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Long-term fusion strategy in Europe

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Abstract

The aim of the European fusion programme is to make available, as early as possible, fusion power as a source of electric energy. At a significantly earlier date, however, this programme should provide conclusive information on the practicability of fusion power production, and on its safety, environmental and economic aspects. Due to the associated long lead times, technologies which can possibly come to fruition only in a second generation of power plants must also be studied now. Here we describe the necessary elements of such a forward directed strategy, whose next step is based on the integration of fusion physics and technology in the ITER device, and on a material development programme, including a neutron irradiation facility with a fusion relevant spectrum.

1. Introduction

The European fusion programme is clearly oriented towards a fusion reactor. This guides the physics programme in the associations and on JET, as well as the technology programme as coordinated by EFDA and executed by the associations and by European industry. The political discussions in preparation of the 6th Framework Programme have shown the importance of outlining the long-term prospects and of defining programme milestones transparent to the general public. The roadmap for fusion outlined in the following is, in the opinion of the authors, consistent with the scientific and technical requirements, but does not constitute an official European planning guideline.

The first milestone, which the fusion programme must achieve, is the demonstration of the feasibility of a fusion power plant. For this we have to demonstrate at least one workable solution for all critical physics and technology questions, demonstrate the favourable safety and environmental properties of a fusion power station, and provide a basis for assessing the economy of a power plant based on these solutions. This will allow the inclusion of fusion as a well-defined element in long-term energy planning. It would also have a more immediate consequence, as the availability of a follow-up technology such as fusion, would allow an immediate shift from abundant, but polluting technologies such as coal, to cleaner, but scarcer resources, such as gas.

A second milestone would be the delivery of net electrical power to the grid, in a plant, which was also fully self-sufficient with respect to tritium supply. The power plant conceptual design studies (PPCS) conducted within EFDA show that the device in which this should occur, the successor to ITER, would differ only in details from an economically viable first generation power plant. Operation of the first power plants will not, of course, imply the end of the development stage of fusion power. Research and development towards even more efficient and economic use will continue. As some of these developments, notably in the material sciences, have long lead times and require extensive testing programmes, certain R&D which would come to fruition only in a second generation of power plants should be initiated now. In addition, the substantial economic benefits of even small increases in efficiency or availability provide a role for the ongoing research into other toroidal confinement configurations, which could enter the development line after ITER.

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EFDA conducts two programs, which aim at outlining more clearly the requirements of a future power plant and give guidance to physics and technology R&D. In a top–down logic, the socio-economic research on fusion (SERF) identifies the requirements for fusion power to become an integrated and socially accepted part of the European and global energy supply systems. In this framework we study the direct costs of electricity production by fusion [1] the additional costs to society arising from its impact on the environment during the entire life cycle of power plants (externalities) [2] and develop, using standard economic models, general energy supply scenarios for the second half of this century [3]. The PPCS develop models for first generation and advanced power plants, and serve to define the required physics and technology R&D [4]. Superposed on all these programs, and clearly of primary importance, are studies of the safety aspects of ITER and future power plants [5].

The three device generations: (i) ITER, (ii) DEMO ($\simeq$ first generation power plant), (iii) advanced (second generation) power plants, correspond to three levels of physics and technology requirements, and require consistent solutions in the key areas of (1) plasma performance, (2) divertor physics and technology (including plasma facing materials and cooling system), (3) breeding blanket, (4) structural and functional materials and (5) other technologies (e.g. tritium handling). Beyond their own design requirements, ITER (and possibly, to some extent, also DEMO) will also serve as a test bed for more advanced physics and for modules or components of more advanced technology.

2. Key research and development issues

2.1. Plasma performance

ITERs baseline operating mode – the so-called ELMy H-mode – is well established in all divertor tokamaks [6]. R&D, conducted in particular on JET, ASDEX-Upgrade and DIII-D, and design modifications between the 1998 ITER design and ITER-FEAT have improved the robustness of these predictions during the last two years. On the physics side, in this regime ITER can therefore concentrate on exploring the novel, specific impact of fusion heating by α-particles, with regard to both the unusual response of heating power to changes in temperature and the presence of a large, isotropic population of energetic particles, with a birth velocity exceeding the Alfven speed. A number of phenomena – sawtooth oscillations in the core, stability of Alfven-modes, and the general dynamics of profile development – are expected to be qualitatively affected by this, and the operating range of ITER, in particular the design value of $Q = 10$, will ensure that these differences can be adequately explored.

For the purpose of power plant studies, physics performance is usually measured by four dimensionless figures of merit, derived by normalizing the dimensional quantities of energy confinement time, τ_E, average plasma density, n, total plasma heating power (fusion or other), P_{heat}, and total plasma pressure, p, to the predictions of empirical fits or first principle theories. A set of such parameters is implicitly defined by the relations [6]:

$$\tau_E = H_{98y,2} \times 0.056 I_{\text{p}}^{0.93} B_{\text{t}}^{0.15} P^{-0.69} n^{0.41} R^{1.39} a^{0.58} \mu^{0.19} \kappa^{0.78}, \tag{1a}$$

$$n = (n/n_{\text{G}}) I_{\text{p}}/(\pi a^2), \tag{1b}$$

$$P = (P/P_{LH}) \times 2.84 B_{\text{t}}^{0.82} n^{0.58} R a^{0.81} \mu^{-1}, \tag{1c}$$

$$p = \beta_N I_{\text{p}} B_{\text{t}}/(80\,\pi a) \tag{1d}$$

which introduce the H-factor, $H_{98y,2}$, the Greenwald density, n_{G}, the L to H-mode threshold power, P_{LH}, and the Troyon parameter, β_N, using the plasma current, I_{p}, toroidal field, B_{t}, major (R) and minor (a) plasma radius, elongation of the poloidal plasma cross-section, κ, and isotope mass number, μ. The quantity $H_{98y,2} = 1$ corresponds to the expectation value for energy confinement in the standard ITER regime (so-called ELMy H-mode), $n/n_{\text{G}} = 1$ to the attainment of the empirical Greenwald density limit, and $P/P_{LH} > 1$ constitutes an empirical requirement for access to the favourable H-mode energy confinement regime. The definition of the Troyon parameter, on the other hand, is well founded on theoretical stability analyses, but the attainable limiting value can vary somewhat, depending on profiles, plasma shape, and wall proximity.

