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controlled fusion research

NUCLEAR FUSION DEVELOPMENT IN EUROPE

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CONTENT

A separate document on long-term programme will be prepared later jointly by the leaders of the NET group and of the INTOR European delegation.

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INTRODUCTION AND ABSTRACT

In several Decisions concerning the fusion programme the Council of Ministers of the European Community has recognized that "controlled thermonuclear fusion could be of benefit to the Community, particularly in the wider context of the security of the long-term energy supply."

It is trivial today to emphasize the importance of energy, nevertheless a few facts are worth mentioning:

- In the last year the European Community consumed 10^7 GWh ($\sim 10^9$ toe) of primary energy an important fraction of which was used for the generation of 1.2×10^6 GWh of electricity.

- The figures are similar for the USSR while they have to be doubled, approximately, for the U.S.A. and halved for Japan.

- The annual primary energy consumption in terms of GWh/km^2 of land surface was 7 for the Community in 1979, while for U.S.A. the figure was 2 and 12 for Japan; for comparison the average annual geothermal flux over the earth's surface is 0.5 GWh/km^2 and the solar flux is 3000 GWh/km^2.

In the EC, the present expenditure for imported fuel (oil, gas, coal) is about 250 MEUA per day.* This situation can only become worse: in the short term due to well known political difficulties; in the long term due to a short-fall in resources.

In the long term, apart from coal which (with an improvement in methods of extraction and utilization) could last a long time, there are only three possible energy sources; solar, fission breeder and fusion. Solar energy is certainly abundant even if its large scale utilization needs

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* which is somewhat more than the total annual expenditure for fusion research in the European Community.

an equally large scale development. Unless one takes into account the use of geo-stationary satellites the cost of solar energy will be dependent upon latitude, climate and the required power density. All these conditions appear rather unfavourable for the EC on average.

Fission breeder reactors already exist, but require even further improvements, and their real or imaginary ecological disadvantages are certainly amplified by the density of population. Fusion is the most uncertain and least developed of the three, but in principle has potential advantages which could be particularly valuable for Europe.

The primary fusion fuels (D, Li) are widespread and cheap (1 g natural lithium could produce 15 MWh); both those fuels and helium, the final product of the reactions, are stable. The fusion reactor could be made very safe from the nuclear point of view and the doubling time for breeding new fuel could be very short. However, these advantages only exemplify the fusion system's potential, it is not known presently to what extent they could be realised. The cheapness of the fuel could be offset by the cost of construction, the ecological attractiveness by the difficulty of containing large quantities of tritium within the system and by the activation of the structure. Engineering constraints could limit severely the attainment of a satisfactory breeding ratio. It is very difficult to make a prognosis today. The fusion-fission hybrid system could be exploited, if required, to provide a solution to fuel problems arising in the fission reactor industry.

Fig. 1 shows a schematic arrangement of a possible fusion reactor. Every small part of the picture contains difficult problems - some of them, such as those involving the plasma, are completely new, others, for instance in the field of materials, are quantitatively at or beyond the limits of present technological development. So today it is difficult or impossible to make an accurate estimate for the economic and social cost of fusion energy. Nevertheless, the scale of the energy problem is so large that even small differences can produce enormous advantages. It seems reasonable, therefore, to promote the development of the three major options up to the point where realistic comparisons are possible, and in any case, it is very important to secure enough diversification among the main energy resources for the future.

In the United States, in spite of the uncertainties and difficulties outlined above, fusion has recently been declared a national objective. Following the recommendations of the 'Buchsbaum Panel', and their endorsement by Congress in a Public Law signed by President Carter in October 1980, a further acceleration of the magnetic fusion programme can be expected. In particular, the construction in the 80's of a fusion engineering device (FED) is part of this national objective and this might be opened to international participation.

Japan has shown its confidence in going ahead with enlarging its own programme and has also made heavy financial investment in the U.S.A. experiments.

In Europe, from the start of Euratom, fusion has had a significant role in the Community research programme. Today it is the most important part. From the very beginning, Euratom decided, instead of trying to build its own fusion laboratory, to support and coordinate the activities of the national laboratories which were just being established in the member States.

In 'the Sixties', the main problems were connected with plasma confinement and to a lesser extent with heating. The nature of the research work was basic and exploratory, in this situation it was not wise to build very large machines, and it was normal for the initiative to come from the research teams in the various associated laboratories. The role of the Commission was to help avoid, where reasonable, the duplication of experimental ideas; to promote the exchange of "know-how" particularly on problems of general interest such as diagnostics and the techniques connected with the construction of apparatus. It should be remembered that during these years the experimental results gave little reason for optimism: the open configuration was demonstrating its limitations and the results coming from stellarators were not encouraging. In Europe the existence of a common programme helped to overcome this period of depression and it was during these years that most of the laboratories presently active in the fusion field were established and extended.

