separation system and many auxiliary systems, such as a tritium transfer system, analytics and tritium retention systems. TLK has a licence to handle 40 g tritium (presently 24 g on site). Ten glove boxes with a total glove box volume of 125 m^3 are currently available to serve as a second containment for tritium facilities. Tritium analytics is being done with 3 calorimeters, 3 gas chromatographs (GC), 1 omegatron, 2 quadrupole mass spectrometers, 2 liquid scintillation counter (LSC) systems and 1 IR spectrometer. The TLK is a multi-purpose R&D facility; it can easily be adjusted to new R&D tasks due to its modular setup.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">- Systems for processing of low, middle and highly tritiated water
- He+T test facility for tritium extraction from large He flows
- Additional analytics: IR spectroscopy of hydrogen isotopomers at about 20 K, Laser Raman spectroscopy of gaseous hydrogen isotopomers, micro-GC for fast measurements</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">TLK participates in the EU fusion programme since the fusion technology project was launched at FZK in 1983. In accordance with its mission TLK has developed a DT fuel cycle for fusion reactors including storage, plasma exhaust processing, water detritiation and isotope separation as well as appropriate analytical devices. The 3-step CAPER process for plasma exhaust processing is now the reference process for ITER. The pilot plant TRENTA (water detritiation system (WDS) combined with isotope separation system (ISS)) provides the necessary experimental background to deal with the EU procurement package “WDS-ISS” for ITER.. Another important item is the successful operation of a closed tritium cycle at TLK to enable multi-purpose R&D with tritium (purity up to 99%, up to 40 g). The experience gained during 14 years of tritium operation with the TLK facility is the basis of both, the future design work for ITER and the future R&D for DEMO.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

Five year perspective:

- ITER: TLK will have the leading role with regard to design and management of the EU procurement package WDS-ISS for ITER (under ITER credit through F4E). TLK will perform additional R&D as required in the framework of the procurement package.
- DEMO / fusion roadmap: *Reliable Tokamak Operation* (mission 2) as well as the development of key components for the tritium fuel cycle (mission #6, *Materials and Components for Nuclear Operation*) requires the knowledge of reliable and safe handling of tritium and the long term experience in design and construction of fuel cycle components. Based on its experimental background of up to now more than 14 years of tritium operation with more than 20 g tritium TLK will identify and deal with open items related to DEMO fuel cycle (e.g.: general design, procedures for safe handling, quality assurance, upgraded fuel cycle components, especially R&D on “He and T test facility” for recovering tritium from large He streams).

10 year perspective:

- ITER: continuation of 5 years perspective with the aim to finish the installation and commissioning of the EU procurement package WDS-ISS at ITER site.
- DEMO: Taking leadership for the design of the whole tritium fuel cycle of DEMO. Performing R&D on all related fields in particular with regard to advanced requirements for DEMO compared with those for ITER for the key components of the fuel cycle to ensure *Reliable Tokamak Operation* (mission 2). In this context TLK will be upgraded to house a “He and T test facility” to enable integrated tests of coolant purification system and tritium extraction of future helium cooled breeding blankets. Tests in such a facility will make sure that a reliable and sufficient tritium recovery is available (mission 6, *Materials and Components for Nuclear Operation*). Another aspect is the minimization of tritium release into the environment by optimization of the WDS-ISS process, since at the moment, the major release of tritium into environment is being done via the WDS.
- Training of scientists, engineers and technicians at the EU wide unique tritium laboratory.

Longer term perspective:

- DEMO: Tritium technology will be a crucial item towards the development of a fusion power reactor. Low tritium emissions, safe and reliable components are a pre-condition for the acceptance of fusion technology in the future. Taking into account the advanced requirements for DEMO a conceptual design of the tritium fuel cycle for the DEMO reactor will be developed until 2017 followed by a detailed design of the tritium fuel cycle.

FORWARD PLANNING

- T0+5 y:
  - Performing R&D according to the requirements of the EU procurement package WDS-ISS
  - R&D for processing of water with high tritium concentration not covered by ITER WDS-ISS (up to 1.4 MBq/kg) using either PERMCAT (permeator combined with catalyst) reactor or VPCE (vapour phase catalytic exchange) column. The goal is to install at least one loop at laboratory scale at TLK and demonstrate the feasibility of the process.
  - Helium cooled breeder blanket: Assessment of DEMO requirements and comparison with existing R&D results to identify open items for further R&D; already clear is the necessity to setup and operate a “He and T test facility” because full scale experiments with realistic DEMO helium throughputs (3000 kg/s) have not been done up to now. Continuation of former R&D work with scaled down mock-ups (cold trap, cryogenic molecular sieve bed) to get input for design of the “He and T test facility”.
  - Performing of laboratory scaled test experiments to optimize tritium fuel cycle components and processes with regard to DEMO (in particular analytical tools, waste management, accountancy)
- T0 +10 y
  - Completion of work with regard to EU procurement package WDS-ISS for ITER
  - Experiments with highly tritiated water to get experience and data for scaling up to DEMO
  - Design and construction of “He and T test facility”
  - Design of DEMO fuel cycle: adjustment of processes to DEMO requirements to cope with the increased requirements of DEMO with regard to throughputs, inventories and releases into environment; simplification of components to ensure long life time and high availability, which is necessary for a reliable tokamak operation; in addition, the usage of robust and simplified components leads to reduced costs.
  - Release of tritium into environment: Minimization of tritium release from WDS-ISS by optimization of the process.
  - Waste management: Development of accountancy methods for wastes (e.g. use of calorimetry)
  - Establishment of a regular fuel cycle trainee programme to ensure the availability of a sufficient number of experts for a safe and reliable tritium operation within DEMO

- QA and tritium safety: Defining guidelines and working procedures based on own long term experience in operation of a large and versatile tritium laboratory.

<table>
<tr><td>FACILITITY</td><td colspan="3">TOSKA, Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in euros:
1982 TOSKA with 50 t crane and 300 W refriger. 5900 k€
1988 Power Supply 30 kA & Switching Circuits 1200 k€
1993 Power Supply 50 kA & Crane 80 t 1300 k€
1993 Refrigerator 2 kW with purifier & gas storage 10000 k€
1993 TOSKA Upgrade 5000 k€
2000 Switching Circuit 80 kA 1000 k€
2000 Power Supply 20 kA 600 k€
Total: 25,000 k€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros:
2007/2008 TOSKA Upgrade 2700 k€</td></tr>
<tr><td colspan="3">OPERATION (TOSKA was slowed down 2002 after TFMC test and reactivated in 2007 to prepare W7-X coil testing)
- average number of operation days/year (over the past 4 or 5 years): 90 d/y
- yearly cost of operation in 2007 euros (including ppy): 2.3 Mio €
- yearly manpower for operation in ppy: 11 ppy</td></tr>
<tr><td rowspan="7">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users:
TOSKA is not a user facility but used for the execution of the FZK fusion R&D program in the frame of EURATOM</td><td>Yearly integrated equivalent full time facility users: -</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 5</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>9</td><td>5</td></tr>
<tr><td>2005</td><td>6</td><td>5</td></tr>
<tr><td>2006</td><td>7</td><td>7</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: CRPP, CEA, ENEA, IPP, Universities Polito Torino, Udine, Twente, Bratislava
Collaborations outside EU: NIFS (Toki, Japan), LLNL (Livermore, USA)
Number of experimental contributions to ITPA: -</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">TOSKA was built in 1982 to host the test of the EURATOM LCT coil that was constructed in the frame of the Large Coil Task.
In the following years the capabilities of TOSKA were subsequently upgraded.
Key parameters are actually:
– 2 KW refrigerator 4.5 K He, 300 W refrigerator down to 1.8 K He;
– 80 kA power supply and switching circuit; 50 kA, 30 kA, 20 kA power supplies
– 120 t crane
– Large scale cryostat (free inner dimension: diameter 4.3 m, 6.5 m height)
TOSKA can be used for large scale experiments that need extensive cryotechnical equipment and/or high current capabilities, e.g. tests of large flow supercritical (4.5 K) He circuits or tests of large superconducting magnets.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">– High resolution data acquisition and control (e.g. 40 channels with 500 kHz/16 bit and dedicated channels with 24 bit resolution)
To speed up the coil testing frequency, integration of fast evacuation system and new internal current joint pads is foreseen</td></tr>
<tr><td>PROGRAMME ACHIEVEMENTS</td><td colspan="3">• Test of EURATOM LCT coil (10 kA @4.5K)
• Test of Poloidal Model coil (POLO) (30 kA@4.5 K)
• Test of LCT coil at 1.8 K (19 kA @1.8 K)
• Test of W7-X Demo Coil (16 kA @ 4.5 K)
• Test of ITER Toroidal Field Model Coil (TFMC) alone and TFMC with LCT (ITER EDA L-2 Project) (80 kA for TFMC and 16 kA for LCT in parallel)
• Demonstration of HTS Current Lead for ITER (80 kA max. current with 50 K Helium, 68 kA with Nitrogen cooling)</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