To assure access to the regime dominated by α-particle heating, ITERs baseline operation mode has been based on well-established values for these figures of merit. Fusion power plant designs, on the other hand, assume higher values of the dimensionless plasma pressure parameter, β_N, and are usually based on continuous, rather than pulsed, operation. These two facts are connected, as higher β_N produces not only a higher power density (at given magnetic field and current, and in the optimum operation temperature range, fusion power density varies $\sim \beta_N^2$), but raises also the 'free' contribution to the plasma current produced by a thermoelectric effect: the so-called bootstrap current. Continuous operation can then be achieved by using a smaller fraction of power-consuming additional current driven by neutral beams or RF techniques. Table 1 gives the values of these figures of merit required for a range of devices prototypical for the present state of the art

Table 1
Underlying physics assumptions of different ITER operating scenarios and power plant designs

	State of the art	ITER-steady state		DEMO		
				First generation power plant		Second generation PP
	ITER-baseline	$Q = 5$ reference	$Q = 30$	ITER-size[a]	PPCS[b]	ARIES-RS
H	1	1.6	2.2	1.2	1.2	1.4
β_N	1.8	3.1	4.4	3.9	3.5	5
n/n_G	0.84	0.8	1.2	1	1.1	1.1

[a] R. Toschi et al., SOFT 2000.
[b] Power plant conceptual study (provisional plant model).

(ITER baseline [7]) and first generation [4,8] and advanced power plant designs [9] (P/P_{LH} is not shown, as the required values are rather conservative).

ITER has, as a second reference mode, continuous operation at $Q = 5$, and is expected to expand its performance envelope further in the course of its operational life. Reaching $Q = 30$, as assumed in the third column of the table, would in fact correspond to entering the performance regime of advanced power plant designs. Starting from the present experience of tokamak experiments, the required progress consists less in the improvement of the numerical parameter values (the required value of $\beta_N H$ for $Q = 30$ operation in ITER has, for example, already been achieved in DIII-D, albeit only for very short pulses [10]), but rather in the extension of this operation to useful pulse lengths, and ultimately to steady-state, in a form compatible with dominant α-particle heating.

Two main problems have to be overcome in developing this mode of operation. Very high β-values can only be achieved by exploiting the stabilizing influence of conducting walls on MHD modes. This is particularly true for the flat, or even hollow, plasma current profiles which are expected in the steady-state scenarios, where the bootstrap current will provide the major non-inductive contribution. Stabilization by walls is effective, however, only on time scales below the resistive wall diffusion time (unless a way is found to maintain a high plasma rotation speed) [11]. The situation bears a strict analogy to the axisymmetric, vertical displacement instability, which we have learnt to handle well: in that case the presence of the walls reduces the growth rate from that associated with the plasma-inertia to the value determined by wall resistivity, thereby allowing adequate time for intervention with an active feedback system. The only – however, significant – difference lies in the more complex mode structure involved in controlling MHD modes. ITER is provided with a set of suitable control coils, and several existing devices have, or are preparing to install, similar systems to test the relevant control principles.

The second issue involves the control of the plasma profiles, which have to satisfy strong constraints to yield good energy confinement and stability to MHD modes other than the largest scale ones discussed above. The control of these profiles will become significantly more complicated in the truly steady-state, α-heated situation, as both the current profile (through the bootstrap current) and the heating profile will be dominated, in the main, by intrinsic, rather than externally controlled, contributions. The primary method of control is the external current drive, which must be used sparingly, however, to avoid degrading the energy balance of the plant. Due to the strong link to the α-heating issue, the conclusive experiments in this regard will have to be conducted on ITER, but they can benefit greatly from exploratory research performed on smaller devices.

2.2. Plasma wall interaction

Ensuring that the plasma facing components have an adequate lifetime is a prime example of the need for integrating technology development and physics research [12]. Whereas the latter activity had focussed for decades on protecting the plasma from impurities, the advent of ITER has raised the converse issue. Lifetime of the components and, in addition, inventory control of tritium and dust have become the dominating issues. The problem requires an integrated approach, since, for example, one obvious solution of the power handling problem – to convert the heating power into impurity radiation once it has fulfilled its mission of maintaining a temperature gradient between the plasma core and the periphery – has to be made compatible with the requirement of low core radiation losses and acceptable fuel dilution.

