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R&D Needs and Required Facilities for the Development of Fusion as an Energy Source

Report of the Fusion Facilities Review Panel

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Report

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R&D Needs and Required Facilities for the Development of Fusion as an Energy Source

Report of the Fusion Facilities Review Panel
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Foreword

In December 2007 the European Commission established an independent Panel for a review of the R&D visions and the required facilities[i] of the European fusion programme. This review which is stipulated by the EURATOM FP7 Specific Programme on Fusion Research[ii] has the "motivations to support the rapid and efficient development of fusion as an energy source and to maintain in the programme the facilities needed to fulfil its medium and long term objectives". A vision of the R&D required to make fusion energy production ready for commercial exploitation shall be developed and all significant facilities, existing or under construction including proposed or considered upgrades, shall be reviewed. The required facilities should be incorporated in a road map; and prioritised according to the corresponding benefits, costs and risks. Non priority facilities should be identified. The full terms of reference to the panel are given in Annex I.

Following its constitution in December 2007 the Panel, composed of five European members with a professional background outside fusion and four international members with experience in fusion R&D[iii], took up its work in a first meeting on February 26/27, 2008. The Panel was provided with input papers established by EFDA[iv] and the Associated Fusion Laboratories and received during its meetings and visits in-depth presentations by representatives of the Commission, EFDA, Fusion for Energy (F4E), ITER and the Associated Laboratories of the Programme. Furthermore, coordinated by EFDA, comprehensive written information was provided in response to specific questions raised by the Panel. The Panel met for four plenary meetings in Brussels, held several video conferences[v], and delegations visited most of the major laboratories for which the input documents refer to substantial intended investments in facilities[vi]. The Panel concluded its work in October 2008 unanimously endorsing the present report.

When conducting its work, the Panel took account of the particular situation of fusion research as a long-term endeavour of great promise and substantial challenges. The assessment is based on the programme's objective to achieve the ultimate goal of enabling the entry of fusion into the commercial regime in a fast track approach with the creation of prototype reactors in approximately 30 or 35 years. Specifically, the periods until completion of ITER construction and the first phase of ITER operation are in the forefront of deliberations and developments which can maximise the success of ITER. When assessing research needs and necessary facilities, the Panel had to acknowledge the multidisciplinary nature of R&D tasks and the resulting complexity of the programme. Furthermore, account had to be taken of the specific organisational structure in fusion R&D where the overall programme management at European level is ensured through committees, the Commission services as well as F4E (for dedicated ITER, Broader Approach and DEMO work) and EFDA, (for R&D on JET and coordinated research by European task forces and joint topical teams which rely on the strength and resource basis of the national associated laboratories).

The Panel was deeply impressed by the progress achieved in fusion R&D, the scientific-technical quality of the work being undertaken, the sharing of tasks among the partners and the commitment of all parties in the Programme towards achieving the goal of useful fusion power.

During the course of its work the Panel has been made informally aware of near-term funding constraints which possibly might have an impact on the EURATOM fusion programme or parts of it. The Panel was of the opinion that any consequences from this information on its work do not alter the analysis and the conclusions contained in its report if a rapid and efficient development of fusion as an energy source, as stipulated in the Terms of Reference, *shall continue to be the programmatic objective of Europe's effort in fusion research. Should it not be possible in the foreseeable future to pursue this objective then some of the conclusions would have to be re-examined.*

The Panel wishes to thank the European Commission, EFDA, the Associations, Fusion for Energy and ITER for their constructive support, the quality of the input provided and their readiness in responding to the questions which were raised by the Panel.

Brussels, October 8, 2008

Prof. Dr.-Ing. Thomas Hartkopf
Chairman of the Panel

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[i] For the purpose of this review, "facility" is taken to mean any device or installation, including high performance computers, built and operated for the purpose of fusion R&D, and funded through the fusion programme.

[ii] "At an early stage of the Framework programme, a review will be carried out of the facilities in the programme, examining the possibility of phasing out existing facilities, and considering the need for new devices in parallel to ITER exploitation. The review will be used a basis for the possible support of new or upgraded devices in order to ensure that the programme will maintain an adequate set of fusion facilities for the relevant R&D."

[iii] The Panel membership is listed in Annex III.

[iv]EFDA: European Fusion Development Agreement. Fusion for Energy (F4E): Joint Undertaking, European domestic agency for ITER. An explanation of acronyms is provided in Annex V

[v] Meetings of the Panel in Brussels took place on February 26/27, May 27/28, July16/17 and September 23/24, 2008. Additionally four video conferences were arranged with international members during visits to facilities and meetings.

[vi] Visits by delegations of the Panel were undertaken to FZK (Karlsruhe, DE, December 2007), JET / MAST (EFDA, UKAEA, Culham Laboratory, UK, April 2008), ASDEX-UG (IPP Garching, DE, May 2008), Wendelstein 7-X (IPP Greifswald, DE, June 2008), TORE SUPRA (CEA Cadarache, FR, June 2008), FTU (ENEA Frascati, IT, July 2008)

Executive Summary

Security of energy supply and mitigation of Global Warming require vigorous efforts with a strong boost to energy research. Fusion energy using the ubiquitously available deuterium and lithium fuel could make a major contribution to future environmentally friendly and safe electricity generation without long-term storage of radioactive waste.

The long-term goal of European fusion research is the joint creation of prototype reactors for power stations[1]. The Panel acknowledges the remarkable progress in fusion R&D already achieved. Still, pivotal developments have to be achieved on the path towards reliable and efficient fusion electricity production at high availability.

The Panel is impressed by the quality of the research community and the coherence of the programme, and supports its thrust for a rapid and efficient development towards the ultimate objective. The launch of the ITER project in an international partnership is the most recent major advance, one which introduces a step change in the European programme requiring reorientation and concentration on core activities with an increasing emphasis on fusion technology. The European fusion laboratories will play the essential role for the necessary R&D, for fusion engineering, and for providing skilled staff and a pool of expertise. Interaction with industry, based on the large-scale industrial contributions to ITER construction, should be developed in the longer term towards industrial leadership.

I. An integrated vision on the required R&D

The Panel strongly supports the determination of EFDA, F4E and the fusion laboratories in pursuing a streamlined, vigorous fusion energy oriented programme and addressing consistently the complex range of interconnected R&D objectives in fusion physics, technology and engineering. This necessary core programme can be structured in seven interrelated missions along the path from ITER towards DEMO and a commercial fusion power plant[2]. **For this core programme the Panel, with a particular view to optimise the progress on ITER and accelerate the development of fusion, provides the following recommendations.**

During the period of ITER construction the key strategic R&D emphasis should be on

A1) Supporting ITER construction and preparation for operation specifically by

- Accomplishing outstanding R&D issues and exploiting recent R&D progress for the design and construction of ITER systems and components;
- Resolving ITER physics issues which might limit the performance, constrain the accessible parameter space and/or impact on the operational reliability;

[1] For details see Annex I.

[2] First it is necessary to achieve (1) burning plasmas in ITER which must then be shown (2) to allow reliable operation; the future devices must be equipped with suitable (3) first wall materials and should operate in (4) long pulses and (ultimately in) steady-state. Furthermore, it is simultaneously necessary to (5) predict fusion performance for ITER and the further development steps towards a commercial plant and to develop and qualify (6) materials and components suitable for the ultimate full-power nuclear operation. Progress on ITER and by accompanying R&D should be brought together in a (7) DEMO integrated design oriented towards high availability and efficient electricity production.

- Preparing the rapid start-up of ITER, targeting promising operational regimes;
- Strengthening diagnostic and modelling capabilities and fostering developments for improved solutions in specific areas of fusion physics and technology.

A2) Preparing DEMO design, simultaneously carrying out long lead R&D by

- Strengthening a coherent materials research programme for DEMO and future fusion power plants and establishing experimental means for validation;
- Advancing Tokamak concept improvements and pursuing the Stellarator line for optimizing the path towards DEMO and a commercial fusion power plant;
- Establishing soon a group for proceeding towards the definition of a conceptual DEMO design, steering the DEMO R&D programme and preparing industrial involvement.

During the following decade the focus must shift towards

B) Preparing for DEMO construction, based on ITER and the accompanying R&D, with increased involvement of industry and utilities, by focusing on

- Achieving the goals of ITER in DEMO relevant conditions with emphasis on steady-state aspects;
- Developing a blanket and auxiliary systems compliant with DEMO conditions;
- Optimising and validating suitable materials and components for DEMO;
- Assessing concept improvements for the Tokamak and the potential of the Stellarator for optimizing the path towards commercial fusion power;
- Developing a "numerical burning plasma device" for the detailed prediction of fusion performance and assistance in the definition and design of DEMO;
- Establishing the engineering design for DEMO.

These ITER and DEMO priorities must be complemented by

C) Pursuing innovation for

- Improving theoretical understanding based on first-principles;
- Developing novel solutions in physics and technology and unifying experimental and theoretical knowledge from different confinement schemes;
- Extending fusion physics and technology, and the relevant data bases.

New ideas from science driven R&D, access to professional and academic excellence in the fusion disciplines and adjacent fields and intensive links to universities will be vital for optimal progress in the long-term development of fusion energy.

D) Maintaining and renewing the staffing basis of the Programme

- The training of young scientists and engineers is a particularly important task for the long-term future of the programme.

II. Required Facilities for supporting the envisioned R&D

ITER will be the key device for the decades to come. The substantial time and cost requirements for operation, and even more for modifications and maintenance, impose that ITER campaigns must be carefully targeted for optimum scientific value, and be validated for a high prospect of success. For these tasks a well-defined set of specialised, more cost/time efficient and flexible fusion devices as well as technology and computing facilities are needed and should be adapted to the new requirements.

The Panel recommends the following roadmap for facilities prioritized according to the corresponding benefits, costs and risks and listed according to category and size. For each facility the priority for ITER and DEMO is indicated[3].

Fusion devices: In assessing the R&D required for DEMO, it is important to understand that no single machine, including ITER, can address all R&D missions; the first fully integrated experience will be in DEMO itself.

- **JET** [vh, m] is the most relevant device for support to ITER until new devices with improved capabilities become available. Currently being enhanced for urgent ITER tasks, JET needs to operate until 2014/15 at least and would benefit from an early installation of an ECRH system. Depending on the JT-60SA schedule JET operation for a few further years should be foreseen.
- **JT-60SA** [h, m[4]], currently scheduled to start operation in 2016 in Japan, will go beyond JET's steady-state capability. Should the device be equipped with an ITER relevant first wall and more current drive an extension beyond five years of the European share in the exploitation should be considered.
- **European satellite** [h, vh]. Considering the importance of accompanying ITER with a strong satellite programme, and taking note of the FAST proposal, Europe should develop a design for a device accessing the relevant parameter space and complementary to JT-60SA.
- **Wendelstein 7-X** [m, vh[5]] should investigate steady-state operation of relevance to ITER as well as demonstrate the reactor potential of the Stellarator concept.
- **ASDEX UG** [vh, m], with appropriate enhancements, should support ITER well into the next decade and possibly longer unless a similar device, at somewhat higher current, would become available to the programme e.g. through international collaboration.
- **TCV** [m,m] and **MAST** [m, m[6]] should access a wider multidimensional parameter space beyond the ITER-like Tokamaks. Reasonable upgrades should enhance their relevance.
- **Academia-based devices,** in particular **COMPASS** [l, l][7], have a valuable role for the cost-efficient development of dedicated diagnostics, control tools and fundamental

[3] vh = very high priority, h = high priority, m = medium priority, l = low priority, - = not applicable. Thus JET[vh, m] means that the priority of JET is very high for directly ITER, and medium for directly DEMO, oriented R&D.

[4] The priority would be higher if a more DEMO-relevant wall and higher current drive capability would be installed.

[5] With respect to steady-state operation in a reactor perspective

[6] The priority could be high or very high depending on the assessment of a Components Test Facility perspective.

studies. They are also important as regional attractors for young scientists to the European fusion programme.

Other devices[8] should complete their missions in the coming years. These priority tasks include inter alia, operating an ITER-relevant lower hybrid antenna and investigating plasma wall interaction in a steady-state environment (**Tore Supra** [m, l]), using ergodic magnetic fields to manage heat loads (**TEXTOR** [m, l]), controlling the plasma boundary (**RFX** [m, l]), assessing the liquid lithium walls (**FTU** [l, m]) and preparing Wendelstein 7-X operation (**TJ-II** [l, m]).

Fusion technology facilities:

- **Test beds and laboratories confirming solutions for essential ITER components within the European contribution** [vh, l]. The most substantial ones are the neutral beam test facility and magnet cold testing facilities.
- **IFMIF** [-, vh]. The finalisation and validation of the design for the International Fusion Materials Irradiation Facility (IFMIF). An extensive programme for testing and validating materials being required, it is imperative to make IFMIF available for preparing the DEMO engineering design and construction.
- **Facilities for the preparation of DEMO components and technologies**[9] [l, h]. Some additional facilities and upgrades are needed for crucial R&D on materials, fuel cycle and remote handling; here the Technofusion proposal aims at contributing to fill the gap.
- A **Components Test Facility** [-, m[10]] could be desirable for risk reduction for DEMO associated to the qualification of nuclear technology components. Its usefulness should be assessed and the feasibility should be explored, for instance on an upgraded MAST device if a Spherical Tokamak option is chosen.

In some areas of ITER and DEMO oriented R&D the facility basis should be consolidated in due time. These include low power gyrotron test beds, some blanket and neutral beam related facilities and, used on a customer task-by-task basis, facilities for the examination of non-irradiated materials as well as facilities for non-specific neutron irradiation with low fluence.

Computing facilities: Advanced modelling of plasmas and materials is of rapidly growing importance and requires high performance computing facilities.

- In the short-term **a dedicated 100 Tflops facility** [h, vh] with professional support staff in numerical and modelling techniques should be implemented and soon be followed by larger efforts aiming at the realisation, in particular, of a numerical fusion device and numerical materials development.

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[7] These priorities relate, in line with the ones of the other devices and facilities, to directly ITER and DEMO oriented R&D and not to the valuable generic relevance addressed in the text. COMPASS is among these devices the most relevant in shape, size and equipment, giving it for some aspects a major relevance in contributing to ITER R&D.

[8] The priorities for the five devices in this paragraph refer to the priority tasks still to be executed.

[9] These relate to areas such as heating and current drive, plasma facing materials / exhaust systems (some fusion devices have also an important role), magnet and superconductor development, materials research, fuel cycle and blanket, maintenance and remote handling, licensing and safety.

[10] The priority would be very high in case a full test of a breeding blanket would be considered mandatory for DEMO. The Panel attributes a very high priority to the study of the usefulness of a CTF which should be executed soon.

III. General Aspects

When adapting the programme to the recommended vision of R&D needs and the required facility basis of the new ITER era the following points should be considered:

- Europe as the main investor should draw maximum benefit from ITER, in particular by ensuring efficient and optimum progress of ITER from a DEMO perspective. This requires extensive support by accompanying R&D in co-ordination with the other ITER Parties;
- Europe should maintain and further strengthen the integrated character of its fusion programme. Care must be taken not to lose competence as well as R&D and training capacity of the national Associates when reorienting their programmes;
- Existing support schemes for staff mobility, education and training need to be continued and expanded;
- Intensive mutual collaboration on the major facilities of the European and the international fusion programmes should be developed in parallel to the recommended adaptation of the European facilities base;
- Decisions for important upgrades and new facilities, in particular IFMIF, should be taken in coherence with a fast track approach of the programme;
- Innovative research must be vigorously encouraged;
- The recommended strong efforts for making ITER a success and the preparations, in parallel to ITER construction and exploitation, for the DEMO design and construction will require a growth of resources over time. Securing the substantial investments that have already been made in the priority facilities requires keeping these facilities up-to-date.
- A growing role of industry and utilities in DEMO and its definition will be mandatory for the ultimate success of commercial fusion power development and should be actively developed.

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I. Introduction

A. *Motivation for fusion research*

Global warming is perceived as a significant long-term threat to mankind and the transition to new carbon-free or carbon mitigated energy systems is understood as being mandatory. An even more imminent problem for the European Union and many other parts of the world is the security of energy supply at affordable prices. The surge in demand for fossil fuels, limitations in world oil and gas reserves and their geographically uneven distribution create a situation which has led to the recent dramatic increase in fuel prices and pose major threats to society and economy.

The importance of drastically changing this situation by introducing new and improved energy technology is in striking contrast with the public investment in energy research. This holds also for the European Framework Programmes where the share of funds for energy R&D dropped from 66% in the first (1983-1986) to 10.5% in the seventh Framework Programme (2007-2013)[11]. This bleak situation is further aggravated by the fact that the average support of R&D in Europe is still well below the target of the 3% (of Gross Domestic Product) set out for 2010 in the Lisbon R&D strategy[12].

A vigorously increased effort in energy R&D is indispensable if the defined targets of CO_2 reduction and security of energy supply at affordable prices shall be met[13].

Fusion power is a potential major new source of base-load electricity with attractive features: no greenhouse gas emissions, abundant and world-wide distributed sources of fuels (deuterium from water and lithium[14]) and inherent safety features which ensure that no need of evacuation of population may arise from a fusion power station. If built with low-activation materials fusion power will have the advantage that radioactive waste can be recycled within a hundred years and most of it even on a shorter time scale.

B. *State-of-the-art in fusion R&D*

Fusion R&D has made impressive progress over the past decades and is now at the turning point towards realizing reactor-size burning plasmas. Europe has played a major

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[11] Renda et al. CEPS 2008: Evaluation of the EU funding of Research in the fields of nuclear fusion and Aerospace/Aeronautics, Study commissioned by the Budget Committee of the European Parliament

[12] "Working together for growth and jobs A new start for the Lisbon Strategy" COM (2005) 24 Communication by President Barroso to the Spring European Council 2005 (2.2.2005)

13 "It is essential to address the mismatch between the sheer magnitude of the energy and climate change challenge and the current levels of research and innovation effort": SET-Plan, p.12, COM (2007) 723 final (European Commission)

[14] In a fusion power plant deuterium and (radio-active) tritium are burned, the latter is produced inside the reactor in a lithium-containing blanket. Apart for initial start-up only deuterium and lithium need to be transported to the plant site.

role in this development which today concentrates essentially on the Tokamak[15] and, less developed, the Stellarator[16] lines.

The Joint European Torus (JET), so far the world's largest fusion device, is routinely achieving temperatures in the >100 million degree range and has generated fusion power in the megawatt range for seconds. JET has demonstrated a fusion power amplification of 0,67 in a real fusion fuel (a 50:50 deuterium-tritium mixture). More recently, in deuterium discharges, equivalent break-even (where the expected fusion power equals the heating power) has been achieved and similar results were obtained with the Japanese JT-60U. Numerous other achievements, established by the world's fusion laboratories, have been experimentally demonstrated. There is a solid R&D basis for ITER and a high confidence that ITER will achieve its objectives, in particular a burning plasma with a tenfold energy amplification and demonstrate the feasibility of fusion at a power plant scale[17]. Beyond ITER the demonstration of the economic viability of fusion as a large-scale power system has to be addressed and needs pivotal developments. Even with a highly successful development it will take still several decades before fusion power can be taken into consideration for the electricity generation market. However, the rate of progress and the promise of fusion have led all major industrial countries of the world to pursue substantial R&D programmes for the development of fusion power and to join the ITER project.

JET has incorporated many developments which originated from work on smaller facilities in the associated research laboratories. Examples are the development of the magnetic divertor, the "H-mode" operation with a significantly improved confinement above the standard or "L-mode", internal transport barriers which lead to further improvement of confinement and, associated to this, negative shear operation where the current profile is manipulated to produce optimized confinement. This "step ladder" of R&D is effective since smaller machines save time and money for the exploration of new physics. JET has also developed sophisticated control instruments and mechanisms which permit to guide plasma discharges within, but close to the borders of operational regimes. Based on numerous experimental results in different devices, notably the ones with similar plasma shape (Compass-D, ASDEX-UG, DIII-D, JT60U. JET), empirical scaling laws have been derived which allow predicting many features of ITER with reasonable confidence, including the operational performance of ITER.

Fusion technology has made also essential progress in many areas. Because the Ohmic heating by a plasma current strongly diminishes with increasing temperature, additional heating systems have been developed for injecting megawatts of neutral particle beam or

[15] Over the past decades concepts of toroidal magnetic confinement have proven to offer the best development perspective towards a fusion power plant and many other ideas which were initially investigated were more or less abandoned. Within the toroidal magnetic confinement class, it is the Tokamak which has progressed fastest. A strong toroidal magnetic field produced by external coils, a poloidal field generated by a large toroidal plasma current and an additional poloidal configuration (vertical field) produced by a second set of external coils provide the confining magnetic field. The simple axisymmetry makes the Tokamak easier to construct and facilitates the interpretation of physical effects.

[16] An alternative concept is the Stellarator where the basic magnetic confinement configuration is fully provided by currents in external coils. This enforces a more complex symmetry which is more difficult to calculate and to explore and the development of the Stellarator lags well behind the Tokamak. However, the Stellarator promises significant advantages, compared to the Tokamak, for stable steady-state operation of high performance plasmas.

[17] Power systems such as fossil, fission or renewable plants can be built at rather small scales. However, the underlying physics imposes that a base-load fusion power plant cannot be conceived at a unit size of much less than one GW electric. ITER, or in fact any other device with a "burning" plasma (i.e. a plasma where a substantial fraction of the heating power is provided by fusion reactions) must therefore be close to the size of a future fusion power plant. This makes each step in the development of fusion power comparatively expensive and implies long construction times.

electromagnetic power into the plasma. Temperature and density profiles can be controlled with these systems and advanced fuelling techniques.

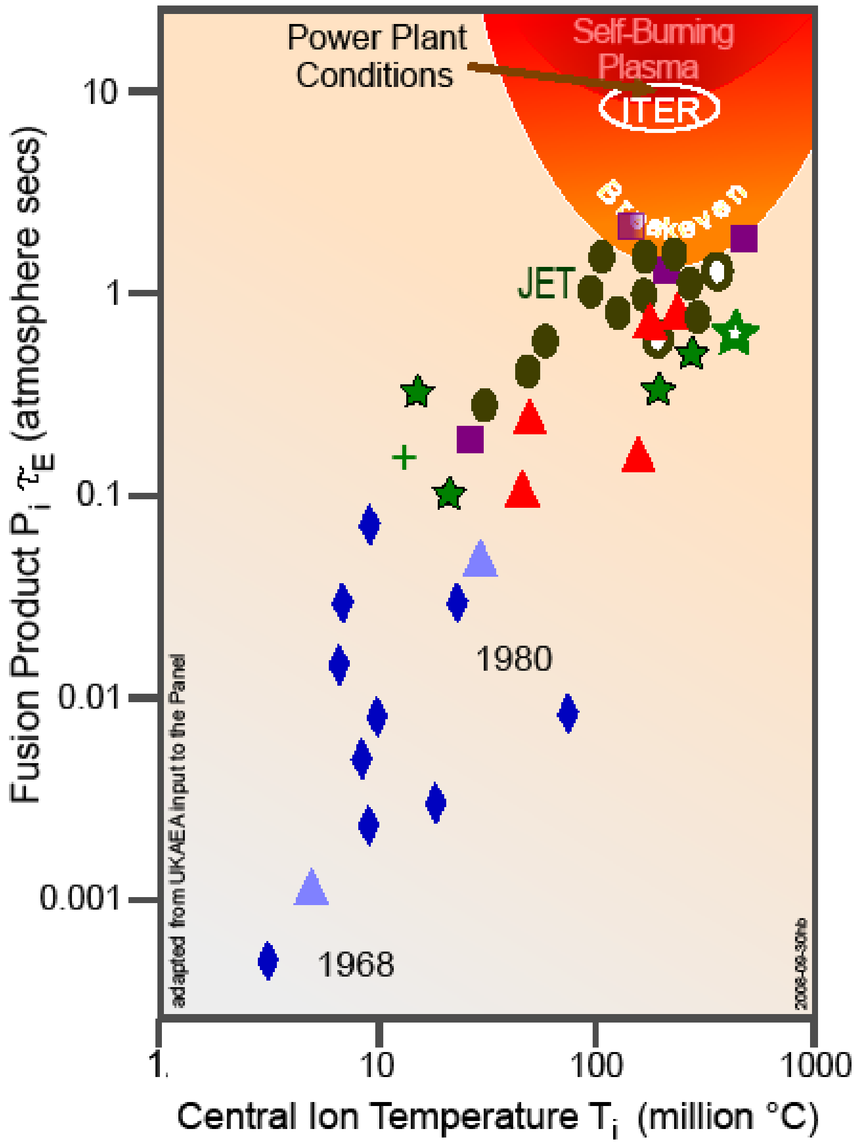

Fig 1: Progress in fusion R&D towards the burning plasma conditions of a power plant displayed as progress for the confinement quality (the product of plasma pressure times energy confinement time) versus the central ion temperature. (Adapted from input to the Panel by, and courtesy of, UKAEA).

Materials research has provided promising alloys for reduced/low activation structural steels, high heat flux tiles, and ceramic insulators, etc. which provide a basis for the development of reactor relevant structural and functional materials.

Fusion theory is progressing, but the coherent understanding of the complex plasma and plasma-wall interactions is still insufficient and (semi-)empirical methods are prevailing for the extrapolation from known to new territory. However, theory and simulation have seen a strong increase in capability and perspectives are being conceived for increasingly realistic first principle based numerical codes of fusion plasmas and their interaction with the plasma facing components.

C. Aims and Strategy of the EURATOM Fusion Programme

The long-term overall objective of the EURATOM fusion programme, embracing all the fusion activities in the Member States and associated third countries, is "developing the knowledge base for, and realising ITER as the major step towards, the creation of prototype reactors for power stations which are safe, sustainable, environmentally responsible, and economically viable" [18].

ITER shall "demonstrate the scientific and technical feasibility of fusion energy for peaceful purposes. To do this, ITER will have to create study and control a particular state of matter, plasma, which will produce a fusion power of some 500 million watts during extended duration and then nearly continuous discharges. For the first time in the world, this power will greatly exceed, by a factor of ten, the power applied to the plasma. ITER will also have to test components and technologies which are vital to a future industrial reactor and shall demonstrate their integration in one unit"[19].

ITER will not be the last development step before a commercial fusion power plant can be constructed. Both with regard to improvements in physics and technology but even more so in efficiency, reliability and availability further steps have to be made in a demonstration stage (DEMO) which will be the essential link towards the commercial regime[20].

Industry which today acts mainly as build-to-print supplier in fusion will have to become deeply involved in fusion, starting with substantial contributions to the ITER construction. Its role in DEMO must evolve towards the role of an architect-engineer and finally industry is expected to take over on its own responsibility the construction and further development of commercial fusion power plants. **Early interaction with utilities and industrial power plant constructors as well as licensing authorities is indispensable for defining the requirements which must guide R&D towards a first commercial fusion power plant.** For utilities it appears mandatory that design solutions must ensure steady-state electricity production at high efficiency, availability and reliability and at economically feasible prices. Construction companies will prefer known technologies and reduction of complexity. Licensing will require that all major elements of physics and technology must have been proven before construction can be started. Commercial power plant studies are essential for developing a coherent vision of these requirements which must adequately be taken into account when defining the design of a DEMO device which indeed shall accomplish the transition to the commercial regime.

In response to the demand for a rapid rebuild of our energy system towards sustainability it was recommended in 2001 to strive for an early entry of fusion into the electricity

[18] Council Decision for FP7 (Official Journal of the European Union L 54/21(en), 22.2.2007)

[19] COM (2003) 215 final

[20] The original strategy of the programme, conceived when JET was constructed, foresaw beyond JET a "Next Step" which now is being realised with ITER, followed by two further development steps before the commercial phase would be reached, first a DEMOnstration reactor (with some external control power still applied) for integrating all physical and technological features of a future fusion power plant into one device and subsequently a Prototype commercial reactor with a fully self-sustained plasma demonstrating the integral compliance with economic requirements, in particular the achievement of the necessary efficiency and reliability.

market by pursuing a "fast track" scenario[21]. ITER should demonstrate the production and extraction of sustained fusion power and serve as an enabling research machine regardless of the specific design of later commercial reactors. A DEMO machine, as the only and last step after ITER before entering the commercial phase, would have to demonstrate and validate all aspects of a future fusion power plant in an integrated facility. This comprises, in particular, also the technical and materials aspects: in time therefore for the DEMO construction an enhanced materials R&D programme must develop and qualify the first wall and structural materials which shall be used and which are expected to be close to the materials of future commercial reactors. Among other requirements, a materials irradiation facility must be built in parallel to ITER to enable the validation of the necessary materials in time. In this ambitious scenario the construction of DEMO should be started within 20 years (ten years after commissioning of ITER) and the inroad of fusion into the commercial market could be expected around the mid of this century. This projection depends crucially on rapid progress towards DEMO with ITER and with required complementary physics and technology R&D which would have to be fostered with correspondingly enhanced funds and without delays in political decision making. Intensive international collaboration on major facilities will be important for optimizing progress and use of resources.

Recently, the 2007 Strategic Energy Technology Plan of the European Commission[22] has identified the completion of the construction of ITER as a key benchmark for the next ten years and highlighted the importance of an early involvement of industry. A further option for accelerating progress has also been proposed in conjunction with this plan involving an (additional) early DEMO device (eDEMO) possibly to be constructed in parallel to the ITER first phase of operation based on near-term available technology and already largely developed physics scenarios. In this report this option will not be assessed.

The Panel's terms of reference demand "to develop a vision of the R&D required to make fusion energy production ready for commercial exploitation" i.e. to take into consideration the entire perspective towards the ultimate goal. Clearly, the choice of strategy is essential and must be discussed in the context of the current and future funding available to the fusion programme. A rapid and effective approach towards fusion energy does imply a vigorous parallel development in all necessary areas of fusion R&D i.e. ITER, materials technologies, components development and concept improvements in preparation for DEMO. **The Panel, following the Terms of Reference and being convinced that it is mandatory to bring fusion to the market within a useful time span, bases its considerations on a "fast track scenario" expecting that the funding situation will make this approach feasible. In doing so, the analysis presented in this report will focus on the essential core of the programme.** Additional activities and facilities would be desirable for further time and risk minimization such as e.g. suggested in the SET-Plan initiative.

[21] Conclusions of the Fusion Fast Track Experts Meeting (chaired by Sir David King) held on 27 November 2001 on the initiative of Mr. De Donnea, President of the Research Council. Based on the progress already achieved it was recommended to merge the DEMO and Prototype into one development step "that should be designed as a credible prototype for a power-producing fusion reactor, although in itself not fully technically and economically optimised". This scenario takes into account that nowadays it is understood to be favourable to operate a fusion power plant, rather than thought earlier in a fully self-sustained mode, in a controlled mode with a continuous external control power for additional heating and current drive systems. The importance of progressing in parallel with the major projects in fusion physics and in technology is highlighted: "The two major international ventures on fusion energy development, i.e. ITER and IFMIF should proceed in a co-ordinated way, with the realization of ITER starting in parallel with the detailed engineering design of IFMIF".

[22] "A European Strategic Energy Technology Plan (SET-Plan) 'Towards a low carbon future'" COM (2007) 723 final / SEC (2007) 1508 …1511

D. International collaboration in Fusion R&D

The EURATOM fusion research is a long-standing example for a European Research area and an excellent international cooperation. A substantial exchange of scientists between the major laboratories has been a hallmark. Internationally, the most prominent frame of collaboration is the ITER initiative for the next major step in fusion which started, after political incentives in 1985, with conceptual design activities in 1988 under the auspices of the IAEA and has eventually led in 2006 to the international agreement for constructing and exploiting the device[23]. In support of ITER the International Tokamak Physics Activity (ITPA) among the leading programmes and laboratories was established under the auspices of the IFRC[24] (IAEA) and has recently been invited to operate under the auspices of ITER[25]. Specific topical collaborations in fusion R&D are pursued in the frame of nine IEA implementing agreements. Among bilateral agreements the "Broader Approach" Agreement with Japan is of particular relevance (see below). Framework agreements for facilitating collaboration have been concluded between EURATOM and most other countries with interest in fusion research[26].

1. European participation in ITER

Europe has been chosen to host ITER and has committed itself to site ITER in Cadarache (France). The objectives of ITER and the shares and obligations by the Parties[27] are defined in the international ITER agreement. Europe, being the host of ITER, is responsible for delivering about 45% of the contributions. The international construction of ITER represents major technical and organisational challenges. A Joint Fund has been established which covers the expenditure of the international team and about 10% of the construction cost. The remaining 90% will be undertaken through in-kind contributions by the parties. The European domestic agency responsible for these contributions is Fusion for Energy (F4E, recently set up as a Joint Undertaking according to the EURATOM Treaty which is sited in Barcelona, Spain). F4E will develop, place, finance and supervise industrial contracts on behalf of EURATOM. The European contributions include a major or even sole responsibility for items in the following areas:

- toroidal and poloidal field magnets;
- the main vacuum vessel;
- the divertor;
- vacuum cryopumps and leak detection systems;
- remote handling equipment for the divertor and neutral beam system;
- in-vessel viewing and metrology systems;
- hydrogen isotope separation and water detritiation;

[23] Originally construction was foreseen among the initial partners (Europe, Japan, Soviet Union, USA) to start by the end of the 1990s. Delays were incurred by a redesign towards a smaller device, the temporary withdrawal of the US from the ITER initiative and complex competitions inside Europe and between Europe and Japan on the siting of ITER.