TOSKA is needed to support ITER and the R&D work which is necessary for DEMO.

In the sense of mission 2 "*Reliable Tokamak Operation*", it is indispensable to use reliable and efficient He (4.5 K, supercritical) pumps. For ITER He pumps are necessary which have a high efficiency of >70%, a large massflow >2 kg/s and a large pressure head (increase) of 1.5 bar. A pump which combines these features is not available on the market and needs R&D for development and specific qualification tests. Such a special pump has to be optimized in efficiency to minimize refrigerator power and save electrical operating costs.
A test facility has to provide a complex and large He cryo-infrastructure which is available at TOSKA.

In the sense of mission 7 "*DEMO Integrated Design*", TOSKA can support DEMO by contributing to the coil development and testing:

- To check the potential of classical superconductors, the construction of a test coil using advanced Nb3Sn is actually discussed. Such a coil should be tested in TOSKA using the existing infrastructure (no upgrade necessary).
- The potential of high temperature superconductors is under investigation for use in coils for DEMO. After the development of cabling/bundling techniques for fusion relevant cables and conductors, a demonstration-solenoid shall be designed and constructed. This solenoid can be tested in TOSKA using the Nb3Sn coil discussed above to create a background field.
- After decision whether classical or HTS superconductors shall be used to build coils for DEMO, the TF model coil will be designed and constructed. This coil can be tested in TOSKA.

Five year perspective: Until 2009 test of W7-X coils, 2010-2012 Test of ITER He-pumps
10 year perspective: Tests of advanced Nb3Sn demonstrator coil, preparation for HTS demonstration- solenoid
Longer term perspective Test of HTS demonstration-solenoid, test of TF model coil for DEMO

FORWARD PLANNING

Test of newly developed ITER He pumps (2010 →)
Preparation and test coil with advanced Nb3Sn cable (2012 →)
Preparation and test of HTS demonstration-solenoid (2015 →)
Preparation and test of TF model coil for DEMO (2020 →)

<table>
<tr><td>FACILITITY</td><td colspan="3">HYDEX - Hydrogen and Dust Explosion Facility -, Forschungszentrum Karlsruhe</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros: 2004 – 2007; 2500 k€ (re-using equipment that newly bought would have amounted to ~ 6 M€)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Year of foreseen investment, cost in 2007 euros: 2008 -2009: 650 k€ for large test vessel (220m3), new instrumentation (laser diagnostics and other), 200 k€ for test tube extensions
2009: 300 k€ for facility modification for sub-atmospheric tests and addition of a high performance evacuation system, 150 k€ for test tube instrumentation</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 160
- yearly cost of operation in 2007 euros: 1000 k€ (including personell)
- yearly manpower for operation in ppy: 9.0</td></tr>
<tr><td rowspan="8">USE OF FACILITY (for magnetic confinement devices and, when appropriate for technology facilities)</td><td colspan="2">Number of facility users: HYDEX is not a user facility but is used for the FZK R&D programme on safety of fusion reactors and hydrogen safety.</td><td>Yearly integrated equivalent full time facility users: (ppy) N. a.</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 6</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td>2</td><td>2</td></tr>
<tr><td>2006</td><td>3</td><td>3</td></tr>
<tr><td>2007</td><td>2</td><td>3</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: CEA-Saclay, Univ.of Karlsruhe, Univ. of Munich
Collaborations outside EU::Kurchatov Institute, Moscow
Prospects: Collaboration with CEA-Saclay on dust/hydrogen combustion mitigation systems for ITER</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">HYDEX is an ensemble of facilities for dust/hydrogen combustion studies, ranging from a 20 l experimental vessel to a 100 m^3 volume large scale explosion chamber, allowing measurements of important phenomena like dust mobilization, dust combustion, and hybrid dust/hydrogen explosions on very different geometrical scales. In particular, it provides unique experimental capabilities for investigation and control of combined hydrogen/dust explosions in potential ITER accident scenarios, e.g. air ingress into the plasma chamber. Test results on a large ITER relevant scale are needed for the validation of three-dimensional computer codes which are being developed for predictive analysis and mitigation of reactive events in ITER.
<u>20-l-sphere</u>: up to 30 bar overpressure; determination of standard explosion indices like maximum explosion overpressure and pressure rise rate for dust/hydrogen/air mixtures. <u>Tube facilities of 35 l and 300 l volume</u>: Design pressures of150 and 300 bar, respectively. Measurement of flame propagation velocities and overpressures in turbulently mobilized and burning dust/hydrogen/air mixtures. <u>100m^3 explosion chamber</u>: 12 m long, 60 bar static pressure capability; ITER relevant scale experiments; code validation.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">A new larger test vessel with 220 m^3 of volume will be added (2008) and equipped with innovative instrumentation (2009) for time and space resolved measurement of the complex 3D flow and combustion processes. The experiments will provide important data like flame propagation speeds as well as practical results like local pressure loads on the confining structure (the ITER design pressure is 2 bar). The capability for large scale tests at sub-atmospheric pressures will also be developed (2009). Tube extensions to 50 l and 1200 l volume, resp. are planned for 2008 and 2009.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">- Validation of three-dimensional computer codes for simulation of dust mobilization, dust-air combustion and combined hydrogen-dust-air explosions in ITER (on-going).
- Measurements of dust and hybrid H_2-dust reactions in 3 small and medium scale facilities (part of HYDEX), employing different graphite, W and C-W dust mixtures.
- Determination of limiting oxygen concentrations needed to suppress fast combustion modes in hydrogen-dust-air mixtures, use of the results for design of a potential mitigation system for ITER.
- Currently, dust explosion processes with graphite and tungsten dusts of different size distributions with and without hydrogen have been investigated in three small and medium scale facilities which are part of HYDEX.</td></tr>
</table>

PROGRAMME: ADDRESSING THE PROGRAMME NEEDS

This research addresses Mission 2, reliable tokamak operation, because the Generic Site Safety Report for ITER has identified accident sequences which can lead to co-existence of combustible substances (hydrogen, dust) with oxygen (air ingress). Since an ignition is difficult to exclude, a potential for combustion processes exists, which could lead simultaneously to mobilization of radiological substances (T, dust), increased leakage areas (breaks), and transient pressures above 1 bar (venting of plasma chamber). Combustion could therefore cause potentially high radiological source terms including gaseous tritium and dust particles. For risk management and investment protection a detailed understanding and theoretical modelling of hybrid dust-hydrogen combustion is needed. 3d codes for numerical simulation of dust mobilisation, transport and combustion are also required to identify feasible and effective mitigation measures.