The end of 'the Sixties' was characterized by the success of the Tokamak in the U.S.S.R. So, from the programme point of view, both in the U.S.A. and in Europe the problem was to fill the gap and to extend the Russian results. In order to encourage and help the laboratories in the construction of the machines, the idea was conceived, at this time, to provide Preferential Support for projects involving large capital investment.

The necessity to cover a wide range of parameters meant that a number of different experiments had to be built, and in order to make all these diverse experiments available to all laboratories, the principle that each machine should be accessible to all partners was specifically mentioned in the Council Decision. This principle was simultaneously at the origin of the arrangement for the easy transfer of staff under the Mobility Agreement.

As a consequence of the concentration on Tokamaks, it was necessary to abandon some of the alternative lines, those remaining (stellarators, reversed field pinch) being exploited in a few selected laboratories. It became evident, certainly for stellarators and almost unavoidably in tokamaks, that strong additional heating methods would have to be employed. Therefore, in order to ensure success in the development of adequate systems, it was agreed that heating techniques should be developed jointly and the tasks distributed between a few special laboratories. All this gave the programme a greater Community character and also required the Commission and the programme's coordinating bodies to undertake a larger responsibility for the management and coordination of these parts of the programme. This character has been recognized officially in the words of the Council Decision, stating that the fusion programme shall be:

> "a long term cooperative project embracing all work carried out in the Member States in this field, designed to lead in due course to the joint construction of prototypes with a view to their industrial production and marketing".

"The Seventies" have been fruitful for fusion in Europe as well as world-wide. In Europe, as in other programmes, many Tokamaks were built: some of them (TFR; FT) to extend the parameter range and performance, others for special purposes. The latter include: DITE and more recently ASDEX as divertor experiments; PETULA, WEGA, ERASMUS and DANTE for RF heating and refuelling studies, and TEXTOR, a Tokamak built specially for the investigation of plasma/wall effects. Some of these experiments have been conducted as collaborative ventures between several associated laboratories. The culmination of such cooperative activities has been the design and the start of construction of JET. JET was the first to be proposed of the large machines (JET, TFTR, JT-60) now under construction, and was, from the start intended as a combined effort by all the associated laboratories. Good progress has also been made in the alternative lines, the stellarator and reversed field pinch. Another effective area of collaboration has been in the mutual exchange of heating and diagnostic equipment between various associations.

In general therefore, concerning the applied plasma physics one may say that the objectives have been reached, even if there have been some occasional delays. The overall situation in this field is satisfactory, though we no longer hold some of the records held earlier in the decade. This achievement has been in spite of the fact that in Europe fusion enjoys weaker support than that in the U.S.A. and Japan, and in spite of the long delays in gaining a Council Decision (2 1/2 years for JET, 2 years for the current programme), which introduced difficulties in the management and execution of the programme.

In the area of Fusion Technology, the enormous magnitude and diversity of tasks make the choice of priorities difficult and it is not easy for the traditional plasma physics laboratories to divert a part of their effort in this direction.

Added to this are the problems of involving those relevant areas of expertise lying outside the fusion associations and in some cases of facing political difficulties. For these reasons, the situation for technology appears less than favourable, though mention must be made of the exception in superconducting magnet development where there is a strong and forward looking collaborative activity.

For the 1980's, the major commitment will be the completion of the construction of JET, and its operation in extended performance. Hopefully, this will enable ignition conditions to be reached, but in any case a substantial extension of the parameter range will be achieved. Concerning the nature and exact timing of the various phases of the JET experimental programme, the appropriate decisions can only be taken according to the circumstances pertaining at the time, but everything should be provided to prevent any further delays. The collaboration between JET and the associated laboratories has to be improved. Psychological and material conditions should be created so that each laboratory can consider JET to be its major experimental machine, while at the same time recognizing that JET has its own well defined function to perform.

In addition to JET, the laboratories should have to study the behaviour of plasmas with their existing research facilities (possibly upgraded). The aim should be to gain further understanding and improvement of characteristics such as $n\tau$, β and purity levels, as well as develop diagnostics and heating techniques. Any proposals for the construction of further medium-large size experiments should be carefully examined, taking into account among other things, that the construction and operation of such experiments would engage, for 10-15 years, a large fraction of the skilled manpower and resources of the laboratory concerned.

Some alternative lines, possibly in an international context, should be explored further to the point where their potential can be assessed.

Further common ventures will be encouraged, and even the larger laboratories must now envisage the possibility of devoting a large part of their effort to a joint project which might be realized elsewhere.