[24] IFRC: International Fusion Research Council (IAEA, Vienna). IAEA also publishes the journal "Nuclear Fusion" and supports collaborations on Atomic and Molecular Data for Fusion and Fusion Research using Small Tokamaks.

[25] This proposal has been endorsed by the ITER Council in June 2008

[26] Currently these are Japan, Kazakhstan, South Korea, Russia, Ukraine, USA as well as Argentina, and Canada. Negotiations are ongoing with China and India. Some activities with institutions in the former Soviet Union are also supported through the ISTC and STCU programmes.

[27] The ITER parties are so far, besides Europe, China, India, Japan, South Korea, Russia and the US.

- components of the three heating systems foreseen on ITER: ion cyclotron, electron cyclotron and negative ion neutral beam injection;
- a number of plasma diagnostics;
- buildings and services.

The technologies needed for building ITER have been developed essentially by the four initial ITER parties. In Europe this was undertaken through EFDA[28] jointly with associated research laboratories and industry. Full-size prototypes or major scaled mock-ups were constructed during the past decade for several essential components with a strong participation of Europe. After conclusion of the ITER agreement a design review was requested which has been finalised in 2008. It incorporates into the design a number of R&D results which were achieved after the final engineering design was presented in 2001 and also addresses remaining technical issues which should be resolved in time before the corresponding components are on the critical path. **These R&D needs in support of the European share in the ITER construction and the corresponding required facilities are of high priority.**

2. *European share in the Broader Approach Agreement*

In the context of the ITER site negotiations with Japan an initiative for collaboration between Europe and Japan has been agreed which encompasses not only support to ITER but also steps towards DEMO. This "Broader Approach Agreement" foresees European participation in

- The Engineering Validation and Engineering Design Activities (EVEDA) of the International Fusion Materials Irradiation Facility (IFMIF) which are being coordinated in Rokkasho (Japan) and shall be concluded in 2011;
- A superconducting upgrade of the Japanese JT-60 Tokamak, called JT-60SA ("Super Advanced"), which will act as a "satellite Tokamak" in preparation and support of ITER;
- An International Fusion Energy Research Centre (IFERC) in Japan which will include a DEMO design and R&D coordination centre, a computer simulation centre and an ITER remote experimentation centre.

Financially, these activities are mainly supported by voluntary contributions from several European Member States (currently France, Germany, Italy, Spain) and Switzerland[29]. **The Panel points in particular to the importance of the IFMIF related activities. For the IFMIF EVEDA some additional R&D activities in support of those which were agreed with Japan have been identified and should be pursued by the European laboratories.**

3. *The international fusion programmes*[30]

Over the past decades the prime international fusion programmes, besides the EURATOM programme, were the Japanese, the Russian and the US programmes. Meanwhile substantial fusion R&D programmes have been established in China and South Korea. India, Brazil and others are following. All ITER parties are engaged in major fusion experiments and technological developments for a future fusion power plant.

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[28] Until the end of 2007, when these tasks were transferred to F4E

[29] The total of European financial contributions is estimated at around 350 M€. Belgium is considering to join the contributors

[30] The following paragraphs are based on information by the international members of the Panel and, regarding S. Korea and India, on ITER news and other public sources.

China has recently constructed the superconducting EAST Tokamak (ASIPP, Hefei) and operates the smaller HL-2a (the former ASDEX device from Europe at SWIP, Chengdu) and superconducting HT-7 (ASIPP, Hefei) devices. Furthermore, there are 3 small Tokamaks in universities supporting, with other specialised facilities, a strong education and training and basic plasma science programme. The HL-2a Tokamak shall be upgraded to capabilities similar to ASDEX-UG. The HT-7 device has achieved 400s discharges at currents of ~50 kA which were sustained non-inductively by ~100 kW LHCD for about 150 – 200 s. Investigations on these smaller devices are oriented towards general Tokamak physics. The EAST Tokamak is a 1-1.5 MA class superconducting device with a strong shaping capacity and heated by 10 MW (2010) / <25 MW (2015) LHCD, ICRH and NBI systems. The device is equipped with actively cooled plasma facing components. The present carbon surface shall be replaced by Tungsten in the divertor and in about 6-8 years a transition to a full actively cooled metallic wall is planned. The EAST physics programme is oriented at realizing Advanced Tokamak steady-state operation and comprises hardware and tool development. It aims at demonstrating ultimately 1000s shots with $ß_N > 4\text{-}6$. Internationally the team is strongly involved in the ITPA. With regard to magnetic confinement fusion energy production a gap analysis regarding necessary R&D has been undertaken and a conceptual design for a DEMO concept is underway including R&D for key technologies. For the Chinese R&D on electricity from DT fusion a roadmap has been established which foresees apart from low energy gain devices, ultimately a fusion power test reactor (FDS-II) similar in size to ITER with the objective of delivering 2500 MW with Q~30.

India joined ITER in December 2005. It operates since a good decade two small Tokamaks and made a major step with the decision, in 1995, for the construction of the SST-1 Superconducting Steady-state Tokamak which, in international comparison is still a smaller device, however capable of high elongation and triangularity and equipped with LHCD, ECRH and NBI systems. SST-1 is now in the final stages of manufacture at the Institute for Plasma Research (IPR) near Gandhinagar. Fusion Technology activities relate essentially to ITER procurements as well as to components and systems for SST-1. Indian fusion laboratories have collaborated with Europe mainly under the IAEA small Tokamak Implementing Agreement.

Japan is engaged strongly in Tokamak research and reactor studies at the JAEA[31] and on concept improvements (in particular the Stellarator but also the Spherical Tokamak, Reversed Field Pinch and the Mirror) in NIFS[32] and universities as well as on general fusion science in a large number of universities. Along the Tokamak line the flag ship is the JT-60U device (having a similar function as the European JET, but no DT capability) which has achieved record values in equivalent Q and, in reversed shear operation, bootstrap current fractions of ~70% for 8 s (not yet fully steady-state controlled). Recently the device has been upgraded, inter alia, in long pulse neutral beam power. Its operation will terminate in 2008 when its R&D missions have been accomplished and the device shall be upgraded / replaced by the larger superconducting JT-60SA for which the design is under finalisation. This device, expected to start operation in 2016, has a crucial mission in the Japanese programme aiming at contributing to the early realization of fusion energy by supporting the exploitation of ITER and addressing key R&D issues for DEMO. As mentioned before, Europe is involved, through the Broader Approach Agreement in the construction and for a limited time (five years) in the exploitation of this device.

[31] Japan Atomic Energy Agency, formerly JAERI

[32] National Institute for Fusion Studies, a compound of university institutes

DEMO concepts are actively developed and a strong effort is ongoing in fusion simulation projects supported by high performance computers. At the Rokkasho site, originally foreseen by Japan for ITER, an International Fusion Energy Research Centre (IFERC) has been founded in 2007 where the Broader Approach Activities will be carried out within the next ten years. The second strong line is Stellarator research. The world-wide largest and highest performing Stellarator currently under operation is the Large Helical Device (LHD, 30 m^3 plasma volume - only the European W7-X, planned to be commissioned in 2014 will be of similar size). The verification of initial objectives in hydrogen operation is close to completion and it is planned, likely with increased heating power, to start soon deuterium experiments. A long-standing collaboration in a wide variety of topics between the Japanese and European programmes has been taking place under bilateral and IEA implementing agreements such as on the Stellarator, Plasma-Wall and Large Tokamaks. For the latter, coordinated experiments are conducted on JET and JT-60U. The Broader Approach will substantially intensify partnership between Japan and Europe.

Russia: The Tokamak is an invention of scientists in the former Soviet Union. In the years following 1968/9 when first high temperature confinement was confirmed in the T3 device a world-wide reorientation took place towards the Tokamak concept. The Tokamak line has seen a continuing development in Russian laboratories where several devices are under operation, T10 (Kurchatov Institute, Moscow, a large aspect ratio device with R/a~5), the superconducting T-15 (Kurchatov Institute, currently starting operation), T11 (Triniti, Troisk) and the smaller TUMAN 3 and 4 devices in the Ioffe Institute, St. Petersburg where also the Spherical Tokamak “GLOBUS” is under operation in an intensive worldwide cooperation in this area. There is also a small university based Stellarator. At the Budker Institute (Novosibirsk) research on the Mirror / Gas-Dynamic Trap concept is being pursued aiming at developing a potential candidate for a volumetric fusion neutron source. While Russia puts great emphasis on its involvement in ITER, some efforts are also on conceptual developments for a DEMO. Significant activities are underway in fusion technology with leading contributions in the superconductor and the ECRH gyrotron development as well as in liquid metal wall technology and materials research. Collaborations with the EURATOM fusion programme exist in many areas; particularly successful are the gyrotron development and the activities on, and for, JET which concentrate on diagnostic systems, pellet injection, modelling and transport physics but there are also activities e.g. in the areas of materials modelling and materials irradiation. There is strong interest in further intensifying collaborations, inter alia, possibly, with a major ECRH project for JET.

South Korea has started fusion research in the 1970’s and constructed small-scale Tokamaks and a Mirror during the 1980’s and 1990’s. In 1995 a National Fusion Programme was established and the construction of a major superconducting Tokamak, KSTAR, decided. This device went now successfully into operation and shall be the basis for the national fusion R&D plan in support of South Korea’s international participation in ITER. KSTAR shall have flexible heating systems and should serve over the coming years as South Korean ITER pilot plant, providing useful knowledge and data, and contribute to steady-state oriented advanced Tokamak research (the device has pulse lengths up to 300 seconds). KSTAR shall also be a facility for strongly enhancing the country’s involvement in other international collaboration in fusion R&D.

USA: The first substantial DT experiment on a Tokamak was successfully carried out in the US device TFTR, built in parallel with the European JET. The US programme is today oriented towards advancing plasma science, fusion science and fusion technology, the knowledge base needed for a future fusion energy source. The US was founding partner of the ITER initiative but withdrew in 1998 and re-joined in 2003. Apart from its contributions to the ITER construction the major emphasis of the domestic programme is on the Tokamak line with the medium-size, ITER-shape devices DIII-D (General Atomics, San Diego, similar in capability to ASDEX-UG) and Alcator C-Mod (MIT, Boston, a compact, high magnetic field device) as well as the Spherical Tokamak NSTX (PPPL, Princeton). In a five years' perspective the experimental programme on DIII-D is proposed to evolve with ITER's needs which will require a range of substantial upgrades for heating, active RMP coils, wall armour for divertor and vessel, pellet and jet injection and diagnostics. Work on Alcator C-Mod which is complementing this strategic approach focusses on physics aspects namely transport, H-mode pedestal, plasma-wall interaction, wave-plasma and macroscopic stability. Furthermore, a small Stellarator[33] and MST, a substantial Reversed Field Pinch experiment, are being pursued and a number of alternative concepts are investigated on a smaller scale. In combination with a strong programme on fusion plasma simulation the experimental activities shall provide the basis for a decision on the next step in fusion energy development in the US. Initiatives under consideration in the US fusion community are a "Fusion Development Facility" (General Atomics, a DT steady-state low Q machine using copper magnets), a Components Test Facility (Oak Ridge, based on the Spherical Tokamak concept for DT operation at extended pulses (100's of seconds) for nuclear components testing) and a Spherical Tokamak focused on plasma wall interactions at PPPL. The US laboratories have since long very intensive collaborations in particular with JET where activities focus on diagnostics, ICRH (antenna), and a wide range of coordinated experiments. There are also intensive interactions with many European laboratories both in fusion physics and technology. A substantial inertial fusion programme is carried out, however with a major orientation towards other applications than fusion energy for electricity generation.

Of particular interest for international collaboration regarding R&D in support of ITER are the future JT-60SA to which Europe is already affiliated through the Broader Approach Agreement, the EAST Tokamak in China, the upgraded T-15 in Russia, KSTAR in South Korea and the US devices DIII-D and Alcator C-Mod. The new superconducting devices have the potential of contributing significantly to long-pulse physics in ITER / DEMO relevant geometry. These devices will have advantages over circular long-pulse Tokamaks in particular if they will be equipped with reactor-relevant wall materials and with sufficient current drive capability. In the case of EAST tungsten coverage of the wall tiles is being contemplated for a later operational phase and T-15 will investigate apart from graphite also tungsten and lithium wall elements. Besides the satellite-class device JT-60SA also these modern medium-size devices have a high potential for synergy with R&D interests of the European fusion programme.

Overall, there is a high interest in the further development of fusion R&D among all these major international players. For all of them ITER is the most important step towards commercial fusion power which needs now to be brought to success and the national programmes are strongly oriented towards supporting their involvement in ITER. **The Panel recommends that the European fusion programme should seize opportunities for fostering existing international collaborations and for establishing new ones for participating in the exploitation of the new generation of devices in the international programmes.**

[33] A medium sized Stellarator construction project, NCSX at Princeton has recently been cancelled.

Table I.1: Major Tokamaks in the world								
Device	Opera-tion since	Country	Configu-ration	Steady-state capability	I_p (MA)	B_t (T)	R (m)	a (m)
JET	1983	EU	Divertor		4-5	4	2.96	1.00
JT-60U	1991	JP	Divertor		3	4	3.40	1.00
D III-D	1986	US	Divertor		1-3	2.1	1.66	0,67
TORE SUPRA	1988	FR	Limiter	SC Magnets. Actively cooled first wall	2	4.2	2.4	0.75
KSTAR	2008	KR	Divertor	SC Magnets	2	3.5	1.80	0.5
Alcator C-MOD	1993	US	Divertor		2	8.0	0.67	0.22
FTU	1990	IT	Limiter		1.6	8.0	0.93	0.30
ASDEX-Upgrade	1991	DE	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	CN	Divertor	SC Magnets, actively cooled first wall	1(1.5)	3.5(4)	1.85	0.45
T-15	2008	RU	Divertor	SC Magnets	1	3.6	2.43	0.42
TCV	1992	CH	Divertor		1	1.54	0.88	0.25
TEXTOR	1981 (1994)	DE	Limiter / ergodic divertor		0.8	3.0	1.75	0.47
HL-2a	2002	CN	Limiter		0.5	2.8	1.64	0.40
HT-7	1993	CN	Limiter	SC Magnets	0.3	2.5	1.22	0.27
COMPASS	2008/9	CZ	Divertor		0.35	2.1	0.56	0.2
SST-1	soon	IN		SC Magnets	0.2	3.0	1.1	0.2

Table I.2: Key Characteristics of ITER and JT-60SA, both decided for construction								
Device	Foreseen Operation	Country	Configu-ration	Steady-state capability	Ip (MA)	B_t (T)	R_0 (m)	a (m)
ITER	2018	INT	Divertor	SC Magnets	15	5.3	6.32	2.02
JT-60SA (2008 data)	2016	JP (in collab. with EU)	Divertor	SC Magnets	5.5	2.3	~2.95	~1.15

II. Research needs for a rapid and efficient realisation of fusion energy

ITER, the internationally agreed project for the first magnetic confinement device with a burning fusion plasma, is currently starting construction in Europe and is expected to be commissioned in a decade from now. A substantial R&D programme has been performed for ITER during the design phases including the validation of major design solutions on mock-ups and prototypes. Still there are R&D issues to be solved during construction which are to be contracted by the domestic agencies (F4E in the case of Europe) to industry and fusion laboratories. Beyond this range of R&D projects directly related to ITER construction the Panel has identified a substantial need for accompanying work in order to support and accelerate the start-up of ITER exploitation and the further progress on ITER. Work extends to many areas of physics and technology. An example is plasma operation where the base-line scenarios for ITER are well assessed but where there is a highly interesting scope for improved scenarios which should already be explored and tested on the existing Tokamaks during the ITER construction period.

DEMO, in a fast track scenario, shall demonstrate that a fusion device can operate and produce electricity in a way which can be directly extrapolated towards commercial power plants. This is an ambitious step beyond ITER and the Panel notes that numerous improvements in physics and technology need to be developed within the period of ITER construction and its first phase of operation in order to enable a DEMO design and to prepare for construction within the desired time horizon. With regard to the near-term, the Panel highlights in particular the required R&D for ensuring capability for steady-state operation and materials research. Solutions for the former must be achieved via development of advanced physics features and improved components for advanced control, fuelling, exhaust, heating and current drive and for the latter by a vigorous materials development and testing programme.

Ultimate success in fusion R&D requires the development of continuously operating, efficient and reliable fusion power plants. Achieving this goal still depends on pivotal developments in physics and technology which will require very substantial efforts.

A. R&D needs for specific missions

For discussing and managing research needs it is necessary to use a consistent structure of the overall R&D scope. While a topical frame might be most straightforward it may not be most suited for the complexity of fusion research where important interrelations exist between the different physics and technology fields which must be taken into account. The Panel will follow a scheme introduced by EFDA and the fusion bodies which defines a coherent set of seven missions, covering the entire topical range. Under these missions the research players from the various relevant physics and technology fields are grouped together for accomplishing solutions to complex R&D issues. These are discussed below in a sequence which is more or less describing the path from ITER towards DEMO and the commercial fusion reactor: it is necessary to achieve (1) *burning plasmas* in ITER which must be shown (2) to allow *reliable operation*; the future devices must be equipped with suitable (3) *first wall materials* and should operate in (4) *long pulses and (ultimately in) steady-state*. Furthermore, it is simultaneously necessary to (5) *predict fusion performance* for ITER and the further development steps towards a commercial plant and to develop and qualify (6) *materials and components suitable for* the ultimate full-power *nuclear operation*. Progress on ITER and by accompanying R&D should be brought together in a (7) *DEMO integrated design oriented towards high availability and efficient electricity production*.

This structure does not describe a sequence of temporal priority. For example, the development and qualification of suitable structural and functional materials for the first wall and the structural components do imply very long lead times and progress is urgent. Also, initial steps for an integrated design of DEMO must be started early – as it cannot be excluded a priori that possible issues may be identified which would require a reorientation or re-balancing of near-term R&D actions.

1. Burning plasmas

The first and foremost task of a fusion device is to confine the fusion plasma at sufficient density by magnetic fields and to heat it to temperature regimes of some 100 million degrees where fusion reactions in the "burning plasma" will release energy[34]. In the absence of theoretical ab-initio models progress in confinement is based on empirical scaling. JET, the largest European fusion device and the only one presently capable of operating with the fusion fuel foreseen for a future reactor (a mixture of deuterium and tritium), has generated, in 1997, 16 MW of fusion power for seconds in a magnetic divertor configuration adapted from the ASDEX-Upgrade experiment. The ITER plasma configuration is based on this design line and is supported by an experimental scaling involving all major Tokamaks in the world.

Mission 1: Milestones
Start of ITER operation:
- Predictive capability for burning plasma physics,
- Control capability for pulse management and for off-normal events

End of ITER Phase I
- Burn control capability
- Burn optimisation,
- development of diagnostics, actuators

The gross behaviour of burning plasmas is usually extrapolated based on empirical scaling laws. With growing experimental and theoretical insight into the complex physics new challenges for the stable confinement of the plasma were identified which require R&D efforts. Major issues are to optimize the fusion burn while steering plasma current, density and temperature profiles within stringent stability and operational limits and to control the plasma-wall interaction.

In ITER, DEMO and future fusion power plants the fast alpha particles resulting from the fusion reactions must be well confined inside the plasma while slowing down in order to heat the plasma[35]. Mechanisms are known which could result in instabilities and premature loss of fast particles. The domain of fast particle effects and their control will be a core area of plasma physics investigations on ITER and should be explored and tested as far as it is possible on JET-class devices. Much of these physics aspects are also relevant, and can be transferred, to the Stellarator line.

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[34] The most adequate measure of the confinement quality of the hot fusion fuel is the product of density (n), temperature of the plasma (T) and the confinement time (τ_E). This "triple product" must attain for a burning plasma a value $nT\tau_E \geq 10^{21} keVs/m^3$. In experiments performed so far, the triple product has increased over many orders of magnitude close to the required value.

[35] So far, the largest fusion power in a magnetic confinement fusion device was generated in JET. Albeit a power amplification factor of Q~0.6 was achieved, the alpha particle fraction (20% of the fusion power is carried by alphas) in the JET plasma is still too small to fully assess non-linear feed-back effects of these fast particles on stability transport, current generation, heating in the plasma and on the fusion reactivity profiles. This will be different in ITER and even more so in a fusion power plant. Nevertheless, a range of fast particle effects has been explored in JET and other devices by experiments with artificial creation of a fast particle population and through numerical modelling. Effects are complex and the modelling of burning plasmas needs to be extended (see mission 5).

2. *Reliable Tokamak operation*

Mission 2: Milestones

Start of ITER operation:

- Wall conditioning methods
- Actuator requirements for operational control
- Pulse management and control
- Off-normal event control

End of ITER Phase I

- DEMO relevant operation and control management system

Due to the large toroidal plasma current the Tokamak possesses a substantial internal free energy which can be released in instabilities and a reliable plasma control is a major challenge. Some of the instabilities are of a global nature such as disruptions (including vertical displacement events) which may lead to the termination of the plasma discharge and the commutation of a large part of the current from the plasma to conducting structures of the vessel and in-vessel components. Thereby large forces may occur. The larger the fusion device the more important it is to avoid such off-normal events. Fully developed disruptions must become very exceptional. In most cases the development of disruptions can be intercepted at an early stage and a number of techniques for the detection of precursors and for mitigating the evolution of disruptions have been identified. Their development towards reliable control tools is a high priority where medium size devices must contribute substantially before developed techniques should be transferred to JET (and future satellite devices) and ITER. Considering the importance of minimizing disruptions on ITER and mitigating their impact the Panel emphasizes to devote sufficient efforts on this R&D topic.

Off-normal events are not the only issues for plasma control. Starting with plasma-wall interactions and the plasma break-down at the beginning of a discharge up to its termination there are many aspects which require operational control in order to optimize stability, confinement or transport and to avoid negative impacts on the device or its performance. Control techniques for reliable operation comprise diagnostics and actuators with feed-back systems for position and shape control, current drive, heating and fuelling of the plasma, for exhausting the helium which is generated by the fusion burn, etc.

The power flux from the boundary plasma to the wall is naturally unstable, i.e. intermittent and local, which may lead to unacceptable local heat loads to the divertor and the first wall. If these “Edge Localized Modes” (ELMs) are left uncontrolled they may provoke premature erosion and corresponding deterioration of the first wall lifetime. Recent studies indicate that magnetic perturbations in the edge plasma or dedicated shallow pellet injection can induce more benign i.e. higher frequency ELMs with lower amplitudes. These and other techniques must be further explored in existing devices and be developed towards reliable control tools compliant with ITER burning plasmas.

The Stellarator which shares most of these aspects with the Tokamak is less demanding for control of current driven instabilities since (in a Stellarator of the W 7-X / Helias type) there will be no net-toroidal current. Consequently, this configuration lends itself more easily for a steady-state fusion power plant since no substantial current drive or “bootstrap current” with associated steep pressure profiles is necessary.

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

Eighty percent of the fusion power is carried away by 14.1 MeV neutrons whose energy is absorbed in the blanket. The remaining twenty percent is delivered to the plasma by alpha particles which are finally exhausted in the divertor. The associated power flow is expected to be of order 1 MW/m^2 to the first wall of the main chamber and up to 10 MW/m^2 in the divertor for ITER. Materials and first wall components have been developed for JET and other devices which can master these power flows with acceptable lifetime for ITER. However, for DEMO these requirements may be even higher and further developments are needed.

Mission 3: Milestones
Start of ITER operation:

- Erosion, deposition, dust diagnostics
- T-removal technologies
- Validated modelling capability for plasma wall interactions (PWI)
- Compliance of first wall capabilities and plasma scenarios for ITER

End of ITER Phase I

- Capability for detailed gas inventory
- DEMO relevant PWI diagnostics
- Compliance of first wall capabilities and plasma scenarios for DEMO, including nuclear aspects

The first wall and the divertor are also exposed to powerful charged and neutral particle fluxes, to fusion neutrons and to a wide spectrum of electromagnetic radiation up to the ultra hard x-ray domain. A range of physical and chemical processes may lead to the release of impurities which could be of concern for the power balance of the plasma. Interaction of the neutral beam and radio wave heating systems with first wall areas add to the problem. An integral and important component of this mission is to develop plasma scenarios which are compatible with the first wall and exhaust requirements.

In principle surface materials of low atomic number which, when entering the plasma, will be rapidly fully ionized should result in lower radiation losses. However, for surface areas which are exposed to very high power flux, high-Z materials (coatings) with low sputtering yields such as tungsten can be more advantageous. Concepts for the first wall must be further developed for compliance with the requirements in ITER and DEMO, for the latter the higher nuclear requirements are a significant additional constraint.

Another requirement is that the first wall material must not lead to unacceptable absorption or co-deposition of the deuterium and tritium fuel since this would imply an undesirable or even unacceptable large inventory of immobile tritium. This problem was identified for uncoated graphite walls. ITER will use Beryllium in the main chamber and tungsten or tungsten coatings for the divertor tiles, at least when moving from the initial phases to DT operation. For a fusion power plant it is still unclear whether solutions with more than one wall material will be advisable. Experiments are underway to assess and optimize solutions with only one material e.g. tungsten. The properties of metallic walls regarding dust, T retention and removal need to be further quantified.

4. *Physics and technology of long pulse and steady state*

A prime requirement for a fusion power plant which is aiming at the production of base-load electricity is continuous thermal output from the fusion core. Under normal operation a fusion power plant should maintain the burning plasma, if not in steady state, then for very long pulses[36]. ITER shall achieve long pulse (ultimately steady state) operation with a power amplification Q=5 or better and DEMO should demonstrate plasma operation consistent with continuous electricity output in a future fusion power plant.

By principle the Tokamak is not a steady-state device because the required large toroidal plasma current (15 MA in ITER) is generated by transformer action. For long pulse or steady-state operation in a Tokamak fusion power plant non-inductive current drive systems[37] must be used and a high fraction of the required current shall be generated by a "bootstrap" effect linked to the steep radial density and temperature profiles in the plasma and the remainder of the current by non-inductive current drive systems. The bootstrap effect was predicted in the 1970's and has since been confirmed in many experiments. However, for the desirable high fraction of this self-generated current the profiles are less robust to MHD instability and this problem must be assessed and remedies found. This will be a major task for smaller experiments and for subsequent application in ITER.

Mission 4: Milestones

Start of ITER operation:

- Development of hybrid and steady-state scenarios for ITER
- ITER relevant Lower Hybrid Current Drive demonstrated
- Coupling of Ion and LH Current Drive to high performance plasmas
- Control for advanced scenarios
- Confirmation of advanced Stellarator configuration and physics

End of ITER Phase I

- Definition of DEMO long pulse and steady-state scenarios
- Proof of Heating and Current Drive systems for DEMO
- Demonstration of steady-state high performance Stellarator operation

The Stellarator does not have a large net toroidal plasma current since all elements of the basic confining magnetic field configuration are generated by currents in external coils. Thus, as already mentioned under Mission 2, the Stellarator lends itself for steady-state operation.

There are several key R&D areas for this mission such as superconducting technology, actively cooled high-power load in-vessel components, fuel control systems, heating and current drive systems and the domains of diagnostics and control. The superconducting coils represent a major investment item with high technical challenges. If the use of novel high-temperature superconducting materials for the magnets would become feasible, technical constraints would be alleviated substantially both for the Tokamak and the Stellarator.

[36] There is a large thermal inertia in the steam cycle of a power plant: the high energy stored in the phase transition from water to steam is able to bridge short gaps (of order one quarter of an hour) in the fusion power production. Correspondingly, any interrupted plasma operation should be resumed within this time interval. The Panel notes that pulsed operation is, however, possible only if the associated cyclic stresses do not lead to an unacceptably short facility lifetime.

[37] Auxiliary systems are those which are needed to fuel, heat and control a burning fusion plasma. For ITER, and likely also for future fusion devices, the workhorses for heating the plasma are so-called neutral beam injectors (NBI). These are very substantial components in which negative fuel ions are accelerated to energies between 500 keV and 1 MeV, neutralised and injected into the fusion plasma. Here they undergo ionization and slowing down by interaction with the plasma and thereby replenish and heat the fusion plasma. Besides NBI, radio and high-frequency electromagnetic waves are used for heating the plasma, applying (localized) current drive and for providing control of certain instabilities. Heating systems will apply up t0 +++60 MW to ITER in pulses. In the longer term they must be made fit for cw operation.

A major factor for the re-circulating power and the overall efficiency of a fusion power plant is the amount of needed non-inductive current drive and heating by auxiliary systems. Adapting the performance and enhancing the efficiency of these systems are therefore important development targets. Simultaneously the development of physics scenarios must be carried forward which combine a high bootstrap current fraction with stability and good confinement at high plasma density[38]. So-called advanced scenarios for the Tokamak would be most desirable but pose the largest challenges. Hybrid scenarios with less demanding profile and stability requirements have been identified which could be used for very long pulse operation. Such scenarios must be explored on smaller machines and then be scaled up to ITER while simultaneously attempting to minimize the necessary heating, current drive and control tools.

5. Predicting fusion performance

> **Mission 5: Milestones**
> **Start of ITER operation**:
> - First principle based predictive capability for the H-mode pedestal in ITER
> - Coherent numerical Tokamak for gross planning and evaluation of experiments on ITER
> - Numerical models capable of efficiently utilizing then actual computing systems
>
> **End of ITER Phase I**
> - Availability, for DEMO design, of a numerical Tokamak and Stellarator which has been validated on the essential devices and ITER burning plasmas.

Enhancing the capability of modelling fusion plasmas must go hand in hand with progress in experiments. Currently, many of the salient features of a Tokamak discharge are modelled in semi-empirical approaches lacking consistency and ability for novel predictions. In the first place it is therefore the theoretical understanding of the burning plasma physics which, although much progress has already been achieved, must be further improved taking into account all the features arising from confinement geometry, stability, turbulence, plasma wall interaction, fast particle effects, isotope and impurity behaviour, etc. Progress in this direction should allow to develop first-principles-based models capable of covering the complexity of a fusion plasma in a more adequate manner than today. These models must be transformed into numerical codes which can utilize efficiently present hundreds-of-teraflop, and even more so future multi-petaflop, installations for progressively mapping the full plasma volume in three dimensions at all relevant scales. Once this has been achieved, reduced models can be derived for specific purposes requiring less computational effort. As a result a “numerical Tokamak” (followed by a numerical Stellarator) would emerge which, integrating all relevant physics features could be used for detailed predictions regarding novel scenarios. Equally, for real-time control of plasma discharges, specific routines can be developed e.g. by parameterizing the relevant code simulations. Advances in modelling could lead to substantial time and cost savings for future experiments on ITER and other large fusion devices: experiments could be better targeted to optimal operation scenarios while operational risks (e.g. by exceeding stability boundaries) could be minimised.

The Panel emphasizes the importance of developing a “numerical Tokamak”. This involves a substantial theoretical / numerical effort and will require extensive

[38] Fusion power strongly increases with plasma density. Tokamak experiments exhibit usually an empirical upper density limit (“Greenwald” limit) which is proportional to the plasma current. Understanding the underlying physics could lead to ways of overcoming this limit as it is possible in Stellarators.

benchmarking against the widest possible range of operational conditions and types of fusion machine, in particular against the burning plasmas of ITER, in order to validate the correctness and completeness of the implemented physics. During its development intermediate targets must be to further enhance understanding and the predictive capability of major features in existing devices (H-mode threshold, ELM cycle, Greenwald density limit, Alfvén stability etc).

6. Materials and components for nuclear operation

All components close to the plasma including parts of the mechanical structure of the reactor will be exposed to intense 14 MeV neutron irradiation which leads to embitterment, swelling due to hydrogen and helium production and to transmutation into other isotopes and elements. For plasma-facing materials the heat load and interaction with the plasma constituents is an added constraint.

Mission 6: Milestones

Start of ITER operation

- Validation of reference EUROFER for TBMs
- Availability of divertor armour materials (W?)
- Pre-selection of DEMO Divertor and Blanket concepts

End of ITER Phase I

- Qualification of structural and functional materials for DEMO
- Validation of first wall material
- Clearance and recycling of DEMO activated materials ensured

The objective of materials research is that despite the harsh environment the structure of the plant should withstand the neutron impact for the required design lifetime without unacceptable deterioration of its functional quality and without generating a radioactive inventory with long decay times. The Panel was presented with the development of steels which under irradiation with currently available neutron sources show a potential to withstand for a reasonable time the damage which will be acquired in DEMO.