The development of such 3d simulation tools will require experimental data on different geometrical scales to validate the theoretical models and to prove the scaling ability for ITER conditions. Several small and medium scale facilities for investigation of hydrogen-dust explosions have been constructed at FZK and integrated into the HYDEX facility. In addition, HYDEX contains several large scale test vessels which allow obtaining data on ITER relevant scale for hydrogen-air combustion, dust mobilization, dust-air explosions, and combined hydrogen-dust-air reactions.

Experiments in HYDEX are necessary to clarify the governing physical phenomena on large scales and to verify the theoretical models and numerical tools for reliable ITER predictions. They are an important contribution of the European Fusion R&D programme for the licensing of ITER in the coming 10 years and safe operation afterwards.

FORWARD PLANNING

2008: addition of a new larger vessel ($220m^3$)
2009: purchase and installation of instrumentation, preparation for tests at sub-atmospheric initial conditions
2010: large scale experiments on dust mobilization and dust-air combustion
2011: large scale experiments on combined hydrogen and dust combustion,
2012: test of mitigation measures by injection of inert gases or chemically active flame suppressants, catalytic recombiners, deliberate ignition (spark igniters) or combinations of these approaches.

Facility	**Plasmatron VISION I, The Belgian Nuclear Research Centre SCK•CEN**
Resources involved	Original investment and subsequent upgrades: Investment year: 2007 recovered (ISPRA JRC) and will be refurbished in 2008 Cost of first investment: 15 + 255 k€
	Cost of foreseen upgrades: 1) Be/T operation: Year: 2010; Cost: ~100k€ (gloveboxes, safety (tritium, beryllium control), diagnostics, remote control, instrumentation, …) 2) Irradiated materials: later; Cost: up to 1000 k€ (shielding / hotcell, waste, ...)
	Operation: - average number of operation days/year (over past 4-5 years): Not applicable - yearly cost of operation in 2007: not applicable since not yet in operation (estimation: ~188 k€ in the future) - yearly manpower of operation: not yet defined (estimation: ≤1 ppy +1 tpy) - estimated yearly maintenance cost: 41 k€
Use of facility	not applicable since not yet in operation
Collaborations	Foreseen prospects: open EU facility, strong interaction with JET (mixed, Be, T contaminated materials), Collaborations with PISCES-B in US and Magnum-PSI in Netherlands, Possible contributions to ITPA for mirrors/windows (effect of Be)
Present technical capabilities	Factual set of parameters (from archive papers): quasi-steady state plasma simulator ▪ Originally designed for PWI studies in fusion ▪ Various gas mixtures ▪ Volume: 18 litres ▪ Target diameter: ~25cm ▪ Target temperature: RT - 600°C (heating/cooling) ▪ Cold self-sustained volumetric plasma ▪ Ion energies: 20 - 500 eV ▪ Magnetic field: 0.2T ▪ Pulse duration: steady state (above 100 sec) ▪ Flux density target: ~ 10^{20}-10^{21} ions/m^2.s ▪ Installed in glove box
Future technical capabilities	New after upgrade: include planning - Be/Tritium materials capability: 3-5 years (or more if needed) - irradiated materials capability: 5-10 years (start date depend on shielding requirements)
Programme: achievements	Not applicable
Programme: addressing the programme needs	7 R&D missions and fusion roadmap: Strong contributions directly to missions 3 & 6. <u>Mission 3</u>: FW materials & compatibility with ITER/DEMO relevant plasmas <u>Mission 6</u>: materials and components for nuclear operation (PFM have plasma as added constraint) The **Plasmatron VISION I** it is capable to investigate: - influence of long pulse operations on PFM - Be/T and irradiated materials - mirrors/windows and diagnostic studies (erosion/redeposition/dust) - erosion/redeposition (dust) studies - tritium retention/inventory/trapping/implantation/aging/… - tritium/codeposited (dust) in-situ removal methods **Five to ten years:** - ITER alternative divertor armour material study (i.e. Tungsten), including research on irradiated materials (erosion, embrittlement, retention, ... under plasma contact) - Implication of the mixed materials on erosion/tritium retention/… - Diagnostic technique (mirrors/windows) studies for erosion/deposition (dust) for ITER operation and the influence of plasma and neutrons on their performance **Long term:** - Qualification of structural and functional materials for DEMO - selection of n-irradiated materials that are optimized for PWI (tritium retention, embrittlement, erosion, …) for DEMO
Forward planning	Summary of key elements of timetable and planning - Steady-state plasma simulator (high flux, ion energies 20-500 eV): end of 2008 - Be/T materials capability: from end 2009 - irradiated materials capability: depends on shielding requirements (simple or hot cell)

<table>
<tr><td>FACILITY</td><td colspan="5">Gamma irradiation facilities, The Belgian Nuclear Research Centre SCK•CEN</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="5">Original investment and subsequent upgrades:
Investment year: 1977 Cost of first investment (Rita): 125k€; Brigitte: 250k€;Geuse: 125k€; Kirsten 40k€;</td></tr>
<tr><td colspan="5">COST OF FORESEEN UPGRADES: Brigitte 50k€;</td></tr>
<tr><td colspan="5">Operation:
- average number of operation days/year (over past 4-5 years): 360 days/year
- yearly cost of operation in 2007: 284 k€ (pay attention: not used only for fusion)
- yearly manpower of operation: 5 man-months
- estimated yearly maintenance cost:35 k€</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: 14</td><td colspan="3">Yearly integrated equivalent full time facility users: n.a.</td></tr>
<tr><td colspan="5">Number of PhD/ diploma thesis using experimental data from the facility in the last 5-10 years:
5 diploma thesis ; 5 PhD thesis</td></tr>
<tr><td colspan="5">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td colspan="2">Journals</td><td colspan="2">Conferences</td></tr>
<tr><td>2004</td><td colspan="2">3</td><td colspan="2">2</td></tr>
<tr><td>2005</td><td colspan="2">2</td><td colspan="2">2</td></tr>
<tr><td>2006</td><td colspan="2">3</td><td colspan="2">2</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="5">Collaborations inside EU: CEA, EDF, AREVA, EADS, EFDA…
Collaborations outside EU: JAERI (Japan) and Kurchatov (Russia)
Number of experimental contributions to ITPA: in average 2 contributions / year
Sharing facility with other fields of research: Fission; Space research;
Prospects: Waste research</td></tr>
<tr><td rowspan="9">PRESENT TECHNICAL CAPABILITIES</td><td colspan="5">The main characteristics of the various gamma irradiation facilities are as follows:</td></tr>
<tr><td>Name</td><td>Dose rate kGy/h</td><td>Main dimensions(mm)</td><td>Environment</td><td>Characteristics</td></tr>
<tr><td>Geuse II</td><td>0.01-0.5</td><td>900 X Φ380</td><td>air, inert gas, vacuum</td><td>Temperature control
On line meas.</td></tr>
<tr><td>Brigitte</td><td>0.5-25</td><td>900 X Φ220</td><td>air, inert gas,</td><td>Temperature control
On line meas.</td></tr>
<tr><td>Rita</td><td>1.5</td><td>600 X Φ380</td><td>air, inert gas, vacuum</td><td>Temperature control
On line meas.</td></tr>
<tr><td>Kirsten</td><td>80</td><td>200 X Φ35</td><td>air,</td><td>Off line</td></tr>
<tr><td>CMF Ax</td><td>0.5-20</td><td>900 X Φ80</td><td>air, inert gas,</td><td>Temperature control
On line meas.</td></tr>
<tr><td>Cal</td><td>2</td><td>Gamma Beam Line</td><td>air</td><td>Temperature control
On line meas.</td></tr>
<tr><td>CLARA</td><td>-</td><td>2 m^3</td><td>Humidity /
T°C -40 => 200</td><td>Climate chamber</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="5">Irradiation under active vacuum (10^{-5} mbar) now available.</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="5">- lots of irradiation of diagnostics components and materials (incl. optical components and FO)
- in-situ testing of Remote handling components
- in-situ testing of diagnostics materials with on-line testing
- various potential environment (gas, water, liquid metals, vacuum) and fine temperature control</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="5">The facility address mainly the mission 2 (reliability of operation), 3 (plasma facing components, for windows e.g.), 6 (materials and components under radiations) and 7 (assuring high availability of components and systems)
<u>Five year perspective:</u> continue testing functional materials (glass, fiber optics, insulating materials, etc), and develop a strong knowledge on the qualification and testing of fusion components; Contributing to the design of IFMIF instrumentation and diagnostics.
<u>Ten year perspective</u>: Further involvement in innovative components for DEMO and future power plants. Increasing testing of remote handling components and systems.</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="5">Irradiation under spent fuel depends on the availability of the BR2 reactor, foreseen to work until 2016</td></tr>
</table>