The key physics contribution to the solution of the divertor problem has been the experimental verification and the modelling of the phenomenon of 'detachment', which occurs when radiation losses in the scrape-off layer (SOL) from (intrinsic or added) impurities and hydrogen, together with momentum transfer from the plasma to neutral particles, reduce the energy flux and the plasma pressure at the target plates far below the midplane values on the corresponding flux surfaces. The problem becomes more severe when proceeding from

Table 2
Estimated divertor heat loads

	P_{fus} (GW)	R_o (m)	P_{rad}/P_{heat} [a]	$q_{div,nom}$ [b] (MW/m^2)
ITER-ref.	0.5	6.2	0.8	5
ITER-SS	0.36	6.2	0.8	5
PPCS	>4	>7.5	0.9	14
ARIES-RS	2.2	5.5	0.9	11

[a] $P_{heat} = (P_{fus}(1/5 + 1/Q))$.
[b] $q_{div,nom} = (P_{heat} - P_{rad})/(4\pi R_o \lambda F)$ with geometry factor $F = 10$ and mid plane heat flux width scaling like $\lambda = 0.003 R_o^{0.5}$.

present devices to ITER and is further accentuated in power plants, in essence because the thickness of the SOL, measured by the power flux decay length in the midplane, λ, is observed to increase less than linearly with the device dimensions ($\lambda \sim R^{\gamma}$, $\gamma < 1$), and as power plant designs have significantly higher volumetric power densities than ITER. An estimate of the expected divertor heat flux density is given in Table 2 for a number of representative designs, where the estimate is based on an 'average' scaling with $\gamma = 0.5$, symmetric power distribution on the in- and outboard divertor legs, and a radiated fraction which is increased to 0.9 for the two power plant versions (from an assumed ITER value of 0.8). The geometry factor, F, is the average ratio between the divertor area wetted by the SOL and the area of toroidal intersection between the latter and the midplane. A further increase in the radiated power fraction would not directly lead to a proportional reduction of $q_{heat,nom}$, as the part of the radiated power which emanates from the divertor region itself, starts making an increasing contribution to the target plate heat load. The handling of such heat loads and the associated particle fluxes poses a problem for the integrity of the surfaces on which they are incident and is also very demanding for the steady state cooling of the components. Whereas mock-ups and test stands are efficient R&D tools for the latter problem, study of the plasma wall interaction itself requires a plasma environment corresponding to that of a burning plasma. In fact, by providing a unique combination of correct geometry, appropriate plasma conditions, long pulse lengths and high duty cycle, ITER constitutes the optimum platform for reactor tests, particularly as higher target power loads could be realized by reducing the impurity content.

ITERs starting configuration foresees three different plasma-facing materials, each chosen so as to satisfy the requirements of differing regions. CFCs are selected for the highest heat flux components due to their capability to withstand impulsive heat loads. They are expected, however, to be subject to strong chemical erosion and, through co-deposition, to be the major contributing factor to the in-vessel tritium inventory. Tungsten appears as the most promising long-term option for ITER and the power plant, but only if the divertor temperature can be routinely kept at or below about 10 eV, and if impulsive heat loads from abnormal events (large ELMs or disruptions) can be reliably suppressed. Tungsten's deployment for complete divertor coverage on ITER could follow the first 'exploratory' stage of operation, provided experience with the partial coverage already foreseen for initial operation, as well as that arising from experiments with tungsten coatings which have already been initiated in existing devices, is positive. To minimize radioactive inventory, afterheat, and loss of breeding capability, in a power plant tungsten would be applied in the form of layers on both divertor and first wall.

The simple issue of power removal poses a considerable technical challenge for the divertor design. For a power plant, this question has to be discussed in combination with the breeding blanket design, as one would like to minimize the number of different cooling concepts, while the simultaneous presence of large quantities of beryllium and water in the vessel would appear to be prohibited on safety grounds. These issues are only of limited concern for ITER where the test blanket modules (TBM) will cover approximately one percent of the first wall surface. One of the two test blanket options will, however, be Europe's candidate for the breeding blanket in DEMO and the first generation of power plants, and the compatibility issue will therefore be important. In the water-cooled lithium lead (WCLL) option, a water-cooled divertor could be a natural further development of the ITER design, with an increase of the water temperature up to 320 °C (PWR condition) to improve the thermal efficiency. The helium cooled pebble bed (HCPB) would also require helium cooling for the divertor, which, for the 10 MW/m^2 rating postulated above, is a difficult design problem. For the second generation of power plants we will strive for still higher thermodynamic efficiency and hence higher blanket temperatures. This will necessitate helium or liquid metal cooling and the development of compatible divertor cooling systems (or the acceptance of multiple cooling concepts in the vessel).

2.3. Breeding blanket

The choice of the breeding blanket concept has traditionally been the key decision in conceptual power plant studies. Although the studies coordinated by EFDA have revealed the importance of an integrated approach, the critical role of this selection is undisputed. An overview of blanket concepts both for DEMO/first generation and for advanced power plant designs is given in the review by Giancarli et al. [13] from which we have extracted the information in Table 3. WCLL and I-HCPB are the two European candidates for the DEMO blanket, which are also employed in two corresponding

Table 3
Overview of blanket concepts

	WCLL	I-HCPB	Dual-coolant	TAURO	A-HCPB
Structural material	EUROFER/ODS	EUROFER/ODS	ODS	SiC/SiC	SiC/SiC
Coatings	Anti-corr.		SiC/SiC inserts		
Breeder/multiplier	Pb–17Li	Li-ceramic/Be	Pb–17Li	Pb–17Li	Li-ceramic/Be
T_{max} (breeder)	550 °C	880 °C	750 °C	900 °C	920 °C
Coolant	H_2O	He	Pb–17Li and He	Pb–17Li	He
Purge	n.a.	He	n.a.	n.a.	He
$T_{coolant}$ (inlet/outlet)	265/325 °C	250/550 °C	460/700 (Pb–17Li) 300/480 (He) °C	450/860 °C	350/700 °C
NWL (peak)	6.6	4.4	5.0	3.5	3.5
Net efficiency	33	37	44	43	45

first generation power plant designs developed within the PPCS. The second generation of blankets (TAURO and A-HCPB) distinguishes itself primarily by higher net efficiency, and hence higher coolant temperature, based on the assumed availability of new structural materials, notably SiC/SiC. The dual-coolant concept, employing SiC/SiC only as inserts, corresponds to an intermediate technology, which could possibly even be applied in the first generation of power plants.