To limit the activation of materials requires a suitable optimisation. Predictions are that materials could be feasible which allow reprocessing about 100 years after shutdown with available industrial techniques. This would allow to essentially re-utilize these materials. Still, such materials have to be further researched, and reliable fabrication and joining technologies be developed. Validation is an important task, however, it is associated with long lead times. The availability of a suitable high fluence neutron facility with a fusion relevant spectrum is an urgent issue which is being tackled with the IFMIF EVEDA (Engineering Design and Validation Activity). IFMIF is a major enterprise for which crucial technological challenges have to be overcome during EVEDA. The R&D plan is fixed by the Broader Approach Agreement. However, when working on these tasks some additional needs were identified for accomplishing the European part[39]. The Panel considers the additional R&D tasks to be justified and

[39] These comprise: (1) Validation of energy absorption, in 1-6 MeV electron beam experiments, of the required interaction of the ion beam with the liquid metal target to eliminate the risk of turbulence or cavitation (estimated 3 M€); (2) Verification of the capability of non-linear beam optics to transform the circular cross section of the IFMIF deuterium beam at the end of the accelerator into the rectangular foot print needed on the target (estimated ~4 M€); (3)Experimental confirmation of a novel compact variant of the Drift Tube Linac accelerator which could provide savings for the IFMIF building (estimated ~15 M€).; (4)Improvement of the diagnostic capability in the high energy beam section for better operational control and safety (estimated ~3 M€); (5) Establishment of a comprehensive data base for the (mechanical) properties of relevant IFMIF materials on existing irradiation facilities (estimated +++ M€).

recommends their execution[40]. Measures for ensuring a rapid time schedule of IFMIF and minimising the risk for the construction of this facility should be taken.

Another priority is the development of materials and components for the fuel cycle (in particular the blanket for tritium breeding and thermal energy extraction) and the power and particle exhaust from the plasma. Altogether this mission has a scope well beyond ITER (where a test blanket module will provide first information on the choice of technologies); it is crucial for enabling DEMO and must lead on DEMO to the demonstration of full power plant relevance of all materials and components.

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

The high investment and long time span needed for the further steps in fusion R&D and the high integration of the components of the core of a fusion power plant require a strong forward-looking guidance. The next major step beyond ITER, DEMO which shall demonstrate the commercial viability of a fusion power plant, must be defined carefully. A DEMO design will be also important for focusing ITER R&D on those issues which are crucial for the feasibility of a future power plant. On the other hand, an integrated design of DEMO will depend on the full physics picture which still has to be developed and confirmed in experiments on ITER and can therefore only comprehensively be addressed once ITER has undertaken long-pulse high-power DT experiments. For example, the further development of scenarios with a high bootstrap current fraction mentioned above is of particular relevance for the efficiency (and low recirculating power) of a future power plant.

Mission 7: Milestones

Short term

- CTF feasibility study

Start of ITER operation

- Completion of DEMO conceptual design study
- Feasibility of DEMO maintenance procedures

End of ITER Phase I

- Completion of DEMO engineering design activity and supporting R&D: readiness for licensing and construction
- Assessment of engineering feasibility of Stellarator power plant after confirmation of potential

A significant part of the overall effort for such an integrated DEMO design is related to component design (blanket, heating and fuelling systems and fuel cycle, remote handling and maintenance / refurbishment systems, plant control etc.). In particular, improvements of the heating and current drive systems and efficient remote handling technology are important for the minimisation of maintenance times and optimisation of machine availability. Industry must play a major role in this work and this has been acknowledged already by the community[41], both for ensuring that feasibility according to industrial capabilities and standards is ensured, in particular in the area of nuclear

[40] except for the novel more compact accelerator structure for which first a cost-benefit and risk study should be undertaken

[41] "Industry should be involved in all phases of DEMO development, starting with the conceptual design, then in detailed design and construction, and finally in support for operation …. Industry should be charged with systems and plant engineering, component design and fabrication, remote handling procedures and equipment, balance of plant engineering and licensing support. Moving towards construction the responsibility of industry should increase and project management should increasingly include industrial experts with experience in plant construction management." [excerpt from SETP report (quoted after [1] p.9)]

technologies and for becoming acquainted with the system requirements of a fusion power plant. This would be a significant step, beyond industries' involvement in ITER construction, towards a full industrial leadership for the realisation of future commercial power plants.

Compared to ITER the DEMO design will have to achieve a new quality by incorporating plant efficiency and availability standards at close-to-commercial level for which the utilities' views should be sought. Reference points could be e.g. modern fission reactors. This is a substantial challenge since a fusion power plant will be of higher complexity, involving a much larger number of components. In due time conceptual power plant studies should carry forward the characteristics of the DEMO design and benchmark fusion against other reactor systems.

Over the coming years an important task for optimizing the Programme's approach towards DEMO is to identify the key characteristics for a conceptual DEMO design. This must be done in a balance between requirements derived from power plant studies and possible technical options within a reasonable extrapolation of the present state-of-the-art in physics, technology and materials R&D, involving the competence of the future commercial partners in the construction and use of fusion power plants. **The Panel recommends that a small DEMO group should be established soon for steering and assisting progress towards the definition of a conceptual design of DEMO, and for providing feedback to the programme for the optimisation of the ITER and accompanying R&D towards the needs of DEMO. This group should also develop and manage the interaction with industry (for defining the requirements of builders) and utilities (for defining the specifications of the customers and operators of future fusion power plants) for the definition of DEMO.** This DEMO group should gradually evolve to a full design team in about a decade from now.

B. An integrated vision on R&D needs

Assessing the programme, the Panel became highly impressed by the quality of the research executed, the progress already achieved and the vision of managers for an integrated research programme aiming at the ultimate target of useful fusion power. However, the Panel equally realized that there are still very substantial and challenging research and development efforts ahead and that a well-coordinated programme of this scope and duration must also include extensive measures for risk reduction and innovative research which in the end may turn out to provide essential benefits. The major projects of the programme, related to ITER and DEMO must be managed in a clear project driven way but in parallel concept improvements and efforts for enhancing the fundamental understanding in plasma physics, materials research and other areas of relevance need to be undertaken. In other words: targeted core projects, pursued in close interaction with industry and, where advantageous, in

Major ITER current research needs

- Burning plasma physics, scenarios, size scaling, modelling
- Control strategies, control of MHD instabilities (ELMs, disruptions, etc.)
- Tritium technologies, tritium retention and removal, dust characterisation
- Selection of plasma facing components
- Heating and current drive technologies
- Diagnostic systems
- Fuelling and vacuum technology
- Remote handling
- Support to licensing

international collaboration, must be embedded in a fusion R&D programme with tight connections to the academic environment and incorporating strong efforts on training and education. The Associations, in particular also the smaller ones are essential for the European-wide links to universities.

1. *Period until start of ITER operation*

Within this perspective, based on the identified research needs and missions, the Panel sees the key strategic relevance for fusion R&D for the period until start of ITER operation on:

A. **Supporting ITER construction and preparation of operation.**
   For further progress towards DEMO ITER will be the crucial fusion device. The immediate and highest priority must therefore be to make ITER a success. Besides the internationally agreed commitment to the construction of the device, further R&D efforts are needed, and the necessary resources must be secured, for the support of systems and components development and, beyond construction, for the preparation and support of ITER experimentation. Europe has attracted the ITER site to Cadarache (France) in international competition and has committed itself to the largest share (~45%) in ITER construction in order to acquire comprehensive know-how in all areas essential for the construction of a burning fusion device. With this outstanding investment Europe must have a particular interest in getting an optimum return from ITER.

   During the coming decade R&D tasks for ITER comprise***:***

   Accomplishing outstanding technology R&D issues and exploiting recent R&D progress for the design and construction of ITER systems and components: Some R&D in support to specific industrial construction tasks allocated to Europe is required. While the major core components of the device are now being procured, items such as the plasma facing components, the heating systems, diagnostics etc. which will be constructed in a couple of years, require in part finalisation of development and/or testing and validation with corresponding mock-ups and facilities. Furthermore, progress which has been achieved since finalising the design or which will result during the coming years should be incorporated. The former is expected to be taken into account by the current ITER design review; the latter should be demonstrated and validated in order to ensure that up-to-date requirements and design choices are implemented in the construction wherever still possible. This pertains e.g. to the choice of material for the initial ITER first wall or the tritium breeding test blanket module.

   Resolving ITER physics issues which might limit the performance, constrain the accessible parameter space and/or impact on the operational reliability: Several issues such as high intermittent heat loads to the divertor, fast particle effects in the core of the plasma or MHD modes must be controlled in order to avoid negative impacts on the desired operational regimes for ITER or limitations of the maximum performance. Measures which uplift or mitigate these constraints or reduce the operational risk should be investigated. Examples cover a wide range including the control and mitigation of disruptions, ELMs or Neoclassical Tearing Modes, operation at low torque of the plasma, aspects like the flux consumption during start up of the plasma, the ability to launch a discharge with metallic walls, improvement of the radiative power fraction in the divertor or measures for dust and tritium retention control.

Preparing a rapid start-up of ITER, targeting promising operational regimes: ITER foresees a programme of 20 years for exploitation, initially starting with operation in hydrogen followed by deuterium and, after a shut down, presently foreseen in the fifth year for a change from the initial carbon wall to a metallic wall, proceeding to deuterium-tritium operation. The detailed planning for ITER commissioning (for the different phases) and exploitation foresees a period of nearly 300 days for this refurbishment including restart, for ECRH commissioning an integral time of 350 days is expected and for other measures such as making ELM control techniques functioning experimentally etc. further significant operational time is allocated. In order to accelerate the start-up and exploitation of ITER substantial experimental, theoretical and modelling efforts would be needed and validation of scenarios close to the ITER conditions, including the first wall, plays the most important role[42]. Given the significance of reducing time and cost for ITER tasks and in particular shut downs, all reasonable efforts should be made to minimize modifications of ITER and to accelerate exploitation by preparatory work.

Strengthening diagnostic and modelling capabilities and fostering developments for improved solutions in specific areas of fusion physics and technology: A significant effort is required for the development for ITER diagnostics which need to comply with much more rigid requirements and conditions than the ones on present day devices. Also diagnostics must be considerably improved to qualify for ITER. Modelling, already today being of high importance will progressively obtain a key role for all developments in fusion research such as burning plasma physics, components design and materials development.

Of particular near-term relevance are R&D aspects contained in missions 1 (simulation of fast ion effects), mission 2 (operational limits, plasma control, ITER qualified diagnostics), mission 3 (mitigation of erosion and T retention), and mission 5 (burning plasma physics predictive capability).

B. Preparing the DEMO design, simultaneously carrying out long lead R&D. During the same period, in parallel to, and notwithstanding the importance of, ITER support, important long-lead physics and technology issues for DEMO and the reactor must be vigorously pursued.

Strengthening a coherent materials research programme for DEMO and future fusion plants and establishing experimental means for validation: Indispensable for the ultimate success of fusion R&D is to ensure the availability of suitable materials for the hostile environment in the core of a fusion power plant[43]. The development, qualification and validation of candidate structural and functional

[42] During the coming decade JET will be internationally the only satellite device operating at high performance and with comprehensive equipment. JT-60U will be closed down soon for allowing construction of JT-60SA which, with a foreseen start of operation in 2016, will likely only achieve relevant ITER results in parallel, not prior, to ITER exploitation.

[43] Neutron damage is strongly dependent on the fluence and the energy of the neutrons which impinge on the material. In solids, there are two key effects; first the neutrons can displace atoms from their positions in the lattice inducing changes to the mechanical and electrical properties of the material. This mechanism increase with the neutron fluence and is measured in displacements per atom (dpa) while the resulting effects (creep, hardening, ...) may saturate under certain conditions. As a second effect the neutron can induce nuclear transmutations. The main result of interest here is the liberation of alpha particles which then become neutralized. These helium atoms diffuse in the lattice and aggregate on grain boundaries. Typically, under fusion conditions about 10 appm He/dpa are expected for steels and under substantial irradiation the material exhibits “helium swelling”. Testing materials under a fusion relevant neutron spectrum is therefore essential for their qualification.

materials is on the critical path if a fast track approach to a fusion power plant shall be pursued and it is of utmost urgency to establish the necessary means[44]. A major R&D target is therefore the accomplishment and validation of the design for the International Fusion Materials Irradiation Facility in a timely fashion. Several components must be tested and beyond the Broader Approach some additional R&D be carried out. For materials research the modelling on high performance computers is of increasing importance: ab-initio materials models should become available which should progressively assist the definition and selection of candidate materials such as for radiation-resistant low activation steels.
Linked to this point is the important ITER test blanket development programme. Mission 6 is most relevant for the mentioned aspects followed by mission 3.

Advancing concept improvements for the Tokamak and pursuing the Stellarator line for optimizing the path towards DEMO and the commercial fusion power plant: The main Tokamak line towards the fusion reactor is being developed along plasma configurations of similar shape and dimensionless parameters in different size. Beyond this design line a wider space should be explored for the choice and detailed optimization of the DEMO configuration and the fusion power plant. The Stellarator with its intrinsic advantages for steady-state operation should be further developed. In addition Spherical Tokamaks, Stellarators and Reversed Field Pinches are capable of exploring parameter regimes of interest, but inaccessible, to the standard Tokamak.

Establishing soon a group for steering the DEMO R&D programme, preparing the definition of a conceptual DEMO design and optimizing R&D on, and in parallel to, ITER: Conceiving a burning fusion device requires finding numerous compromises between conflicting design optimisation aspects. Preparations for a conceptual design of DEMO, in conjunction with fusion power plant studies and feed-back with utilities and construction companies, should enable to identify areas where present technology is insufficient. Solutions must be developed which meet requirements within reasonable extrapolation of proven physics and technology, and dedicated R&D must be stimulated in areas where these do not yet exist. Modelling capabilities must be strongly enhanced in order to develop a comprehensive predictive capability for the DEMO design activities. The group should develop and manage the interaction with industry and utilities for taking into account their capabilities and requirements, respectively, and contribute to guiding the R&D programme to be carried out on ITER, and other fusion and technology facilities towards optimum DEMO relevance. This group should gradually evolve towards the future DEMO design team.

2. *Period of ITER operation*

Preparing for DEMO construction

During the first decade of ITER exploitation it will be decided whether DEMO can be constructed. ITER results and accompanying R&D for DEMO will form the basis of adapting conceptual DEMO design elements towards an engineering design.

As mentioned before, being the main investor, Europe should draw commensurate benefit from ITER results. This is by no means automatically ensured. Only with a

[44] The neutron fluence on ITER is expected to be up to ~3 dpa (displacements per atom) over the lifetime which is far inferior to DEMO (up to ~70 dpa/fpy) or a future fusion power plant (up to 150 dpa/fpy). ITER can therefore be built with austenitic steels (316L) and no blanket is foreseen while blanket modules will be tested. For these, as for DEMO structural materials the present candidate structural material is EUROFER operating between 300° and 550° C.

strong fusion R&D programme which is accompanying ITER will Europe be able to impact on ITER for pursuing a programme optimally oriented towards DEMO relevance, to absorb the results generated on ITER and to transform them into progress towards the development of commercial fusion power. Adequate funds and an adequate facility basis of international relevance are needed to achieve these objectives.

Industrial participation, and beyond this period, leadership, is becoming mandatory in all R&D areas, in particular where systems integration and components development is involved. **Mechanisms must be set up to consolidate and further develop industrial interest building upon the large ITER contracts**. Targeted, well-contained and time-limited projects for DEMO prototyping should evolve; where possible, competition must be encouraged. **The fusion laboratories working on components technology should be strongly motivated to develop partnership with relevant industries**: the successful integration of industrial and public R&D players will be crucial for the development of DEMO and the further development of fusion energy.

Topically, for preparing DEMO construction emphasis must be on:

Achieving the goals of ITER in DEMO relevant conditions with emphasis on steady-state aspects: The most prominent ITER targets are to achieve burning plasmas with an energy amplification Q>10 (and Q>5 for extended pulses). In order to demonstrate the feasibility of progressing towards DEMO, these performance results must be achieved under conditions which are as much as possible relevant for DEMO, i.e. which can be extrapolated under the requirements imposed by the DEMO design, such as a reactor-relevant wall (albeit operated at lower temperature), suitability for efficient steady-state thermal power output and robustness against instabilities.

Developing a blanket and auxiliary systems compliant with DEMO conditions: DEMO will be the first device with a tritium breeding blanket which shall close the tritium cycle. Among the possible options water, helium, metal cooled and dual-cooled concepts have been investigated for further development. Tritium self-sufficiency of fusion requires a breeding ratio sufficiently above unity. Furthermore, since the efficiency of a power plant increases with the operating temperature, the blanket should be operated well above the temperatures foreseen for ITER. Limitations are linked mainly to the coolant and to materials compatible with the coolant. Materials development and validation is playing an essential role. The ITER Test Blanket Module Programme and work on IFMIF are of high relevance for this area.

Besides the blanket other components and auxiliary systems need to be further developed for being compliant with the more challenging conditions in DEMO. Substantial work is required for the divertor, the first wall as well as for diagnostics, control systems and actuators. Among them in particular the heating and current drive systems need to be improved. Lifetime, safety, reliability, operability and maintainability are becoming major issues for all systems and components in the nuclear environment of a high performance steady-state operating DEMO.

Optimising and validating suitable materials and components for DEMO: Testing micro samples in IFMIF will provide essential information on the suitability of the materials chosen[45] and its results are needed for the qualification and licensing of

[45] IFMIF is expected to provide testing volumes of ~0.5 litres with 20-55 dpa/year, ~6 litres with 1-20 dpa/year and >8 litres with <1 dpa/year. Test materials will be inserted as micro samples.

all DEMO components exposed to significant irradiation such as first wall tiles, blanket or structural elements. It should be assessed whether a volumetric neutron source as a “Components Test Facility” (CTF) with the proper fusion neutron spectrum would be useful for risk reduction regarding the qualification and validation of nuclear components and could impact in time for the DEMO construction decision. If this would be confirmed, a proof of feasibility for a CTF, e.g. along the Spherical Tokamak or the mainline Tokamak concept, would become an urgent task.

Assessing concept improvements for the Tokamak and the potential of the Stellarator for optimizing the path towards commercial fusion power: The progress and success of ITER depends on the quality of preparation of operational scenarios, novel features, control mechanisms etc. on smaller devices and their validation in satellite class devices before application to ITER. The foreseen research programme of 20 years on ITER must be guided towards providing the optimum information for the further progress towards DEMO and a fusion power plant. This requires an accompanying programme which is focused on DEMO and the ultimate goal while strongly liaising with, and impacting on, ITER R&D. ITER is designed to allow upgrades to its capability throughout its lifetime, but long lead-times for the ITER systems (including licensing) require early conceptual definitions of upgrade paths. Main aspects are the heating and current drive systems, diagnostics, the control capability and the development of the plasma facing components towards the requirements of a reactor. Research must focus on the advanced Tokamak scenarios with their potential for high bootstrap and non-inductive current drive.

Another important area is to confirm the potential of the Stellarator with its intrinsic capability for steady-state plasma operation for a reactor. Given the importance of continuous operation of a fusion power plant, it is important that the Stellarator option with its intrinsic steady-state capability is advanced in parallel to the Tokamak.

Developing a “numerical burning plasma device” for the detailed prediction of fusion performance and assistance in the definition and design of DEMO: In a decade from now the numerical and modelling capabilities should have been sufficiently progressed so that a comprehensive description of a Tokamak (and later a Stellarator) plasma discharge can be derived which must be validated against burning plasmas results of ITER. This should greatly assist the experimental progress and help for the design of DEMO.

Establishing the DEMO engineering design: Construction of DEMO should start, under the fast track scenario, by the end of the considered period. Taking the ITER design experience as reference, the engineering design of DEMO needs to start at the beginning of this period i.e. around the time of the expected ITER commissioning. The DEMO design will require the development of mock-ups and prototypes for the essential components. This will be a substantial effort which should emanate from the R&D activities on components and systems described before. Contrary to ITER which is conceived for allowing upgrades and modifications throughout its life, DEMO should be an optimised point design with a reduced set of targeted diagnostics and optimized control systems for the few anticipated operational modes. Possibly the project could allow for one major refurbishment between two phases of operation, an initial one for confirming the extrapolations from ITER and accompanying R&D, the second one for demonstrating feasibility of electricity production with high efficiency and reliability for the following commercial development of fusion. Need for components and systems optimisation in DEMO at a later stage should be limited.

Mission 7 is key to this long-term priority; all other missions are linked to it, notably Mission 4 regarding the exploration and testing of stable high performance steady-state scenarios and the development of the Stellarator and with regard to missions 5 (and 6) modelling. The time horizon of all DEMO R&D lines is dictated by the desired start of DEMO construction which, in the fast track scenario should be within 20 years.

3. Pursuing innovation

While recommending for the core projects a clear top-down R&D planning[46], the Panel is aware that the overall research field is still too wide to rely exclusively on such an approach. As a complementary part of the programme, there must be an effort of sufficient weight to explore more tentative perspectives for innovations and improvements in areas where R&D options are still too open for exclusively pursuing target-oriented top-down approaches. Unifying the knowledge from the experimental and theoretical work on different confinement schemes (in particular the Tokamak and the Stellarator, but also others such as the Spherical Tokamak and the RFP) will be an interesting area for the exploration of possible innovative concept improvements. The Panel emphasizes that scientific curiosity and interest for seizing novel scientific and technological opportunities must remain important motivations for R&D. Also links must be established and maintained with adjacent scientific and technological fields for incorporating essential knowledge[47] generated in these research domains.

The dedicated ITER and DEMO oriented work should therefore be complemented by innovative R&D aimed at opening new options and improving present modelling by i*mproving the theoretical understanding based on first principles, developing novel solutions in physics and technology and extending fusion physics and technology knowledge and the relevant data bases.*

Some areas where progress should be especially rewarding are:

- Fundamental theory of high temperature plasmas
- Study and modelling of toroidal magnetic confinement fusion concepts and their comparison and keeping in touch with other concepts
- Physics of low temperature plasmas, high power plasma wall interactions, erosion, deposition, retention of hydrogen in materials
- Diagnostic methods for high temperature plasmas in a nuclear environment
- Physics and technology of negative ion beams
- Physics and technology of high power electromagnetic radiation generation and interaction with plasmas and wall structures
- General theoretical and experimental materials research
- Innovative superconductor physics and technology[48]

[46] It should be stressed that, in a top-down R&D planning for the core projects, the problem definition, the rationale of the research plan and the specific research steps must be established jointly by EFDA and the individual involved Associated Laboratories.

[47] A recent example is the development of solid state Hall sensors which earlier appeared to have too many disadvantages but where superior performance under fusion conditions is now expected and implementation is foreseen for ITER magnetic field measurements.

[48] To take superconductors as an example, the present low-temperature superconductors are difficult and costly in production and require helium temperatures of 4-5°K which makes operation expensive due to the high cooling requirements. Recent developments show promises that novel high temperature superconductors may become available with high critical currents and magnetic field capabilities as required for fusion applications. This research is still in the fundamental phase and in the largest part beyond the scope of the fusion programme. However, it is important that the breadth of the programme ensures that information and technology flow be ensured such as to permit application of these promising technologies at the earliest useful moment.

- Nuclear reactor technology

These activities should be closely linked to the mission oriented R&D tasks under priorities 1-3. Furthermore, they should support missions 1 to 6 and it may turn out that progress in these underlying activities could, in the end, be of particular relevance for mission 7.

4. A concise development plan

Analyzing future R&D in terms of aggregated missions must include an assessment of whether there are major issues which could entail problems or even develop into show stoppers. The time horizons when solutions are needed must be identified in a consistent overall R&D plan. The Panel received a gap analysis provided by EFDA which it has analyzed. In order to visualize the resulting scope of the R&D Vision described in the previous paragraphs, Fig. 1 shows a concise strategic planning schedule[49] for the major topical issues which must be addressed and solved.

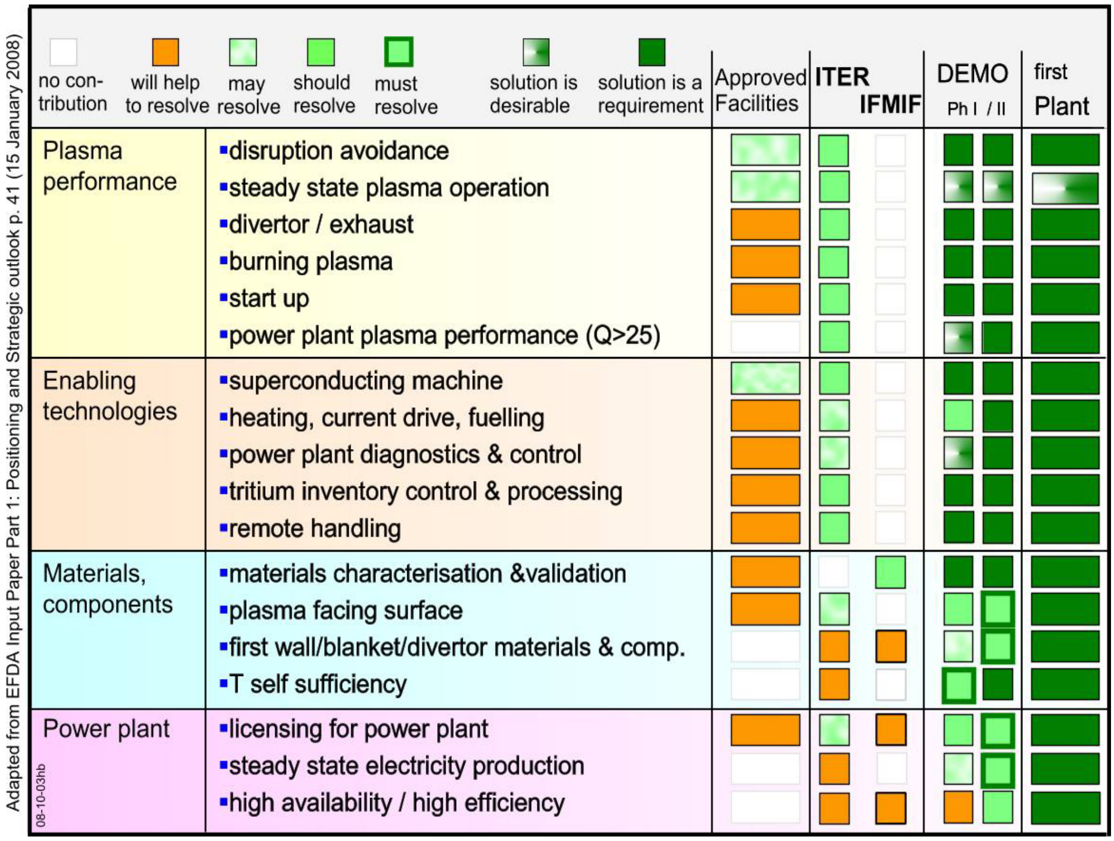

Fig. 2: Major R&D issues and expected / required evolution. Targeting the expected / required solutions is displayed in a time frame exemplified by the present devices ("Approved facilities"), "ITER and IFMIF", "DEMO", with two assumed phases for a blanket selection and R&D programme (Phase I) and a high performance demonstration programme (Phase II), and finally a "first plant" i.e. the start of the commercialisation phase of fusion power where all R&D items must have found satisfactory answers not excluding, of course, further optimisation. (Adapted from EFDA, Input Paper Part I, p. 41)

[49] This graph is following the classification of EFDA, however with a simplification of presentation and adjustments resulting from the analysis undertaken.

III. Facilities required to fulfil the R&D needs

A. *Introduction*

In the previous sections the main R&D needs and the seven missions to be accomplished for the development of fusion energy were analyzed. Which facilities are then needed to support the envisioned R&D? Before addressing the question of a facilities development plan, the three classes of facilities which are needed for fusion development will be separately considered:

- *Fusion devices* for the investigation of plasmas of thermonuclear interest,
- *Technology facilities*: test facilities for components, systems or processes of a future fusion device such as ITER or DEMO and other technology facilities which serve for the testing of materials under the irradiation or particle and heat fluxes expected in future fusion devices or the qualification before and after such exposure,
- *Computing facilities* for the modelling of fusion plasmas, scenarios and materials calculations.

Most of the installations under the first two bullets are specific to fusion R&D albeit not dedicated entirely only to one of the integral R&D priorities identified in the preceding chapter. Nevertheless the following analysis will be structured according to ITER and DEMO relevance. As mentioned above, Innovative and Accompanying R&D is in practice strongly interlinked with ITER and DEMO work and is pursued on the same facilities. In the following, therefore, facilities for this priority are not separately addressed.

B. *Fusion devices*

The essential fusion facility for the coming decades will be ITER, then followed by DEMO. These devices will confirm, or not, the promise of fusion[50].

ITER needs to be supported by large "satellite" devices like JET now and JT-60SA in the future, and also these cannot be efficiently operated without the continuing assistance of smaller devices. Any change of hardware which is desired for optimizing their performance requires very substantial financial and time resources on ITER and large satellite devices. Their nuclear quality (even when only operated for some time with pure deuterium) requires that all in-vessel maintenance and work is done fully by remote handling. Any malfunction or deviation from established stable operational regimes may result in risks of damage for which repair would be costly and time consuming. Optimisation and performance enhancement on ITER and the satellite devices must therefore be restricted, wherever possible, to regimes which have been well explored, prepared and tested on smaller devices.

Also within explored regimes the sheer difference in cost (and time) between a shot on ITER, on JET or a smaller device strongly recommends to undertake, where feasible,

[50] Both ITER and DEMO are not subject to this assessment: ITER since it is a well-defined international device for which design and research missions are established (i.e. ITER is considered to be a fixed item to this review), DEMO because it is currently more a programmatic step and not yet defined as a proposal for a specific device. Also the Japanese device, JT-60SA, planned to start operation in 2016, which, as an ITER satellite, links to the scope of a European facilities' assessment. This is due to the fact that under the Broader Approach Agreement European partners are contributing funds to its construction and (for a limited time) exploitation. In return, a sharing of the use has been agreed for five years of high performance operation. The European use and the definition of the device's layout, being essentially agreed, will therefore also be considered to be boundary conditions to, and not subject of, this review.

R&D tasks first on small devices and then to validate them on satellites and finally on ITER. For the time- and cost efficient exploitation of ITER and satellites the smaller devices are therefore essential.

It is not yet clear whether DEMO should be a direct further extrapolation from ITER in extension of the stepladder (COMPASS) - ASDEX-UG – JET – ITER or whether results from ITER or accompanying R&D will recommend modifications to the plasma shape, to other major parameters, to complementing features or even a transition to the Stellarator. The physics and technology of Tokamaks and Stellarators share many, if not most, features but the Stellarator, when optimized to eliminate a toroidal current, has, as pointed out before, no current-driven instabilities and an intrinsic steady-state capability. For preparing the DEMO step it is therefore important to conduct research also with devices which extend the exploration of the multidimensional parameter space beyond the one of ITER-like devices.

In the following sections information on the existing facilities, JT-60SA and the proposal for a European satellite are provided, followed by a study of their capabilities, remaining apparent gaps and costs with regard to the seven R&D missions, first in an ITER oriented perspective and then on a DEMO oriented time scale. Subsequently the individual fusion devices are categorized with regard to their relevance for ITER, for DEMO.

1. *ITER satellite class*

ITER “satellites” are fusion devices of sufficient performance to provide directly relevant information for ITER and beyond for DEMO. Key requirements are that dimensionless parameters are close to ITER / DEMO[51], the pulse length should be substantially longer than the current redistribution time and, for equilibration with the wall, several minutes or even hours. Typically these devices need to be in the 5 MA regime (“JET class”) and can be significantly smaller than ITER, more versatile and time and cost efficient in investment, operation and maintenance. Their missions are to test and validate, again in conjunction with smaller devices concepts and scenarios for ITER, to expand the information obtained on ITER into adjacent regimes (closer, e.g. to suspected operational boundaries or to DEMO conditions), to complement ITER in the testing of innovative technology for DEMO which is not foreseen to be implemented in ITER and, in general, to support the scaling, by modelling, towards DEMO conditions. In assessing the R&D required to implement the fast track approach to DEMO, it is important to understand that no single machine, including ITER, can address all R&D missions simultaneously; the first fully integrated experience will be in DEMO itself. By the same token no single satellite (or smaller) device can address simultaneously all issues in the full set of ITER physics parameters.

A “mainstream” satellite (JET in parallel to ITER construction, JT-60SA during ITER operation, both with a large volume and moderate magnetic field) will be useful for aspects of plasma control over relevant physics time scales and for the extension of results towards reactor relevant stationary operating regimes but would

[51] The plasma size must be sufficient for slowing down of fast particles mainly by electron collisions and to meet relevant finite-orbit fast ion dynamics, the thermal plasma pressure should be high enough to see relevant fast ion and thermal plasma stability aspects, electron and ion temperatures should be sufficiently balanced, the collisionality should be adequate to reproduce relevant thermal plasma stability properties, the density should be close to the Greenwald density and the radiated power fraction should correspond to ITER / DEMO values. For relevant steady-state scenarios the normalised pressure must be close to the range $ß_N$~3.5 - 4.5.

not be able to operate at high power loads to the divertor without violating pressure limits. On the other hand, a compact, high field device would allow attaining the power load regime of ITER and DEMO and therefore probably display similar ELM behaviour within accessible plasma regimes. In such a device also some relevant aspects of fast particle physics could be studied. For this purpose, in the absence of substantial fusion alpha production, fast ions have to be generated by radio wave or neutral beam heating / current drive systems. However, the fast ion distribution produced by these means is anisotropic, such simulations are therefore only partially relevant to the case of a burning plasma where the fast ion (alpha) distribution is isotropic.

DT operation is a key asset for JET in preparation for ITER, enhancing its value greatly with respect to non-DT devices. However, once ITER is operating in DT and will attain fusion power significantly above the break-even level, the tritium capability of a satellite (which, due to its size always will be limited around the break-even level) will become of lesser interest and must be seen in a balance with the practical and economical advantages of a comparable machine working without tritium.