<table>
<tr><td>FACILITY</td><td colspan="3">Tritium laboratory , The Belgian Nuclear research centre SCK•CEN</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
Investments 2005-2009: 1.2 M€</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:</td></tr>
<tr><td colspan="3">OPERATION:
- average number of operation days/year: 200 days/year
- yearly cost of operation in 2007 euros: 250 k€/year
- yearly manpower of operation: 0.6 ppy
- estimated yearly maintenance cost: included in operation</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users:</td><td>Yearly integrated equivalent full time facility users: 6 gloveboxes, 6 fumehoods, 1 processcell</td></tr>
<tr><td colspan="3">Number of PhD/ diploma thesis using experimental data from the facility in the last 5-10 years:
17 diploma thesis ; no PhD thesis</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td>4</td></tr>
<tr><td>2005</td><td>2</td><td>2</td></tr>
<tr><td>2006</td><td></td><td>3</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: JET-CSU (UK), CEA-Cadarache (Fr), FzK-TLK (D), ICIT (Ro), IFIN HH (Ro)
Collaborations outside EU: Kinetrics (Ca), Brush Wellman (USA)
Number of experimental contributions to ITPA: not applicable
Sharing facility with other fieds of research: Potentially Be research
Prospects: Nuclear research institute from St. Petersburg</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">At this moment one 50 m² room is still in operation whilst another room (80 m²) is being refurbished. We are currently licensed to handle 37 TBq (1000 Ci) of tritium. There are 2 fume hoods and a walk in - process cell with a floor area of 11.5 m² for working with tritiated compounds. The extraction rate capacity is maximally 10700 m³/h. There is a continuous working tritium monitoring system for the work atmosphere and a discontinuous system to monitor the release to the stack.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">When the first phase of the tritium lab refurbishment is completed (by mid 2008) we will have the following capabilities: 80 m² controlled area ;1 process cell with a ground surface of 15 m²; 2 gloveboxes; 2 fume hoods; a ventilation system allowing for an extraction rate 15,350 m³/h; a license for a max. tritium inventory of 370 TBq.
When the refurbishment is completed (mid-end 2010) the SCK•CEN tritium laboratory will have following capabilities: 130 m² controlled area; 1 process cell 15 m²; 6 gloveboxes (one for working with tritiated beryllium); 5 regular + 1 walk in fume hoods; extraction rate 15,350 m³/h; license for a max . tritium inventory of 370 TBq</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">A catalyst for detritiated water detritiation had been developed and operated successfully in a pilot installation based on the principle of counter current liquid phase catalytic exchange. This has been reported in following EFDA JET tasks FT 2.3; FT 2.20 and FT 2.21; FT 2.22.
Other achievements for fusion by the SCK•CEN tritium lab have been delivered through EFDA JET tasks: FT 2.15, FT 2.22, FT 2.25, FT 2.26, FT 2.28 and following EFDA Garching tasks: TSW-001/D2, TSW-002 and TSW-003.</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">We target to address the needs for mission 3. More specifically the tritium laboratory can contribute in the research for the dust and tritium removal techniques comprised in the 10 year milestones.
<u>Five year perspective:</u> Within 0.5 years the laboratory will be partially refurbished. In 2010 the refurbishment should be completed and the laboratory will be licensed to work with 370 TBq tritium (vs. 37 TBq now). A number of our gloveboxes will be dedicated for tritiated dust (beryllium) research. The first research programs on tritiated dust, Be and W, detritiation and diffusion of T should be concluded.
<u>Ten year perspective</u>: The process cell with a ground surface of 15m² allows us to built larger scale pilot test installations for dust and tritium removal (ex-situ) detritiation techniques that have been evaluated on a smaller scale in the preceding years. A strong interaction between the VISION I plasmatron and the tritium lab is foreseen.</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Until mid 2008: continuing support in a 50 m² lab with a max tritium inventory of 37 TBq.
As of 2008: 80 m² lab room commissioned for 370 TBq tritium; decommissioning of the 50 m² lab.
By mid/end 2010: Refurbishment completed. Tritium laboratory is 130 m² for handling 370 TBq.
Research programs on tritiated dust and detritiation of PFC's are about to start and last until 2013.
Currently research is ongoing on detritiation of non facing plasma components and tritium leak and diffusion out of vessels for tritium containing waste.</td></tr>
</table>

FACILITY	**Material test reactor BR2, The Belgian Nuclear Research Centre SCK•CEN**
RESOURCES INVOLVED	Original investment and subsequent upgrades: Investment year: 1961-63 Cost of first investment: ~ 25 M€ (~10e9 BEF of 1963); 1st change of Be matrix (1979-80) : 2nd change of Be matrix and global refurbishment (1995-96)
	COST OF FORESEEN UPGRADES:
	Operation: - average number of operation days/year (over past 4-5 years): 120 days - yearly cost of operation in 2007: 15 000 k€ (pay attention: not used only for fusion) - yearly manpower of operation: 680 man-months - estimated yearly maintenance cost: included in operation costs
USE OF FACILITY	Number of facility users: > 20 — Yearly integrated equivalent full time facility users: n.a.
	Number of PhD/ diploma thesis using experimental data from the facility in the last 5-10 years: data not available
	Number of yearly publications based on experimental results from facility: the publications are mostly related to the Post Irradiation Experiments carried out on the irradiated samples.
COLLABORATIONS	Collaborations inside EU: *CEA, EU, IRE, Mallinckrodt, NRG, ….* Collaborations outside EU: *DOE, Japanese Organisations and industries,* Number of experimental contributions to ITPA: not applicable Sharing facility with other fields of research: Fission; radioisotope production; Si doping Prospects: Westinghouse, CEA,
PRESENT TECHNICAL CAPABILITIES	The capabilities and the design of the BR2 are particularly well adapted to the R&D options : - a core with a central vertical 200 mm diameter channel, with all its other channels inclined to form a hyperboloidal arrangement around it. This geometry combines compactness leading to high fission power density, with easy access at the top and bottom covers, allowing complex irradiation devices to be inserted and withdrawn; - a large number of experimental positions of 84 mm with in addition 4 peripheral 200 mm channels for large irradiation devices. Through loop experiments can be installed through penetrations in the top and bottom covers of the vessel; - a remarkable flexibility of utilization: the reactor core configuration and operation mode are adapted to experimental requirements; irradiation conditions representative of those of various power reactor types - neutron spectrum tailoring -; - high neutron fluxes, both thermal and fast (up to 1015 n/cm5.s).
FUTURE TECHNICAL CAPABILITIES	- Industrial production of Silicon doping up to diameter 8". *- MTR fuel qualification EVITA, FUTURE,*
PROGRAMME: ACHIEVEMENTS	- lots of high dose irradiation of materials (structural and functional) for fusion - in-situ testing of structure materials (in-situ tensile, fatigue and creep-fatigue) - in-situ testing of diagnostics materials with on-line testing - various potential environment (gas, water, liquid metals, vacuum) and fine temperature control
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	The facility address mainly the mission 6, for developing and characterizing structural and functional materials withstanding the high neutron fluence in fusion reactors. It addresses also partially mission 7 by analyzing the ways of maximizing the reliability of the components, and mission 3, as it allows synergistic testing of plasma facing materials with neutron and plasma effects (e.g. by a coherent use of the reactor and the plasmatron) Five year perspective: continue testing structural materials, and develop a strong knowledge on one or two materials (Eurofer and W); Contributing to the design of IFMIF test facilities. Ten year perspective: Further involvement in innovative materials for DEMO and future power plants. Increasing testing of functional materials under neutrons
FORWARD PLANNING	In 2008 or 2009, foreseen to operate 6 cycles/year (i.e. 130 days/year) allowing to achieve up to 2.5 dpa/year of irradiation in metals. Operation of the reactor presently foreseen until at least 2018.