Technology is more connected with medium and long term plans. A few years ago the Commission prepared a Long Term Plan containing a detailed description of technological problems and identifying questions to be solved before undertaking the construction of a demonstration fusion reactor. As shown in this Plan, we still think that the effort available in the technology area should be focused on the construction of the Next Step, i.e. the 'Post-JET' machine. In 1978, to this end, we set up a NET group (Next European Tokamak) aiming at the exploration of a conceptual design. This practically coincided with the start of INTOR (International Tokamak Reactor), for the design, construction, and operation of a very large Tokamak, as near as possible to a real fusion reactor, as a joint venture between the four large world fusion programmes (EUR, Japan, USA, USSR) under the aegis of the IAEA. It was decided to give priority to the European participation to INTOR and therefore the NET group has, so far, only been developed in order to support and supply the input information to INTOR. In the U.S.A., priority has been given to the national project (ETF, now FED) and the U.S. support to INTOR has been provided as a by-product of this activity. The evolution of the present world political situation casts some shadow over the future of INTOR and shows the fragility of our position.

Therefore, even for the benefit of our contribution to INTOR, we have to re-inforce the NET group and concentrate on providing a strong nucleus of design effort at one of the associated laboratories or at Ispra, in either case fully supported by all the associations. The appropriate experts should be made available by the associations, in addition to those who will be released from the JET design office as time progresses and those from non-associated centres and industry who have the essential expertise not presently available within the fusion Community. The NET group's designs should provide the target and motivation for the technological developments in the next decade.

Such a decision should not pre-judge the final one of whether to build NET as a strictly European venture or as an international project with other (one, two or three) partners. It is therefore urgent to explore fully the different possibilities of real international collaboration.

This document gives an overall picture of the problems in the field of fusion, of the present state of the art, of the organization and content of the running European programme (1979-83), and of a possible path towards our main medium-term objective - the construction of a Post-JET Tokamak. A discussion of long-term programmes (from Post-JET to the Demonstration reactor) will be presented under separate cover.

The document has been prepared by the staff of the Fusion Directorate in Brussels together with the assistance of experts from the Laboratories. Complementary information can be found in the Annexes, part of which has been prepared by the Laboratories.

I FUSION AS AN ENERGY SOURCE

1.1 FUSION REACTIONS AND REACTOR SCHEME

Fusion is a process in which the atomic nuclei of light elements, such as hydrogen isotopes, combine to form a larger nucleus; the reaction is exothermic. The most accessible of these reactions is :

$$ {}^{2}_{1}D + {}^{3}_{1}T \longrightarrow {}^{4}_{2}He\ (3.5\ \text{MeV}) + {}^{1}_{0}n\ (14\ \text{MeV}) $$

It has a maximum cross-section of 5 barns for a centre of mass energy of about 60 KeV. Tritium, because it is unstable with a small half-life (13 years), must be generated by supporting reactions, such as :

$$ {}^{7}_{3}Li + {}^{1}_{0}n \longrightarrow {}^{4}_{2}He + {}^{3}_{1}T + {}^{1}_{0}n - 2.5\ \text{MeV} $$

$$ {}^{6}_{3}Li + {}^{1}_{0}n \longrightarrow {}^{4}_{2}He + {}^{3}_{1}T + 4.8\ \text{MeV} $$

The primary fuels for a reactor (Fig. 1) based on this scheme would then be deuterium and lithium; they are both abundant and cheap. Tritium would be bred inside the reactor, in a lithium blanket surrounding the fusion region. For natural lithium, the breeding ratio, ignoring losses, is ideally about 1.5 and could be improved by the use of (n,2n) reactions, in beryllium for instance. The fuel doubling time could then be expected to be very short (months to years). Fusion energy would be extracted from the blanket through heat exchangers.

Many other light element fusion fuel cycles are possible and can be classified as deuterium or hydrogen based. The deuterium based fuel cycles include the reactions D(D,n)^{3}He, D(D,p)T, ^{3}He(D,p)α and D + ^{6}Li, which has many branches. Examples of proton based cycles are ^{6}Li(p, ^{3}He)α and ^{11}B(p,α)2α. All these cycles are based on commonly available isotopes, (except for ^{3}He) but will require higher ion energies than D-T and better confinement. Reactors based on such cycles could offer advantages related to less complex blanket designs, lower (or non-existent) levels of tritium in the gas handling system, improved material compatibility and lifetime, increased plant lifetime and maintainability, and decreased levels of long-term induced radioactivity. But the more stringent physics conditions required make them very remote, so that present technological developments are all based on the D-T reaction.

1.2 IGNITION CRITERIA

Beam-solid target fusion reactions, which are of common use in neutron sources, have been shown to be inapplicable to reactors: their overall efficiency is smaller than unity for fundamental reasons. The only possible way seems to be heating the fuel (for instance a 50/50 mixture of DT) to a temperature high enough for the fusion cross-section to reach a sufficient value. At these high temperatures the fuel is a fully ionized plasma.