JET is the currently largest and most powerful magnetic fusion device and the only one which is capable of using DT fuel. The value of JET for validating ITER physics issues has been outstanding and will so remain for the foreseeable future. JET will be the first choice for validating ITER features and ITER system solutions until another device with more advanced capabilities will become available. The size of JET – in linear dimensions about half the one of ITER – and its heating systems already permit a realistic assessment of fast particle confinement and effects. Over the years a remarkably comprehensive set of diagnostic has been installed which greatly contributes to the relevance of this device. JET is unique for its burning plasma diagnostics. Tests of components such as the one which currently is imminent for the ITER-like ICRH antenna should be directly transferable to ITER. In order to test the wall conditions of the ITER DT phase JET will be equipped with beryllium walls in the main chamber and with tungsten in the divertor (the latter based on the experience gained on ASDEX-UG). JET will be able to make substantial contributions to Missions 1-5 and with its tritium technology also to Mission 6. ITER licensing preparations strongly rely on JET with its tritium facility. Compared to all other European devices JET is outstanding in operating cost, due both to its size and its nuclear capability.

In response to urgent ITER R&D needs, JET technical enhancements (EP2) are underway which will be completed in 2010. The programme which was designed when deciding these upgrades is of utmost importance for ITER and requires JET operation at least until 2014/15, including a DT phase (including operation in pure tritium for investigating retention in metallic walls). Beyond, there are compelling scientific arguments for exploiting JET for a few more years, the exact schedule depending on when JT-60SA will be able to operate at a performance of relevance to ITER. Specifically, JET could work on scenario development and means for facilitating access to the H-mode in ITER, continue work on ELM mitigation and disruption / runaway electron issues and further develop detritiation and dust removal techniques etc.

JET	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	4.5	3.5	2.98	1,25	~80	~40	5-10s (@ 4 MA)
Major upgrades underway (JET EP2 →2010). Upgrades proposed[52] (→2012-2014): ECRH for MHD control, central current drive and electron heating for steady-state operation. LHCD launcher for improved steady-state current drive. ITER like perturbation coils for ELM and resistive wall mode control. Substitution of tungsten-coating by solid tungsten-tiles. A second ICRH antenna. Change from co- to balanced neutral beam injection.							

- Benefits:
  - *ITER:* The value of JET for ITER has been and continues to be outstanding. JET will continue to be the only device which is capable of DT operation until the start of ITER DT phase (~2024), The device can operate with Be walls, has sufficiently large effective dimensions and high plasma current (low ρ*). Fully equipped with heating systems (except ECRH) and diagnostics (unique for studying burning plasma effects). Production of fast ions close to ITER in dimensionless parameters. ITER relevant ELM and NTM study and control. Presently enhanced for ITER relevance with regard to heating, unique for validating ITER scenarios with a relevant wall (W and Be).
  - *DEMO:* Contributions to the exploration and assessment of DEMO operational regimes in conjunction with the ITER oriented programme. On a time scale relevant for the DEMO engineering design JET is no longer expected to contribute.
- Drawbacks, Risks:
  - *ITER:* Without an ECRH system (should be implemented) limited ability for e.g. NTM control.
  - *DEMO:* Although the lifetime of the magnet, other key components and the neutron budget are used up only to order of ten percent, risk of age-induced technical failures may become high in a decade from now. JET has no superconducting coils, hence pulse lengths and access to steady state at high performance are limited. In the longer term, JET should therefore be substituted by JT-60SA and/or possibly by a new European satellite device.
- Cost:
  - Investment planned: 60.4 M€ (ongoing for EP2 enhancements). Cost of an ECRH system (<10M€, if a sufficient level of international participation is achieved). Cost for additional proposed upgrades for a programme beyond 2014/15 are estimated at a few tens of M€. Preparations would have to start soon and investments would be required before 2013 in order to implement upgrades before the planned tritium campaigns in 2014/5 which would be technically highly advisable.
  - Operation: ~80 M€/y, JET has a high cost fraction to EURATOM (~60-70M€) and absorbs about 70% of the EURATOM funds for

[52] Here and for the following facilities these upgrades are discussed based on the documentation provided by the facilities/institutes to the Panel through EFDA. The Panel notes that this does not imply an endorsement by EFDA or the Steering Committees of the Associations.

the operation and upgrade of the fusion devices in the European programme.

JT-60SA is a Japanese project for an ITER satellite to which European partners contribute in the context of the Broader Approach Agreement. Europe will provide the toroidal field magnet and the cryogenic plant and participate in the procurement of the cryostat, the ECRH system and the power supplies. For the first five years of high performance operation joint exploitation is agreed between Japan and Europe. The objectives of JT-60SA are to optimize ITER relevant plasma scenarios and test new operating scenarios, to test and optimize auxiliary systems, advance the understanding of ITER relevant physics, test improvements and modifications of components and systems before their implementation on ITER and to train, in an international environment, professionals and technicians. Furthermore it should complement ITER results in preparation of DEMO by exploring, through a broad flexibility and low aspect ratio, the role of shaping and active stabilization, explore steady-state operation at high normalised plasma pressure ($ß_N$>3.5) for 100s and more, optimize non-inductive current drive (aiming at up to 70% bootstrap current) and control power fluxes to the walls in steady-state operation regimes. JT-60SA is not capable of operating with DT fuel and will have carbon walls (possibility of a change to W is under consideration). According to the revised baseline schedule the first plasma should be achieved in 2016. High performance operation should be expected to start a few years later. 24 MW perpendicular and 10 MW tangential neutral beam injection and 7 MW ECRH are ultimately foreseen for the additional heating. A stabilizing shell and active sector-coils will be applied for RWM feedback control. Remote maintenance of in vessel components is mandatory after high-power long-pulse experiments in deuterium. Overall, the device is designed to cover a parameter range close to the one of JET with a somewhat lower magnetic field capability but a higher current than the one in JET routine operation.

JT60 SA (Baseline) (Draft 2008)	I_p (MA)	B^t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	{5.5} [5.5]	{2.68} [2.3]	{~3.1} [~2.95]	{~1.15} [~1.1]	{~130} [~140]	{41} [41]	≤100 s @ ≤5.5 MA
Construction expected to be accomplished by 2016**.**							

- <u>Benefits:</u>
    - *ITER:* Expected to become, after phasing out of JET, internationally the major device for assessing and validating ITER operational regimes at parameters closest to ITER compared to all presently existing devices; subject to final design decisions.
    - *DEMO:* The capabilities of the device including its capability to explore Tokamak steady-state operation should allow to make major contributions to the definition of DEMO. Subject to final design and upgrade decisions.
- <u>Drawbacks, Risks</u>:
    - *ITER:* Absence of capability for DT operation will be a drawback for the period until ITER starts operation in DT (~2025). Comparatively low toroidal field (2.68 T, due to using the non-DEMO relevant NbTi magnet technology). Limited agreed period for European participation in the exploitation (five years at full

performance) which may arrive late with regard to preparing for ITER high performance operation. Only neutral beam heating and ECRH foreseen, not ICRH nor LHCD. Only the divertor is water cooled: no wall equilibrium possible. No tungsten wall is presently foreseen, therefore difficult to address issues of high performance steady state operation. Under these conditions the longer-term interest for the European programme would be limited.

    - *DEMO:* European participation in the exploitation is only secured for the first five years. This, in addition to the layout of the device will reduce the usefulness for the European DEMO preparation.

- Cost for Europe:
    - Investment planned: European contribution to construction: 180 M€ (borne by voluntary Member States' contributions).
    - Operation: 7.5 M€ during construction, ~20 M€/y during operation for the agreed partial use during five years exploitation.

Potential for a European satellite class device: The scope for support to ITER and preparation of DEMO is challenging and it is essential to accompany ITER with a strong satellite programme. Taking note of the FAST proposal, Europe should develop a design for a device complementary to JT-60SA which would access the relevant parameter space and which would contribute to drawing maximum benefit from ITER.

FAST is a first proposal, based on a conceptual study for an ITER satellite, that would prepare advanced Tokamak scenarios for ITER and be capable of covering a wider (dimensionless) parameter range than JT-60SA with values closer to ITER. The plasma shape is similar to ITER and transport physics should be relevant since the ratio between energy confinement time and electron-ion equipartition time is comparable to that of ITER and heating will be mainly on electrons (as expected for the alpha particle heating in ITER). Liquid nitrogen pre-cooled magnets shall reach >8T field thereby allowing to have a compact device with linear dimensions between ASDEX-UG and TORE SUPRA. At high field and current (7.5T/6.5 MA) the flat-top time is estimated <20 s while with reduced values of 3.5T and 3 MA a pulse length of 170 s should be attainable at high normalized plasma pressure (β_N). Altogether 40 – 50 MW of heating power would be required for which ICRH, LH, ECRH and (in a second stage) negative NBI are proposed. The design aims at addressing major ITER and DEMO issues in an integral fashion. The power load figure of merit P/R would be higher than for JT-60SA and similar to that of ITER (P/R~22). An ITER/DEMO relevant metallic wall is proposed. If these projections will be confirmed, FAST could be expected to substantially complement JT-60SA and provide a highly relevant added value both for ITER and for DEMO oriented R&D.

The proposal meets compromises in order to benefit from available infrastructure and be comparatively low in cost. Although the size of the plasma is small this is compensated by the high magnetic field (~16 T on the coils) which, however, prevents the use of established superconducting technology and imposes copper coils with resulting limitations in discharge duration.

FAST (proposal)	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	<8	<8.5	1.82	0.64	~20	40 (50)	~20s (@ 6.5 MA, <170s (@ 3MA)
Design allows the later installation of 10 MW neg. NBI (not foreseen for first stage) which would provide for added flexibility (widening) in the radial fast particle density profile.							

- Benefits: (based on the FAST proposal, to be further assessed)
  - *ITER:* The device could be expected to substantially complement the accessible operational regime of JT-60SA approaching closer the values of ITER in several parameters. Actively cooled first ITER relevant metallic wall and divertor components are foreseen.
  - *DEMO:* In addition to the benefits regarding ITER a high first wall load (above ITER values, i.e. closer to DEMO) could be studied in the FAST long-pulse operation mode.
- Drawbacks, Risks: (based on the FAST proposal, to be further assessed)
  - *ITER:* No capability of DT operation would be a drawback for the years until ITER starts operation in DT (~2025). Pulse length at high performance may be marginal for assessing steady-state issues. Compact size may impact on RF heating capability and diagnostic access.
  - *DEMO:* Limited steady-state capability recommends to further improve the proposal.
- Cost: (based on the FAST proposal, to be further assessed)
  - Investment: preliminarily estimated at ~300 M€ (these costs are relatively low because copper coils would be used and existing infrastructure could be partially exploited. A superconducting device would be more expensive). DT capability would add severely to the cost.
  - Operation: a preliminary estimate is 13.5 M€/y plus manpower cost.

2. Medium sized and smaller devices

a) Tokamaks

TORE SUPRA is the largest Tokamak in this class after JET, and the only superconducting one, in Europe. The plasma shape is circular and a pumped limiter is used as exhaust system, features which distinguish this device (as well as the medium size devices FTU and TEXTOR) from the ITER Tokamak line. These circular devices usually cannot access the H-mode which is the standard operational mode in divertor Tokamaks. The edge electron temperature is high and does not easily allow the use of refractory metallic plasma facing elements under high heat load. Due to its superconducting toroidal field magnet (the poloidal magnet is normal conducting) and, unique in the world, the active (water) cooling of the entire inner wall (carbon / CFC) TORE SUPRA can, with non-inductive current drive, operate in very long pulses and attain steady-state with respect to all plasma and plasma-wall time scales at power exhaust levels up to 25 MW. The maximum discharge duration so far was 6 minutes at 3 MW injected power and a non-inductive plasma current of 0.5 MA (i.e. with zero loop voltage). The project is planning to further build up the long-pulse heating and current drive capability and enhance the means for studying core plasma fast particle effects in order to strengthen the potential for a programme, aimed at extending beyond the next 5 years, centred in particular on long-pulse LH and ICRH antenna development, fast particle physics, plasma-wall interactions (perhaps involving high Z and mixed materials) and fundamental turbulence studies[53].

TORE SUPRA	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Circular Tokamak	2	4.2	2.4	0.75	25	20 (24)	360s (@ 0.5 MA)
Major upgrades underway (→2010): Enhancement of LHCD power for steady-state operation; in-situ inspection and remote handling under vacuum by articulated beam. Upgrades proposed (→2012-2014): enhancement of ICRH and ECRH for steady-state operation							

- Benefits:
    - *ITER:* Will provide information on steady-state high heat loads for C, CFC walls which would be relevant for the first phase of ITER operation. Largest medium-size device (25 m^3 plasma volume) with a pulse length capability exceeding all relevant time scales and corresponding advanced real-time control methods. Fully actively cooled first wall allowing extended operation at high power loads and remote handling equipment.
    - DEMO: Technologies for steady-state heating and current drive. In case that carbon based materials would be partly reconsidered for some first wall applications TORE SUPRA would be able to undertake high heat flux tests in a realistic Tokamak environment according to present perspectives.
- Drawbacks, Risks:
    - *ITER:* Relevance and applicability of results to ITER in part mitigated since circular plasma shape, a limiter exhaust instead of a divertor. No NBI heating. The new superconducting international Tokamaks (EAST, KSTAR, T-15) can be expected, over time, to be developed towards similar capability as TORE SUPRA (albeit being smaller) with the advantage of ITER relevant shape, plasma edge and wall materials.
    - *DEMO:* No access to DEMO relevant advanced operational regimes. Non-reactor relevant wall material according to present perspectives.
- Cost:
    - Investment planned: 15 M€ + 10M€ (2013/14, for extensions required for core heating, fast particle physics and technology of steady-state/long pulse at 10 MW/m^2)
    - Operation: 19M€/y

[53] The Panel considers this to be a highly useful programme in case its objectives could not be taken up adequately on other facilities in the foreseeable future.

ASDEX-UG is a medium-size divertor Tokamak with a plasma shape very close to JET and ITER and a scientific programme that investigates a wide range of ITER relevant physics. Since several years a programme has been undertaken to study the feasibility of tungsten as a wall material and the first wall graphite and CFC tiles are now fully coated with tungsten. Coverage with tungsten will be used in ITER for the divertor plates and is currently the preferred choice for the DEMO first wall. The device has NBI, ECRH and ICRH heating systems, delivering together some 30 MW. Extensions of the ECRH heating system to 4 MW / 10s is underway; about half of the foreseen investment of 12.7 M€ has already been spent. Following the successful demonstration of enhancing plasma control by saddle coils in COMPASS and DIII-D (as well as RFX), ASDEX-UG will replace the existing internal stabilizing conductors by a stabilizing shell and new saddle coils. Installation of a lower hybrid current drive system is contemplated. If the vessel volume would be fully used the plasma volume could increase from 14 to 25 m^3 and allow attaining plasma currents in excess of 2 MA.

ASDEX-UG	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	1.6	3.1	1.65	0.4	14	30	10s (@ 1.2 MA)
Major Upgrades underway (2009/10) ECRH with fast steerable launchers for feedback mode control, (2010/12) Major Upgrade proposed (2010-2012) advanced feedback control by internal saddle coils and conducting shell, (2014) enhancement of LHCD for current profile tailoring and demonstration of ITER relevant PAM lower hybrid antenna, modification of the ICRH antenna for W-compatibility							

- <u>Benefits</u>:
  - *ITER:* After JET closest in physics aspects to ITER and therefore most useful for exploring ITER plasma physics at lower but still relevant scale and for stepladder scaling.
  - *DEMO:* Device which is among its class in Europe closest to DEMO, if the ITER line is extended. Full tungsten wall (as is presently anticipated for DEMO). Could be modified to larger volume / higher performance.
- <u>Drawbacks, Risks</u>:
  - *ITER:* Limited pulse duration. Extension required for closer access to ITER parameters.
  - *DEMO:* Limited steady-state capability since not superconducting and no Lower Hybrid Current Drive system. Plasma current limited (could be enhanced to ~2 MA if volume would be fully utilised).
- <u>Cost</u>:
  - Investment planned: 12.7 M€ (ongoing), 6.9M€ (2010/12) + 11M€ (2009-2014). Longer term: no investment costs available (tentative enlargement of volume).
  - Operation: 9.75 M€/y

FTU is a Tokamak of the circular limiter type like TORE SUPRA, however much more compact. It operates with pre-cooled copper coils and attains for limited pulse durations very high toroidal magnetic fields. Therefore, despite its much smaller dimensions, the plasma current and the confinement aspects can be similar to the ones in ASDEX-UG or TORE SUPRA. The device is an all-metal machine

(Molybdenum) investigating a liquid lithium limiter allowing to operate up to 10 MW/m2 heat load and to undertake corresponding plasma wall and boundary plasma studies. Additional heating is by EBW, ECRH and LH providing means for high density current drive but also for start-up and ramp-up studies and real-time control of a wide range of MHD phenomena including disruptions. The ECRH system has been equipped with real-time steerable launchers for mode control and checking an "ITER enabled" collective Thomson scattering system.

FTU	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Circular high field Tokamak	1.6	8	0.93	0.3	1.7	4.1	1.5s (@ 1.6 MA)
Major Upgrades underway (2009) 60° liquid limiter sector, ECRH with fast steerable launchers for mode control and ITER enabled CTS demonstration Major Upgrade proposed 2x 850 kW/1s additional gyrotrons							

- Benefits:
  - *ITER:* High-field compact device with ITER relevant densities. High electron heating.
  - *DEMO:* Explores the principle of liquid metal for contributing to the control of plasma-wall interaction.
- Drawbacks, Risks:
  - *ITER:* Relevance and applicability of results to ITER in part mitigated since circular plasma shape, a limiter exhaust instead of a divertor and a non-ITER wall material. No NBI and ICRH heating. Extension in heating needed for accessing optimum beta values.
  - *DEMO:* No access to DEMO relevant advanced operational regimes and non-reactor relevant wall material (Molybdenum).
- Cost:
  - Investment planned: 6 M€
  - Operation: 2.5 M€/y

TEXTOR is a circular Tokamak designed for the study of plasma wall interaction and good diagnostic access which has pioneered wall conditioning techniques. It is used as a facility for studying plasma-wall interactions and has a unique feature of two air-interlocks with gas feed, external heating and active cooling as well as comprehensive diagnostics for the study of large wall samples (15 cm diameter). Thereby samples can be exchanged without breaking the vacuum. Extreme power fluxes (<200MW/m^2) can be applied for studying e.g. material migration (in the context of fuel retention and removal) or melt layers on tungsten tiles. In addition to a toroidal pumped limiter the device can operate with a unique Dynamic Ergodic Divertor allowing to study transport and stability aspects of resonant magnetic perturbations near the wall and to establish a helical divertor configurations similar to the ones expected in Wendelstein 7-X, thereby also being (so far) a unique tool for code benchmarking. As a diagnostic facility a number of ITER relevant tasks (CXRS diagnostics, JET ITER like wall, dispersion interferometer, collective Thomson scattering, laser induced desorption spectroscopy for tritium retention and dust) and Wendelstein contracts (VUV spectrometer, Bragg X-ray spectrometer, diagnostic neutral beam) are being executed with time horizons between 2010 and 2014.

TEXTOR	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Circular Tokamak	0.8	3.0	1.75	0.47	7	9	12s (@ 0.8 MA)
Upgrades foreseen: enlargement of the air-interlocks for accommodating larger samples, laser systems for in situ fuel desorption, material ablation and dust detection, system for in-situ W coating techniques							

- Benefits:
    - *ITER:* Similar in size and field to AUG (but half the current), particularly equipped for plasma wall interaction studies.
    - *DEMO:* Exploration of modifications to the scrape-off layer in view of understanding and improving the particle and power flows to the exhaust system.
- Drawbacks, Risks:
    - *ITER:* Relevance and applicability of results to ITER in part mitigated since circular plasma shape, a limiter exhaust instead of a divertor and a non-ITER wall material.
    - *DEMO:* No access to DEMO relevant advanced operational regimes and non-reactor relevant wall material. No LH heating.
- Cost:
    - Investment planned: 2 M€
    - Operation: 4.9 M€/y

TCV is a highly variable Tokamak built for realizing and investigating a wide range of different plasma shapes around the JET/ITER type plasma configuration and thereby is able to provide significant information regarding the dependence of confinement and other properties on configuration parameters (triangularity, elongation, double null plasmas). The device has a two-frequency (2nd and 3rd harmonic) ECRH system (higher-frequency for the third harmonic considered) capable of feedback controlled localised heating and current drive. Discharges with full non-inductive ECCD or with 100% bootstrap current have been obtained and the control of local shear allows sawtooth (de-)stabilisation, NTM control etc.

TCV	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	1.02	1.52	0.88	0.25 / 0.7	1.3-3	4.5	4s (@ 1 MA)
Major upgrades (2010-2011): active ergodisation coils, low power Alfvén wave antennas, enhancement of power handling capability of low field side tiles, (2011-13) neutral beam injector for torque control, X3 heating.							

- Benefits:
    - *ITER:* Highly versatile in covering a wide range of shapes including ITER shapes. Advanced ECRH heating system. The proposed active ergodisation coils and enhancements in injected power (in particular NBI for ion heating) and power handling capability will provide considerable added relevance to this device.
    - *DEMO:* Exploration of shape variations around the ITER-like shape and the flexibility in heating will provide essential contributions to a consistent data base for DEMO optimisation.

- Drawbacks, Risks:
    - *ITER:* Comparatively smaller device, currently not reaching optimum beta operation due to limited heating power.
    - *DEMO:* Results on non ITER-like shapes can be expected to require validation on a larger device capable of operating with that plasma shape before they can be applied to DEMO.
- Cost:
    - Investment planned: 10 M€
    - Operation: 6 M€/y

COMPASS is a compact Tokamak with an elongated JET/ITER-like shaped plasma under commissioning at IPP-CR (Prague) which previously was operated by UKAEA (Culham). The device has a special set of saddle coils. It belongs to the "stepladder" approach and has contributed extensively to the understanding of the impact of "error fields" on the plasma behaviour (in particular the plasma rotation). In its new location it is planned to be used initially for ELM mitigation studies and generally for edge plasma investigations but will be equipped with a neutral beam system allowing the investigation of higher performing plasmas and a LHCD system for fast electron physics. Among the Academia-based smaller devices it is the largest and most relevant one for contributing to the physics investigations of the ITER oriented larger Tokamaks.

COMPASS	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Divertor Tokamak	0.25	2.1	0.56	0.23	0.5	1	1s
Major upgrades: 2009: neutral beam system, 2010: LHCD system, both integral to the basic equipment of the device							

- Benefits:
    - *ITER:* Small Tokamak of ITER shape. Equipped with systems for active mode control. Useful for low-cost contributions to exploratory scaling studies, diagnostics development. Basic fusion physics and training device.
    - *DEMO:* Useful for low-cost scoping studies in DEMO relevant shaped plasmas (for the current ITER step ladder approach).
- Drawbacks, Risks:
    - *ITER:* Comparatively smaller device therefore limitations in disentangling core and boundary physics, Only NBI and ECRH heating foreseen.
    - *DEMO:* Limited DEMO relevance with regard to performance and disentangling core and boundary physics. Only NBI and LH systems. Graphite wall and divertor.
- Cost:
    - Investment planned: 3.9 M€ (ongoing)
    - Operation: 0.85 M€/y

ISTTOK is a small Tokamak (originally constructed by FOM, Rijnhuizen) located in IST (Lisbon). It is used as a training and development facility for real time plasma control and diagnostic work of IST staff which has become strongly involved with its acquired competence in these fields in many major laboratories. The device has recently been used as a test bed for the fundamental physics investigation of the behaviour of a limiter system using flowing liquid metal and the study of its impact on the plasma.

ISTTOK	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Circular Tokamak	0.08	0.6	0.46	0.085	0.065	Ohmic	.05s
No upgrades foreseen							

- Benefits:
  - Small Tokamak, useful for specific exploratory studies such as MHD and impurity effects of liquid limiter concepts and the development of control tools. University device for basic fusion physics and training.
- Drawbacks, Risks:
  - Circular limiter machine, due to size and construction strong limitations in plasma and confinement quality.
- Cost:
  - Investment planned: -
  - Operation: 0.175 M€/y

b) Spherical Tokamak

MAST is a Spherical Tokamak (ST), i.e. with a much lower aspect ratio and more compact shape than the main line Tokamaks. This makes the physics different in several aspects. While in a standard Tokamak the toroidal field is about ten times larger than the poloidal field, here it is of similar amplitude and MAST's plasmas can show a strong paramagnetic enhancement inside the plasma. A very high ratio of plasma pressure to magnetic pressure can be achieved in this configuration; values are in the order of 30 – 40% of the magnetic pressure (to be compared with typically 4-12% in other Tokamaks and Stellarators). MAST provides important contributions to the physics understanding of toroidal plasmas in domains not accessible by larger aspect ratio devices. Drawbacks of a Spherical Tokamak for a reactor perspective are the tight space in the centre bore of the configuration which, at the desirable low aspect ratios, does neither permit to shield the centre solenoid against the neutron flux nor to employ in this part of the device a breeding blanket and would require operation with an exchangeable central column. The geometry of the device also leads to a much smaller effective surface of the divertor and therefore to a very high power load which may be difficult to master in high power plasmas close to burn conditions unless the divertor would be relocated to a larger radius location. Due to MAST's rectangular cross-section of the vessel and toroidal field coils, it could be possible to implement such a "long-leg" divertor with a considerable flux expansion and substantial reduction of the specific power load to the divertor plates and it would be interesting to study this option[54].

While a reactor perspective is difficult to envisage, it is contemplated that the ST could be developed into a fusion neutron source with low power amplification for a

[54] A similar divertor configuration is currently being studied in conjunction with the US concept of a National High-power Advanced Torus Experiment (NHTX).

Components Testing Facility (CTF). Compared to a standard Tokamak a ST volumetric neutron source is expected to have a large usable volume with high fluence at low tritium consumption (estimated at ~ 1 kg/fpy). However, the problems mentioned above, issues of stable steady-state operation and others would have to be resolved and, depending on the confirmation of the usefulness of a CTF, a feasibility study would be required if a ST based volumetric neutron source were to be built. The experimental validation of feasibility could be a major objective for MAST with the proposed substantial upgrade.

MAST	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Spherical Tokamak	1,4	0.52	0.85	0.65	~10	5.4	0,5 s (@ MA)
Major upgrades proposed (→2014): Enhancement of heating systems, divertor, centre column, pellet injector to access high performance, steady-state (~5s) regimes							

- Benefits:
  - *ITER:* contributing to assessing aspect ratio dependence of physical effects. Very good access for plasma diagnostics providing unique information. Substantial contributions for model testing and development.
  - *DEMO:* Substantial contributions to a consistent database for DEMO optimisation. With proposed upgrades the feasibility of a ST based CTF could be experimentally tested. The study of the novel concept of a long-leg divertor could be of major interest.
- Drawbacks, Risks:
  - ITER: limited contributions in the core parameter regimes of ITER
  - *DEMO:* while a CTF may be feasible, the concept is difficult to be extrapolated to a fusion power plant without invoking still speculative technical solutions.
- Cost:
  - Investment planned: 37 M€
  - Operation: 6.5 M€ (after upgrading 8 M€/y)

c) Reversed Field Pinch

RFX is the world's largest Reversed Field Pinch (RFP). This is a configuration where the Plasma is allowed to relax in a (minimum energy) state with reversed toroidal field at the plasma boundary and the toroidal field at the coils is much lower than in the Tokamak or Stellarator, allowing very high engineering beta values. The configuration does not show disruptions and can be brought into a single helicity state with improved confinement characteristics relative to the standard mode of RFP operation. The toroidal loop voltage in RFX is about 20 V (an order of magnitude higher than in a Tokamak) which provides an impressive ~30 MW Ohmic heating of the plasma. No other heating system is therefore installed. Research on RFX is contributing to the understanding of a range of physics features in toroidal plasmas. Generic RFP research on RFX links with work of relevance to Tokamaks such as the study of relaxation phenomena (turbulent core transport, electrostatic and strong magnetic shear, edge and pedestal turbulence), density limit and beta limit studies, momentum transport. Of particular interest is active real-time MHD mode control of the plasma boundary for Resistive Wall Mode control and in view of disruption prevention in Tokamaks. This work enabled RFX staff to make significant contributions to MHD work in the Tokamak domain. Plans foresee to study these phenomena in the coming years in a range of 1.5 – 2 MA plasma current close to the maximum specifications of RFX.

RFX	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Reversed Field Pinch	<2 MA	0,7T (axis)	2	0.5	10	30 (Ohmic)	0.5s (@ 1.5 MA)
After completion of the recent refurbishment no major upgrades are currently foreseen.							

- <u>Benefits</u>:
  - *ITER:* Contributions to the identification of MHD phenomena at parameter combinations not accessible with Tokamaks in particular for the plasma boundary. Particularly equipped for the study of active mode control by magnetic actuators.
  - *DEMO:* The same features as the ones of interest to ITER, study of physics phenomena at high RFP performance widening the DEMO database.
- <u>Drawbacks, Risks</u>:
  - *ITER:* Usually indirect contributions since the RFP/RFX parameter space is remote from the ITER one.
  - *DEMO:* Extrapolations of the RFP concept towards a reactor have been studied but, based on present knowledge, they do not appear competitive and are not considered in the reactor perspective of the European fusion programme.
- <u>Cost:</u>
  - Investment planned: after the recent refurbishment none are foreseen for the next years.
  - Operation: 2.5 M€/y

EXTRAP-T2R is a small Reversed Field Pinch with active saddle coils for feedback control of MHD linked to a real time digital controller enabling various feedback routines and feed forward mode control. The device has provided the first demonstration of full active control of multiple resistive wall mode instabilities in an RFP and results on measurements of resonant field error amplification and on statistical properties of turbulence in the edge region.

EXTRAP-T2R	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Reversed Field Pinch	0.15MA	0,1T (axis)	1.24	0.18	0.8	~3 (Ohmic)	0.05s
After completion of the recent refurbishment no major upgrades are currently foreseen.							

- <u>Benefits</u>:
  - Contributions in the area of MHD modes, control and turbulence. University based device for basic fusion physics and training.

- Drawbacks, Risks:
  - Due to size and confinement type limited direct contributions to an ITER /DEMO relevant data base.
- Cost:
  - Investment planned: none foreseen.
  - Operation: 0.55 M€/y

d) Stellarator

WENDELSTEIN 7-X is a large Stellarator with modular coils which is optimized for eliminating a net toroidal current inside the plasma. The device, presently under construction is planned to be commissioned in 2014. Building on results from its predecessor W7-AS and the LHD device (Japan, helical coils, different optimisation, similar in size to W7-X) W 7-X should demonstrate in particular the expected good confinement of fast particles and more generally plasma confinement at a level comparable to Tokamaks of similar size and field as well as the necessary impurity control and exhaust capability. The fully superconducting device is designed for a reasonable flexibility of the magnetic configuration around its design point and will in a second phase, foreseen to start in 2019, be capable of long-pulse/steady-state high power discharges with active cooling of the divertor and all in-vessel components. R&D on W 7-X will be pivotal for determining the prospects, in particular for steady-state operation, of a Stellarator DEMO or power plant.

W 7-X	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Stellarator	-	3	5.5	.53	30	20 (33)	1800
Start 2014, major upgrade 2019 (cooled in-vessel components, divertor, heating, diagnostics)							

- Benefits:
  - *ITER:* W7-X being under construction, contributions are made by modelling efforts to ITER relevant activities but no experimental input from the device is foreseeable for the ITER construction period. Later the device will be of interest for exploring experimentally the dependence of physics effects on the (absence of) plasma current.
  - *DEMO:* In its full layout expected to be capable of steady-state operation with high wall power load, the device should provide essential information on the potential of an optimized Stellarator in the perspective of DEMO and a fusion power plant. This would include in particular the confirmation of steady-state operation without significant external current drive, absence of disruptions, the potential for higher density operation (resulting in lower relative alpha particle pressure) than the Tokamak.
- Drawbacks, Risks:
  - *ITER:* no direct contributions to ITER planned. The time schedule will permit experimental information at full performance only in parallel with ITER operation.
  - *DEMO:* The Stellarator development is 1-2 generations behind the Tokamak and may not meet the DEMO schedule as envisaged for a

fast track along the ITER Tokamak line.

- Cost:
    - Investment planned: basic device (ongoing): 346 M€, Diagnostics (ongoing up to 2019): 26 M€
    - Stage II heating (after 2019): 38 M€
    - Operation: 49 M€ including personnel, facility infrastructure and running investments (preliminary estimate)

TJ-II is a Stellarator of the "flexible HELIAC" type (with a core circular coil made of two helical windings around which the vacuum vessel with the plasma are helically arranged) allowing particularly wide variations of the equilibrium configuration and study of the effect of the rotational transform (the device has low shear), and in general the magnetic topology, on transport, improved confinement, turbulence and plasma rotation. The team has acquired, in conjunction with its experimental work on TJ-II a strong position in the areas of turbulence and transport within the European and international fusion programmes. Recently the device has been equipped with a system for coating the wall with evaporated lithium providing substantial better performance. The foreseen installation of a divertor system is expected to have high relevance for W7-X.