<table>
<tr><td>FACILITY</td><td colspan="3">MAGNUM-PSI (FOM, in collaboration with TEC partners)</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">TOTAL INVESTMENT:
10 Million Euro (Status: fully granted)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
Minor upgrading: 0.3 MEuro/yr, continuous.
Major upgrades: not yet detailed. Typically few MEuro every 5 yr.</td></tr>
<tr><td colspan="3">OPERATION
0.5 MEuro/yr operation cost; 1 MEuro/yr basic scientific team.
(status: basic research programme funded until 2015)</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users:
Aim = team of up to 40 working with Magnum-PSI and associated program (incl numerical physics).
Core team TEC 25; Collaborators from other associations.</td><td>Yearly integrated equivalent full time facility users: (ppy)
No reliable estimate yet. Scientific team total 40.</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: 1
aim = 10 PhD students in team – 4 year course → 2.5/year.
Similar number of undergraduate students, final experimental 1 year stay.</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:
Estimate based on productivity in similar groups at FOM: > 50 papers/year.
Table not applicable (facility in construction phase)</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td></td><td></td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: Presently already active :
TEC; IPP-Garching; U. Cuza, Romania; U. Innsbruck; JET ; Eindhoven University of Technology.
Collaborations outside EU: MIT; ASIPP
Number of experimental contributions to ITPA (if applicable) : 2 in Jan 2008
Sharing facility with other fields of research (if applicable): possibility will be considered. Esp energy research: solar cells (fast deposition) and hydrogen storage in metals (parallel to H retention).
Prospects: collaborations with SCK-Mol, UKAEA-Fusion, CIEMAT, Slovenian association, being developed; more to come.</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">Forerunner Pilot-PSI, reaches flux density at target exceeding 10^{25} H^+ /$m^2 s$, power flux density <50 MW/m^2, at T_e=1-6 eV, B < 1.6 T, in Ø=1 cm beam with pulse duration 4 s @ 1.6 T, minutes @ 0.4 T.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">Steady-state plasma jet with Ø=10 cm, B<3T; power fluxes <10MW/m^2, particle flux density<10^{24} m^{-2} s^{-1}; Te~1-7 eV, ne<10^{21} m^{-3}
0.6 x 0.1 m^2 cooled target, under variable angle.
In-situ plasma and surface diagnostics
Ex-situ surface analysis tools (XBS, IBA, TDS, etc)</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Pilot-PSI used for high flux chemical erosion studies of CFC; H retention in W; test of first mirror material;
Magnum-PSI under construction.</td></tr>
<tr><td>PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">The programme; integration in the EU PWI research programme.
Magnum-PSI is a world-wide unique facility, reaching ITER divertor-relevant conditions by combining high particle and power flux density with low electron temperature and high magnetic field, in steady state. It is designed for maximal diagnostic access to the plasma surface interaction area, while the samples can be retracted – in vacuo – to an analysis station. The research programme links experiment to extensive modelling, and is integrated with PSI research on tokamaks in the European PSI research programme coordinated under the EU Task Force Plasma Wall Interaction.</td></tr>
</table>

	Relevance to the Missions. Magnum-PSI is central to **Mission 3** (1st wall + compatibility with plasma) Addressing: T-retention in CFC and metals, isotope exchange; erosion, redeposition and migration, use of scavenger techniques to mitigate T-retention; mixed materials; melt layer physics; physics of the magnetized sheath; dust formation, detection and mitigation. Development of in-situ diagnostics, development of specific techniques for e.g. tritium recovery, divertor PFC concepts, test of plasma facing mirrors, etc. In parallel to all of those: benchmark environment for codes. Programme is complementary to PSI research in tokamak, carried out in one integrated PSI research programme. Magnum-PSI can further make important contributions to **Mission 4** (long pulse/ steady state) by providing a steady state test of PFC's in relevant conditions, and **Mission 5** (prediction of tokamak performance) by providing a well-controlled and relevant environment to benchmark the numerical codes for SOL and PSI. These will be important factors in the 'numerical fusion reactor'. **Mission 6** (materials & fuel cycle) by addressing the problem of T retention and providing a test environment for development of diagnostics and recovery techniques. (below we follow as much as possible the milestones in the 'core document') Five year perspective. **contribute to:** • Contribution to final ITER divertor material choice. • Provide basis for extrapolation T retention in metals. • Start benchmarking codes in dedicated experiments. • Exploration of dust issue. • Tests of PFC concepts in relevant conditions. 10 year perspective (if applicable): **contribute to:** • diagnostic techniques for erosion, deposition, dust formation etc. (Mission 3) • dust and tritium removal techniques (Mission 3) • Predictive capability for all aspects of plasma surface interaction (erosion, migration, redeposition, T retention, mixed materials, sheath physics etc.) benchmarked on experiments (Mission 3). • plasma scenarios compatible with high-Z and mixed materials, and suitable for long-pulse and Steady-State Scenarios (Missions 3 & 4) • availability of a 'numerical tokamak' for planning and analysing experiments on ITER (Mission 5) Longer term perspective (if applicable): **contribute to:** • hydrogen/deuterium inventory data to establish the basis for ITER DT operation and divertor materials optimisation (Mission 3) FOR DEMO: • confirmation of DEMO physics basis (Missions 1 to 4) • selection of appropriate diagnostics (Missions 1 to 4), • Availability of a 'numerical tokamak' (Mission 5) • final selection of Divertor and Blanket concepts (results from IFMIF, TBM etc. and validated modelling) (Mission 3 & 6) • Selection of dedicated structural, functional and plasma facing material(s) (Missions 3 & 6)
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above) The programme at Magnum-PSI includes the plasma physics, the PSI and the effects on the material, with a special mention for the dust issue. Magnum-PSI provides a divertor relevant plasma, steady state, with relevant B and beam dimension. The device provides ample access for in situ diagnostics. It will be a testbed for the development of dedicated divertor and PFC, dust and T retention diagnostics, including the modelling for the interpretation. It will also provide a relevant testbed for techniques such as tritium recovery, scavenger and dust mitigation. Very important is the integration of the numerical modelling with the experimental facility. Magnum-PSI provides a well-defined benchmark environment for PSI and material codes. The research on Magnum-PSI is foreseen to go hand-in-hand with PSI research on confinement devices. Magnum-PSI is relevant to both ITER and DEMO/reactor.