The two European DEMO test blanket designs will also be tested in the form of TBM for ITER, with the characteristics given in Table 4. Both module types will be subject to a detailed test programme during the first 10 years of ITER operation, starting (e.g. for electromagnetic tests) during the hydrogen phase of ITER. The use of several slightly modified versions of each blanket type is planned to test different aspects of the designs.

2.4. Structural materials

The structural materials to be used in a fusion reactor should have a low activation cross-section and a short activity decay time, maintain favourable mechanical properties under large neutron fluence, and allow for high operating temperatures [14]. The targeted development of the last two decades has given confidence that these requirements can be met with ferritic steels, whose development and testing could be completed in time to be used as ITER test blankets, for the construction of DEMO and for a first generation of power plants. From the radiological side, a reasonable target is to arrive at a surface gamma dose rate allowing remote controlled recycling (and possibly do better) within 100 years of the date of removal from the power plant. As can be seen

Table 4
European DEMO test blanket modules for ITER

	WCLL	HCPB
Structural material	EUROFER	EUROFER
Breeder/multiplier	Pb–17Li	Li-ceramic/Be
T_{max} (breeder)	459 °C	<900 °C
Coolant	H_2O	Helium
Purge	n.a.	Helium (0.13 MPa)
Coolant pressure	15.5 MPa	8 MPa
$T_{coolant}$ (inlet/outlet)	315/325 °C	250/500 °C
NWL	1.1 MW	0.8 MW

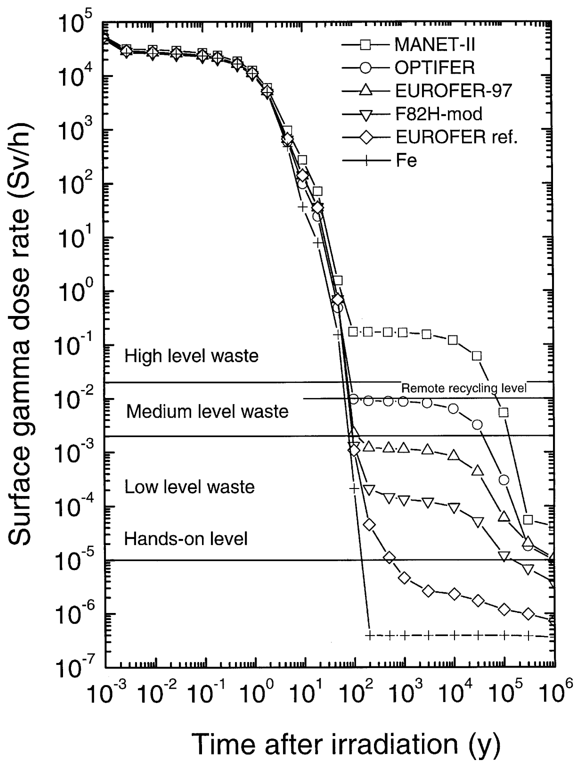

Fig. 1. Surface γ-dose rate for fusion-developed ferritic steels, after irradiation corresponding to 12.5 MWa/m² first wall (from [15]).

from Fig. 1 [15], this target has been already met by EUROFER-97, and the planned further reduction of certain impurities (impacting only on the cost of the material production) would further reduce the long-term activation [16].

The progress represented by EUROFER – albeit tested so far only up to 2.5 dpa – was obtained by the lowering of the ductile to brittle transition temperature (DBTT) by about 50 °C compared to, e.g. F82H. The DBTT tends to rise under irradiation, but this control of this feature has now also been improved: the best data currently available for fusion-developed materials refer to F82H, where the rise of the DBTT exhibits a first saturation after some 2 dpa. We expect to have EUROFER irradiated up to 35 dpa by April 2002, and up to 70–80 dpa by December 2004. Post-irradiation tests will be conducted in the year following each of these milestones. From experience with fast-breeder steels of similar composition, it seems plausible that EUROFER should be capable of maintaining adequate properties up to the 150 dpa target considered desirable for power plant use. The irradiation tests in fission plants, together with an improving modelling capability for radiation damage, will give confidence and will be useful for fast screening of further candidate materials. However, a conclusive proof of the suitability of these materials for use in fusion power plants will require corresponding tests with a fusion-like neutron spectrum, to ensure the relevant ratio of the two damage processes (atomic displacement and He-production).

The mechanical properties of materials at high temperature are decisive for the permissible coolant temperatures and hence the thermodynamic plant efficiency. Tests with oxygen dispersion strengthened (ODS) EUROFER alloys have indicated that these materials would allow a 100 °C increase in the operating temperature and that their development and testing for power plant suitability could be accomplished in time for DEMO, and possibly even for some test blanket applications on ITER.

On the time horizon of a second generation of power plants, we are striving to develop structural materials, which allow hands-on manipulation of the material, or exemption from radioactive waste regulation, after a 100-year cooling time, combined with blanket operating temperatures in the 700–900 °C range. Vanadium based alloys and, in particular, SiC/SiC composites are the prime candidates.