TJ-II	I_p (MA)	B_t (T)	R (m)	a (m)	V (m^3)	P (MW)	τ (s)
Stellarator	-	1	1.5	0.2	1.2	2.6	1
Major upgrades underway (→2010): Electron Bernstein Wave heating, Divertor system							

- Benefits:
    - *ITER:* Contributions to the study of turbulence and plasma effects at low current operation which facilitates to identify the role the plasma current in the underlying physics.
    - *DEMO:* As the only active Stellarator in Europe particularly useful for preparation, and in anticipation, of W7-X. Contributing to the general Stellarator data base.
- Drawbacks, Risks:
    - *ITER:* Only indirect contributions to the Tokamak data base
    - *DEMO:* When the large W7-X comes into operation the device will lose relevance.
- Cost:
    - Investment planned: 4 M€
    - Operation: 2.6 M€/y

3. Contributions of fusion devices to the R&D needs and remaining gaps

JET and the equipment in the associated laboratories form a powerful set of facilities for the study of fusion plasmas. However, the identified R&D missions for the core programme require reorientation and further experimental capabilities. While JET and some of the smaller devices will, with adequate upgrades, be able to address efficiently these ITER and / or DEMO R&D needs during the next decade, some other machines are losing relevance within the foreseeable future, in particular also when taking into account the evolving international facilities base. This section will analyse how the facilities will be able to cover the identified research missions.

a) **Support to ITER** (concentrating on the period until 2018, including preparation for DEMO design): For the necessary R&D for ITER the JET facility is the internationally most relevant one even if the smaller European devices provide, also in international comparison, significant contributions as well. Out of the seven missions the first five are relevant to fusion physics on a time horizon of ITER commissioning and these are addressed with regard to ITER aspects but it should be noted that DEMO relevant work is an integral part of the R&D programme of the major facilities.

Mission 1: Key issues are linked to ensuring that ITER can establish, maintain and control a stable burning plasma. This requires developing heating strategies in conditions where fast ions can be produced and their interaction with the bulk plasma be studied under conditions sufficiently close to ITER. Capabilities are available for fusion alpha particles with JET (which in DT operation has produced 3.5 MeV alpha particles, albeit in a low fraction but with isotropic distribution) and for externally driven fast particles with devices which can produce these burning-plasma relevant fast particle populations by minority heating (in particular JET and ASDEX-UG but also TORE SUPRA, TEXTOR and MAST, the latter, due to its low magnetic field, with velocities well above the Alfvén velocity). Gaps exist in particular on JET as the most relevant device where, for reaching a relevant normalised alpha particle pressure, the heating systems must be upgraded. This is currently under preparation.

Mission 2: Here exists a complex demand scenario at conditions sufficiently close to ITER: improvements in wall conditioning, pulse management, plasma scenario control tools with regard to operational boundaries, off-normal events and wall protection. Capabilities are available in particular with JET and ASDEX-UG which are closest to ITER; instability control at operational boundaries with ECRH can be studied on ASDEX-UG and TCV. For disruptions also other Tokamaks can contribute, in particular those with elongated cross-section. For break down and start up additionally MAST, TCV (ECRH) as well as FTU and TORE SUPRA (ECRH and LHCD) are useful (for these early phases their limiter configuration and a circular cross section are not a concern). On RFX specific studies for the active MHD and plasma wall control can be undertaken. Gaps: The full range of ITER bulk and fast particle physics cannot be studied. On JET the upgrade of heating systems is necessary for reaching relevant high plasma pressure boundaries and for extending control capabilities an ECRH system should be installed. Active coils for the control of resistive wall modes will be an important tool for scenario control. They are now foreseen for ITER and should preferably be also installed in JET, ASDEX-UG and other relevant devices.

Mission 3: Central to this area is the optimisation of the first wall where carbon, as used so far in most devices will not comply with the power fluxes and the required low tritium retention targets in ITER and DEMO. Tests of a metallic wall (coating) under ITER relevant power fluxes (P/R>20) and wall temperatures (200°C for ITER, higher for DEMO) are necessary Furthermore, plasma scenarios must be validated for a relevant set of plasma facing materials. Capabilities are available, for a tungsten wall (however, at lower temperatures and limited discharge durations) on ASDEX-UG and under preparation in JET. Full, elaborate remote handling is only available on JET which is also the only device for testing tritium retention in a high-performance Tokamak environment. TORE SUPRA so far has unique capabilities for the investigation of steady-state aspects and wall equilibration under high heat loads. MAST and TEXTOR can access relevant heat fluxes, the latter as a plasma-wall-interaction facility for components up to 15 cm diameter (doubling is foreseen). The liquid-metal limiter installations in FTU (and, regarding fundamental MHD aspects, ISTTOK) can explore novel possibilities of extending the operational range beyond the one of solid target plates for withstanding stationary or intermittent heat fluxes. Gaps relate to JET until the wall will be modified to a W and Be coverage and to other medium-size devices which do not operate with ITER relevant walls (note that in circular devices the plasma edge temperature is usually too high for long-pulse operation with high-Z metallic limiters/walls).

Mission 4: The study of long-pulse and steady-state operation demands improvements in physics and technology aspects of devices and their components. For the Tokamak the achievement of a sufficient bootstrap current fraction is essential, requiring substantial control capabilities (for high $ß_N$ plasmas) which are in part addressed in Mission 2. In addition the current drive and heating systems must be capable of steady-state operation and must have good coupling properties for the required plasma scenarios. Furthermore, the exhaust has to be adequate for maintaining clean, stable plasmas over the desired long pulse durations and ultimately in steady state. Capabilities are available in TORE SUPRA which currently is still unique in long pulse operation and power handling ability. JET and ASDEX-UG (the former with LHCD, the latter with ECRH) are capable of developing ITER relevant plasma scenarios at limited pulse duration. FTU, TCV and MAST contribute to this mission. RFX can offer a contribution to the physics and control of resistive wall modes and to feedback systems and TEXTOR investigates means for influencing the power flow to the wall by ergodic fields. Gaps exist since TORE SUPRA, the only European device with superconducting coils, cannot access fully ITER relevant scenarios (circular limiter plasma and non-metallic wall), JET and ASDEX-UG should preferably be upgraded with active control coils for RWM stabilisation, the latter should be equipped with LHCD while JET needs an ECRH system for NTM control and its LH system should be upgraded for testing an ITER relevant launcher. MAST could be upgraded to better access steady-state regimes.

Mission 5: An intensive activity in theoretical work and numerical modelling must interact with comprehensive experimental studies in a wide parameter range in order to validate the capability of codes to predict fusion performance. Capabilities exist experimentally in particular by the “step-ladder” facilities COMPASS - ASDEX-UG (and DIII-D, Alcator C-Mod) – JET (and JT-60U) and by devices which can access different shapes and features, in particular MAST, TCV, the circular Tokamaks TORE SUPRA, FTU, TEXTOR and the Reversed Field Pinches RFX (and EXTRAP) as well as the Stellarator TJ-II. The complexity of the physics involved in a “numerical fusion device” gives to these latter facilities which explore parameter regimes beyond the ones of the ITER step ladder devices a dedicated role for disentangling parameter dependencies (toroidal magnetic field, toroidal plasma current, aspect ratio etc.). These would be more difficult to assess in a data base from main-line devices only. Gaps exist in several of the mentioned devices with regard to their capabilities to address features of Missions 1-4, such as high ß regimes at high fields and current, and specific diagnostics.

Mission 6: ITER will still be built with conventional materials, nevertheless its nuclear operation requires special preparation, in particular also for licensing and safety aspects. Capabilities: JET provides a ground-breaking, but limited, starting point for the nuclear technology on ITER and for refurbishment and maintenance procedures. Gaps: During the period until start-up of the nuclear phase of ITER there will be no other Tokamak after JET which could provide information on nuclear components. Dedicated efforts are needed to support the licensing requirements for ITER.

b) Preparation for DEMO (for the period until start of DEMO construction): DEMO oriented work is being pursued in parallel with work for ITER and, once in operation, ITER itself shall provide the core information on components, scenarios and physics for a DEMO design. Work for DEMO on concept improvements, and also ITER related work in the DEMO perspective, is an integral part of the facilities' programmes considered in the previous section and aspects, relating especially to the period until ITER commissioning, will not be repeated here. Rather, this section concentrates on the time horizon of the early years of ITER operation, i.e. from the commissioning expected in 2018 until the time span of about 2025-30 when major results from the DT operation on ITER should become available. This period will be of particular relevance for the finalisation of the DEMO engineering design under a fast track scenario. For this still distant period the work on smaller devices cannot be predicted with much confidence but it is obvious that the arguments of cost- and time efficiency given before impose that some smaller devices will be available for exploring/testing novel features and operational conditions in assistance to the satellite devices and ITER itself.

Regarding satellite devices, JET cannot be expected still to operate during the period under consideration. With JT-60SA a successor to JET should become available around 2016. High performance operation will start several years later. According to present understanding its use for the European Fusion Programme may be rather short in time[55] and would not extend (much) in parallel to the relevant phases of ITER high performance nuclear operation. The device as currently planned, would be targeting the present JET operational regime, however, having a relatively low magnetic field, it would have a somewhat limited capability for assessing a sufficient range of regimes of DEMO interest. Also it is not yet clear whether it will have an ITER and DEMO relevant first wall, nor if the current drive and heating systems (ECRH, NBCD) will fully adequate for accessing the high performance regimes with dominant electron heating which should be of main interest for establishing a DEMO design point. Another satellite device, which, within reasonable financial constraints, should be designed to optimally complement JT-60SA and cover a wider DEMO relevant (non-dimensional) parameter space, should therefore be considered for the European fusion programme, having taken note of the FAST proposal. For supporting progress towards a risk-optimized DEMO design this device should become available within a time window prior to, or with, the early years of ITER operation.

Mission 1: ITER will have to provide the key information on burning plasmas. Capabilities are provided primarily by ITER. Besides ITER, a stepladder of devices is needed for exploring, testing and validating physics and operational features for ITER in a cost and time efficient way. Assessing the fast particle

[55] The present Broader Approach Agreement foresees European participation in the JT-60SA exploitation for five years at full performance.

confinement in a Stellarator configuration should be addressed by Wendelstein 7-X once it has been equipped with full heating. Gaps, as of now, exist since ITER may not fully assess the DEMO relevant range of power amplification (Q = 30 – 40) with the corresponding higher fraction of fast alpha particles in the core of the plasma and impact on possible fast particle instabilities. This issue likely will remain open. With regard to the important satellite function JET is unlikely to be able of performing its current role as top device of the stepladder approach on the envisaged time horizon around 2020 and beyond. In this context it should be noted again that the DT capability of JET or another satellite will be less significant and would not be cost efficient, once ITER will have entered high performance DT operation.

Mission 2: Capabilities: The main information is expected from ITER, although higher efficiency as demanded in DEMO requires operation at higher normalised plasma pressure and density. For this mission JT-60SA is expected to be able to provide information similar to what could be expected by JET, however with the advantage of having an ECRH system (on JET the early availability of such a system would be of very high relevance for the remaining exploitation). Smaller European devices, at least one of them in the range of ~2 MA, will be a necessary complement. For the Stellarator approach Wendelstein 7-X should conclusively address wall conditioning and operational control aspects. Gaps will exist since the scope of this mission is very large and JT-60SA will not be able to address all features relevant for reliable operation (also, under current planning the device does not have ICRH or LHCD systems and, more importantly no reactor-relevant first wall). If these limitations remain, another more adequate device in a current range around 5 MA would be of high relevance for the European fusion programme under this mission.

Mission 3: Capabilities: Again, ITER should deliver the key information for DEMO preparation. JT-60SA is foreseen to have comprehensive remote handling capabilities. Reliable Stellarator divertor operation has to be developed on Wendelstein 7-X for which the implementation of an actively cooled divertor is foreseen in 2019. Preparatory studies will be done on TJ-II in the near future. MAST could examine the potential of a long-leg (flux expansion) divertor and the corresponding impact on first wall elements. Gaps: As mentioned before, the carbon wall of JT-60SA, as likely foreseen during the operational phase in which Europe is participating, will be a severe limitation to its DEMO relevance and other devices, including one in the same class, should be available which can address the compatibility of the first wall with relevant plasma conditions. DEMO will have an even higher power flux (P/R value) than ITER, a feature which will be difficult to match in a smaller main-line device. A compact high field device like proposed with FAST could approach this condition more readily. In addition, for efficiency of the thermal cycle, DEMO will have to operate at higher wall temperatures (400-500°C) and it would be highly desirable to explore operation with such wall temperatures on a device in the coming decade.

Mission 4: Capabilities: ITER should approach long-pulse operation and ultimately steady-state conditions. JT-60SA as a superconducting device with active MHD control capacity will be able to study the steady-state / long pulse regime and the associated long-pulse plasma control. The superconducting Wendelstein 7-X, equipped with actively cooled first wall elements after the first operational phase, should demonstrate, around 2020-5, stable DEMO relevant steady-state operation. Gaps: ITER, according to present planning will not (yet) have an LHCD system. This may compromise its ability to approach steady-state operation. Limitations in its heating and current drive systems compromise the ability of JT-60SA for accessing high performance high bootstrap current operation close to operational boundaries. This challenging regime, in fact, will be a major issue for the preparation of DEMO which must demonstrate efficient and reliable plasma operation permitting steady-state electricity generation. No European superconducting Tokamak will be available during the required period. If a European satellite would be decided it should have sufficient capability for addressing this mission with pulse durations sufficient to cover the relevant plasma physical time scales.

Mission 5: As for the other missions also for the experimental support and validation of modelling the "stepladder" approach should be retained with a basis spanning a sufficiently wide parameter range. Capabilities: ITER itself will provide the experimental validation for modelling the high performance burning plasma regime. For extending the experimental information and the validation of predictive codes to wider parameter regimes it should be expected that in addition to satellite device(s) a sufficient range of medium and smaller facilities will be operating in parallel to ITER – at least when taking into account the world-wide facility basis[56]. Gaps: Considering the presently foreseen operational range of ITER and JT-60SA a complementary device in the same class would be of high relevance for this mission.

Mission 6: ITER will be the first Tokamak to undertake extended operation with DT fuel, creating a strong nuclear environment. Still, the envisaged activation will be considerably lower than in DEMO for which materials and components must qualify to demanding standards of irradiation, power handling, efficiency and reliability. Capabilities: ITER should provide, with regard to the integration of nuclear materials and components in a Tokamak environment, their handling, maintenance and refurbishment, safety and licensing, the essential information needed for DEMO and also the Test Blanket Module Programme will be mandatory. Gaps: A Components Test Facility could be particularly relevant for this mission for reducing the development risk of components exposed to the neutron flux and it should be assessed to which extent it would enhance the possibilities provided by IFMIF. Whether such a Components Test Facility can be constructed using the Spherical Tokamak (or another) concept also has still to be assessed[57].

Mission 7: Work under the above six missions will determine the physics and technology of the DEMO design. With regard to fusion facilities the capabilities and gaps which were identified under the other missions apply also for this mission. In addition it still has to be assessed whether an early DEMO[58] could significantly contribute to the development of fusion. If time schedules would permit, the merit of such a device, which would be built on near-term extrapolations of physics and technology, is proposed to be a reduction of risk

[56] Access to the world facility basis will be easier for the purpose of Mission 5 than for the other missions where a much stronger interplay with the operators and influence on the operational and machine development programmes would be required.

[57] Critical issues for the Spherical Tokamak approach appear to be, apart from the core plasma physics aspects (start/ramp up, core stability at high performance, off-axis current drive, impurity control), in particular the capability for a high duty factor, the feasibility of a centre column with sufficient lifetime /exchangeability and a divertor which would have to handle significantly more power than in ITER. For the latter issue the recent ideas for a long-leg divertor might be attractive.

[58] as discussed in the SET Plan.

along the further development path of fusion. As mentioned before, the Panel has not included this option into its assessment.

The following figures show the positioning of major fusion devices with regard to ITER and DEMO parameter spaces in dimensionless coordinates.

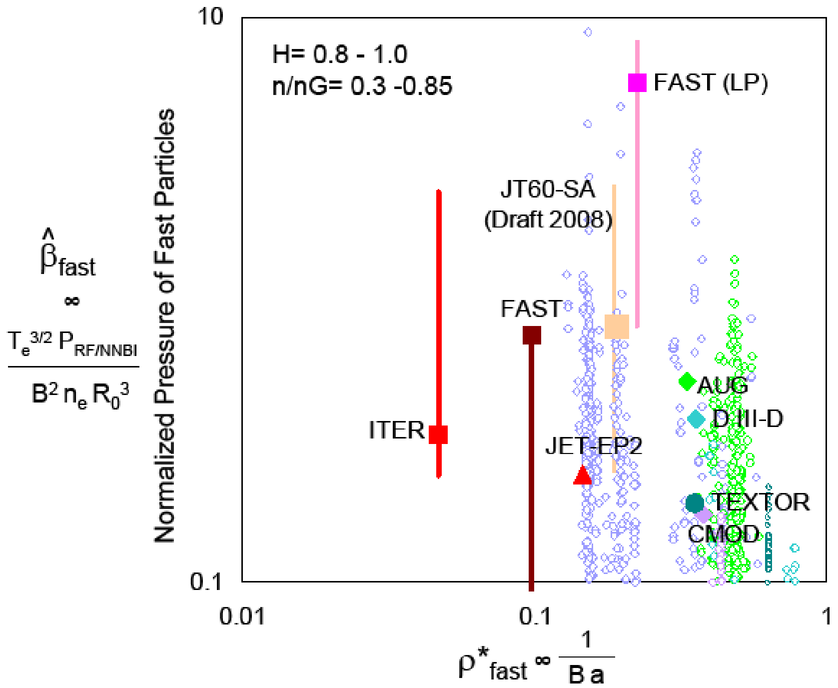

Fig. 3: Approximate operating regimes of fusion devices in the plane of (normalised) fast particle pressure (fusion alpha particle pressure for ITER) and (normalised) gyro radius of fast ions (source: EFDA)

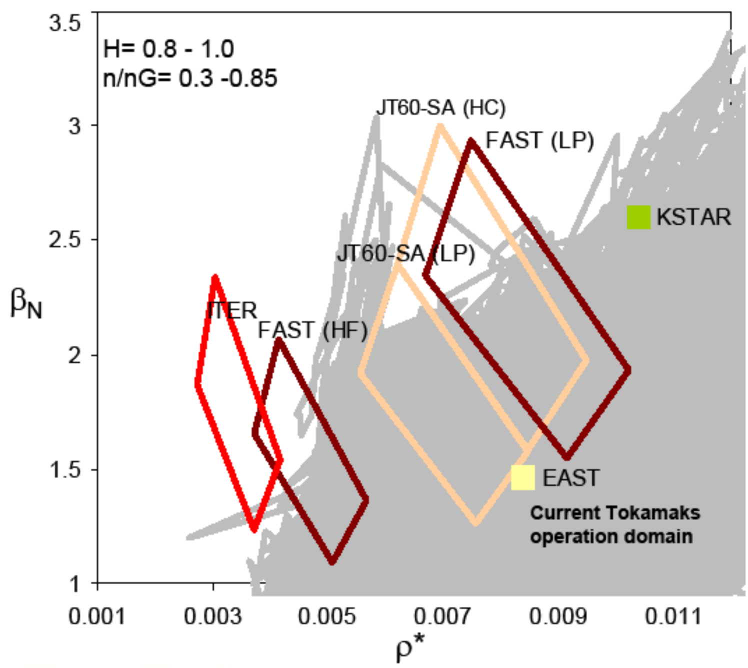

Fig. 4: Approximate operating regimes for fusion devices in the plane of (normalised) plasma pressure and (normalised) gyro radius of thermal ions (source: EFDA)

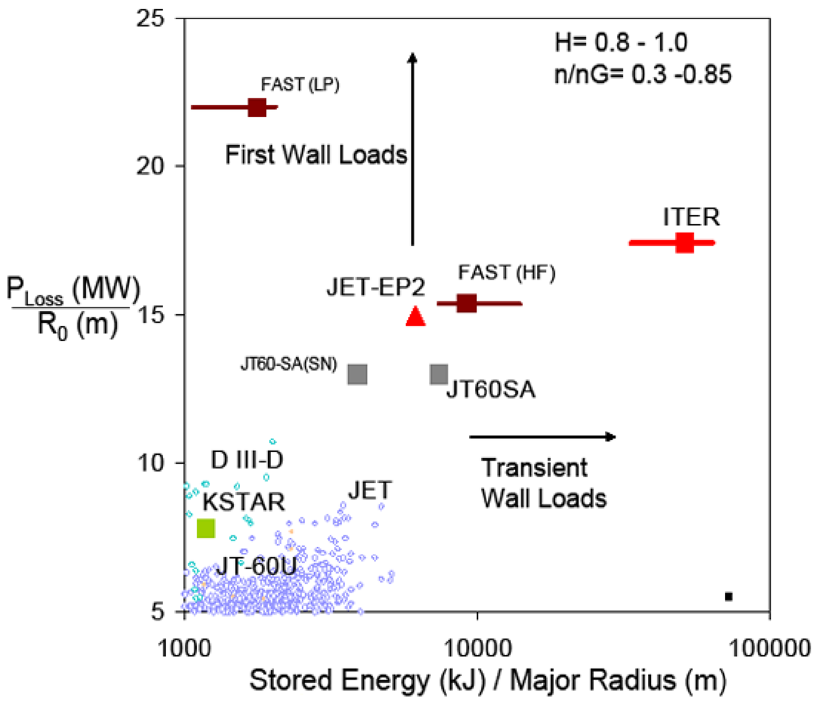

Fig. 5: Approximate operating regimes for fusion devices in the plane of (normalised) loss power to the wall and (normalised) stored plasma energy (which may be released in transient events) (source: EFDA)

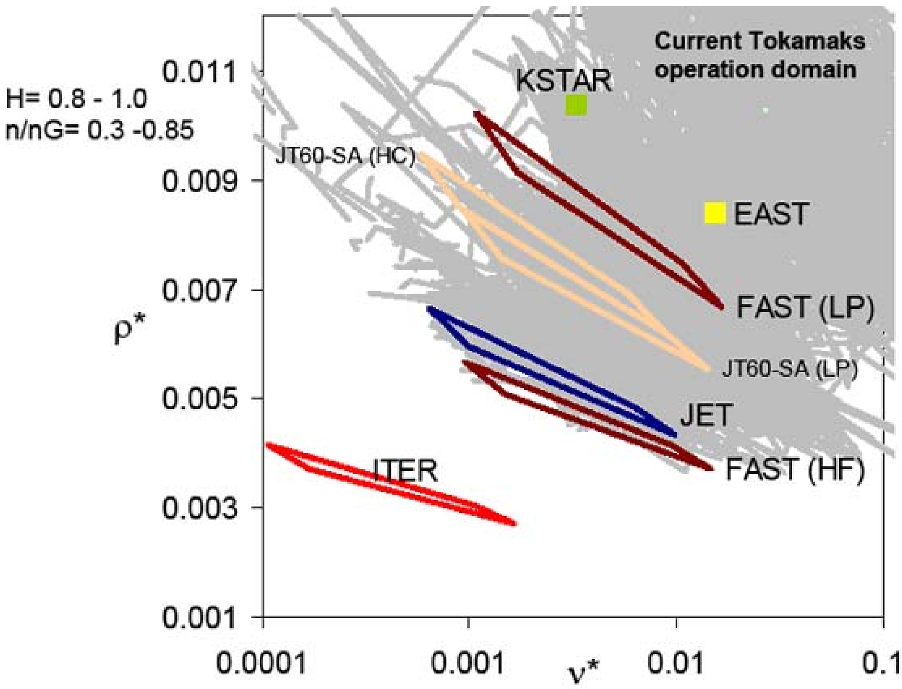

Fig. 6: Approximate operating regimes for fusion devices in the plane of (normalised) ion thermal gyro radius and (normalised) collision frequency indicating how well devices can map ITER conditions for the scaling of bulk plasma transport. The bottom-left region in the graph is the most critical one. (source: EFDA)

C. Fusion technology facilities

The first and foremost fusion technology facility for the further development of fusion will be ITER. Its design, components, systems and experience in developing partnership with industry will provide the most relevant information for DEMO and the further path towards commercialisation of fusion power. The satellite devices and, in particular in the areas of first wall materials and components, also the other fusion devices are contributing as well to fusion technology R&D. Furthermore the development of heating, current drive and fuelling systems, diagnostics etc. for the existing fusion devices has generated the knowledge at the basis of the corresponding activities for ITER. Another example is knowledge from the construction of the superconducting coils for Wendelstein 7-X which has been incorporated mainly in the testing procedures for the ITER poloidal and toroidal field coils[59].

When speaking here about the fusion technology programme and technology facilities, however, the additional and important range of tasks and specialised facilities are addressed which are needed in view of ITER construction and for progressing towards DEMO construction. These are development and testing facilities for components (e.g. magnets, heating systems, breeding blanket, divertor, diagnostics, remote handling), irradiation and high heat flux facilities for structural and functional materials (e.g. steels, insulators, plasma facing materials and coatings), pre- and post irradiation inspection and other test facilities and mock-ups.

1. Supporting ITER

For the major components of ITER substantial developments and tests were performed during the Engineering Design Activity phase. Extensive testing of systems and components is required for the construction in order to qualify manufacturing technologies and to minimize the risk of any failure of components and systems which could be detrimental if discovered too late Furthermore, there are components for which some R&D is still required or where the procurement schedule allows to benefit from anticipated improvements by further R&D. For these categories of tasks facilities are needed which are listed below in Table II.1[60]. The continued operation and upgrade of technology facilities and laboratories to meet the ITER requirements with adequate capacity is a top priority. Among new facility the most significant one is the Neutral Beam Test Facility which will be built at Padova and where the ITER NBI system will be developed and tested.

[59] E.g. the Paschen tests of the coils against insulation problems under over-voltage.

[60] The two tables in this chapter are based on table A.4.2 of the EFDA input paper, Part II “Facilities” and have been updated on the basis of input received by F4E.

Table II.1: Technology facilities for ITER R&D and testing			
Purpose	Capabilities: gaps or surplus capacity	Facilities (* = used on customer task-by-task basis)	Comments, benefits, risks, upgrades needed
Superconductor strands	ok	*Walter Spring (Durham, UKAEA), *Pacman, *Tarsis (both FOM)	Used on a customer task-by-task basis
Superconductor cables	ok	Sultan, Edipo (both CRPP), Twente Press (FOM)	With the recent setup of EDIPO capacity is now expected to be adequate
Superconducting coils	Gap Capacity is insufficient for TF and PF windings cold tests	TOSKA (FZK), New facilities to be built for the TF winding packs cold tests on the industrial site, and for the PF coils on the ITER site	TOSKA will be too small for ITER coil tests. Estimated cost for new cold test facilities: 50M€. Unclear whether the coils will be built at one or several industrial sites. To be determined whether only one facility should be built at the ITER site.
Heat flux Be compatible components	Gap	BESTH (IPP-CR), *ETA-BETA (ENEA), JUDITH (FZJ)	Small upgrade for BESTH Estimated upgrade for Judith ~0.75 M€ (new beam control unit) These facilities are Be compatible, but their capability is insufficient for series production and acceptance tests for ITER first wall panels. This would require a dedicated e-beam facility capable of handling the dimensions of the ITER first wall panels. Estimated cost ~ 3 M€
High heat flux component	OK with upgrade of JUDITH	JUDITH /HML (FZJ), GLADIS (IPP), *FE200 (AREVA/CEA)	JUDITH: New hot coolant loop, new hot cell building needed: ~7.5M€
ECRH	Gap and partly surplus	ECRH test beds - CRPP, - FZK, -IPP-Greifswald -Test stand in industry	Gyrotron test facility (some upgrade needed) Prototype testing of antenna (some upgrade needed)
ICRH	ok	-JET, -CEA	Some upgrades are required (~1M€)

NBI	Gap Basic experience from (lower power, lower voltage) facilities at JET, IPP Garching and CEA Cadarache, also source development. Partly surplus after construction of the NBTF	New Facility: - Neutral beam test facility NBTF (ENEA-CNR-INFN, Padova)	Estimated cost of this new facility is ~100 M€ depending on technical specifications. After development and test of the NBI prototype, two systems will be built and installed in ITER Later use of facility is still to be decided pending DEMO requirements.
Cryopumps	Ok with upgrade	TIMO (FZK)	Upgrade ~2M€
Port plugs	Gaps	In the EU and/or in an ITER on-site facility.	Diagnostics, ECRH and ICRH. Systems integrated into port plugs require testing. New facilities needed. Cost estimated about 10M€. To be determined in collaboration with other ITER parties.
Fuel cycle system	ok	TLK (FZK), JET AGHS	Capacities have been established to fulfil this important testing need. For the tritium breeding modules to be installed in ITER (if required) an upgrade may be necessary (~8 M€ in 2015)
Divertor remote handling	OK with upgrade	Divertor Test Platform 2, VTT Finland	Full scale mock-up of ITER divertor region, simulation of in-vessel maintenance and RH. Costs estimated ~6 M€
NBI remote handling	ok	NBTF Padova	Task will be included in the NBI facility under construction.
Transfer casks	Gap		Test facility needed for transfer cask transport and docking and in-cask operations. Cost estimated ~4M€
In Vessel Viewing System	Gap	IVVS	Facility to test the In Vessel Viewing System, simulating the ITER environment. Cost estimated ~5M€
Safety issues and procedures	Ok, with upgrade	Facilities at FZK and CEA	1.3 M€ for upgrade of HYDEX (FZK)
Dust and tritium measurement and removal	Possibly gap	Tokamak devices	Possibly need for dedicated facility demonstrating for the complex ITER vacuum vessel geometry feasibility of dust measurement, mobilization and removal. Cost estimated ~5M€
Test blanket module	Ok with upgrade	Facilities - HELOKA (FZK) EBBTF (ENEA), *Latvia Other facilities for specific tasks	1.2M€ for upgrade of EBBTF (ENEA).

The aggregated costs for these ITER oriented investments on technology facilities, including a few useful additional demands[61] amount to about 180 M€ whereof 100 M€ are required for the NBTF facility and estimated 50 M€ for the toroidal field and poloidal field windings cold test facilities.

It should be noted that the current planning for ITER foresees to use a carbon divertor for the initial hydrogen and subsequent (first nuclear) Deuterium phases and to exchange for a tungsten divertor only for the Deuterium Tritium (D-T) phase. While this may be useful for starting the device with the best known divertor conditions the consequence is that ITER will delay the fully nuclear operational phase in D-T which generates a risk for delivering comprehensive information in time for a fast track DEMO construction target.

As a consequence of this planning there are still substantial ITER tasks which address carbon divertor issues (high heat load, erosion, dust, co-deposition of hydrogen isotopes, tritium removal techniques, safety) and which require corresponding facilities. Should ITER take a decision to start right from the beginning with a metallic divertor (tungsten), a corresponding adaptation of ITER tasks and facilities should follow without delay.

2. Preparation for DEMO

Beyond the specific tasks in conjunction with ITER construction and the programme to be executed on ITER once in operation there is still a demanding range of technology R&D tasks for which ITER will not be able to contribute the key information and which must be executed on specific technology facilities, listed below in Table II.2, in order to acquire the capabilities needed for advancing towards DEMO and the commercial fusion power plant. Key areas are the further development of components (Breeding Blanket etc.) and structural and functional materials (Ferritic Martensitic steels, ODS steels, SiC-SiC and other high temperature materials, ceramic insulators, neutron multipliers and breeder materials). Indispensable for the testing and qualification of materials is an irradiation facility with a fusion relevant neutron spectrum as presently prepared under the IFMIF EVEDA. The realization of this facility is of utmost urgency for filling the most essential gap with regard to mission 6.

Table II.2: Technology facilities for DEMO oriented R&D and testing

Purpose	Gaps and surplus capacity	Facilities (* = used on customer task-by-task basis)	Comments, benefits, risks, upgrades needed
Plasma-wall interaction simulators	Gap	- MAGNUM PSI (FOM) - PWI facility (IPP), - VISION (SCK-CEN), - Techno Fusion (CIEMAT) International: PISCES, *plasma guns (Russia)	Investments: MAGNUM ~3M€, Investments: VISION 1.1 M€, Technofusion (new proposal) 8 M€
Plasma facing components testing		*SATIR (CEA)	Acceptance of joints, non-destructive inspection etc.

[61] These are listed in the EFDA Input Paper to the Panel, part II, p. 25 and refer to some further useful enhancements of facilities in the area of high heat flux component, ECRH, cryopump system, TBM and safety related testing.