HFR & PALLAS for ITER-DEMO R&D missions 3 and 6

Author: Bob van der Schaaf, vanderschaaf@nrg-eu

Version: 080104

FACILITY	**High Flux Reactor, FOM-NRG, Petten, The Netherlands**
RESOURCES INVOLVED	ORIGINAL INVESTMENT: In the year 1962: 25 Mhfl == 12 MEuro
	COST OF FORESEEN UPGRADES: PALLAS in the year 2016 the sum of 300 MEuro (rough estimate)
	OPERATION: HFR: 285 FPD/year (peak positions: 7 dpa/year). From 2016: PALLAS: > 300 FPD/year (>15 dpa/year; in boosters over 25 dpa/yr) Yearly operation & manpower cost for Fusion Technology depending on neutron irradiation volume needed for EFDA missions 3 & 6, and F4E projects.
USE OF FACILITY	HFR & PALLAS used for: 1. Fusion power plant materials and component development. 2. Fission power present & Generation-4 materials and component development. 3. Medical & technical isotope production.
COLLABORATION	Collaborations inside the EU: FZK, FZJ, CEA, ENEA, SCK & CRPP Collaborations outside the EU: JAEA, Kurchatov, PNL, UCSB
PRESENT TECHNICAL CAPAPBILITIES	Neutrons: The mixed neutron spectrum of the HFR provides good conditions for simulations, though without the helium effects to damage levels of 20 dpa in a few years. Materials: 1. Rigs for sample, specimens and coupon testing of metallic materials up to 800 ^{o}C with sizes a few mm up to CT blocks 60*60*12.5 mm 2. Rigs for carbon and SiC materials samples up to 1200 ^{o}C. 3. Log-term rigs for endurance testing of lithium ceramics and beryllium pebbles. Components: 1. Gas-cooled sub-modules simulating cutouts of blankets 2. Cyclic operation of first wall modules in a neutron flux. 3. Lithium lead devices for tritium generation and lead behavior studies. Cross fertilization irradiation devices for fission - fusion developments: EXTREMAT
FUTURE TECHNICAL CAPABILITIES	1. Speeding up of irradiation times with a factor three in PALLAS compared to HFR. 2. Widening operation windows for sample test rigs. 3. More degrees of freedom for component testing. 4. Extension of auxiliary equipment for rig and component control & data collection
PROGRAMME: ACHIEVEMENTS	1. ITER materials data base for irradiation effects on 316LN plate & welds 2. Materials data base for irradiation effects on RAFM steel products & EB, Laser and TIG welds for TBM in ITER and DEMO preparations. 3. Procedure and mechanisms for Irradiated weld technology under remote control For ITER applications. 4. Mechanisms for Radiation Effects on steels and their welds. 5. Materials data and tritium release: Li-lead and lithium ceramics. 6. Mechanisms for tritium release in lithium ceramics and lithium lead. 7. Mechanical and physical data of : SiC ceramic composites and graphite and carbon fibre re-inforced graphite CFC, and Cr, W. 8. Thermal shock effects on irradiated graphite and CFC. 9. Modeling support for texture effects of fibers in composite structures 10. Verification of models for blanket behavior prediction.
PROGRAMME: ADDRESSING PROGRAMME NEEDS	The mixed neutron spectrum of the HFR provides good conditions for ITER & DEMO relevant component simulations, though without helium effects. IFMIF must provide the input for 14 MeV neutron helium generation effects in materials. The core of the HFR and PALLAS provide sufficient volume for component testing with relevant sections of components such as divertors, blankets and first wall structures. PALLAS will offer a similar neutron spectrum as the HFR, but at with a two to threefold neutron flux. The PALLAS fluxes are speeding up irradiations to DEMO relevant levels of 70 dpa to durations of a few years, acceptable for the fast track development. **1. Mission 3: First Wall Materials** *1.1. First Wall materials* Effects of radiation damage on surface, mechanical, and physical properties in structural materials such as low activation steels, and for the longer term: structural tungsten alloys and SiCSiC ceramic composites.

	Behavior of coatings with low or high Z-number in a high density neutron field, under thermal cycling with high heat fluxes. Combined gas effects from particle irradiations for gas pick -up and re-distribution of gases, voids and bubbles under neutron radiation. *1.2. Divertors* .Tritium pick-up and release as affected by neutron fluxes in tungsten with an armor function, and in structural materials SiCSiC ceramic composites and nano-micro-structured ODS EUROFER. Radiation damage effects on surface, mechanical, and physical properties in those materials will be treated under first wall materials programs. **2. Mission 6: Materials and Components** *2.1. Fuel cycle* *Materials* Tritium release/retention, dimensional stability, and integrity will be measured for lithium ceramics, and beryllium as multiplier with the purpose of preparing design and safety handbooks with properties and damage mechanism equations and explanations. Lithium lead tritium release and helium effects will be quantified and analyzed for handbooks. *Components* Sub-modules with lithium ceramic and lead, derived from the DEMO design outcome, will be "cut to size" and tested under appropriate conditions. Results will improve design and lay-out of blankets, and contribute to the safety cases for licensing authorities. *2.2. Structural materials* Steel development will aim for lower activation and broader operating temperature window, reducing the lower and increasing the higher limit to the ultimate low activation ODS RAFM steel. Radiation damage effects reduction on mechanical properties will be the foremost target, together with the assurance that fabrication processes do not affect the properties adversely. SiCSiC application in advanced blankets with high operating temperatures requires developments assuring the limited effect of radiation damage and helium. Tungsten alloy developments with application for divertors and possibly advanced blankets will require lots of efforts to limit the radiation effects on its' properties *2.3. IFMIF test device simulations* The testing modules in IFMIF are exposed to high radiation damage levels. To predict their operating lifetime component testing in HFR/PALLAS is most effective. Heater behavior and compatibility are on the list of investigation. Components in the target vicinity could be other test objects relevant for the safe and reliable operation of IFMIF.
FORWARD PLANNING	Within 10 years with HFR: *Mission 3:* Quantitative results on radiation effects on low & high Z coatings. Neutron effects on the tritium retention in tungsten and EUROFER. *Mission 6:* Lithium ceramics design data with radiation effects for ITER ODS-RAFM handbook, including joints, with radiation effects to 15 dpa IFMIF test component cut-out demonstration test to 15 dpa Within 15 years with PALLAS: *Mission 3:* Gas effect data for tungsten and SiC up to 20 dpa Neutron effects on advanced divertor module designs. *Mission 6:* Selection of reference nano ODS RAFM steel with 40 dpa data. Intermediate results tungsten alloys and advanced SiC composites & joints Blanket module operational tests with lithium ceramics and lead to 20 dpa Within 20 years with PALLAS: *Mission 3:* Divertor testing with advanced W and SiC up to 50 dpa Advanced coating testing completed up to 70 dpa *Mission 3:*Advanced blanket module testing up to 100 dpa completed Handbook ODS RAFM steel with 100 dpa data on radiation effects Selection of reference structural W and SiC composites with 40 dpa data

<table>
<tr><td>FACILITITY</td><td colspan="3">Irradiation Facilities
CIEMAT</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: 3 M€ in a period of 40 years</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES: No significant upgrades foreseen</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 200
- estimated yearly cost of operation around 0.2 M€ (2007)
- yearly manpower for operation in ppy: 1 ppy + 2 tpy</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: 5-10</td><td>Yearly integrated equivalent full time facility users: (ppy) N/A</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 10 years: 4</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>8</td><td>15</td></tr>
<tr><td>2005</td><td>7</td><td>17</td></tr>
<tr><td>2006</td><td>10</td><td>22</td></tr>
<tr><td>COLLABORA TIONS</td><td colspan="3">Collaborations inside EU: SCK-MOL, CEA, UKAEA, ENEA, FZK, FZJ, Latvia, Rumania, UAM
Collaborations outside EU: Uni. Sendai, JAEA, Oak-Ridge
Number of experimental contributions to ITPA (if applicable) :
Sharing facility with other fields of research (if applicable):</td></tr>
<tr><td>PRESENT TECH. CAPABILITIES</td><td colspan="3">(factual set of parameters)
The facility includes two different irradiation equipments:
- A 2 MeV VdG electron accelerator (currents up to a few microamperes) with the capability to make measurements during irradiation (conductivity, optical absorption, radioluminescence, dielectric properties, permeation,…) at different temperatures
- A gamma-irradiation facility (dose rate up to 10 Gy/m) with the capability to make irradiations under controlled atmosphere and in the RT-250 ºC temperature range.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">(if applicable: new technical capabilities after foreseen upgrade(s); including planning)</td></tr>
<tr><td>PROGRAMME