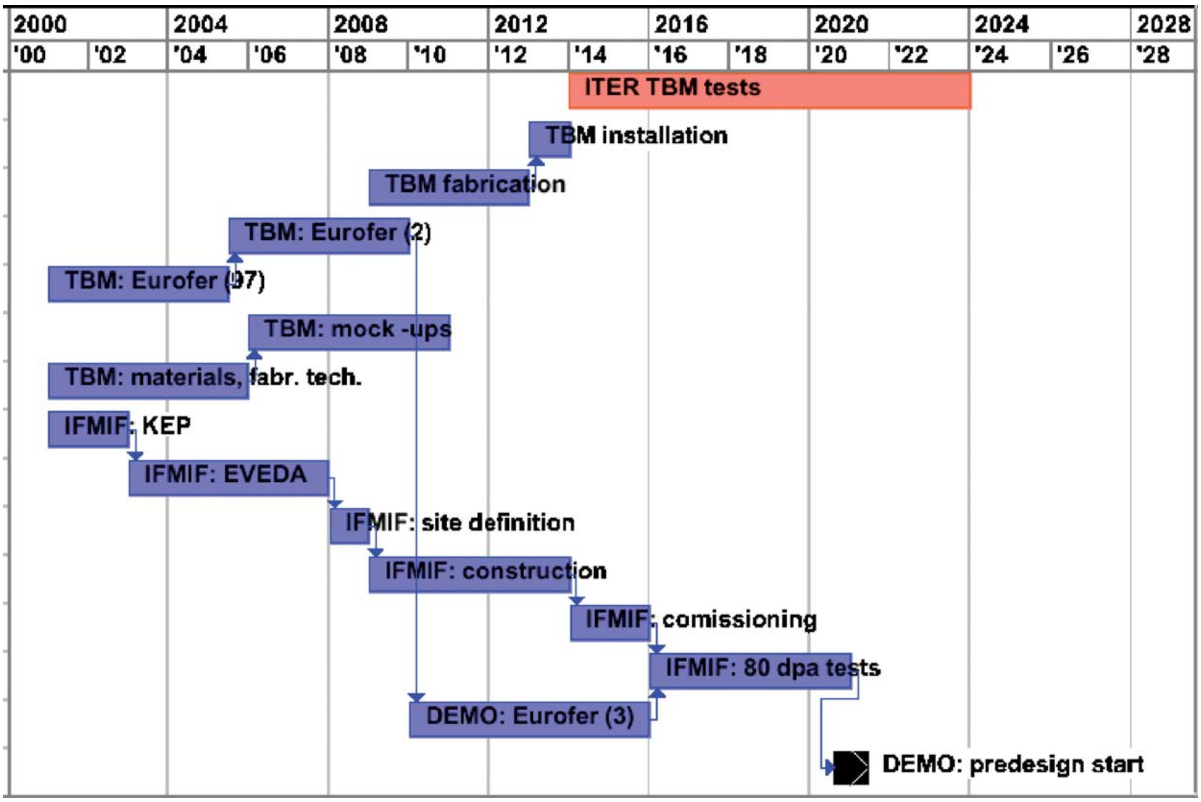

Fig. 2. Development plan of the present long-term technology programme.

3. Development plans

3.1. Present long-term technology programme

The blanket tests on ITER should start with the operation of this device. A well-defined timetable for the R&D needs and the fabrication schedule has been identified for the TBM. These modules will be fabricated from a EUROFER alloy, but considering the relatively low neutron wall loading and fluence with respect to DEMO, their qualification to power plant fluences with a fusion spectrum is not required. However, to be ready to proceed with the design of DEMO, the qualification of the materials up to a neutron fluence constituting a meaningful minimum target for DEMO-operation (80 dpa) should be accomplished by the time the critical ITER information on fusion physics and technology integration becomes available. This necessitates the timely availability of a fusion-relevant irradiation source, with the characteristics of neutron flux, spectrum and irradiation volume such as provided by the IFMIF [16] project.

The development of the TBMs, including irradiation tests of two development stages of EUROFER in fission plants, R&D on the materials fabrication techniques, and the fabrication of mock-ups, as well as the radiological qualification of a possible third development stage of EUROFER to DEMO specifications in a suitable fusion relevant neutron source, constitute the two legs of the present long-term technology programme (Fig. 2). To comply with this timetable, the present key engineering planning phase of IFMIF would have to be followed up by an engineering validation engineering design activity during FP 6, with construction starting in FP7.

3.2. A road map to commercial fusion power production

Starting with the ITER construction and operation schedule outlined in the ITER-FEAT outline design report [7] and the long-term technology planning, one can attempt to construct a roadmap leading the achievement of the milestones defined in the introduction as quickly as possible. Evidently such an attempt must be based on the assumptions that no significant delays occur in decisions dependent on the political process and that all experiments and R&D work can be brought to a successful conclusion in the estimated time. The baseline schedule, presented in Fig. 3, attempts to minimize risk by proceeding with the major part of the design of each device only after all required results from the preceding device generation are available. For the timetables shown here the hydrogen, deuterium and low performance DT operating phases of ITER were