Materials characterisation (non-irradiated)	Surplus, gaps	*Various	Surplus for conventional mechanical testing, gaps for high temperature tests and microstructure analyses (TEM, ATP).
Post-irradiation examination		*Hot cells (partly multipurpose)	Capabilities and capacities to be determined.
Heating and Current drive test beds (efficiency and long pulse R&D)	Possibly Gap	ITER relevant facilities for NB, EC, IC, LH systems NBTF (Padova) ECRH test stand (CRPP)	Partly depending on novel concepts and technical solutions NBTF will also be essential for DEMO work if NBI will be applied. For the Heating and CD systems the demand of specific facilities needs to be assessed in due time.
Neutron irradiation of plasma facing and structural and functional DEMO materials	Gaps, Surplus (low fluence fission reactors)	*Fission reactors, *accelerator based neutron sources. The key required facility is IFMIF If feasible at reasonable conditions, a CTF could contribute substantially for risk reduction.	Fission reactors do not have the right energy spectrum and few have the proper testing environment (temperature window, instrumentation). They are only stopgaps until IFMIF and/or another facility with a high fluence fusion neutron spectrum will be available. (for IFMIF and CTF: see text). In addition to IFMIF research fission reactors with high fluence continue to be needed. Low flux 14 MeV sources are useful for benchmarking of models.
Charged particle beam irradiation		Dual and triple beam charged particle accelerators *JANNUS (CEA) *Rossendorf, - Technofusion (CIEMAT) (proposed)), - FNG (Frascati) (with proposed cyclotron).	Charged particle beams are useful mainly for validating modelling and simulating fusion He/dpa ratios. Deeper penetration would be desirable and could be achieved on Technofusion (19 M€). Upgrade FNG: 7M€
Breeding Blanket development and He cooled Divertor	Possible surplus	-HELOKA (FZK) -EBBTF (ENEA) Other facilities for specific tasks	Upgrade HELOKA ~16 M€ With narrowing down of blanket options surplus expected except for tritium and helium facilities and irradiation facilities
Tritium / fuel cycle	Gap	-TLK/FML (FZK), -Tritium laboratory (SCK/CEN) -JET Active Gas Handling System is the only facility of this kind integrated in a Tokamak.	Upgrade TLK ~8 M€ and FML Hot Cells 1 M€ Continued availability of tritium laboratories is essential. Need for integrated tests of processes and components, dependence on cooling medium.
IFMIF construction and test programme	Gap	-Additional facilities	Requires a number of specific facilities, prototypes see text. Cost estimate of this preparatory R&D effort: ~31M€

In-vessel maintenance procedures – remote handling	Gap	-Technofusion proposal (CIEMAT) includes a RH laboratory	A major remote maintenance development laboratory in response to requirements from a DEMO design will be needed, costs estimated at 10-20M€
Balance of plant	Gap	-Large He-loop facility which is Tritium compatible	Depending on DEMO design
Improved strand and conductor development, advanced model coil design	Ok Possible gap (tbd)	-Facilities for strand testing (the ones used for ITER: EDIPO, TWENTE Press, facility for coil testing)	Depending on superconductor materials and technology development

The aggregate for the costed enhancements of facilities for DEMO oriented work listed above is about 88 M€. This does include preparatory work for IFMIF in the frame of the EVEDA but not the construction costs of the facility which are addressed separately below. Table II.4 provides an overview over the major technology facilities and their cost for original investment, operation and proposed upgrades.

a) International Fusion Materials Irradiation Facility (IFMIF)

Only with an irradiation source having a neutron energy spectrum representative of the 14 MeV fusion neutrons at high fluence is it possible to test and validate materials for the application in the nuclear environment of DEMO or a future fusion power plant. At present most material examinations are undertaken in (fast) fission reactors where the neutron energy is a factor 10-20 lower than in a fusion environment. These lower energy fission neutrons create by more than an order of magnitude less gaseous transmutations. In particular the generation of high amounts of He changes significantly the properties of irradiated materials. There are accelerator-based 14 MeV sources which, however, have very low fluence and are used for (modelling) benchmarking purposes rather than to irradiate many samples up to several dpa.

A source capable of delivering the necessary neutron flux needed for accomplishing relevant irradiation levels within a couple of years has been conceived based on a deuterium - lithium beam-target reaction producing a neutron spectrum generating damage and He production very similar to the ones expected in a fusion environment. The design for this source is currently developed by Europe and Japan as International Fusion Materials Irradiation Facility (IFMIF) under a engineering design and validation activity (EVEDA) in the frame of the Broader Approach Agreement with a total budget of ~150 M€. IFMIF is being designed to achieve in a volume of 0.5 litres a fluence of 20-55 dpa/full power year (dpa/fpy). Thus, DEMO materials could be qualified within a few years of irradiation on this facility. Furthermore, there will be a volume of 6 litres in the range between 1 – 20 dpa/fpy available and a larger testing volume (≥8 litres) albeit at much lower fluence (<1 dpa/fpy).

No decision for the siting and construction of this facility has yet been taken. In the frame of the EVEDA which started in 2007 and is scheduled to last six years until the end of 2013, the critical components of the facility need to be assessed. The following tasks are foreseen[62]:

- Construction and test of the first part of the deuteron accelerator prototype (up to 10 MeV, 125 mA),
- Construction of a mock-up (scale 1:3) of the lithium loop and target area (incident beam energy 1 GW/m2);
- Construction of High Flux Test Modules for validation of the sample concept and irradiation of parts in fission reactors.

EVEDA will also refine the estimates for the construction schedule which presently foresees to realize by 2018 a first beam and by 2021 the second beam and full power (40 MeV, 125 mA, 10 MW). Also the overall cost will be more precisely assessed which are presently estimated at 540 M€ for construction, 232 M€ for installation and commissioning and ~80 M€/y for operation at full power. The European share will depend on the international sharing.

IFMIF trespasses into new territories with respect to accelerator technology and in the beam target system and qualifying materials with small size sample. Measures to launch the negotiation process for deciding the site and the detailed practical and financial arrangements for the construction and operation of this facility should be urgently undertaken. The construction of IFMIF is mandatory since it will be impossible to license the structural and functional materials for DEMO construction without relevant irradiation validation – and there are no other conceivable means[63]. **In a fast track towards DEMO the realization of IFMIF is indispensable with an effective start of irradiation tests, using a first beam (half the ultimate flux), in parallel to the start of ITER operation**.

b) Materials test laboratories

The Panel recommends to strengthen the materials programme. Laboratories are needed in which candidate materials will be developed and explored which then shall be ultimately tested on IFMIF or, for high heat flux materials in ITER and dedicated facilities. Significant knowledge on materials is available, or under development, in other countries (US, Russia, China, South Korea, Japan) albeit not all is of relevance for the European programme. Although several laboratories exist in Europe and international collaboration on major research topics is growing, an enhancement of the capacity of European materials laboratories is needed for the development of new materials, preparation of samples, pre- and post irradiation examination and a strong accompanying science and modelling programme. The proposal for a new integrated materials test laboratory in Spain, called Technofusion, aims into this direction. Such a facility would be in particular important to reduce gaps in the IFMIF testing programme in the areas of high-energy multi-beam irradiation facilities plasma wall interaction simulators and studies for cooling loops of liquid metal blankets. Computing facilities for modelling materials behaviour under irradiation and relevant operational conditions will play an increasingly important role in materials research.

[62] In support and complement to these tasks it is proposed to undertake, beyond the EVEDA agreement, validation tasks and diagnostics developments for IFMIF on European facilities.

[63] It should be noted, that also in the future fission reactors are needed for materials testing. They will be used for pre-characterisation of the fusion materials whereas IFMIF will be fully required for the “final qualification”.

D. Computing facilities

Extensive modelling requirements have been identified under many missions which are of high importance to fusion research, in particular in the areas of plasma physics, operational scenarios for ITER / DEMO and, last but not least, materials research. (They have been discussed in particular under mission 5). Limitations in the presently dominating shared-use of multi purpose computing facilities recommends to proceed, beyond the establishment of integrated task forces for modelling work, to fusion-dedicated high performance computing centres with high quality professional support for numerical and computational work. One such centre will be established with European participation in the International Fusion Energy Research Centre within the Broader Approach Agreement in Rokkasho (Japan). Present planning foresees for 2012 a 1 Pflop machine. There is ample scope for another, European high performance computing centre (HPC) which could be equipped initially with a machine of order 100 Tflops which should be adequate for establishing the capabilities and means for the numerical and computational methods. It is essential to have both a, team of physicists and numerical modellers and a strong competence of parallel computing and network specialists integrated in this facility. In the medium term perspective numerical models can be expected to require computers deep in the Pflop range or beyond which, by then, should have become available. **A rapid increase of the capabilities of the European fusion programme in the advanced computing and modelling domain is mandatory to keep pace with the international development**[64].

E. An integrated facilities road map

When assessing facilities and identifying the required ones for a rapid and efficient approach to fusion energy, the Panel distinguishes, where adequate, between relevance for ITER and for DEMO and makes reference to the implications of a specific time schedule. Generally a fast track approach towards DEMO is the basis for the analysis.

1. Fusion devices

The present set of fusion devices, which were optimally designed and targeted according to the state of the art of their time and kept relevant by continuing significant upgrades, constitutes a very substantial investment and continues to be highly useful. These devices, together with those of the other major fusion programmes, have prepared the grounds for ITER and are currently being used predominantly for a wide range of ITER relevant tasks, for the preparation of DEMO and to a minor extent for fundamental fusion science. For the future, the Panel recommends a shift of emphasis towards the most relevant devices supporting ITER and DEMO[65]. International collaboration should be strongly exploited particularly for the satellite devices and the new medium-size superconducting international Tokamaks.

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[64] In this context it is interesting to note that in the US a "Fusion Simulation Project" has been launched by the Department of Energy with the overall objective "to produce a world-leading predictive simulation capability" for predicting "the behavior of plasma discharges in toroidal magnetic fusion devices on all relevant time and space scales" aiming at computing power at petaflop scale and, for DEMO predictive simulations, exaflop scale (W.M. Tang, Journal of Physics: Conference Series 125 (2008) 012047)

[65] This shift towards larger devices bears the risk of losing some of the basis for education, training and attracting young talented scientists and engineers to fusion. Parallel efforts for maintaining sufficient links to the academic environment, efforts for training and education must therefore be planned.

a. **Support to ITER** (in particular on a time horizon of ITER commissioning): The range of facilities oriented towards ITER support, must contain devices in the satellite class and at smaller size for the reasons of flexibility, risk reduction and cost and time efficiency[66].

For the satellite devices the scope of physics exploration and testing is substantial and it is advisable to have, in parallel to ITER internationally more than one device. For the next decade, however, JET will be the unique device in this class. Its current programme requires operation at least until 2014/15. A substantial tritium operation should be undertaken around this time in order, inter alia, to validate the retention behaviour of the metallic wall and the new, extended operational regimes which will only be accessible after installation of the present enhancements. JET should provide input to final adaptations of ITER systems and for the preparation of ITER operation. Beyond the 2014/15 time frame the scope of work and JET's capabilities would continue to be of high benefit for saving time and cost on ITER, where its relevance would highly benefit from an ECRH system (which should preferably be already very soon installed in order to strengthen already the programme up to 2014/5). Some further improvements, among them an upgraded lower hybrid current drive system would be desirable as well. The Panel recommends, if technically feasible, to extend the operation of JET for a few more years beyond 2014/15 until new satellite facilities will take over[67].

JT60-SA will assume, after starting operation (expected in 2016), for some time the role of the leading fusion device until ITER will start plasma operation. Among presently existing or agreed projects, JT-60SA would become the only satellite device in support of ITER after termination of JET. The involvement of Europe in its construction is important and its exploitation should provide the expected benefit. Participation in the use of the device should therefore be secured for a sufficiently long time. The device cannot cover the whole parameter space of relevance to ITER and DEMO. Aimed at exploring for ITER (and DEMO) steady state operation in pulses up to 100s, its heating and current drive capabilities and its non-reactor relevant wall (according to present planning) are important limitations[68].

The Panel, taking note of the FAST proposal as an example for a satellite showing attractive features in a compromise with comparatively low costs, recommends that Europe should develop a design aiming at optimally accessing the ITER / DEMO relevant parameter space taking into account the final developments for JT-60SA such as to provide a high complementary benefit in particular for missions 2 – 5.

Among the medium size devices, ASDEX-Upgrade is most suited for efficient support of ITER and the ITER satellites and it can contribute with a wide range of

[66] These have been addressed before and can be illustrated by an example: the annual operation costs of ITER (~280 M€) give a figure well above 200 K€ per shot while on the same basis for the proposed FAST or ASDEX-UG a shot is in the range of order 20 k€ or 7 k€ respectively and on COMPASS a few hundred Euro. Any savings on exploratory, e.g. low-power shots on ITER by scoping preparation on smaller devices will therefore be highly advantageous.

[67] When comparing time scales it must be taken into account that any major device will need a few years for reaching its expected performance, operational flexibility and diagnostic capability.

[68] Currently a design review of JT-60SA is underway which has not yet been concluded. Even if the capabilities of the device are optimized , a strong case for a complementary facility would still exist.

dedicated studies and exploratory work. Active coils and a LHCD system should be implemented.

The other Tokamaks in this mid-size range, TORE SUPRA, TEXTOR and FTU have substantially contributed to the development so far, but in the longer term they are not expected to contribute at equal level to detailed ITER physics preparatory studies since they have the disadvantage of circular shape and limiter (instead of divertor) and correspondingly there are differences in physics features such as higher edge electron temperatures and their inability to routinely access H-modes. Therefore it is more difficult to operate with tungsten walls as envisaged for ITER during the deuterium-tritium phase (if not earlier). While TORE SUPRA is unique in its long pulse capabilities and actively cooled walls, in some years the new ITER-like shaped superconducting international Tokamaks can be expected to supersede its relevance for ITER if, as can be expected, they will be equipped adequately. Nevertheless, as explained before, these facilities execute currently programmes of ITER relevance or high generic interest and should continue for completing these missions within the coming years.

In complement to ASDEX-Upgrade the smaller versatile TCV and the Spherical Tokamak MAST are able to extend studies into different parameter regimes and are highly valuable for the period under discussion. MAST is also of interest for the possible development of a Components Test Facility (see below under Technology Facilities) and the investigation of the novel concept of a long-leg (flux expansion) divertor.

The Reversed Field Pinch RFX is contributing with results in otherwise inaccessible parameter ranges and there is, for the next years, scope and interest for these studies.

The Stellarator TJ-II, presently of relevance, inter alia, for the Stellarator divertor development, will likely lose its R&D interest for the Stellarator development once W7-X has become operative. Beyond, the device, or another one, could be considered as European regional training and competence attractor undertaking more fundamental fusion physics studies.In this context of confinement schemes varying from the Tokamak and in-depth studies of plasma features through accessing a wider parameter space also the unification of knowledge from the research for W7-X with that of the forum of Tokamaks is highly valuable, in order to propel understanding of key research issues in particular in missions 2, 4, and 5.

In the area of the smaller academia based devices the COMPASS Tokamak with an ITER-like plasma shape will be the most relevant one. The device will enter scientific operation in 2009 and should work, e.g. in conjunction with ASDEX Upgrade, on features for which scaling at identical shape is of interest, such as on ELM control and other plasma wall and exhaust issues. This device, and even more EXTRAP-T2R and ISTTOK, can be operated, by comparison, at very low-cost but they are highly useful as regional facilities for generic studies, diagnostics, control and data acquisition development and for the attraction of young scientists and as training facilities for the European fusion programme.

b. **Preparation for DEMO:** In a perspective up to the completion of the first ten years of ITER operation, JET will likely no longer be the device available or to be recommended for the function of a satellite device. D-T operation may no longer be a distinctive physics advantage (at least in parallel to ITER D-T operation) and adds to complexity and cost of operation. Aging of components will become eventually a problem which must be considered in a balance with the advantages and the excellent equipment of the device. JT-60SA, and, if decided, a European satellite, will have achieved high performance during the period under consideration. The former, according to the present design, should overlap closely with JET in its physics parameter space, while extending the capabilities in particular towards steady-state operation (albeit full performance is expected to be limited to flat-top pulse lengths of ~100s). Also with regard to the DEMO relevance of JT60-SA to the EURATOM fusion programme an extension of European participation in its exploitation should be secured for a sufficiently long time.

   With respect to medium-size devices, during the time under consideration one can expect that the new non-European superconducting devices, in particular EAST, KSTAR and T-15, will have been fully equipped and will provide highly valuable R&D contributions[69] for ITER and DEMO. In Europe also several Tokamaks should be available, among them a device of the ASDEX-UG class from which most benefit (in comparison to cost) should be drawn. In the long run a somewhat higher plasma current than now available on ASDEX-UG would be desirable and studies for enhancing the plasma current capability to the range of 2 MA by upgrading this device or by developing another solution, e.g. through international collaboration, should be undertaken in due time. Depending on results achieved during the next decade, also somewhat smaller devices, complementing the main ITER line such as do now MAST and TCV, would continue to be desirable for the Programme.

   A sufficiently broad base for fundamental fusion physics studies and training, like now existing with COMPASS and other small devices in addition to the major ones, should be considered.

   For the Stellarator line Wendelstein 7-X should have become, with LHD, the leading device world-wide. The assessment of the Stellarator perspective for DEMO and the longer term should be met around the mid 2020’s in order to be able to liaise with the data base developed for the Tokamak.

[69] These devices will have a role similar to TORE SUPRA today for steady-state investigations but in ITER like plasma configurations.

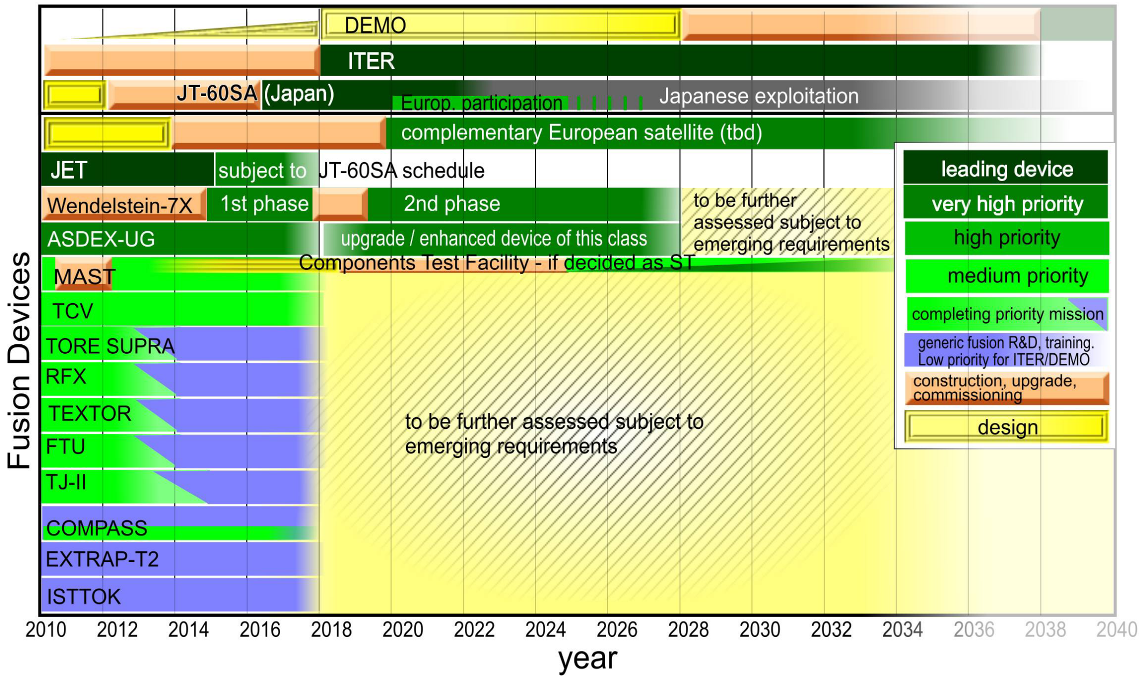

Fig. 7 Time evolution for the relevance of fusion devices with regard to ITER and DEMO. In the upper box DEMO, ITER and JT-60SA are illustrated for reference. The bar for JT-60SA may change subject to further negotiations between EURATOM and Japan. In the lower part the European devices are listed. The priority of JET in the years after ~2014 depends on the schedule of JT-60SA. Included are also proposals, possibly to be developed, for a European satellite (FAST being an example) and a Components Test Facility which both need to be assessed, the former with regard to optimally serving DEMO purposes and complementing JT-60SA, the latter for its usefulness and feasibility along the Spherical Tokamak configuration. The exact points in time for the change in relevance for the devices 4-8 (counted from the bottom) depend on the respective schedules of the ongoing tasks and for reorienting resources towards new priorities.

2. *Technology Facilities*

A substantial range of technology facilities, listed in previous sections, is required for ITER R&D needs and for DEMO oriented R&D. The largest near-term investments are needed for the Neutral Beam Test Facility and the magnets cold testing facilities. The Panel understands that the latter are currently expected to be located, for the toroidal field coil winding packs, at the industrial site in order to undertake the tests before the packs will be embedded in the coil casings. For the larger poloidal field coils a facility shall be constructed at the ITER site. There are some other gaps related to ITER facilities which need to be filled.

The facilities which are indispensable for accomplishing tasks for ITER construction need to be supported as long as these tasks are underway; a typical time horizon is 7-10 years and depends, in part, still on choices to be made by ITER. Thereafter their usefulness for DEMO must be assessed which may result in mothballing for a certain time some of them and abandoning others.

For the preparation of DEMO considerably more gaps exist. The facility of utmost importance and with outstanding investment requirements is IFMIF. Solving the remaining technical issues in the EVEDA frame[70] and preparing siting and construction decisions are now urgent tasks if the fast track time table shall be pursued.

In the DEMO perspective a Components Test Facility could have a significant value for risk minimization in relation to the qualification of the nuclear technology components provided it would become available early enough. The usefulness of such a facility should be assessed. Pending the positive outcome and a feasibility study that confirms the potential advantages of a Spherical Tokamak concept relative to that of a Tokamak with more conventional aspect ratio, an upgraded MAST would be able to validate the feasibility of the Spherical Tokamak concept for such a facility.

Surplus capacity of fusion specific facilities exists in a few areas such as low-power gyrotron testing and neutral beam aspects (once the new Neutral Beam Test Facility will be available). Other surplus capacity relates to non-specific facilities such as non-irradiated materials characterisation which are used on a customer task-by-task basis. A specific area is the neutron irradiation of materials where, before IFMIF is available but also in parallel to it, high fluence fission reactors have to be used for the pre-characterisation of materials. An apparent surplus capacity related to low-fluence fission research reactors is drastically reduced if the requirements for testing fusion relevant instrumentation, in-vessel components (including sensors and cables) are taken into account. For the foreseeable future the irradiation to high doses requires continued access to Russian reactors.

3. Computing Facilities

The Japanese IFERC research and computing centre within the Broader Approach Agreement will provide, in a few years, a substantial computing capability in which Europe can participate for some time. In preparation of the exploitation of this facility and other future high performance computing systems a dedicated European computing and modelling centre should be created for the development of advanced highly parallelised codes, equipped, in the short term with a capacity in the range of 100 Tflop and aiming at progressing rapidly to higher performance with the state-of-the-art in computing power aiming at the realisation of, in particular, a numerical Tokamak (followed by a Stellarator) and an integrated numerical approach to materials development. Investments in this domain are comparatively small but need, in order to be useful, to be accompanied by sufficient professional numerical and computational support.

[70] With the supporting activities mentioned before.

IV. General aspects

A. *Increasing role of Industry*

Entering the ITER construction phase sets a new paradigm for fusion R&D. The work in physics and technology needed in support of this spearheading project is substantial and must be accompanied by efforts for harnessing the know-how from technological developments, components construction and later from ITER operation, and must be transformed into progress towards DEMO. This requires that the pool of scientific knowledge is preserved in the research laboratories by a far-sighted staff policy and that the industrial know-how is maintained by a development schedule which ensures continuity and fostering of industrial involvement in fusion R&D. In anticipation of the completion of ITER components manufacturing and assembly a plan should be developed which meets these requirements in particular by a programme of industrial prototyping of DEMO components in interaction with the relevant fusion laboratories.

When progressing towards DEMO and the subsequent commercial phase of fusion industrial involvement in fusion cannot remain confined to manufacturers of components and systems and providers of services. Industrial entrepreneurship must develop so that industrial consortia will assume for DEMO the role of architect-engineers. Also the future customers, the electrical utilities, have to be involved as soon as possible, and to be associated to the DEMO design in order to ensure a fusion energy development which will meet the market requirements as close as possible. This task should be actively pursued by the proposed DEMO steering group.

Eventually industry will have, in its own commercial interest, to take over all the projecting, design and construction work for the first generation of commercial power plants. Based on results from DEMO, industry would then interact with research institutes as a customer, rather than a builder and service provider, regarding further R&D on critical issues and concept improvements of systems and components.

B. *Project work, innovation and scientific excellence*

The Panel considers that the structure of the EURATOM fusion programme has been well adapted to the new challenges. The project-oriented efforts for dedicated ITER construction needs (and in the medium-term future increasingly for DEMO tasks) are managed by the F4E Joint Undertaking. The broader scientific and technological R&D for ITER and DEMO in the associated laboratories including the joint exploitation of JET is undertaken under EFDA by pan-European (and international) task forces and topical groups[71]. EFDA, in collaboration with the Associated Laboratories, has an important role within the core project activities, particularly in the problem definition and in determining the direction of research. The aim of further enhancing the trans-national and international character of scientific teams on fusion devices and technology facilities should be strongly pursued.

The substantial demand for R&D from ITER / F4E has been described in this report both with regard to supporting R&D for European construction tasks and to physics R&D for resolving remaining critical issues of components to be installed at a later

[71] Also on ASDEX-Upgrade, RFX and other facilities the programmes for forthcoming periods of exploitation are set up, by international calls for proposals with the criterion of optimally serving the established R&D objectives. Subsequently they are executed by corresponding international task forces.

point in time (first wall, antennae, diagnostics …) and/or in preparation of ITER operation. Most of these tasks will attract interest due to their programmatic and scientific relevance. However there are tasks which are of lesser academic scientific merit such as testing of manufactured components, database work or, e.g., also certain investigations of off-normal events under ITER relevant conditions etc. Where possible, such tasks should be linked to a wider context of scientific interest. This is essential for the involved research laboratories since they are integrated in a national research environment where the necessary (dominant) national financial support share usually depends on high scores in scientific excellence.

Sufficient breadth for complementary bottom-up incentives and generic research is of high importance for a long-term endeavour such as the fusion programme. Innovations are essential and will in part be generated in conjunction with research in (and take up from) adjacent fields. A broader perspective also can contribute to a deeper fundamental scientific understanding and to a faster progress. Without allowing the project-oriented core activities to connect to a range of complementary "autonomous" R&D there could be a significant risk of losing best performing scientific staff which frequently, while driving core activities, is linking this project work to fundamental scientific questions and thereby adds significantly to the scientific quality of fusion R&D and its long-term prospects.

C. Staffing

Fusion R&D ranges from the physics and technology of advanced superconducting materials to high temperature plasma physics, the development of high frequency power systems to novel structural materials with outstanding characteristics, from new diagnostic systems to advanced numerical code development. Over the past decade on average a total of about 2000 professionals have been active in the EURATOM fusion programme, with roughly equal numbers of physicists and engineers. Broken down into individual disciplines the staff numbers become fairly small and are an important limiting factor for progress and breadth of R&D. This tight staffing basis must be managed carefully when major changes in emphasis occur such as is the case now with the start of ITER construction and a refocusing of activities with a progressive shift of weight towards fusion technology.

EFDA is expecting that the overall programme can be managed with a rather flat integral staff profile over the coming two decades, showing an increase of only 15-20% over the coming years. This is based on EFDA's assumption that about half the staff currently engaged in the operation and exploitation of the Association's fusion facilities will become soon available for covering the new tasks which must be achieved in the coming decade.

The Panel considers that the staffing basis anticipated by EFDA is tight for mastering the identified R&D needs and should be expanded when possible for fostering the desired rapid progress towards DEMO and the ultimate goal of the programme. An efficient managing of the staff resources by redirecting competence and workforce to the new projects of high relevance will be essential and will require clear choices. In particular, if the design (and, if decided, subsequently the construction) of a new satellite device shall be feasible, the adaptation of the exploitation schedules of devices with decreasing priority must be realized as assumed in the staff planning of EFDA. A careful steering must ensure that existing highly skilled staff is not lost but motivated towards engaging in the new priorities.

With respect to qualification a critical gap exists notably for experienced engineers. The European involvement in ITER construction will require in the Associations and industry a build-up to about 600 engineers from a present capacity of ~240. The timely construction of IFMIF (assuming a strong European participation) will add to this demand as will the design of a satellite device (and initially to a small extent a DEMO steering group). With the accomplishment of these design and construction tasks staff resources need, in about a decade from now, to be oriented towards the starting DEMO design and prototyping work with a rising demand towards the anticipated construction. Progressively, this staff build-up will have to take place in industries involved in the preparation of the DEMO construction.

Another area where long lead-times require a timely increase of efforts is R&D on structural and functional materials. This work, together with realising IFMIF for the materials validation, is of crucial relevance for establishing the capability of constructing DEMO. Currently about 50 professionals are engaged in this area – this effort should be about tripled within the coming five to eight years.

On the operational side, several years before commissioning of ITER the teams for exploitation must be prepared and staff for the operation be trained. In the case of JET a clearly defined and proven qualification scheme is in place. Senior experimental plasma physicists from fusion laboratories undergo a training of typically three years' duration before they are entrusted as session leaders with the responsibility for the experimental operations. For ITER a similar training programme must be conceived in anticipation of operation, starting on JET and continuing later on future satellites.

With theory, modelling and computational code work aiming for the DEMO design at a comprehensive capability for a "numerical burning plasma device" and for materials modelling, there must be a significant strengthening of the professional staff resources in line with the build-up of the high performance computing capacity. As mentioned before, synergy by linking high quality staff from different groups into European task forces should substantially contribute to enhancing the efficiency in this area.

Despite the attractive scope of physics and technology involved in fusion R&D the inflow of high-quality new young staff to fusion laboratories has been lower than desired. Tight links to academic research in related fields are essential for accessing a sufficient education and training potential. The decentralized structure of the EURATOM programme with associated laboratories in all regions of the European Union and Switzerland and involvement of staff from many of these laboratories in the major R&D projects is highly useful for solving this problem. **The Panel welcomes the dedicated educational schemes for fusion physicists and engineers with specific curricula and trans-European educational agreements which have been recently initiated. They should be further developed** and facilities must be made, and kept, available for their support[72].

An essential tool for fostering networking among the laboratories is the "mobility scheme"[73] providing support to travel and subsistence for working stays of researchers who participate in joint tasks. This scheme is fundamental for enabling the exploitation of devices by European task forces, for knowledge transfer and for maintaining and improving the cohesion of the fusion programme. It will have a

[72] The highly valuable role of cost-efficient fusion devices as well as and technology laboratories for the regional attracting and training of young staff for the European fusion programme has been highlighted before. It should be noted that experience shows also for theoretical groups that they strongly benefit from close contact to the experimental activities.

[73] Agreement for the Promotion of Staff Mobility in the Field of Controlled Thermonuclear Fusion

crucial impact on the effectiveness of deploying the tight staffing resources towards the identified priorities of the programme. **The Panel strongly recommends to make the best use of the mobility scheme and to further expand its use.**

D. Finances

A very substantial investment in facilities and personnel has been made over the recent past which should be optimally used for the identified priorities of the Programme. It should be secured and its value be maintained by the recommended upgrades. Where facilities are reoriented, efforts should be made to saveguard this investment by using these resources, to the extent possible, on other devices.

The Panel has become very positively impressed by the aim-oriented long-term partnership in the European fusion programme with its mechanism of co-funding by EURATOM and national partners. The institutional involvement of national partners in European projects outside their own country (which is particularly demonstrated with JET but also by the European participation in other facilities such as W7-X or COMPASS) should be further developed.

In this context the Panel notes that the relative financial involvement of EURATOM in JET is the most significant one. 75% of operation costs are paid from Community funds. These are about 70-80 M€/y and would represent during the coming years, excluding ITER, about 60% of all funds dedicated to the support of fusion devices (including the proposed upgrades). Hence the savings to EURATOM from reducing the base of these other fusion devices are rather modest. Nevertheless the Panel strongly supports the continued JET exploitation as described earlier due to its programmatic importance. However, the Panel would welcome an increased involvement by international and national partners like in ITER and the Broader Approach projects (albeit the latter was triggered by the particular circumstances of the ITER negotiations): economies might be achieved for JET. Internationalisation will be essential for sharing the costs of IFMIF and should also be sought for a European Satellite and/or a Components Test Facility, if their construction would be decided. Other collaborations e.g. on the new international medium-size superconducting Tokamaks and on technology facilities are highly recommended by the Panel for their scientific-programmatic value but it is not expected that with such partnerships significant financial savings could be realized.

Being requested to provide a vision on how *to support the rapid and efficient development of fusion as an energy source,* the Panel's recommendations on priorities in this report cover the minimum actions necessary for maintaining the impetus of the programme. The Panel was informed that the costs of the fusion programme to EURATOM are currently around 180 M€ per year. This includes support to the programme in the Associations (~60 M€/y), JET (~70-80 M€/y), staff, mobility, training, fellowship actions, EURATOM staff salaries and administrative expenditure but excludes support to ITER R&D which is funded through F4E. The Panel is concerned that it may not be possible to fund all of the ITER and DEMO priority actions unless the ceiling would be adjusted or the national partners take over additional financial responsibility[74]. It is understood that R&D by Associated

[74] While this discussion is beyond the remit of the Panel's charter it may be noted that the financial contribution by EURATOM to the general activities and investment in the Associations is the essential lever for the coordination of this important part of the programme and for focussing it towards the key priorities. EURATOM support, presently at rates up to 20% or 40% (for priority investment) should not be marginalized but remain attractive incentives.