ACHIEVEMEN TS</td><td colspan="3">During the last 30 years the facility had a significant impact on the characterization of the behaviour of insulator materials for heating systems and diagnostics.
Some of the research areas that were developed in the facility are the following:
- Discovery and characterization of the RIED effect
- First measurements of dielectric properties during irradiation
- Characterization of RIC in insulator materials
- Characterization of RIEMF effect
- Degradation of mirrors under gamma irradiation
- Development of radiation resistant windows
Presently it is a European Reference lab for insulator materials</td></tr>
<tr><td rowspan="2">PROGRAMME

ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">(Is the facility an “ITER” facility, paid or to be paid under ITER credit through F4E.)
(how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions)
Five year perspective:
The facility will have significant impact on the R&D Mission 2 due to its contribution to the development of diagnostics systems required for ITER operation
10 year perspective (if applicable):
The facility will have significant impact on the R&D Mission 2 due to its contribution to the development of diagnostics systems required for ITER operation
Longer term perspective (if applicable):
The proposed facility can have a significant impact on the R&D Mission 6 (Materials and Components for Nuclear Operation)
The facility can have also a significant impact on R&D mission 7 (DEMO integrated design) due to its contribution to the viability studies of heating systems and diagnostics</td></tr>
<tr><td colspan="3">(How the facility addresses/will address the Satellite Tokamak requirements)
Diagnostics under irradiation from DD</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
</table>

<table>
<tr><td>FACILITITY</td><td>National Centre for Fusion Technologies (TechnoFusion)
CIEMAT</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td>ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES:
N/A</td></tr>
<tr><td>COST OF FORESEEN UPGRADES:
Foreseen investment around 60 M€ in 6-8 years
<table>
<tr><th>Year</th><th>T0</th><th>+1</th><th>+2</th><th>+3</th><th>+4</th><th>+5</th><th>+6</th><th>+7</th></tr>
<tr><td>Investment (M€)</td><td>2.5</td><td>10</td><td>14</td><td>15</td><td>10.5</td><td>5</td><td>2</td><td>0</td></tr>
</table>
.With high probability T0 will be year 2008</td></tr>
<tr><td>OPERATION
- average number of operation days/year (over the past 4 or 5 years): N/A
- estimated yearly cost of operation around 5.4 M€
- yearly manpower for operation in ppy: 30 ppy + 35 tpy</td></tr>
<tr><td rowspan="4">USE OF FACILITY</td><td><table><tr><td>Number of facility users:
N/A</td><td>Yearly integrated equivalent full time facility users: (ppy) N/A</td></tr></table></td></tr>
<tr><td>Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: N/A</td></tr>
<tr><td>Number of yearly publications based on experimental results from facility:
N/A</td></tr>
<tr><td><table>
<tr><th>Year</th><th>Journals</th><th>Conferences</th></tr>
<tr><td>2004</td><td></td><td></td></tr>
<tr><td>2005</td><td></td><td></td></tr>
<tr><td>2006</td><td></td><td></td></tr>
</table></td></tr>
<tr><td>COLLABORATIONS</td><td>Collaborations inside EU: to be explored
Collaborations outside EU: to be explored
Number of experimental contributions to ITPA (if applicable) :
Sharing facility with other fields of research (if applicable): To be explored
Prospects: fission, spallation sources, materials technology medical applications, …</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td>(factual set of parameters (engineering as well as physics parameters) and other technical capabilities)
N/A</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td>(if applicable: new technical capabilities after foreseen upgrade(s); including planning)
The proposal is to create a users-facility called National Centre for Fusion Technologies that will include a number of significant laboratories (open to Spanish and European users). The presently foreseen ones are:
- Accelerator complex: including a cyclotron, two linear accelerators and an electron accelerator. They should be able to accelerate a wide range of ions (going from H to W) with energy enough to penetrate in the materials at least few tens of microns. With them it will be possible:
1) To make triple beam irradiations in a wide range of materials
2) To make single and double beam irradiations in a wide range of materials
3) To make irradiations under magnetic field in a wide range of materials
4) To study radiation induce synergy effects in the interaction of liquid metals and materials

- Materials characterization lab: including a wide range of experimental techniques (microstructure properties –TEM, SEM, AP, SIMS,…-, mechanical properties- including micromechanics-, physical properties –light emission, electrical conductivity, diffusion, thermal conductivity,…-) to be able to make measurements in-situ during irradiation and after irradiation. A special effort will be made to get significant mechanical properties from the small irradiated region).</td></tr>
</table>

	- Plasma-Wall Interaction lab. A facility to exposure of reactor PFC to fluences of few 1024 s-1 at relevant impact energies (< 10’s of eV) and simultaneous exposure to transient loads in a range of 0.1 – 1.0 MJ m-2. It should be able to perform studies of material performance degradation under reactor-like plasma exposure, fuel retention and exploration of combined effects of plasma exposure and radiation by studies of radiation damaged materials (simulated by ion irradiations in-lab). It will include a Plasma Linear Device and a Plasma Gun that can be operated simultaneously/sequentially + associated diagnostics The possibility of coupling the Plasma Linear Device to a mock-up of ITER divertor in order to study dust production is also under consideration. - A liquid metals lab including a medium size flexible loop (focused on Li and/or PbLi technology) with several experimental areas: free surface area (maybe including ions from one accelerator), corrosion area (maybe including electrons from the electron accelerator), MHD and B effects area, purification and monitoring area, H studies area (maybe including electrons from the electron accelerator) permeation studies, … - Remote Handling lab including a building high enough –present estimate: 50x20x15 m3) to host different RH facilities foreseen to be needed in the future (TBM RH facility, Diagnostics RH facility, IFMIF RH facility, maybe others) and a lab able to qualify operating RH submodules (1-3 m3) under ionizing field (produce by the electron accelerator by bremsstrahlung). - A materials processing lab able to produce prototype quantities (a few tens of kg) of advance materials (ODS steels, W or similar grades) including characterization. - A computational simulation lab to assemble a team of experience people working jointly in a common effort to fill the gap between experiments and computer simulations.
PROGRAMME: ACHIEVEMENTS	(a summary of key scientific achievements) N/A
PROGRAMME: ADDRESSING THE PROGRAMME NEEDS	(Is the facility an “ITER” facility, paid or to be paid under ITER credit through F4E.) (how does the present (future/upgraded) facility addresses the 7 R&D Missions and the fusion roadmap and contributes to the development of basic understanding in support of the Missions) Five year perspective: 10 year perspective (if applicable): The proposed facility can have some impact on the R&D Mission 2 mainly due to the Remote Handling Facility that can help to the improvement of ITER operation capabilities The facility can have also a significant impact on R&D mission 3 (First wall materials & compatibility with ITER/DEMO relevant plasmas) due to the proposed PWF Longer term perspective (if applicable): The proposed facility can have a significant impact on the R&D Mission 6 (Materials and Components for Nuclear Operation) mainly due to the accelerator complex proposed (including the characterization techniques), but also due to the liquid metals lab and the materials processing and simulation labs. The facility can have also a significant impact on R&D mission 3 (First wall materials & compatibility with ITER/DEMO relevant plasmas) due to the proposed PWF The facility can have also a significant impact on R&D mission 7 (DEMO integrated design) due to the proposed Remote Handling Facility
	(How the facility addresses/will address the Satellite Tokamak requirements: for JET, JT60SA and satellite tokamak proposal(s) only) Five year perspective(when relevant): 10 year perspective (when relevant): Longer term perspective (when relevant):
FORWARD PLANNING	Summary of the key elements of timetable and planning (if not already adequately addressed above) 1) The agreement to sign the consortia presently under preparation (signature foreseen before summer 2008)