For sustained fusion reactions to take place, the fuel should be heated and maintained at a temperature near the "ignition temperature" (temperature for which the power delivered to the plasma by the alpha particles produced by nuclear reactions balances the plasma losses by radiation, diffusion, etc.). The ion temperature T_i obviously depends upon the plasma losses, usually characterized by the so-called "plasma energy confinement time" τ. A convenient figure of merit indicating the quality of plasma confinement is the product $n\tau$, where n is the plasma density. In the $n\tau$, T_i plane, the ignition curve (Fig. 2) has a minimum which, for D-T reactions, puts $n\tau \simeq 2.10^{14}$ (ions, cm^{-3}, sec) and corresponds to an ion temperature of about 30 KeV.

Ignition then requires simultaneously:

$$T_i \geqslant 10 \text{ KeV}$$
$$n\tau \geqslant 2\text{-}5.10^{14} \ (cm^{-3}, \text{sec}), \text{ depending upon } T_i.$$

This ignition criterion is similar, but not numerically identical, to the well-known Lawson criterion.

To maintain such high temperatures for appreciable periods of time, the plasma must be kept far from outer cold boundaries. Apart from gravitational confinement, which is not accessible to man, there are two known mechanisms for confining a thermonuclear plasma:

- inertial confinement, already used successfully in the H-bomb ;
- magnetic confinement.

1.3 INERTIAL CONFINEMENT

In this approach, plasma temperature and density are raised so rapidly that a significant fraction of the fuel reacts before it has time to disperse. In principle, this can be achieved through the uniform illumination of a spherical pellet containing DT with intense laser beams, electron beams, light or heavy ion beams; the outer layers of the pellet are ablated, compressing and heating the core. As the "confinement time" here is proportional to the pellet radius R, the ignition criterion gives a minimum value for nR, which leads to a minimum value for the pellet energy W_o (proportional to nR^3) at the beginning of explosion. Very roughly:

$$W_o > \left| \frac{n_s}{n_o} \right|^2 \text{(Gigajoule)}$$

where n_o is the pellet density at the beginning of the explosion, and n_s is the solid state density ($n_s = 5.10^{22}$ cm^{-3}). For W_o to be not too large, very high compressions are mandatory (at least 100 times solid state density). Moreover, as W_o should be transferred to the pellet core in a time shorter than the explosion time, beams of extremely high power are required.

1.4 MAGNETIC CONFINEMENT

1.4.1 General

- Charged particles, and thus plasmas, can move along magnetic lines of force, whereas their transverse motion is due to diffusion processes resulting from interactions between particles. Neglecting diffusion, a magnetic field (intensity B) can thus confine a plasma (pressure p) in the directions perpendicular to B. The ratio between plasma and magnetic pressures

$$\beta = \frac{p}{B^2/8\pi}$$

characterizes the efficiency of the utilization of the magnetic field for plasma confinement. In practical devices, Beta values are mainly limited by stability requirements. A high Beta is desirable because the magnetic field is one of the expensive ingredients in experimental devices and future reactors. For an economic power reactor it is estimated that minimum Beta-values of 5-10% could be needed. This fact, together with the technological limitation of quasi-static magnetic fields to about 100 kG, leads to densities of the order of 10^{14} - 10^{15} cm^{-3} for thermonuclear plasmas. The corresponding confinement times (ignition criterion) are $\tau \simeq$ 1-10 sec.

- The most obvious mechanism of plasma diffusion across the magnetic field is provided by binary collisions, mainly between electrons and ions. On this basis, the confinement time should be proportional to $\frac{B^2a^2T^{1/2}}{n}$, "a" being the transverse plasma dimension. At thermonuclear temperatures, this would lead to comfortable large values of τ, even for small devices. Unfortunately, experimental results point to much shorter confinement times, due to enhanced transport phenomena. There is not yet any satisfactory theoretical explanation for this phenomena. On Tokamaks, an empirical scaling law of the form $\tau \sim na^2$ seems to fit with experimental results, although its limits of validity could already have been exceeded in recent high density experiments.

- In order to suppress, or at least to reduce, the leak of plasma along the field lines, there are two main possibilities: to use magnetic mirrors, or to use toroidal devices.