Laboratories required for ITER should, where justified, be adequately supported by F4E or the international ITER organisation.

Linked to the core of the programme some further actions have been highlighted in this report which would be highly desirable for accelerating the future developments[75] and reducing the risks. Besides a more comprehensive preparation for ITER implying in particular an extension of JET operation beyond 2014 for a few further years (requiring a few tens of M€ further investment in the period 2011 – 2014 and the operational costs for the additional years), these include in particular a satellite device as a complement to JT-60SA and, in the DEMO perspective, if feasible, the construction of a Components Test Facility. For a satellite device an indicative cost estimate of about 300 M€ has been provided for the example of the FAST proposal which aims at benefiting substantially from existing infrastructure. No established cost estimate exists for a Components Test Facility.

E. *Further Assessment*

This report examines the R&D needs and required facilities based on the present state of the art. Some questions could not be fully addressed due to the tentative nature of the information available. These relate in particular to the proposal for a European satellite. The Panel, noting the FAST proposal and pointing to the importance of accompanying ITER with a strong satellite base, recommends to develop a consistent proposal supported by specific modelling and design work, which could then be assessed by a scientific-technical ad-hoc group in the light of the knowledge then available on the capabilities of JT-60SA, its ability to contribute to the European fusion programme and the need for a complement. For the proposal of a Components Test Facility the usefulness relating to the fast track time schedule and for complementing IFMIF, as well as the concept to be chosen and the feasibility of construction and licensing need to be assessed.

The Panel is convinced that the implementation of the proposed vision on R&D needs and the adaptation and upgrading of the facility basis can be expected to yield substantial benefits for the programme's ability to support ITER and prepare DEMO on the time horizons considered. The upgrades which are proposed for the fusion devices and technology facilities have been taken note of and the major ones which the Panel considers of particular programmatic relevance are addressed in the context of the respective facilities in this report. The Panel, however, did not assess all proposals for upgrades nor their technical quality, expecting that this should best be done by dedicated scientific-technical expert groups e.g. under the auspices of EFDA.

Before the ITER construction project is approaching completion, the European Fusion Programme and the further fusion development should be assessed again based on the progress then achieved. This should include again a review for adapting the facility basis taking into account the possibilities for international collaboration and developing a vision on how to optimize the steps for the engineering design and the preparation of construction of a DEMO with a view to secure, and further accelerate, progress towards the ultimate goal of the European fusion programme to realize fusion as an energy source.

[75] Ought a strategy of further acceleration to be followed with the construction of an early DEMO facility, overlapping with ITER and before DEMO, then a much higher funding level would be required, the cost of this device being in the ball park of ITER or beyond. As mentioned before, this strategy has not been assessed.

Annex I: Terms of Reference given to the Panel

The EURATOM FP7 Specific Programme stipulates that a review of the facilities in the fusion programme should be carried out. The motivations for this review are to support the rapid and efficient development of fusion as an energy source, and to maintain in the programme the facilities needed to fulfil its medium and long term objectives. The Review will take place in two stages, the first to create a vision of the R&D required to bring about the realisation of fusion energy, and the second to identify the facilities[76] needed in Europe to carry out this R&D. The Commission will use the outcome of the review as a basis for future decision making.

Terms of Reference Stage 1:

The aim of the first stage is to develop a vision of the R&D required to make fusion energy production ready for commercial exploitation.

The R&D requirements should be seen in the context of three phases: ITER construction and commissioning; the early years of ITER operation; the further stages of the demonstration of fusion energy.

Developments which can maximise the success of ITER and accelerate the development of fusion should be given particular attention.

The Panel should seek input from EFDA on the priorities and activities of the European fusion programme, including activities to be potentially carried out in international cooperation; and conduct interviews with other relevant stakeholders such as industry and representatives of the Joint Undertaking.

Terms of Reference Stage 2:

The aim of the second stage is to define the facilities needed to support the envisioned R&D.

The Panel should review all significant facilities in the programme, existing or under construction, including proposed or considered upgrades and taking account of EU international commitments such as ITER and projects within the Broader Approach Agreement with Japan.

Important areas of research that cannot be adequately addressed by existing facilities or those under construction should be identified, and the types of facilities which could fill the gaps should be indicated.

The required facilities should be incorporated into a "facility development" road map, and prioritised according to the corresponding benefits, costs and risks.

Non priority Facilities should be identified but no recommendations for closure are to be made.

The Panel should seek the views of: EFDA on the use of JET and the integration of facilities into a coordinated programme; the Associations (either individually or preferably through clusters) on the role of their existing facilities, possible upgrades and new devices; and other stakeholders.

The Panel will be assisted by an external consultant carrying out a SWOT (strengths, weaknesses, opportunities, threats) analysis of the various facilities.

[76] For the purpose of this review, "facility" is taken to mean any device or installation, including high performance computers, built and operated for the purpose of fusion R&D, and funded through the fusion programme.

Annex II: Legal basis and the 7th Framework Programme (EURATOM)

The legal basis of the EURATOM Fusion Programme and its objectives are defined in the Council Decision for the EURATOM Framework Programmes[77].

The ITER project was agreed in an international framework in 2007[78]. The R&D requirements by ITER and the corresponding R&D plans established by F4E were presented to the Panel. They are considered in general "as given" in the context of this assessment.

EURATOM is also committed to the "Broader Approach" agreement with Japan[79] which comprises (i) the Engineering Design and Validation Activities for the International Fusion Materials Irradiation Facility (IFMIF), currently ongoing, (ii) the European participation in the design and construction of JT-60SA which shall serve as a satellite to ITER and whose R&D potential should exceed that of JET. European participation in the exploitation is currently agreed for the first five years after commissioning. Finally the agreement comprises (iii) an International Fusion Energy Research Center (IFERC) with a high performance computing centre and a DEMO design group. While considering also these facilities and activities under the "Broader Approach" as "given", the Panel discusses their scope, limitations and, for JT-60SA, aspects of enhancing the return-of-investment with regard to the European participation.

The 7th EURATOM Framework Programme Decision covers the period 2007-2011 while the corresponding EC Framework Programme covers the period 2007-2013. The Panel was informed that current planning foresees an interim two-year programme decision (2012-2013) which will have to be taken around 2011 and a further five-year decision for 2013-2018. The near-term visions on the research needs of the fusion programme which the Panel developed are of relevance to these programme decisions but the overall views extend well beyond.

Council Decision for the 7th Framework Programme

The Council Decision for the 7th EURATOM Framework Programme[80] defines for the Fusion R&D programme the overall objective "of developing the technology for a safe, sustainable, environmentally responsible and economically viable energy source". In the Annex of this Decision it is stated: "The long-term goal of European fusion research, embracing all the fusion activities in the Member States and associated third countries, is the joint creation, in approximately 30 or 35 years and subject to technological and scientific progress, of prototype reactors for power stations which meet these requirements, and are economically viable." This definition is based on similar ones of the previous EURATOM Framework Programmes.

For the period of the 7th Framework Programme the objective shall be: "Developing the knowledge base for, and realising ITER as the major step towards, the creation of prototype

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[77] COUNCIL DECISION of 18 December 2006 concerning the Seventh Framework Programme of the European Atomic Energy Community (EURATOM) for nuclear research and training activities (2007 to 2011) (2006/970/EURATOM) (Official Journal of the European Union L 54/21(en), 22.2.2007)

[78] see: COM(2006)240 final, 19.5.2006

[79] BA-Agreement reference

[80] COUNCIL DECISION of 18 December 2006 concerning the Seventh Framework Programme of the European Atomic Energy Community (EURATOM) for nuclear research and training activities (2007 to 2011) (2006/970/EURATOM) (Official Journal of the European Union L 54/21(en), 22.2.2007)

reactors for power stations which are safe, sustainable, environmentally responsible, and economically viable”.

The Council Decision also specifies the necessary steps to meet the overall objective: ”The strategy to achieve the long-term goal entails, as its first priority, the construction of ITER (a major experimental facility which will demonstrate the scientific and technical feasibility of fusion power), followed by the construction of DEMO, a ‘demonstration’ fusion power station. This will be accompanied by a dynamic programme of supporting R&D for ITER and for the developments in fusion materials, technologies and physics required for DEMO. This would involve European industry, the fusion associations and third countries, in particular parties to the ITER Agreement.”

Finally, the activities to be undertaken in the Programme are laid down:

“1. The realisation of ITER: This includes activities for the joint realisation of ITER (as an international research infrastructure), in particular for site preparation, establishing the ITER Organisation and the European Joint Undertaking for ITER, management and staffing, general technical and administrative support, construction of equipment and installations and support for the project during construction.

2. R&D in preparation of ITER operation: A focused physics and technology programme will exploit the relevant facilities and resources in the fusion programme, i.e. JET and other magnetic confinement devices, existing, future or those under construction (Tokamaks, Stellarators, RFPs). It will assess specific key ITER technologies, consolidate ITER project choices, and prepare for ITER operation through experimental and theoretical activities.

3. Technology activities in preparation of DEMO: This entails the vigorous development of fusion materials and key technologies for fusion, including blankets, and the establishment of a dedicated project team to prepare for the construction of the International Fusion Materials Irradiation Facility (IFMIF) to qualify materials for DEMO. It will include irradiation testing and modelling of materials, studies of the DEMO conceptual design, and studies of the safety, environmental and socio-economic aspects of fusion energy.

4. R&D activities for the longer term: The activities will include further development of improved concepts for magnetic confinement schemes with potential advantages for fusion power stations (focussed on the completion of the construction of the W7-X Stellarator device), theory and modelling aimed at a comprehensive understanding of the behaviour of fusion plasmas and coordination, in the context of a keep-in-touch activity, of Member States' civil research activities on inertial confinement.

5. Human resources, education and training: In view of the immediate and medium term needs of ITER, and for the further development of fusion, initiatives aimed at ensuring that adequate human resources will be available, in terms of numbers, range of skills and high-level training and experience will be pursued, in particular in relation to the physics and engineering of fusion.

6. Infrastructures: The construction of the international fusion energy research project ITER will be an element of the new research infrastructures with a strong European dimension.

7. Technology transfer processes: ITER will require new and more flexible organisational structures to enable the process of innovation and technological progress which it creates to be swiftly transferred to industry, so that the challenges can be met to enable European industry to become more competitive.”

Annex III: Input received by the Panel

Input Documents:

- Terms of Reference (CCE-FU 38/8b rev 1, Nov. 2007)
- Council Decision of 18 December 2006 Concerning the Seventh Framework Programme of the European Atomic Energy Community (EURATOM) for nuclear research and training activities (2007 to 2011) (OJ L54/21, 22.2.2007)
- Council Decision of 19 December 2006 concerning the specific programme implementing the Seventh Framework Programme of the European Atomic Energy Community (EURATOM) for nuclear research and training activities (2007 to 2011) (OJ L54/139, 22.2.2007)
- The European Fusion Research Programme: Positioning, Strategic Outlook and Need for Infrastructure towards DEMO
- Part I: Positioning and Strategic Outlook (EFDA, 15 January 2008
- Part II: Facilities (EFDA, 6 May 2008)
- D.V. Bartlett: The European Fusion Programme (Commission Services, DG RTD J6, 19. September 2007)
- Answers to the first set of questions by the Panel (J. Paméla, EFDA, 19 May 2008)
- Answers to the first set of questions by the Panel (D. Gambier, F4E, 16 May 2008)
- Elements of Answers to Facilities Review on Question IV.7 (F. Romanelli, EFDA - JET, 19 May 2008)
- Answers to the second set of questions by the Panel (coord. J. Paméla, EFDA, 10 July 2008)

Presentations and hand-outs prior to, or at first Meeting (26/27 February 2008)

- General introduction to the Review (Y. Capouet, DG RTD J6)
- Terms of Reference of the Panel (Y. Capouet, DG RTD, J6)
- Status and Perspectives of the European Fusion Programme (C. Llewellyn Smith, Chair CCE-FU)
- EFDA (J. Paméla, EFDA)
- Status of Fusion for Energy and Future Perspectives (D. Gambier, F4E)
- ITER Research Needs in Physics and Technology (D. J. Campbell, ITER IO)
- Facility Needs and update JET fiche (J. Paméla, EFDA)
- Stage 1: Positioning and Strategic Outlook – Introduction and Overview (J. Paméla, EFDA)
- Status of Fusion Research – Technology (M. Gasparotto, F4E)
- Seven R&D Missions for the European Fusion Programme (H. Zohm, IPP Garching)
- Roadmap to DEMO and Gap Analysis (D. Stork, UKAEA)
- Status of Fusion Research – Physics (A. Fasoli, CRPP)
- Human Resources (N. Lopes Cardozo, FOM)

Presentations and hand-outs prior to, or at second Meeting (26/27 May 2008)

- Stage 2: Facilities – Introduction (J. Paméla, EFDA)

- The ITER Programme (P. Thomas, F4E)
- JT-60SA (P. Barabaschi, F4E)
- JET: Summary of Key Issues (F. Romanelli, EFDA-JET)
- FAST – a proposal for a facility in support of the development of fusion energy (A. Pizzuto on behalf of the Italian Association on Fusion)
- MAST upgrade for fusion development (A. Morris, UKAEA)
- Strategic importance of long duration discharge studies and the role of Tore Supra (A. Bécoulet, CEA)
- Facilities of the Swiss Association – Missions and Perspectives (A. Fasoli, CRPP)
- The Future Role of ASDEX-Upgrade in the Preparation of ITER and DEMO (H. Zohm, IPP Garching)
- Fusion Technology Facilities mainly related to ITER (M. Gasparotto, F4E)
- Fusion Technology Facilities mainly related to DEMO (M. Gasparotto, F4E)
- National Centre for Fusion Technologies (A. Ibarra, CIEMAT)
- Facilities' Perspectives of the Association EURATOM-FZK (M. Thumm, FZK)

Presentations and hand-outs prior to, or at third Meeting (16/17 July 2008)

- Answers to questions on resources (J. Paméla, EFDA)
- The future role of Wendelstein 7X in preparation of ITER and DEMO (R. Wolf, IPP Greifswald)
- The TJ-II Stellarator Programme (J. Sánchez, CIEMAT)

Documents received after the third meeting:

- Answers on resources (revised) (J. Paméla EFDA)
- Possible savings on the ITER exploitation arising from a JET programme beyond 2014 (F. Romanelli, JET)
- Technofusion:_strategic_role (J. Sanchez, CIEMAT)
- Comments to the Note on the strategic role of the Techno Fusion Facility (M. Gasparotto, F4E)

Presentations and hand-outs at visits to fusion laboratories

- Visit JET / MAST (UKAEA) (14 April 2008, Culham/Abingdon)
- Welcome and Introduction to MAST (C. Llewllyn-Smith, UKAEA)
- Introduction to JET-EFDA (F. Romanelli, EFDA - JET)
- Visit ASDEX-Upgrade (14 May 2008, IPP Garching)
- Welcome and Introduction to IPP (A. Bradshaw, IPP)
- Plasma theory and high performance computing (S. Günter, IPP Garching)
- The Future Role of ASDEX-Upgrade in the Preparation of ITER and DEMO (H. Zohm, IPP Garching)
- Medium Size Technology Facilities at IPP Garching (J. Roth, IPP Garching)
- Visit W7-X (17 June 2008, IPP Greifswald)
- Stellarator physics – what is different? (P. Helander, IPP Greifswald)
- The physics of Wendelstein 7-X and prospects for a stellarator reactor (R. Wolf, IPP Greifswald)
- Steady-state ECRH (H. Laqua, IPP Greifswald)
- Status of the Wendelstein 7-X project (Th. Klinger, IPP Greifswald)

- Visit TORE SUPRA (30 June 2008, CEA Cadarache)
- Tore Supra – a unique facility for the progress of Long Pulse Operation Steady-state Tokamaks (M. Chatelier, CEA)
- Plasma Facing Components: an integrated challenge (A. Grosman, CEA)
- The LHD Enhancement CIMES Project (F. Kazarian, CEA)
- Articulated Inspection Arm for In-Vessel Tokamak Operation (N.N., CEA)
- Tore Supra programmatic aspects: short and medium term perspectives (X. Litaudon, CEA)
- Modelling: from first principles to real-time control (X. Garbet, CEA)
- Visit FTU (1. July 2008 ENEA Frascati)
- The EURATOM-ENEA Association (A. Pizzuto, ENEA Frascati)
- ENEA Fusion Research Activities (A. Pizzuto, ENEA Frascati)
- The Padova Research Unit – Consorzio RFX (F. Gnesotto, Consorzio RFX)

Annex IV: Panel membership

Prof. Roger CASHMORE	Principal of Brasenose College, Oxford and Professor of Experimental Physics in the University of Oxford, United Kingdom
Dr. Jean-Michel DELBECQ	Vice President, Future Nuclear Systems Program Manager, EDF R&D, Clamart, France
Dr. Victor ELSENDOORN	Director Operations Science and Industry, TNO, Delft, The Netherlands
Prof. Dr.-Ing. Thomas HARTKOPF (Chairman)	Professor for Regenerative Energies at the Technical University of Darmstadt, Germany
Prof. Enzo IAROCCI	Professor of Physics at the University La Sapienza of Rome, Italy
Dr. Kimitaka ITOH	National Institute for Fusion Science, Toki, Japan
Dr. Jiangang LI	Director, Institute of Plasma Physics of the Chinese Academy of Sciences, Hefei, China
Prof. Ron PARKER	Professor of Nuclear Science & Engineering, and Electrical Engineering & Computer Science, MIT, Cambridge (MA), USA
Dr. Valentin P. SMIRNOV	Director of the Institute of Nuclear Fusion INF, Moscow, Russian Federation

Scientific Secretary:
Prof. Dr. Hardo BRUHNS — Düsseldorf, Germany

Annex V: Glossary[81]

A

Additional heating: Usually with reference to a plasma which is initially heated by a toroidal current induced in the plasma (Ohmic heating), additional heating designates other means of heating a plasma (absorption of electromagnetic waves or of injected fast neutral particles).

Advanced Tokamak Scenarios: Tokamaks normally generate natural profiles of plasma current and plasma pressure. Using external non-inductive current drive and local control of the current and pressure profiles can allow access to enhanced regimes and even steady-state operation, generally referred to as Advanced Tokamak Scenarios.

ALCATOR C-MOD: High field, high density Tokamak at MIT (USA) with elongated, diverted plasma.

Alfvén waves: A fundamental plasma wave, which is primarily magneto-hydrodynamic in character with an oscillation of the magnetic field and, in some cases, plasma pressure. In Tokamaks, these waves are typically strongly damped. See also fast Alfvén wave.

Alfvén velocity: The velocity of propagation of Alfvén waves in the direction of the magnetic field; it is proportional to the magnetic field strength, and inversely proportional to the square root of the mass density.

alpha particle, or α-particle, He^4: The nucleus of the helium atom, composed of two protons and two neutrons, is one of the two products of the D-T fusion reaction (the other one is a neutron). The α-particles, being electrically charged, are trapped by the magnetic confinement field and therefore can release their energy to the plasma contrary to the neutrons which escape from the plasma and transfer their energy in the blanket surrounding the plasma core. The plasma heating which is provided by these α-particles as they slow down due to collisions is essential for achieving ignition.

Alternative lines: Magnetic confinement development other than the Tokamak.

Anomalous transport: Measured heat and particle loss is anomalously large compared with collisional theory of heat transport in toroidal plasmas.

ASDEX-Upgrade (ASDEX-UG): Medium-sized Tokamak at Garching (Association EURATOM-IPP, Germany) with elongated, diverted plasma and full coverage of the first wall by tungsten.

Aspect ratio: The ratio between the large radius and the small radius of a torus.

Auxiliary heating: See additional heating.

B

BA Agreement: See Broader Approach Agreement

Ballooning instability: A local instability which can develop in the Tokamak when the plasma pressure exceeds a critical value; it therefore constrains the maximum β that can be achieved. It is analogous to the unstable bulge which develops on an over-inflated tyre.

Bernstein waves: see: ion Bernstein waves.

Beta (β): Ratio of plasma pressure to magnetic field pressure. One of the figures of merit for magnetic confinement: the magnitude of the magnetic field pressure determine the cost of the field coil that generates it; since fusion reactivity increases with the square of the plasma pressure, a high value of ß indicates good performance. The highest values achieved in Tokamaks reach 40% (START).

Beta-normalised (β_N): The ratio of plasma current (in MA) to the product of minor radius (in m) and magnetic field (in T) characterises the limit to the achievable β imposed by ideal MHD. Beta-normalised is the ratio of β (as a percentage) to the above ideal MHD parameter. Generally β_N ~3 should be achievable, but techniques for obtaining higher values have been observed experimentally.

Blanket: Structure containing lithium or lithium compounds surrounding the plasma core of a fusion reactor. Its functions are to breed tritium, via lithium-neutron reactions, and to absorb most of the fusion energy to be used for electricity generation.

Bootstrap current: Theory predicted in the 1970's that a toroidal electric current will flow in a Tokamak which is fuelled by energy and particle sources that replace diffusive losses. This diffusion driven "Bootstrap current", which is proportional to β and flows even in the absence of an applied voltage, could be used to provide the poloidal magnetic field: hence the concept of a Bootstrap Tokamak, which has no toroidal voltage. A Bootstrap current consistent with theory was observed many years later on JET and TFTR; it now plays a role in optimising advanced Tokamaks.

Break-even: The fusion performance of a power plant is denoted by Q, which is the ratio of the power released by fusion reactions to that used to heat the plasma. As a convention, scientific breakeven corresponds to Q=1 and ignition to Q=infinity. A fusion power plant would operate at Q≤50.

[81] Original version EUR 17521 European Commission (1996), updated by CRPP and UKAEA. Further adapted to the needs of this report. Courtesy of CRPP and UKAEA ("Glossary of fusion terms" (http://www.fusion.org.uk/info/glossary.htm)

Breeding ratio: The number of tritium atoms produced in the blanket of a fusion power station per tritium nucleus burned in the fusion plasma. In order to replace the burnt tritium and to account for the initial loading of new power plants, the breeding ratio should be somewhat larger than unity.

Broader Approach : The Broader Approach agreement was signed in February 2007 by the EU and the Japanese Government. This cooperation means to complement the ITER project and to accelerate the realisation of fusion energy by carrying our R&D and developing some advanced technologies for the future demonstration fusion power reactor (DEMO).

Burn: The fusion process of consuming D-T fuel in a reactor, releasing energy.

C

CCE-FU: The Consultative Committee for the EURATOM Specific Research and Training Programme in the field of Nuclear Energy, Fusion. Formerly the CCFP.

CEA: Commissariat à l'Energie Atomique, France. Partner in the Association EURATOM-CEA which operates the TORE SUPRA Tokamak.

Centre column: In a Tokamak the component with the primary winding of the transformer which generates the toroidal plasma current and the inner legs of the toroidal field coils.

Charge exchange measurement: Measures the plasma ion temperature. Neutral atoms in the plasma (for example from a neutral beam) donate electrons to hot plasma ions, which are thereby neutralised. These hot atoms are no longer confined by the magnetic field and leave the plasma. Their energy is measured by a neutral particle analyser.

CIEMAT: Centro de Investigaciones Energéticas Medioambientales y Tecnológicas, Spain. Partner in the Association EURATOM-CIEMAT. Operates the flexible heliac Stellarator TJ-II.

Classical transport: Collisions between the individual particles of a plasma allow them to move across the magnetic field. Theories which describe this mechanism are called "classical" (or "neo-classical" when additional effects due to the toroidal geometry are included). The measured heat and particle transport is usually higher than predicted by these theories.

Collisionality: Non-dimensional parameter, which is the inverse ratio of the mean free path of plasma particles between collisions to a characteristic length of the magnetic field configuration.

Compact torus: Class of closed magnetic configurations in which no material elements (coils, conductors or walls) need to link through the bore of the plasma torus. Thus the vessel of compact tori can be spherical or cylindrical. Not to be confused with a "Spherical Tokamak".

COMPASS: COMPact ASSembly, a Tokamak for studies of plasma stability, error fields, at Prague, Czech Republic (Association EURATOM-IPP-CR, formerly Association EURATOM-UKAEA Culham, UK). Originally with circular vessel (COMPASS-C), now with ITER-like plasma shape in a D-shaped vessel (COMPASS-D).

Confinement time: In a fusion plasma neither particles nor energy are perfectly confined. Particle confinement time is the time during which the particles, on average, stay confined. The energy confinement time, which is usually shorter than the particle confinement time, is defined in steady-state as the ratio of the plasma energy content to the total power input to the plasma and is a measure of how fast a plasma would cool if there were no heating.

CRPP: Centre de Recherches en Physique des Plasmas. Fusion laboratories of the Association EURATOM-Swiss Confederation at the Ecole Polytechnique Fédérale de Lausanne and the Paul-Scherrer Institute, Villigen (CRPP-Fusion Technology).

Cryopumps: Used in fusion experiments to absorb the gaseous constituents on (helium-) cooled panels in the vacuum chamber for controlling the density of the plasma and removing impurities.

CU : Comenius University, Slovak Republic. Partner in the Association Euratom – CU.

Current drive (non-inductive): In a Tokamak, plasma current can be driven inductively, with the toroidal plasma acting as a secondary winding of a transformer whose primary coil is at the central column of the device. Continuous current cannot be driven by transformer action. 'Non-inductive' current drive methods are applied either by injecting particles with directed momentum into the plasma or by accelerating electrons by electromagnetic waves so that they carry the current. Also being applied to control instabilities and to optimise confinement by influencing the current profile. The bootstrap effect also drives current.

Current profile (current distribution): The distribution of current density across the minor radius of the plasma.

Current ramp-up (down): The increase (decrease) of plasma current either at the start of operation or during operation.

Cyclotron frequency: Charged particles in a magnetic field have a natural frequency of gyration in the plane perpendicular to the magnetic field - the cyclotron frequency. For electrons in a Tokamak, the cyclotron frequency is typically a few tens of GHz (28 GHz per Tesla), and for ions, a few tens of MHz (7.5 MHz per Tesla for deuterium).

D

DCU: Dublin City University, Ireland. Partner in the Association EURATOM-DCU.

DEMO: Demonstration Reactor (the first device in the European fusion strategy intended to produce significant amounts of electricity) shall be, after ITER, the essential development step towards the commercial regime of fusion power.

Deuterium: A stable isotope of hydrogen, whose nucleus contains one proton and one neutron. In heavy water, normal hydrogen is replaced by deuterium. Sea water contains, on average, 34g deuterium per m^3. Deuterium plasmas are used routinely in present-day experiments; in a fusion power plant the plasma will consist of a mixture of deuterium and tritium which fuse more readily than two deuterium nuclei thereby delivering about 200 times more fusion power under otherwise similar conditions.

DG Research (DG RTD): The Directorate-General of the European Commission, Brussels, responsible for Research and Development**.** Operates the Euratom Fusion Programme.

Diagnostic: Apparatus used for measuring one or more plasma quantities (temperature, density, current, etc.).

Diffusion, thermal (or particle): The random flow of heat (or particles) in the presence of a thermal (or density) gradient.

DIII-D: The largest operating US Tokamak, run by General Atomics, San Diego. It has a flexible configuration and studies core and divertor physics with intense additional heating.

Disruption, Disruptive instability: A complex phenomenon involving MHD instability which results in a rapid release of energy to the wall and strong electromechanical forces in a Tokamak. Plasma control may be lost, triggering a VDE (q.v.). This phenomenon places a limit on the maximum density, pressure and current in a Tokamak.

Distribution function: Describes both the space and velocity distribution of plasma particles.

Divertor: A magnetic field configuration with a separatrix, affecting the edge of the confinement region, designed to remove heat and particles from the plasma, i.e. divert impurities and helium ash to divertor plates in a target chamber. Alternative to using a limiter to define the plasma edge.

Double null: See Single/double null divertors.

Drift kinetic theory: Kinetic theory which describes plasma processes which have spatial scales much greater than the particle Larmor radii.

Drift orbits: Particle motion is tied to straight magnetic field lines. However, electric fields and gradients of the magnetic field give an additional drift perpendicular to the magnetic field creating drift surfaces displaced from the magnetic surfaces.

Driven current: Plasma current produced by a means external to the plasma, inductively or non-inductively.

Driver: In inertial confinement fusion, the laser or particle beam system used to compress a target pellet.

D-T operation: Operation of a fusion device with the deuterium-tritium fusion fuel for producing the proper fusion reactions required in a reactor in contrasts to deuterium or hydrogen operation.

E

ECCD: Electron Cyclotron Current Drive. Non-inductive current drive technique using directed electron cyclotron resonance waves.

ECE: Electron Cyclotron Emission. Radiation emitted by electrons as a result of their cyclotron motion around magnetic field lines. Used to measure electron temperature.

ECH, ECRH: Electron-Cyclotron (Resonance) Heating. Radio wave heating near the resonance frequency (or its multiple) of the electron gyration in a magnetic field. In present and future machines ECH is at typically 60-170 GHz, depending on the magnetic field strength in a machine.

EFDA: European Fusion Development Agreement. An agreement among all the Euratom Fusion Associations to strengthen their co-ordination and collaboration. It provides the organisational framework for the exploitation of the JET Facilities, coordinates Association activities in physics and emerging technology, manages training and career development of researchers, and coordinates European contributions to international collaborations (excluding ITER and the Broader Approach). Technology activities related to ITER and the Broader Approach, previously carried out within the EFDA framework, are being transferred to the Joint Undertaking Fusion for Energy (F4E).

Electron Bernstein wave (EBW) heating: see: ion Bernstein waves

Electron temperature: A measure of electron thermal energy in units of degrees or electron volts (1 eV ~ 10^4 degrees Kelvin).

Electron volt: the energy which is given to an electron when travelling through a voltage drop of 1 V.

ELM: Edge localised mode. An instability which occurs in short periodic bursts during the H-mode in divertor Tokamaks. It modulates and enhances the energy and particle transport at the plasma edge. These peak transient heat and particle losses must be limited in a reactor.

ENEA: Ente per le Nuove Tecnologie, l'Energia e l'Ambiente, Italy. Partner in the Association EURATOM-ENEA.

Energetic particle: In terms of energy, the particles in a plasma can be divided into two classes. The more numerous thermal particles are characterised by a temperature typically in the range 1-30 keV for modern Tokamaks. The less numerous class of energetic particles has significantly higher energy up to several MeV. Energetic particles can be created by electric fields, fusion reactions, neutral beam injection or RF heating.

Error fields: The magnetic coils of a Tokamak are designed to give the desired magnetic field configuration. The finite number of coils and imperfections in their construction lead to unwanted deviations from this configuration known as error fields. These could lead to disruptions and are of particular concern for larger Tokamaks. On the other hand, certain modulations of the magnetic field at the plasma boundary can be used for control purposes.

EXTRAP T-II: External Trap II, a smaller reversed field pinch (RFP) at the Royal Institute of Technology, Stockholm (Association EURATOM-NFR), built for RFP transport and shell stabilisation studies in support of RFX.

EURATOM: European Atomic Energy Community.

F

Fast Alfvén wave: The fast Alfvén wave exists over a broad frequency spectrum, from the ion cyclotron range of frequencies (ICRF) where its character is electromagnetic, down to magnetohydrodynamic frequencies. Its velocity is comparable to the Alfvén velocity. The fast Alfvén wave is used routinely for high-power (~20MW) ICRF heating on JET, as it is efficiently absorbed in the plasma by the mechanism of ion cyclotron resonance. Although usually stable in Tokamaks, the wave can be excited by energetic ion populations.

Fast wave current drive: Current drive produced by a fast wave. The wave can penetrate the plasma more easily than a lower hybrid wave.

Feedback: Use of real-time measurements of a range of plasma and magnetic field parameters to control the parameters, shape or profiles of the plasma to obtain desired conditions.

Field lines, Flux surfaces: Imaginary lines marking the direction of a force field. In a Tokamak these define a set of nested toroidal surfaces, to which particles are approximately constrained, known as flux surfaces.

FIR: Far infra-red (e.g. wavelength ~ 0.2 to 1mm). FIR lasers are used to measure the magnetic field and plasma density.

First wall: The first material boundary that surrounds the plasma. Today, the first wall in all machines is protected by low-Z materials (such as carbon tiles, boron or beryllium coating).

Flat-top current: Constant current during quasi-stationary operating conditions. In Tokamaks the flat top time is in principle limited by the transformer flux but can be extended by non-inductive current drive.

FML: Fusion Materials Laboratory at FZK

FOM: Stichting voor Fundamenteel Onderzoek der Materie (Foundation for basic investigations of matter), The Netherlands. Partner in the Association EURATOM-FOM.

FTU: Frascati Tokamak Upgrade, a high density, high current Tokamak at Frascati, Italy (Association EURATOM-ENEA).

Fusion for Energy (F4E) : The European Joint Undertaking for ITER and the Development of Fusion Energy or Fusion for Energy was established in April 2007 by a decision of the Council of the European Union under the Euratom Treaty for a period of 35 years. The organisation is located in Barcelona and has 3 major objectives : providing Europe's contribution to the ITER project, implementing the Broader Approach Agreement between Japan and EU and preparing for the construction of a demonstration fusion reactor (DEMO).

Fusion triple product: Product of (ion) density, (ion) temperature and energy confinement time. Its value is a measure of the proximity to break-even and ignition.