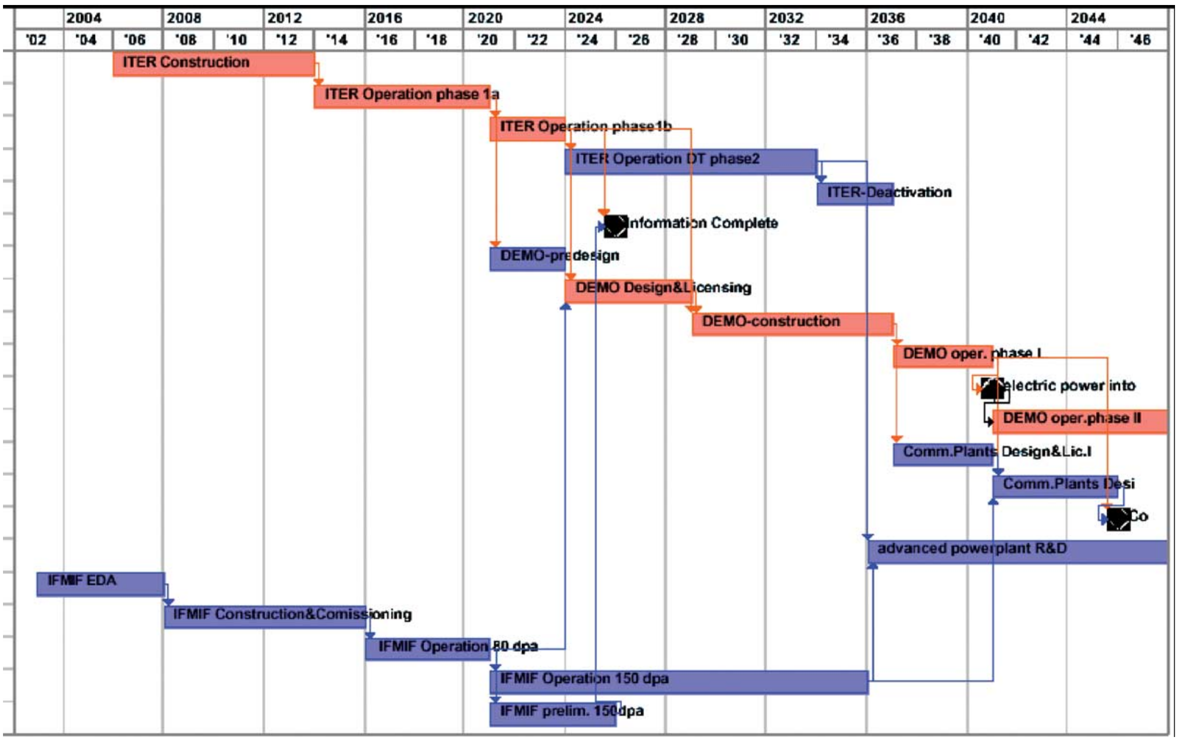

Fig. 3. Road map to commercial fusion power production; reference case.

integrated as phase 1a, with the high performance DT part of phase 1, labelled 1b. In this case, the entire ITER phase 1, together with the full 80 dpa phase of IFMIF, is considered necessary input for the final design (including licensing) phase of DEMO, which is taken to last about 5 years. The determining issue is that the results of the blanket tests on ITER, required for the selection and the final design of the internal components, will only be available at this time. In line with the documented ITER planning, however, the essential physics information is assumed to follow from the first 7 years of ITER operation, and to be available for the start of a DEMO conceptual design phase three years earlier.

The DEMO construction phase, like that of ITER, is assumed to take 8 years, with a further 5 years are taken required to bring the device to full performance. From this moment onwards DEMO should regularly deliver power to the net, with a power-plant relevant energetic efficiency, although probably with a reduced availability.

The European R&D programme is based on the concept that DEMO will be very close to a commercial power plant. The design of a commercial power plant could therefore start immediately as a continuation of the DEMO design work. The second phase, and the licensing activities, would probably commence only when the first operating experience on DEMO and the material tests on IFMIF up to an economically meaningful neutron fluence (150 dpa) have established that an attractive availability can be achieved.

Defining three significant milestones as:

(I) availability of all information needed for construction of a power plant (i.e. demonstration of at least one workable solution to all problems);
(II) first net power delivery into the grid (implying prior full coverage of site power requirements by fusion produced electricity) in a plant which is also fully autonomous in its tritium supply (except for start-up);
(III) start of construction of first commercial power plant;

this schedule would realize (I) by the end of 2025 (when first samples could have been irradiated on IFMIF to 150 dpa, and ITER had concluded operation phase 1), (II) end of 2040, and (III) end of 2045.

The critical path to the demonstration of net power generation (milestone II), and the construction of a commercial power plant (milestone III) in this road map is determined by ITER construction and operation, followed by DEMO design, construction and operation. There is a nominal slack of 3 years in the requirement for IFMIF to meet needed targets, but this time is to be viewed as short compared to the uncertainties in the start of construction. IFMIF operation is on the critical path for milestone (I), as the attainment of 150 dpa could not be achieved before 2025, even for the first batch of samples.

The road map to power production could be accelerated, if a more aggressive design and construction schedule for DEMO were to be implemented. For Fig. 4 it is essentially assumed that construction of DEMO starts before the design of the components internal to the vacuum vessel, in particular the blanket, is finalized. This removes the completion of phase 1b of ITER (involving completion of the tests needed for the qualification of the DEMO blanket design) from the critical path. Implicit in this road map is the concept that the design of the interior components can be sufficiently separated from the rest to make possible the licensing of the basic device and the start of construction before the completion of their design. A second acceleration is obtained by assuming a more aggressive progress of DEMO to full operation, with 3 (rather than 5) years allowed to proceed from the start of operation to net power production.

This accelerated schedule assumes that ITER operation serves primarily to confirm the basic design assumptions of DEMO concerning DT and, in particular advanced mode and steady-state operation. Based on today's know-how in these areas, this would not be possible and so this approach implies intensive physics work parallel to the construction of ITER.

For the further progress to a commercial power plant, the similarity of this device to DEMO is exploited to carry out much of the design and the start of the licensing work in parallel with DEMO design and construction, waiting only with the last 3 years of design and licensing for the confirmation of the successful first operating phase of DEMO, coincident with the first power delivery into the grid.

This schedule would achieve milestone (II) at the end of 2034, and milestone (III) at the end of 2037. Milestone (I), which is critically linked to IFMIF operation, could be moved ahead by a more aggressive construction/commissioning/operation schedule of this device to the end of 2023. With this accelerated schedule for IFMIF, the critical path to the demonstration of electric power and the start of construction of commercial power plants again passes through ITER and DEMO, whereas the completion of necessary information hinges on both the ITER and IFMIF schedules. The time-decisive element of ITER operation is the need to execute the testing programme for the TBM. In a compressed schedule, concentrating on the mandatory experiments, this could probably be carried out in about 2–3 years, among which, however, certainly a major fraction during the high duty (long pulse) phase 1b. This implies that only a more accelerated progress in the physics programme towards this high duty phase – itself scheduled for only 3 years – could lead to substantial savings in the overall time-scale. Assuming a reduction of 1 year in each of