	2) Recent nomination of a Coordinator, with the missions: - to prepare the General Director international search - to prepare the administrative environment - to look for external collaborations - to launch a detailed engineering study - to assure the matching in the EU programme 3) Timetable Conceptual design *2008-2009* Detailed design and prototyping *2009-2011* Buildings and Commercial Hardware *2009-2011* Complex Hardware *2011-2014* Installation and Commissioning *2010-2015* **4) Technical Priorities** –*Provisional.* To be agreed with the EU Programme taking into account availability of equipments, complexity, possible users,…- **First phase** Some characterization techniques (SIMS, Atomic probe), low energy accelerators, Remote Handling Lab, Materials Processing Lab **Second phase** Other characterization techniques, high energy accelerator, liquid metal loop, PWI Facility

<table>
<tr><td>FACILITY</td><td colspan="3">Low Temperature Physics Laboratory (LTP), ATI, TU Vienna</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="3">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros
(details when appropriate)</td></tr>
<tr><td colspan="3">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2007 euros
(details when appropriate)</td></tr>
<tr><td colspan="3">OPERATION
- average number of operation days/year (over the past 4 or 5 years): ~ 220
- yearly cost of operation in 2007 euros:
- yearly manpower for operation in ppy: ~ 5</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="2">Number of facility users: ~ 15</td><td>Yearly integrated equivalent full time facility users: (ppy) ~ 10</td></tr>
<tr><td colspan="3">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~ 30</td></tr>
<tr><td colspan="3">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td>Year</td><td>Journals</td><td>Conferences</td></tr>
<tr><td>2004</td><td>~ 15</td><td>~ 20</td></tr>
<tr><td>2005</td><td>~ 15</td><td>~ 20</td></tr>
<tr><td>2006</td><td>~ 15</td><td>~ 20</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="3">Collaborations inside EU: ICTAS Univ. Linz, IFW Dresden, IPHT Jena, KIT FZ Karlsruhe, INTIBS PAN Wroclaw, IF PAN Warszawa, EIU SAV Bratislava, SAS Kosice, ICMAB and F4E Barcelona, DPMC Geneve, DMSM and IRC Cambridge, EFDA Garching
Collaborations outside EU: Marti Supratec, Huntsman and ETH Zürich (Switzerland), CTD and PS Univ. (USA), ISEM Wollongang (Australia), NIN Xi`an (China), ASC Lab Jiaotong Univ. Chengdu (China), Bochvar Institute
Number of experimental contributions to ITPA (if applicable) :
Sharing facility with other fields of research (if applicable):
Prospects:</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="3">3 SQUID magnetometers (8 T, 7 T, 1 T)
17 T magnet system (ac susceptibility, flux profiles, transport critical currents, magneto-transport)
8 T magnet system
6 T split pair magnet (angular dependence of Jc in tapes and thin films, 2-axis rotation)
Continuous flow cryostat (plus 1.5 T electromagnet or 10 mT split pair system, transport Jc measurements)
Trapped field scanning facility (Hall probe)
Magnetoscan facility
Transverse VSM (vibrating sample magnetometer, 5 T split pair magnet)
100 kN Servohydraulic Material Testing System MTS 810</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="3">--</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="3">Material data base on radiation resistant insulation materials
Irradiation effects on superconducting materials: low temperature superconductors and high temperature superconductors.</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="3">Investigation on radiation resistance of magnet coil components (superconductors, insulation materials)
Five year perspective: Investigations on newly developed insulation materials
10 year perspective (if applicable):
Longer term perspective (if applicable):</td></tr>
<tr><td colspan="3">(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>)
Five year perspective(when relevant):
10 year perspective (when relevant):
Longer term perspective (when relevant):</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="3">Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
<tr><td></td><td colspan="3"></td></tr>
</table>

<table>
<tr><td>FACILITY</td><td colspan="6">Triga (<u>T</u>raining, <u>R</u>esearch and <u>I</u>sotope Production Reactor <u>GA</u>), ATI, TU Vienna</td></tr>
<tr><td rowspan="3">RESOURCES INVOLVED</td><td colspan="6">ORIGINAL INVESTMENT AND SUBSEQUENT UPGRADES: year of investment, cost in 2007 euros
(details when appropriate)</td></tr>
<tr><td colspan="6">COST OF FORESEEN UPGRADES:
year of foreseen investment, cost in 2007 euros
(details when appropriate)</td></tr>
<tr><td colspan="6">OPERATION
- average number of operation days/year (over the past 4 or 5 years): 220 days/year
- yearly cost of operation in 2007 euros: ~ 400 k€
- yearly manpower for operation in ppy: ~ 4</td></tr>
<tr><td rowspan="7">USE OF FACILITY</td><td colspan="3">Number of facility users: ~ 10</td><td colspan="3">Yearly integrated equivalent full time facility users: (ppy) ~ 5</td></tr>
<tr><td colspan="6">Number of PhD/diploma thesis using experimental data from the facility in the last 5-10 years: ~ 20</td></tr>
<tr><td colspan="6">Number of yearly publications based on experimental results from facility:</td></tr>
<tr><td colspan="2">Year</td><td colspan="2">Journals</td><td colspan="2">Conferences</td></tr>
<tr><td colspan="2">2004</td><td colspan="2">~ 20</td><td colspan="2">~ 40</td></tr>
<tr><td colspan="2">2005</td><td colspan="2">~ 20</td><td colspan="2">~ 40</td></tr>
<tr><td colspan="2">2006</td><td colspan="2">~ 20</td><td colspan="2">~ 40</td></tr>
<tr><td>COLLABORATIONS</td><td colspan="6">Collaborations inside EU: F4E (Spain), EFDA, FZK (Germany), Bochvar Inst.
Collaborations outside EU: MartiSupratec, Huntsman (Switzerland), CTD (USA)
Number of experimental contributions to ITPA (if applicable) :
Sharing facility with other fields of research (if applicable):
Prospects:</td></tr>
<tr><td>PRESENT TECHNICAL CAPABILITIES</td><td colspan="6">The TRIGA-reactor Vienna has a maximum continuous power output of 250 kW (thermal). The primary coolant circuit operates at temperatures between 20 and 40 °C. The TRIGA-reactor Vienna can also be operated in a pulsed mode (with a rapid power rise to 250 MW for roughly 40 milliseconds accompanied by an increase in the maximum neutron flux density from $1x10^{13}$ $cm^{-2}s^{-1}$ at 250 kW to $1x10^{16}$ $cm^{-2}s^{-1}$ at 250 MW).
Irradiation devices:
1 four beam holes 15.2 cm in diameter
2 one central irradiation tube (center of core) 3.75 cm in diameter
3 five reflector irradiation tubes
4 one pneumatic transfer system (near core edge)
5 a thermal column with cross section 1.22x1.22 m and length 1.68 m
6 experimental tank with surface area 2.44x2.74 m and depth 3.66 m; connected to the reactor by means of a neutron radiography collimator 0.61x0.61 m in cross section and 1.22 m long.</td></tr>
<tr><td>FUTURE TECHNICAL CAPABILITIES</td><td colspan="6">--</td></tr>
<tr><td>PROGRAMME: ACHIEVEMENTS</td><td colspan="6">(a summary of key scientific achievements)</td></tr>
<tr><td rowspan="2">PROGRAMME: ADDRESSING THE PROGRAMME NEEDS</td><td colspan="6">Irradiation of magnet coil materials like insulation, superconductors etc.
Neutron radiography
Five year perspective: investigations on newly developed insulation materials,
10 year perspective (if applicable): --
Longer term perspective (if applicable): --</td></tr>
<tr><td colspan="6">(How the facility addresses/will address the Satellite Tokamak requirements: <u>for JET, JT60SA and satellite tokamak proposal(s) only</u>)
Five year perspective(when relevant):
10 year perspective (when relevant):
Longer term perspective (when relevant):</td></tr>
<tr><td>FORWARD PLANNING</td><td colspan="6">Summary of the key elements of timetable and planning (if not already adequately addressed above)</td></tr>
</table>