1.4.2 Magnetic mirrors

The ends of a linear system can be plugged by magnetic mirrors (Fig. 3), which are regions of increased magnetic fields. Such a plug is not perfect: particles whose velocity has a large relative component along the field lines escape from the system. Moreover, a simple mirror device is macroscopically unstable. But a succession of new ideas has shown, either experimentally (non-axisymmetric coils to provide a stable minimum-B configuration, tandem mirrors) or at least theoretically (thermal barrier) that end losses can be decreased in such a way that mirror machines could lead to a viable reactor concept. Leakage from the ends of the device, characteristic of open systems, is both a drawback (large losses) and an opportunity (possibility of direct conversion of leaking charged particles to electricity; exhaust of helium, etc.). Another approach to the problem of leaks is the "Bumpy Torus" concept: many simple mirrors are arranged end to end around a circle, so that plasma leaking from the end of one mirror can be trapped by the adjacent mirror. Such a configuration is unstable, but it can be stabilized, at least at low β , by inserting local relativistic electron rings in each of the mirror cells.

1.4.3 Toroidal devices

Here the end losses are eliminated by bending the field lines to close on themselves within the vacuum container. The simplest toroidal magnetic system is that generated by a set of coils wound regularly around a torus. But this toroidal magnetic field is necessarily inhomogeneous and charged-particle drifts lead to the absence of plasma confinement. The drift can be overcome by the addition of a second magnetic field (the poloidal magnetic field) circling around the minor axis of the torus. This gives the resultant field lines a twist so that they constitute a system of toroidally helical field lines lying on nested toroidal magnetic surfaces which form a confinement system. The central line around which the magnetic surfaces are wrapped is called the magnetic axis. The amount of twist of the magnetic lines is characterized by the "safety factor" q, which is the number (not necessarily an integer) of turns a field line makes around the main axis of the torus when it makes one turn around the magnetic axis.

q is approximately given by :

$$q \simeq \frac{B_{toroidal}}{B_{poloidal}} \cdot \frac{\text{Minor radius}}{\text{Major radius}}$$

q plays an important role in plasma stability.

Depending upon how the poloidal field is generated and upon the magnitude of q, the various schemes of toroidal magnetic confinement can be classified. There are three main schemes:

1.4.3.1 The Tokamak is an axisymmetric device in which the toroidal field is constant in time and the poloidal magnetic field is produced by a toroidal current flowing in the plasma itself (Fig. 4). The plasma loop acts as the secondary winding of a transformer whose primary is fed by an external power supply. The primary windings, together with the "Equilibrium coils" which create a weak magnetic field approximately parallel to the main axis of the torus whose function is to control the position and shape of the plasma, are called the Poloidal Field Coils. For Tokamaks to work in steady-state, the plasma current would have to be maintained by other means than a transformer, whose flux swing is necessarily limited; theory and preliminary experiments show that this is conceivable. The value of the plasma current is limited so that inside the plasma q is everywhere larger than 1 (KRUSKAL-SHAFRANOV stability condition), or maybe 1/2 as is suggested by some new theories. This leads to small maximum Beta values (typically 5 to 10 %).

1.4.3.2 The <u>Reversed Field Pinch</u> (Fig. 5) is another axisymmetric toroidal configuration of the same class as the Tokamak but it operates with a poloidal field of the same order as the toroidal field. (The stabilizing toroidal field is thus much weaker and therefore cheaper than in Tokamaks). A particular characteristic feature is that the toroidal magnetic field component has the opposite sign in the outer regions of the plasma, i.e. is reversed, with respect to the field on axis. Because of this the resulting magnetic field possesses a high shear. The device can thus operate at high Beta. An important property of the RFP is that the plasma itself generates the stable high Beta configuration naturally by a process involving relaxation to a near minimum energy (RFP) state. The current is not restricted by stability considerations (as in Tokamaks), so it can be large and so much stronger ohmic heating can be obtained, which would lead to a major simplification. This, together with operation at lower fields and a free choice of aspect ratio (a more open toroidal structure is possible) could result in a less complex reactor design. Possible disadvantages are the use of some kind of conducting shell and the greater technological problems of sustaining large currents for long times, and ultimate DC-operation.

1.4.3.3 The <u>Stellarator</u> (Fig. 6) is a toroidal configuration with nested magnetic surfaces which can be created by external coils only. These coils are best described by usual toroidal field coils with some out-of-plane twist. For the sake of larger experimental flexibility, however, present day devices are equipped with two sets of coils, and a variety of distributions of the field components among the two coil systems are possible (classical Stellarator, Torsatron, Heliotron). In distinction to the Tokamak, the Stellarator is capable of operating without a net plasma current which in principle allows steady-state operation and does not require two independent coil systems. The Stellarator concept can also be combined with the Tokamak system, e.g. for improving the Tokamak plasma stability (suppression of disruptions). The Stellarator is not axisymmetric. This property could introduce some extra loss (not observed in present experiments) and gives rise to asymmetry forces but also offers the potential for greatly improving the overall confinement and perhaps even Beta.

1.4.3.4 Other configurations (Screw Pinch, Extrap, Compact Toroids, etc.) have been proposed and are currently being studied to a lesser extent.