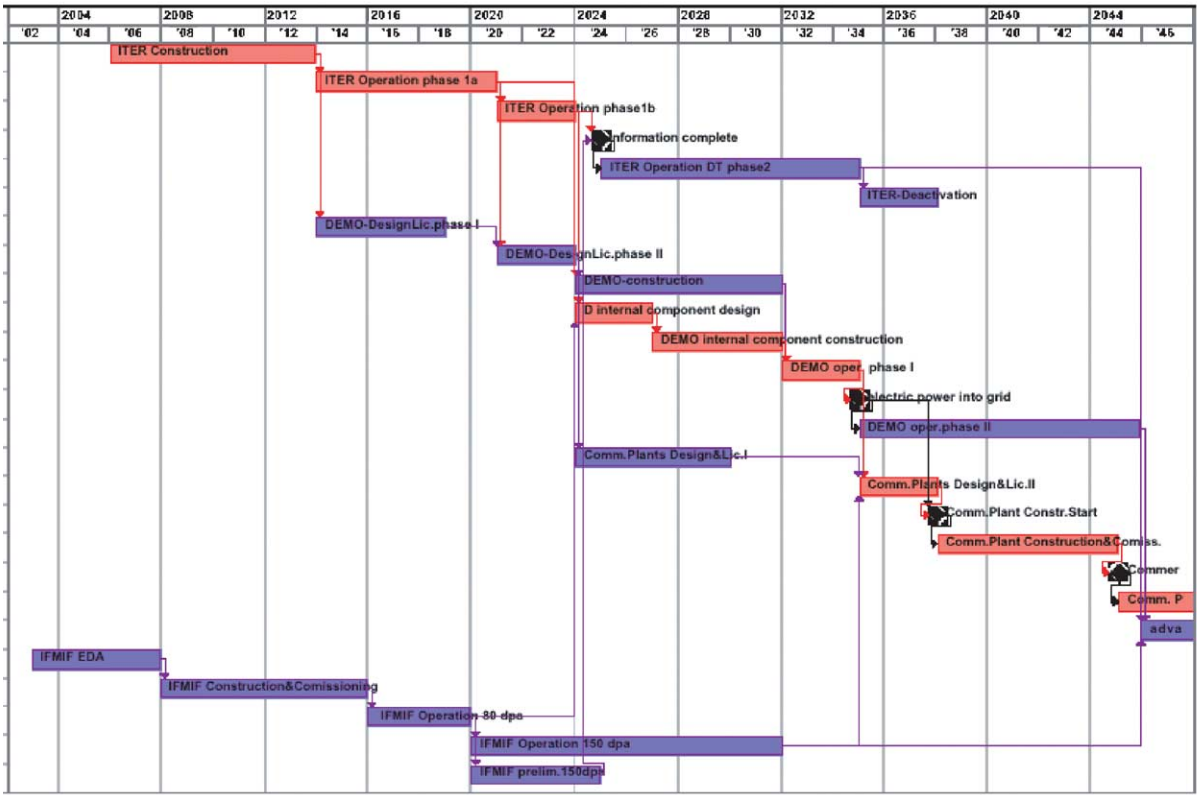

Fig. 4. Road map to commercial fusion power production; accelerated schedule.

phases 1a and 1b would advance milestone II and III by only one year, as IFMIF would again become critical.

The European fusion physics programme also includes a strong effort in other toroidal confinement schemes. Assuming that there is also vigorous progress in the theory of toroidal confinement, the results of the new or upcoming devices, in combination with the burning plasma physics learned from ITER, could still make them an option for a burning plasma technology test bed or first generation power plants. As a case in point, we take here the stellarator and examine the option of making direct progress, after W7-X, LHD and ITER to a stellarator-based DEMO, and evaluate the corresponding impact on the time-scales. For this to be feasible, we must assume the development of an adequate, quantitative theoretical understanding of toroidal confinement that allows the omission of the step involving a DT-burning stellarator, which is substituted by computer simulations. This is a strong, but not unrealistic, assumption. To provide credible extrapolations, these theoretical models would firstly have to be capable of explaining fully the complementary sets of experimental observations obtained, on the one hand, in the non-DT-burning stellarators of the LHD and W7-X generation and, on the other, in DT operation on ITER. The roadmap shown in Fig. 5 therefore explicitly exhibits the development of toroidal confinement theory, which is already ongoing, and foresees a second theory phase – on the critical path – in which the experience arising from operation of the large stellarators and from physics operation with DT on ITER is digested to form the basis for the layout of a stellarator-based DEMO. A sufficiently long operating experience on the LHD/W7-X stellarator generation is also necessary input for this phase, but is much less time-critical than the operating experience with ITER.

Under these important and critical assumptions, the delay relative to the reference scenario for first power production (milestone II) and construction start for commercial power plants (milestone III)) would be surprisingly modest (3 years), essentially because ITER-operation phase 1b – serving to complete the input for the DEMO-blanket design – would be removed from the critical path. Significant additional, stellarator specific, R&D work would be needed, which does not, however, appear here as time-critical, as it could be started relatively early, as soon as warranted by positive results from LHD/W7-X. Milestone (I) would assume a different meaning in this context, as it would be achieved at the same time as under other scenarios (2025), but would