Fusion product: The product of a fusion reaction, for example an α-particle or neutron in a deuterium-tritium plasma.

Fusion reactivity: Fusion reaction rate. For present typical Tokamak conditions, it increases with the square of the density and the ion temperature of the plasma.

FZK: Forschungszentrum Karlsruhe, Germany. Partner in the Association EURATOM-FZK, active in fusion technology and, with the development of gyrotrons, in plasma engineering.

FZJ: Forschungszentrum Jülich, Germany. Partner in the Association EURATOM-FZJ, opera-ting the TEXTOR Tokamak.

G

Gyration (gyro) frequency: the oscillatory (rotation) frequency of charged particles in a magnetic field due to their gyration around magnetic field lines. For deuterium ions it is 7.6 MHz/Tesla and for electron 28 GHz/Tesla

Gyro radius: the radius of the orbit of charged particles in a magnetic field. For a 10 keV fusion plasma in a magnetic field of 1 Tesla the gyro radius of deuterium ions is ~ 14 mm and for electrons ~1/3 mm per Tesla, A fusion alpha particle (3.5 MeV) has a gyro radius of 27 cm at 1 Tesla.

Gyrotron: Device used for generating high power microwaves in the electron cyclotron range of frequencies (50 - 200 GHz). This UHF wave is mostly used to heat the plasma at the electron cyclotron resonance frequency. It also could be used to diagnose the plasma.

H

HAS : Hungarian Academy of Sciences, Hungary. Partner in the Association Euratom – HAS.

Heliac: Stellarator configuration with a central toroidal coil around which the plasma column is wound helically. Because of its high flexibility for investigating a wide range of Stellarator configurations, TJ-II has been constructed as a Heliac.

Helias: Optimised Stellarator configuration, used with modular coils for the large Wendelstein VII-X (Germany, under construction) and SHEILA (Australia).

H-mode: A High confinement regime that has been observed in Tokamak plasmas. It develops when a Tokamak plasma is heated above a characteristic power threshold, which increases with density, magnetic field and machine size. It is characterised by a sharp temperature gradient near the edge (resulting in an edge "temperature pedestal"), ELMs and typically a doubling of the energy confinement time compared to the normal "L" regime. Today, a variety of high confinement modes have been identified in divertor and in limiter configurations (e.g. the I-mode), which, in part, have been obtained by special tailoring of the radial plasma current profile.

H-transition (or L-to-H transition): Transition into the H-regime from the L-regime, usually quite sudden, at a certain threshold power of additional heating and specific plasma parameters.

Halo currents: See Vertical Displacement Event.

Helicity injection: The helicity of a toroidal plasma is related to a linkage of toroidal and poloidal magnetic fluxes, and is approximately conserved throughout a discharge. If additional helicity can be injected, the plasma current could be sustained or even increased.

Helium ash: Fusion reactions in a deuterium-tritium plasma produce energetic α-particles (helium nuclei), which heat the plasma as they slow down. Once this has happened, the α-particles have no further use: they constitute helium ash, which dilutes the fuel and must be removed to maintain a burning plasma.

Hellenic Republic : Partner in the Association Euratom – Hellenic Republic.

High beta (β): Condition in which the plasma energy is a significant fraction of the energy in the magnetic field. An alternative measure is the ratio between the plasma energy and the energy in the poloidal magnetic field, the poloidal β.

Hydrogen: The lightest element; the nucleus consists of only one proton, the atomic shell of one electron. Isotopes of hydrogen, with one or two additional neutrons in the nucleus, are deuterium and tritium respectively.

I

IAEA: International Atomic Energy Agency (of the United Nations), Vienna, Austria. The ITER project is undertaken under the auspices of the IAEA.

ICE: Ion Cyclotron Emission. Observed in JET and TFTR as a suprathermal signal, apparently driven by collective instability of energetic ion populations such as fusion products and injected beam ions.

ICF: Inertial Confinement Fusion. Intense beams of laser light or light or heavy ion beams are used to compress very rapidly and heat tiny target pellets of fusion fuel to initiate fusion burn in the centre. Sufficient fusion reactions must occur in the very short time before the fuel expands under its own pressure. The inertia of the pellet's own mass determines the time scale during which fusion reactions occur, hence the name inertial confinement.

ICRH: Ion Cyclotron Resonance Heating by launching waves into the plasma in the range of the ion cyclotron frequency (radio frequency, typically at several tens of MHz).

ICRF, ICRH: Ion Cyclotron Resonance Frequencies and Ion Cyclotron Resonance Heating 8see there), respectively

Ideal: In the context of MHD, 'ideal' implies that the magnetic field and the plasma always move together. For this to occur, the electrical resistivity of the plasma must be negligible. This is usually a good approximation for fusion plasmas.

IEA: International Energy Agency (of the OECD), Paris, France. Implementing agreements for international collaboration on specific topics in fusion have been set up in the frame of the IEA.

Ignition condition: Condition for self-sustaining fusion reactions: heat provided by fusion α-particles replaces the total heat losses. External sources of plasma heating are no longer necessary and the fusion reaction is self-sustaining. Ignition is not required for energy gain in a power station. Retaining a level of external heating or current drive will be required to control the plasma pressure and current profiles, to optimise the performance, leading to a so-called "driven burn".

Impurities: Ions, other than the basic plasma ion species, which are unwanted as they lose energy by radiation and dilute the plasma.

Impurity screening: The prevention of impurities from entering the plasma.

INRNE : Institute for Nuclear Research and Nuclear Energy, Sofia, Bulgaria. Partner in the Association Euratom – INRNE.:

Internal Reconnection Event (IRE): An instability which breaks magnetic field lines and reconnects them with a different topology to reduce the system to a lower energy state - associated with the operating limits of spherical Tokamaks.

Ion Bernstein wave: A wave which only exists in a hot plasma and is supported by the ions. It propagates at right angles to the magnetic field, when it is undamped, at harmonics of the ion cyclotron frequency. There is also an electron Bernstein wave which propagates at harmonics of the electron cyclotron frequency.

Ion Cyclotron Current Drive (ICCD): Non-inductive current drive using ICRH.

Ion Cyclotron Resonance Heating (ICRH) / Ion Cyclotron Resonance Frequencies (ICRF): Additional heating method using RF waves at frequencies (~ 20-50 MHz) matching the frequency at which ions gyrate around the magnetic field lines.

IPP: Max-Planck-Institut fuer Plasmaphysik, Garching, Germany. Partner in the Association EURATOM-IPP, operating the Tokamak ASDEX-Upgrade. Also has sites in Berlin and in Greifswald, where the construction of the large superconducting Stellarator Wendelstein VII-X (W7-X) is in progress.

IPP-CR: Institute for Plasma Physics – Academy of Sciences of the Czech Republic. Partner in the Association Euratom – IPP.CR

IPPLM : Institute of Plasma Physics and Laser Microfusion, Poland. Partner in the Association – Euratom - IPPLM

IR: Infra Red part of the electromagnetic spectrum.

IRE: Internal Reconnection Event.

IST: Instituto Superior Técnico, Portugal. Partner in the Association EURATOM-IST.

ISTTOK: Tokamak for study of non-inductive current drive, at the Instituto Superior Técnico (IST), Lisbon, Portugal.

ITER: International Thermonuclear Experimental Reactor (the next step as a collaboration originally between EURATOM, Japan, the Russian Federation and the USA, under the auspices of the IAEA). After a conceptual design phase - CDA (1988-1990), an engineering design activities (ITER-EDA, 1992-2001), and an interim phase of Coordinated Technical Activities (CTA) the ITER project with a duration of 35 years (construction, exploitation and decommissioning) was agreed in 2006 between EURATOM (with the largest share), China, India, Japan, Russia, South Korea and the US. The ITER device is being constructed in Cadarache (Southern France) and is expected to be commissioned in 2018.

J

JAEA. Japan Atomic Energy Agency, formerly JAERI. Head-quarters in Tokyo, Japan.

AEC: Japan Atomic Energy Commission, Tokyo, Japan.

JAERI: Japan Atomic Energy Research Institute. Now JAEA.

JET: Joint European Torus. The largest Tokamak in the world, sited at Abingdon, UK. Operated as a Joint Undertaking (JET Joint Undertaking), until the end of 1999. The scientific exploitation of the JET facilities is now guaranteed by the EURATOM fusion Associations within the EFDA framework. The operation of the facility is the responsibility of the Association EURATOM-UKAEA.

JT-60SA: Japan Torus-60 Super Advanced. A superconducting large Tokamak ("satellite class" to ITER) to be built in Naka, Japan with European participation under the Broader Approach Agreement.

JT-60U: Japanese Tokamak at Naka. The largest Japanese Tokamak and second largest operating experiment after JET, but not designed for use with D-T fuel.

K

keV: Kilo-electro-Volt. Energy which an electron acquires passing a voltage difference of 1000 volts. Also used to measure the temperature of a plasma (1 keV corresponds to 11.8 million degrees Kelvin).

Kinetic instability: Oscillation which is unstable as a result of the energy distribution of ions or electrons.

Kinetic theory: A detailed mathematical model of a plasma in which trajectories of electrons and ions are described. More complex than fluid and two-fluid theories, it is necessary in the study of RF heating and some instabilities, particularly when energetic particles are involved.

L

L-H transition: Change from L regime to H regime (usually quite sudden).

L-mode: As opposed to the H mode. Regime with degradation of confinement, in additionally heated plasmas, with respect to plasmas heated Ohmically by the plasma current.

Langmuir probe: Electrical probe inserted into the edge of a plasma for measurements of density, temperature and electric potential.

Larmor radius: Radius of the gyratory motion of particles around magnetic field lines.

Laser ablation: Use of lasers to produce a sudden influx of impurities into the plasma from a solid surface.

Last closed flux surface: The boundary separating those magnetic field lines that intersect the wall (open lines) from the magnetic field lines that never intersect the wall (closed lines).

Lawson criterion: The value of the confinement time multiplied by the ion density (at the required temperature) which must be exceeded in a fusion reactor to reach ignition.

LEI : Lithuanian Energy Institute, Lithuania. Partner in the Association Euratom – LEI.

LH : see Lower hybrid

LHCD : see Lower hybrid current drive

LHRH : see Lower hybrid (resonance) heating

Limiter: A material surface within the Tokamak vessel which defines the edge of the plasma and thus avoids contact between the plasma and the vessel. A pumped limiter can also be used to remove heat and particles and is an alternative exhaust system to the divertor.

Locked modes: MHD modes that cease rotating (though they can still grow).

Low-activation materials: Materials which do not develop high, long-lived radioactivity under neutron irradiation. Similar: Reduced-activation materials.

Low aspect ratio: Low ratio of major to minor radius of the torus.

Lower hybrid current drive (LHCD): Non-inductive current drive using lower hybrid waves.

Lower hybrid heating (LHRH): Plasma heating by radio frequency waves at the "lower hybrid" resonance frequency in the plasma. Typical frequencies are a few GHz.

Lower hybrid (LH) wave: A plasma wave of frequency between the ion and electron cyclotron frequencies. It has a component of electric field parallel to the magnetic field, so it can accelerate electrons moving along the field lines.

M

Magnetic axis: The magnetic surfaces of a Tokamak form a series of nested tori. The central 'torus' defines the magnetic axis.

Magnetic Confinement Fusion (MCF): Confinement and thermal insulation of a plasma within the reactor core volume by the action of magnetic fields. In toroidal magnetic confinement, usually both toroidal and poloidal components of the magnetic field are needed (the field lines are threaded like the filaments of a cable which is bent into a ring).

Magnetic islands: Islands in the magnetic field structure caused either by externally applied fields or internally by unstable current or pressure gradients. See tearing magnetic islands.

Magnetic surfaces (flux surfaces): In toroidal magnetic confinement, the magnetic field lines lie on nested toroidal surfaces. The plasma pressure, but not the amplitude of the magnetic field, is a constant on each magnetic surface.

Major radius: The distance from the Tokamak symmetry axis to the plasma centre.

Marfe: A localised and radiating thermal instability sometimes observed near the edge of Tokamak plasmas.

Marginal Stability: Close to the transition from stability to instability.

MAST: Mega Amp Spherical Tokamak at Culham (Association EURATOM-UKAEA) with a cylindrical vacuum vessel and rectangular toroidal field coils. Began operation in 1999.

MCF: See Magnetic Confinement Fusion

MEdC : Ministry of Education and Research, Romania. Partner in the Association Euratom – MEdC.

MeV: Mega-electron-Volt, unit for nuclear energies. Energy which an electron acquires passing a voltage difference of 1 million volts.

MHD (Magnetohydrodynamics): A mathematical description of the plasma and magnetic field, which treats the plasma as an electrically conducting fluid. Often used to describe the bulk, relatively large-scale, properties of a plasma.

MHD instabilities: Unstable distortions of the shape of the plasma/magnetic field system.

MHEST : Ministry of Higher Education, Science and Technology, Slovenia. Partner in the Association Euratom – MHEST.

Microinstabilities: Instabilities with characteristic wave-lengths similar to the ion Larmor radii, rather than to the Tokamak dimensions. These are thought to be responsible for the fine scale turbulence in Tokamaks, and hence anomalous transport.

Minor radius: Half the small diameter of the tyre-shaped toroid.

Mirnov coils: Pick-up coils at the edge of the plasma for measuring the time variation of magnetic fields arising from instabilities.

Mirror: A linear magnetic confinement concept with a weaker magnetic field in a central region and with strong fields at both ends which reflect contained particles by the mirror effect. Some variants exist to increase the magnetic field in all directions from the centre or to improve the closure of the bottlenecks. The Tandem Mirror confinement concept also involves electrostatic fields.

MIT: Massachusetts Institute of Technology, Boston, USA. Operates the high-field divertor Tokamak ALCATOR C-MOD.

Mode: A resonant wave or oscillation in a plasma. Also used as a synonym for an operating regime.

Mode number: Characterises the wavelength of a mode.

Monte Carlo code: A statistical technique used in numerical calculations where events may occur many times, each with a certain probability.

MSE : Motional Stark Effect: The measurement of shifts and splitting of spectral lines emitted from particles moving in a local electric field. This can be interpreted to give the local magnetic field inside the Tokamak if the particle velocity is known, and is a major diagnostic on some Tokamaks to deduce the current profile.

N

Nb_3Sn, Nb_3Ti: high field, high current superconducting materials used for the magnets in magnetic confinement fusion research.

Negative ion beam: To produce neutral beams, negative ions (obtained by the addition of electrons to neutral atoms) are accelerated and then neutralised before entering the plasma. The efficiency of creating neutral beams from positive ions is too low at the beam energy required for a fusion power station, of the order of 1 MeV.

Neo-classical theory: Classical collisional plasma transport theory, corrected for toroidal effects. The neoclassical theory predicts the existence of the bootstrap current.

Neo-classical tearing mode (NTM): The magnetic island produced by a tearing mode perturbs the bootstrap current which further amplifies the island and degrades confinement or leads to a disruption.

NET: Next European Torus, a design for the Next Step which had been prepared in the 1980's by the NET team (located at the Association EURATOM-IPP in Garching) and which has largely influenced the ITER design. The European ITER contributions in physics and technology were organised by the NET team until 1999.

Neural network: A computer algorithm that uses incoming data to derive plasma parameters, having previously been "trained" on a series of examples of a non-linear input-output mapping.

Neutrons: Elementary neutral particles in the atomic nucleus. Products of Deuterium-Tritium and other fusion reactions.

Neutral beams: Since charged particles cannot easily penetrate the magnetic confinement fields of the plasma, high energy beams of neutral atoms are injected into the plasma for fuelling, heating and current drive. Within the plasma, the atoms of the beam are ionized and are then confined.

Neutron multiplier: The fusion of deuterium and tritium consumes one tritium nucleus per reaction, producing one neutron. Since in the blanket of a power station not every neutron reacts with lithium to produce a new tritium atom, a neutron multiplying element may be used in the blanket to enhance the tritium production so as to make the power station self-sufficient in tritium supply.

Next Step: The next experimental device in the strategy of the European Fusion Programme. Presently pursued via the ITER EDA, with a European activity as a fall-back option. The generic name for an experimental reactor with a long pulse burning plasma at high fusion gain.

NIFS: National Institute for Fusion Science, Nagoya, Japan.

NRIM: National Research Institute for Metals, Sakura-mura, Japan.

Non-inductive heating and current drive: See additional heating and current drive.

NSTX: Spherical Tokamak at Princeton, USA. A similar size to MAST, but of different design. Started operation in 1999.

NTM: see Neoclassical Tearing Mode

O

ÖAW : Austrian Academy of Sciences, Austria. Partner in the Association Euratom – ÖAW.

Ohmic heating (OH): The resistive heating resulting from a current flowing within the plasma corresponding to the heating of a wire by a current flowing through it. Ohmic heating in a Tokamak is insufficient to reach thermonuclear temperatures since, contrary to a wire, the resistance of a plasma decreases strongly with increasing temperature, thus making Ohmic heating weak at high temperatures.

ORNL: Oak Ridge National Laboratory, USA.

Operating limits: See Tokamak operating boundaries.

Optimised shear: Adjusting the current profile to optimise Tokamak.

P

PbLi: Eutectic lithium-lead alloy considered for use as blanket breeding material.

Peeling mode: An edge MHD instability which exists when the current density at the plasma edge is non-zero. It may be associated with ELMs.

Pellet: In inertial confinement concepts, the fuel is contained in tiny spheres, called pellets, which are compressed by laser or particle beams. In magnetic fusion, pellets of frozen hydrogen, deuterium, tritium, accelerated up to several kilometres per second, are used to refuel the plasma and to obtain very high densities.

PIREX: Proton Irradiation Experiment, material test facility (Association EURATOM-Switzerland, CRPP-FT, PSI, Villigen, CH).

Plasma: State of matter above a few thousand degrees where atoms are broken into their constituents, ions and electrons, thereby creating an electrically conducting medium. Plasmas can therefore interact strongly with electric and magnetic fields.

Plasma confinement: Retention of plasma energy or particles within a given region, including the heat and particle losses from the plasma.

Plasma parameters: Physical quantities which characterise the plasma and which must be measured experimentally, such as current, density, temperature, confinement time, beta.

Plasma pressure: Proportional to the product of plasma density and temperature. There is an electron and an ion pressure and the plasma pressure is the sum of the two. In magnetic confinement devices, this pressure is counterbalanced by magnetic pressure.

Plasma shape: Describes the plasma vertical cross-section, circular, elongated, D-shape, diverted, single null, double null.

Polarimetry: Measurement of the rotation of the plane of polarisation of light passing through a magnetically confined plasma; used to measure the local magnetic field and thus the safety factor (see Faraday rotation).

Poloidal field: Component of the magnetic field perpendicular to the toroidal direction and the major radius. The poloidal field is essential for confinement and is generated in a Tokamak by the plasma current and the external coils.

Power threshold: The L-H transition and improved performance regimes related to reversed shear occur when the power exceeds a certain threshold value - the power threshold.

PPPL: Princeton Plasma Physics Laboratory, New Jersey, USA.

Preliminary Tritium Experiment (PTE): Discharges on JET, November 1991, into which a significant amount of tritium was injected for the first time in a Tokamak. The power liberated from fusion reactions (~ 2MW for ~ 2 seconds) was in accordance with expectations. Followed by the more ambitious DTE in 1997.

Profile: Variation of plasma parameters with minor radius.

Profile control: Controlling the profiles of pressure, density or current, in order to control instabilities.

PSI: Paul-Scherrer-Institut, Villigen, Switzerland, active, in muon physics among others fields. The Association EURATOM-Swiss Confederation has their fusion technology activities working in superconductor and materials technology located at Villigen.

Pumped divertor: Divertor field lines directed into a pumped chamber surrounding the target plate.

Q

q, q_{95}: See Safety factor.

Q: Ratio of fusion power to total additional heating power. At Q=∞, no external power is required and the plasma is said to be ignited. A power station should operate with Q~50 to be economical.

R

Radial electric field: Arises when there is a charge imbalance in the plasma.

Radio frequency (RF) heating: Heating with waves in the radiofrequency range at resonance frequencies of the plasma (see ECH, ICRH, LHH).

Reduced activation materials: similar to low activation materials

Reflectometry: Use of reflected microwaves to measure plasma density.

Relaxation: The evolution of a plasma to a lower energy state.

Resistive ballooning modes: A class of ballooning mode which would be stable in the absence of resistivity, but can be unstable in its presence. Related to tearing modes, but topologically different.

Resistive instability: Instability due to diffusion and rearrangement of magnetic field lines. When the plasma resistivity is small, these instabilities have a slow growth rate.

Resistivity: The tendency to resist the flow of electric current, thereby dissipating energy. Plasmas are very good conductors of electric current, so that their resistivity can often be neglected. In this case, 'ideal' magnetohydrodynamics may be applied.

Resonant ions/electrons: Resonance occurs when one of the characteristic frequencies of particle motion in the plasma (for example, the cyclotron frequency) matches the frequency of some applied perturbation (for example, an RF wave).

Resonant magnetic perturbation (RMP): An externally applied magnetic perturbation matched to the spatial structure and optionally the frequency and phase of an instability.

Reversed Field Pinch (RFP): A toroidal magnetic confinement device, similar to a Tokamak, in which the poloidal and toroidal fields are of comparable magnitude. Capable of higher plasma current and pressure for a given external magnetic field. They require a conducting shell close to the plasma for stabilisation.

Reverse (magnetic) shear: In a Tokamak the current density is usually greatest at the magnetic axis, in which case the safety factor increases from the centre to the edge of the plasma. Using non-inductive current drive and/or the bootstrap current the current density can be made to increase away from the centre. In this "reverse shear" case, the safety factor has a minimum away from the plasma centre. Using reverse or low shear ("optimised shear") some Tokamaks, notably DIII-D and TFTR in the US and more recently JT-60U in Japan and JET, have shown greatly improved plasma performance. Reverse shear is an attractive option for advanced Tokamak scenarios.

RF: Radio-Frequency.

RFP : See Reversed Field Pinch

RFX: Reversed Field pinch Experiment at CNR-Padova, Italy (Association EURATOM-ENEA).

RISØ: Forskningscenter Risø, Denmark. Partner in the Association EURATOM-RISØ.

RMP : See Resonant Magnetic Perturbation

Rotational transform: Measure of the ratio of poloidal to toroidal flux defining the pitch of the helical field lines. The q-value of the Tokamak is proportional to the reciprocal of the rotational transform.

Runaway electron: An electron with a very high energy has a decreasing probability of colliding with another charged particle and of losing its energy. Such a particle then gains more and more energy in the electric field of a Tokamak, reaching 10's of MeV.

S

Safety factor: Number of turns the helical magnetic field lines in a Tokamak make round the major circumference for each turn round the minor circumference, denoted q. Has no connection with the ordinary sense of "safety" other than q=1 surfaces are ideally unstable. For diverted plasmas q goes to infinity at the separatrix, so instead q_{95} is used to describe the safety factor near the edge, which is the safety factor of the plasma surface which contains 95% of the poloidal flux.

Sawtooth: A cyclically recurring instability which causes an energy loss from the central region of Tokamak discharges. The temperature periodically falls abruptly, then slowly recovers. The jagged trace produced by plotting temperature against time gives the instability its name.

Sawtooth crash: The rapid collapse of the central temperature in a Tokamak during a sawtooth cycle.

Scaling laws: Empirical or theoretical expressions for how various plasma phenomena (e.g. confinement, power threshold, etc) vary with Tokamak parameters. They are particularly used for predicting the performance of future Tokamaks.

Scrape-off-layer (SOL): The residual plasma between the "edge" of the plasma (defined by the limiter radius or the separatrix) and the Tokamak vessel wall.

Semi-empirical: A theoretical approach in which the behaviour of some key quantities is deduced from experiment, rather than a priori.

SEAFP: The "Safety and Environmental Assessment of Fusion Power", a study conducted by several teams in the associated laboratories, NET/EFDA, industry and the JRC, published in June 1995.

SEAL: The "Safety and Environmental Assessment of Fusion Power Long-term", a programme which was launched in 1995, being undertaken for the European Commission in the framework of the Fusion Programme.

Separatrix: Magnetic surface at which the rotational transform vanishes and the safety factor becomes infinite.

Shear: The safety factor usually varies from magnetic surface to magnetic surface across the plasma cross-section; this variation is measured by the non-dimensional quantity called “shear”. Also refers to the variation of plasma flow (flow shear). If the type of shear is not specified, it usually means magnetic shear.

Single/double null: Points of zero poloidal magnetic field where the separatrix crosses itself are the X-points or nulls. Usually sited above and/or below the plasma. Tokamak divertor configurations have either one or two nulls.

Single fluid model: The set of equations which represent a plasma as a magnetised, electrically conducting fluid with the usual fluid properties of viscosity, thermal conductivity, etc. The possibility of distinct behaviour of electrons and ions (i.e. 2 “fluids”) is excluded.

Small aspect ratio: Same as Low aspect ratio.

SOL : See Scrape-off-layer

Spectroscopy: The detection and analysis of the spectrum of radiation emitted by a plasma. This can yield information about temperatures, impurities, rotation, using different parts of the electromagnetic spectrum (IR, visible, VUV, XUV, etc.)

Spherical Tokamak (ST): A very low aspect ratio Tokamak - it appears almost spherical, though topologically it remains a torus with a centre column. The Spherical Tokamak is being investigated today with medium-sized experiments, in Europe with MAST (UKAEA).

ST : See Spherical Tokamak

Stability theory: The theory of how small perturbations to a system evolve in time. Spontaneous growth is due to instability. Instabilities can saturate at some small amplitude, in which case they may degrade confinement, or grow uncontrollably, in which case the equilibrium is lost leading to a disruption.

Start-up assist: Assisting plasma formation to cross a range of plasma temperature at which impurities radiate strongly, with the aim of minimising the start-up delay and transformer requirements, usually using ECH.

Steady-state power plant: A continuously (as opposed to cyclically) operated power plant.

Stellarator: Closed configuration having the shape of a three-dimensionally distorted ring in which the plasma is confined principally by an externally generated magnetic field (produced by non-planar coils outside the plasma vessel). The coils can be arranged in a modular fashion. Stellarators do not need a transformer; they need an additional heating system for the plasma start-up. Due to the fact that no toroidal plasma current is needed to maintain the confinement configuration, they naturally provide steady-state operation.

SULTAN: Supra Leiter Test ANlage. Large Superconductor Test Facility, CRPP at PSI Villigen, Switzerland (Association EURATOM-Swiss Confederation).

Superthermal radiation: Electromagnetic radiation produced by energetic particles, as opposed to thermal particles.

T

TAE modes: Toroidal Alfvén Eigenmodes. One class of Alfvén gap modes.

Target plates: See Divertor.

TBM: Test blanket module for ITER. ITER will not have a full blanket but will test blanket modules.

TCV: ”Tokamak à Configuration Variable”, for study of elongated and strongly shaped plasmas, at Lausanne, Switzerland (Association EURATOM-Swiss Confederation).

TEKES: Technology Centre Finland. Partner in the Association EURATOM-TEKES.

Tearing mode: A class of resistive MHD instability which has been predicted theoretically in Tokamaks and positively identified in experiments.

TEC : See Trilateral Euregio Cluster

Temperature pedestal: In an H-mode discharge there is a region of steep temperature gradient at the plasma edge. The temperature at the top of this steep gradient region is the temperature pedestal.

Tesla: Unit of magnetic field strength (more exactly the magnetic induction). $1T = 1Vs/m^2 = 10{,}000$Gauss.

TEXTOR: Torus Experiment for Technology Oriented Research. Tokamak at Jülich, Germany (Association EURATOM-FZJ), equipped with an dynamic ergodic divertor for influencing the transport in the plasma boundary.

TFTR: “Tokamak Fusion Test Reactor” at Princeton, the largest US device operating with deuterium-tritium fuel from 1993 – 1997, yielding close to 10 MW fusion power. Ceased operation in March 1997.

Thermal cycling: Successive heating and cooling of materials can lead to cracks or rupture, particularly at boundaries between materials that expand at different rates.

Thermal particles: As a result of collisional energy exchange, the energy of most plasma particles falls within a Maxwellian distribution which is described by a single temperature (typically 1-30keV for Tokamaks). These are the thermal particles, as distinct from energetic particles which lie outside the thermal distribution.

Thomson scattering diagnostic: Diagnostic to measure temperature and density by detecting laser light scattered and Doppler shifted by the thermal plasma electrons.

Tight aspect ratio: Same as Low aspect ratio. See Spherical Tokamak.

TJ-II: A Heliac Stellarator at Madrid, Spain (Association EURATOM-CIEMAT).

TLK: Tritium Laboratory Karlsruhe at FZK

Tokamak: Magnetic configuration with the shape of a torus. The plasma is stabilised by a strong toroidal magnetic field. The poloidal component of the magnetic field is produced by an electrical current flowing toroidally in the plasma. This current is induced via transformer action and, for steady-state, must be maintained by non-inductive current drive and by self-generation of bootstrap current inside the plasma.

Tokamak operating boundaries: The set of plasma parameters, beyond which it is impossible to operate a Tokamak. Careful choice of plasma cross-sectional shapes and current and pressure profiles can increase the operating regime.

TORE SUPRA: Large Tokamak with superconducting toroidal magnetic field coils and a circular plasma cross-section at the Association EURATOM-CEA in Cadarache, France. Equipped with fully actively cooled inner wall for studying high-power steady-state operation.

Toroidal Alfvén Eigenmodes: See TAE modes.

Toroidal field: The component of the magnetic field along the major circumference of the torus. The largest magnetic field component in a Tokamak.

Toroidal stability: Stability analysis taking account of effects due to the toroidal geometry. These are sometimes neglected to identify possible instabilities, but must usually be included for accurate predictions of stability boundaries.

Toroidal turbulence code: A turbulence code which includes effects due to the toroidal geometry.

TOSKA: Large facility testing for superconductors (Association EURATOM- FZK, Karlsruhe, Germany).

Transformer drive: The use of a transformer action to produce plasma current.

Transport: The processes by which particles and energy move across magnetic surfaces.

Transport barrier: In certain operational scenarios (e.g. the H-mode or ITB-mode) a region of low transport exists giving rise to a steep local pressure gradient. Such a region is referred to as a transport barrier.

Transport scaling: The magnitude of heat transport may be expressed, empirically or theoretically, in terms of a simple functional dependence on a few plasma parameters. This allows us to model how the heat transport varies (scales) in response to changes in the value of these parameters.

Trapped particles: The outside (large major radius) of a Tokamak plasma has a lower magnetic field than the inside. Particles with low velocity parallel to the magnetic field compared with the velocity perpendicular to the magnetic field may not enter the higher field (inside) region and become trapped on the outside. They are not free to circulate toroidally but instead bounce back and forth, performing so-called banana orbits.

Tritium: An isotope of hydrogen, whose nucleus consists of one proton and two neutrons. Tritium does not occur naturally, because it is unstable to radioactive decay with a half-life of 12.3 years. Due to its rapid decay, tritium is almost absent on earth. For a fusion reactor, tritium will be produced in the breeding blanket surrounding the core of a fusion power station. Special tritium-handling techno-logy is required for existing (JET) and future fusion machines using deuterium-tritium mixtures as fuel and has been developed on TFTR and JET.

Tritium inventory: The amount of tritium contained in a fusion power station or in a specified part of it.

Turbulence: Randomly fluctuating, as opposed to coherent, wave action. For example, the turbulent water beneath a waterfall can only be described in terms of its averaged properties, such as the scale and duration of fluctuations; whereas a more systematic description can be given to waves on the surface of a still pond.

Turbulent transport: Anomalous heat transport associated with plasma turbulence.

Two-fluid model and multi-fluid model: The extended set of equations which represent a plasma as interpenetrating and interacting fluids of electrons and ions, impurity ions, etc.

U

UKAEA: United Kingdom Atomic Energy Authority. Partner in the Association EURATOM-UKAEA which the spherical Tokamak MAST. Also charged with the operation of the JET facilities under EFDA.

University of Latvia : Partner in the Association Euratom – University of Latvia

V

VDE -Vertical Displacement Event : An event which arises when control of the plasma is lost and the plasma moves vertically. It can lead to a “halo current" in components which surround the plasma resulting in large, potentially damaging, forces on these components. The forces are much larger in larger Tokamaks and are therefore a particular concern for JET and ITER.

VR: Vetenskapsradet (Swedish Research Council) , Sweden. Partner in the Association EURATOM-VR.

VUV: The "Vacuum Ultra Violet" range of the electromagnetic spectrum.

W

Warm plasma refuelling: Fuelling of plasma using medium energy particles or particle clusters.

WENDELSTEIN VII-AS: Advanced Stellarator at Garching, Germany (Association EURATOM-IPP), ceased operation in 2002. Provided the experimental basis for Wendelstein VII-X

WENDELSTEIN VII-X, (W7-X): Large advanced superconducting Stellarator, optimised to produce a reactor-relevant plasma configuration under construction at Greifswald, Germany (Association EURATOM-IPP) with first operation scheduled for 2014.

X

X-point: See single/double null.

XUV: The "Extreme Ultra Violet" range of the electromagnetic spectrum. Shorter wavelengths than VUV.

Z

Z: atomic (charge) number of elements (e.g. 1 for hydrogen and 92 for uranium), is the number of protons in the nucleus and electrons in the atomic shell.

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