1.5 MAIN PROBLEMS ON THE WAY TO THE REACTOR

These problems are reviewed hereafter in the framework of the Tokamak line, which appears today to be the most promising. Some of these problems (e.g. control of a burning plasma) require an absolute solution, whereas others (e.g. high-β) are connected with the efficiency of the reactor and are then problems of optimization.

According to present projections, a Tokamak power reactor will contain, in a toroidal field of about 5 Tesla, of the order of 1000 m^3 of D-T plasma (minor radius 2-3 m), at a mean temperature of 10-15 keV and a mean density of $1\text{-}5.10^{14}$ cm^{-3}. About 100 MW of auxiliary heating power will be required during the first 5-10 seconds in order to reach plasma ignition and self-heating. If the reactor operates in the pulsed mode, as present Tokamaks do, pulse lengths in excess of 100 sec, separated by short off-periods, will have to be achieved in order to avoid severe economic penalties. A reactor unit will produce at least 1 GW electric and burn a few Kg of tritium per day. The density of power generated in the plasma will correspond to a "wall loading" (power flux density through the "first wall" of the reactor chamber) of between 2 and 10 MW m^{-2}.

The main problems can be listed under the following headings:

- Confinement

The influence of plasma density, plasma dimensions, aspect ratio of the torus, shape of the plasma cross-section and q-value on the confinement time τ, has been studied experimentally on many devices. The effect of plasma temperature is beginning to be investigated, using the powerful auxiliary heating methods now available. Anyhow, as theory has not reached the point of providing reliable scaling laws, an experimental approach has to be followed, with devices of increasing size and cost.

- Heating

Efficient heating methods should be developed at the multimegawatt level. These methods belong essentially to two classes: radio-frequency heating, through the absorption of waves at various appropriate frequencies; neutral injection heating, in which the energy is fed to the plasma via intense beams of fast (100-200 keV) neutral particles (charged particles born outside the static confining magnetic field cannot be trapped in the plasma).

- High-β.

In reactor conceptual designs, β values of at least 5 to 10 % (mean value over the plasma cross-section) are considered to be necessary for insuring a reasonable efficiency of the whole system. Recent experiments give confidence that such a target is not unrealistic. Small-scale experiments are conceivable.

- Impurities

A small amount of impurity is sufficient to provide efficient radiation cooling of the plasma. Heavy impurities are obviously the worst. The deleterious effects of plasma-wall interactions should be reduced to an acceptable minimum, for instance through the use of a "divertor": bundles of lines of force located on the outer part of the plasma are taken out of the reactor chamber where the impure plasma they contain can be neutralized and pumped away.

- Long pulses

Mechanical fatigue in the reactor components imposes a limit to the number of pulses during the reactor life-time, and demands a

- minimum pulse-length (steady state operation would be ideal from this point of view). This suggests developing methods of sustaining plasma currents in a non-inductive way, for instance by travelling R.F. waves. The poisoning of plasma during long pulses by impurities and by helium reaction products can be studied experimentally only on very large and advanced devices.

- Control of a burning plasma

  At temperatures of 10-15 keV, the system is thermally unstable: an increase in temperature leads to an increase in reaction cross-section, and then to an increased heating. Regulation mechanisms and shut-down procedures have been proposed, but their experimental study should wait for the existence of future large devices containing hot D-T plasmas.

- Exhaust and refuelling

  During a long pulse, it will be necessary to compensate particle losses (diffusion and burning) by refuelling. Injection of fast solid D-T pellets or external gas-puffing have been suggested. The exhaust of charged reaction products requires an appropriate rate of diffusion of the helium from the central to the outer part of the plasma, and from there to the outside of the reaction chamber, for instance using a divertor.

- First wall

  This will be subjected to substantial radiation damage, as exemplified by 10-100 displacements per atom per year and 100-1000 atom-ppmHe/year expected for the reactor. Alloys with a lifetime of 15-40 MW-years m^{-2} will have to be developed and other attacks on the first wall integrity, e.g. by sputtering erosion or coolant corrosion, need to be overcome.

- Blanket

  For a tritium self-sufficient reactor, a breeding blanket containing lithium will be used. A breeding ratio of 1.02-1.05 will give a comfortable tritium doubling time. The blanket will also accommodate the primary coolant for heat transfer, with about 20 % of the total power being deposited at the first wall and 80 % in the bulk of the blanket. The required blanket thickness is about 1 m, but additional 0.5-1 m of shielding will be necessary to protect the superconducting coils from radiation damage and excessive heat deposition. Several choices can be made for the breeding material (liquid Li, Li-alloys and compounds, with or without ^{6}li-enrichment and/or neutron multipliers) and for the primary coolant (liquid Li, helium, molten salt, water), leaving a wide field for design and experimental investigation.