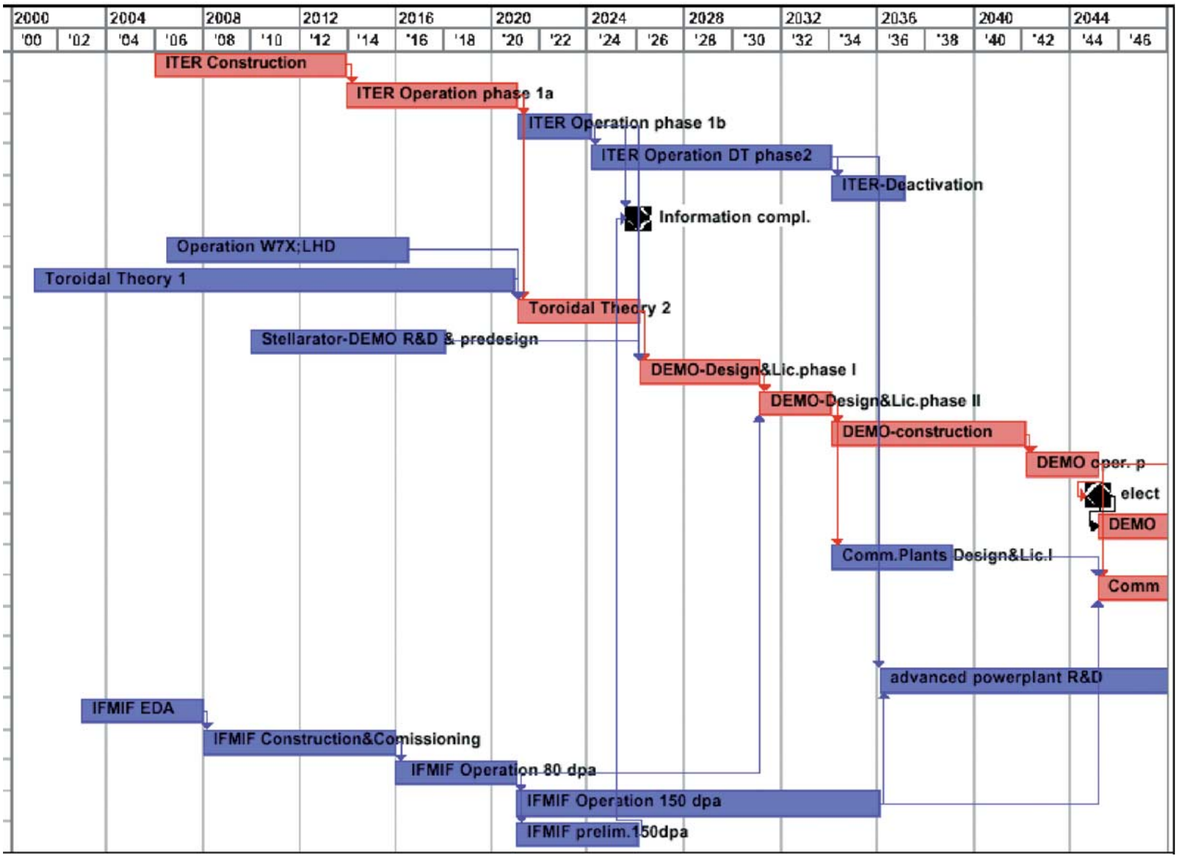

Fig. 5. Road map to commercial fusion power production in a stellarator-based power plant.

imply only the completion of the evidence for one option (the tokamak) which would not be the option further pursued with DEMO. Evidently any scenario requiring a DT-burning stellarator as an intermediate step to a DEMO (either because a tokamak-ITER were not constructed, or because progress in theory did not warrant the extrapolation described above) would lead to a very substantial delay in the time-scale compared to the schedule outlined here.

Besides the assumptions explicitly stated here, a number of other pre-conditions have to be met to make the above roadmaps feasible or realistic targets. One concerns, for example, the further availability of fission reactors for continued material tests, in preparation for, and as a complement to, IFMIF. A second concerns the issue of the tritium availability for start-up, which has to be looked at in the light of the phase-out of existing tritium sources. Scenarios for start-up from zero inventory have been developed, but are probably not completely realistic as they assume that all tritium separations systems work with vanishing inventory.

All three facilities considered in the road map (ITER, DEMO, IFMIF) should continue operation after having fulfilled the critical-path functions outlined above: they will provide information useful for improvements in the efficiency of succeeding devices during the construction and operation phases of such devices, and will serve as development and test platforms for second generation ('advanced') power plant designs.

Care should be taken when interpreting the 'slack' exhibited for some tasks. In the case of the ITER/DEMO and IFMIF schedules, the time intervals implied are small compared to the uncertainties in design and construction schedules, as well as to the possible delays in the decision-making process. In addition, the periods between the completion of one task and the start of a subsequent one would always be filled by work aimed at further reducing uncertainties in the extrapolations and at preparing for steps still further ahead. This is well illustrated by the case of the stellarator reactor studies, where it is indeed probable that critical information required from the LHD/W7-X generation of devices for the planning of a stellarator-based DEMO-reactor would be available earlier than the information needed from ITER, but where the intermediate years would be well spent in carrying out experiments on both devices which would test salient elements of the theoretical models which must be developed.

4. Conclusions

The energy supply scenarios developed within the European SERF programme have shown that electricity generation from fusion power could make a significant contribution to the control of green-house gas emission in the second half of this century. The R&D work undertaken so far, in the Associations, in JET and in European industry, have put us into a position where the technical preconditions for implementing such a role for fusion could be met, provided we maintain the acquired momentum. The future developments need a coordinated and balanced effort in the areas of physics and technology R&D and at their interface. On the critical path are the study of burning plasma and the integration of such a plasma with the technologies required for steady-state operation (including tritium breeding) and the development and test of suitable materials. Development plans leading to the demonstration of fusion power in the mid twenties, delivery of net power to the grid in the mid thirties, and large-scale commercial deployment by mid century require, as critical elements, the timely start of construction of ITER, followed with only small delay by that of a fusion irradiation test facility. These requirements are robust, and would persist even if one were to aim at a DEMO/first generation power plant following a different toroidal confinement principle, such as the stellarator.

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