- Superconducting coils

  All magnetic fields will have to be generated by superconducting coils in a reactor, because of the otherwise formidable power supplies and power dissipation. Toroidal fields on the plasma centerline of 4-5 Tesla are anticipated, which may be beyond the potential of NbTi technology (8 Tesla maximum field at the coil edge) hence requiring the development of Nb_3Sn or other A 15-type superconductors. The bore of the toroidal field coils, usually designed in a D-shape, may be of the order 15/10 m height/width. Very large forces will be exerted on these coils, particularly due to the interaction with the time-varying poloidal fields, and in the case of single coil failure. The problems of AC conductor losses in the poloidal field system must be overcome.

- Tritium

  Large amounts of radioactive tritium will have to be handled in two different systems : the gaseous fuel cycle, with a tritium feed rate of a few Kg/day ; the tritium recovery system from the blanket, particularly critical because it involves tritium inventories of the order of 1 Kg at temperatures of several 100 degrees. Apart from the high reliability required for these systems, containment and decontamination techniques must be developed to ensure the safety of the operating personnel and of the public.

- Remote handling

  The induced activity of the reactor structure will preclude hands-on inspection, maintenance and repair, all of which will have to be done remotely. This appears to require a highly modular design of the reactor as well as the development of tools for remote in-situ inspection and repair, tools for replacement of pieces of reactor equipment, and tools for inspection and repair in hot cells.

- Safety and environment

  All proposed fusion schemes are intrinsically safe against power excursions of exponential type. The nuclear safety problems are only related to the use of tritium and to the activation of the structure. The tritium effluents of a fusion power station need to be kept at acceptable levels, and the disposal of solid radioactive wastes, arising from structure activation, will have to be dealt with properly.

1.6 HYBRIDS

Given the order of magnitude difference which exists between the energies liberated in fusion (17 MeV for DT) and fission (200 MeV for Uranium) reactions, it is tempting to use fusion neutrons to produce extra energy in fissile materials or to convert fertile materials, such as thorium or U-238 which are abundant, into a fission reactor fuel. For this, the fissile or fertile material would be embedded in the blanket of the fusion reactor, which would become a "hybrid" reactor. As compared to pure fusion, the internal power balance of the hybrid system is strongly enhanced, a feature of particular benefit to those confinement systems which recirculate a significant fraction of their thermonuclear power (e.g. mirrors or inertial confinement). Also, fissile generation in hybrids could conceivably contribute to supporting fission systems in case of a shortage of available fuel. However, the technology of hybrid blankets does not appear straightforward from established fission technology, and a careful assessment of the effects of the presence of fissile material on the environmental situation and on the social acceptability of the reactor is required. At present, several well-known physicists are advocating this approach and some preliminary evaluations are made in various countries, but no large programme (with the possible exception of the USSR) has chosen hybrid reactors as one of its main targets.

1.7 OTHER POSSIBLE APPLICATIONS

Two further applications other than electricity generation or fissile fuel production have been considered, both ultimately based on the high energy of D-T neutrons. The first is the utilization of volumetric heating by 14 MeV neutrons to produce hydrogen or synthetic fuels in the blanket ; this could require blanket temperatures in excess of 1000° C which would imply the development of new structural materials. The second is the transmutation of long-lived fission products and actinides from fission reactors into short-lived or non active isotopes ; the required 14 MeV neutron fluxes (10^{16} - 10^{17} n $cm^{-2}sec^{-1}$) are not considered practicable in present fusion reactor designs.

II PRESENT SITUATION IN THE WORLD

2.1 HISTORICAL BACKGROUND

Unclassified fusion research started in 1956. In the earlier, classified work, the main problems in fusion had been correctly identified and many of the present approaches (Stellarator, Tokamak, mirrors, pinches) had already been suggested. But difficulties had been considerably under-estimated : most experts believed that a prototype reactor would be operating within 20 years.

Up to 1968, experimental research on a broad scale involving many different devices of limited size, and a strong effort in theory, laid the foundation of the new physics which was required, in particular in the field of magneto-hydrodynamics. In 1968, reports on good confinement times obtained with the Soviet machine Tokamak triggered a reorientation of most programmes, characterized by a concentration along few lines with increased effort, the lion's share being given to Tokamaks.

In 1972, the conjunction of the appearance of powerful lasers and of numerical calculations showing that compression of plasma up to 10,000 times solid density was conceivable led to a strong expansion of research on inertial confinement.

In 1976, temperatures of thermonuclear interest (6 KeV) were obtained in Tokamaks. This gave renewed confidence that a demonstration reactor, based on the Tokamak concept, was conceivable for the turn of the century.

Large Tokamaks (JET in Europe, TFTR in the USA, JT-60 in Japan) are at present under construction and should within a few years produce and contain plasmas of parameters close to what will be needed in a reactor. It is expected that the Lawson criterion will be satisfied and that near-ignition regimes will be obtained when working with DT plasmas.

In 1978, pre-definition work of INTOR (a very ambitious Tokamak of the post-JET generation) was started on a world scale, under the auspices of the IAEA.

Progress along the main line (Tokamak) is illustrated by Fig. 7.

2.2 STRATEGIC CONSIDERATIONS

The Tokamak line is by far the most advanced. It is hoped that the scientific feasibility of fusion will be demonstrated by the Tokamaks of JET generation in the mid-eighties. Tokamaks of post-JET generation should be test-beds for engineering development, which implies priority shifting from physics to technology.

It is not yet clear whether the Tokamak concept can be the basis of a future reactor, as technological and economical aspects will then be dominant. Reserch on possible alternative lines (which could offer advantages over Tokamaks as fusion reactors), as well as exploratory studies of new concepts, are thus reasonable.

Because of the very strong interconnection between most plasma physics problems (confinement, heating, impurities, high β , etc...), it is often impossible to investigate them separately on small specific experiments. Building large devices is a necessity.

The real alternative to the Tokamak is inertial confinement. Although these two lines have some problems in common (blanket, tritium handling, etc...) they differ so widely that a major difficulty arising, in physics or in technology, along one line would probably not affect the other one.

Within magnetic confinement, open machines (mirrors) and toroidal devices, have many common problems, but still present enough fundamental differences (confinement mechanism, scaling laws, accessibility, etc.) to constitute clear-cut alternatives.

Toroidal devices (Tokamaks, Stellarators, R.F.P., etc...) have some of their basic properties in common. This has both positive and negative aspects. It means that quite a number of technical solutions and experimental or theoretical results can just be transferred from one system to another so that the additional effort required for such alternative lines is rather limited. But it also means that if one of the common basic properties led to a major difficulty, all of these lines would suffer. The differences between lines, which are anyway substantial from the physics point of view, could have important consequences on engineering solutions for, and economics of, the reactor. On the other hand, because of their similarities, these devices are complementary : important results of common interest can more easily be obtained with one or the other, e.g. studies of high-β plasmas with RFP, decoupling of heating and confinement in Stellarators, etc.

All large fusion programmes thus have a common core: Tokamaks and related technology. The options are in the relative weight given to these two items, in the choice of the main alternative lines, and in the percentage of effort (always small) devoted to new concepts.

2.3 ORIENTATIONS OF THE WORLD PROGRAMMES

2.3.1 European Community

- During the last decade, the basic aim has been to concentrate the efforts on the most promising line, the Tokamak. This has been achieved, and Europe has proved to be fully competitive. To give only two examples, TFR, the French device, has been the best performing Tokamak in the years 1974-76, and JET is the largest of the Tokamaks under construction.

- Given the ratio (fig. 8) between the European and American budgets (USA are the best of our competitors), this led to devoting no more than 10% of the activity to a few alternative lines chosen within the toroidal magnetic confinement family, so that they would be both alternative and contributory to Tokamaks. Work on mirrors has been totally discontinued. Efforts made by the Commission in order to support a moderate programme on inertial confinement have substantially failed for political reasons, because of problems linked to possible military implications: less than 2% of the budget goes to laser development and studies on light-matter interactions.

- Within the Tokamak line, a strong effort is being put on heating, where Europe is in fact competitive. The Commission has been constantly trying, with partial success, to increase the effort in technology.

- The integration of JET and of the Associations is fostered. For instance, heating and diagnostics developments for JET are made in the associated laboratories. The operation of JET will require a strong participation of the staff of the Associations.

- The role of European industry has been up to now rather passive: industry acts as supplier of specific objects, with very little involvement in medium-term technical developments.

- The strengthening of international collaboration (with extra-European countries) is a constant preoccupation of the Commission. Concrete results have been obtained in the field of technology (IEA Implementing Agreements), of system studies (INTOR, within IAEA) and are sought on alternative lines (IEA Implementing Agreement) and possibly on Tokamaks.

2.3.2 U S A

The overall programme is large enough to have a much less monochromatic spectrum:

- Tokamaks represent the main activity. Their large device, TFTR, is due to start operating in 1982.
- Inertial confinement gets 35% of the budget, with impressive developments in the field of lasers and light ion beams; heavy ion beams are at present under consideration.
- Their main alternative in magnetic confinement (mirrors) receives considerable support.
- They invest a smaller effort in a second alternative, the ELMO Bumpy Torus, which is a device intermediate between mirrors and toroidal systems.
- They explore new alternative concepts at a very low level of funding.

A strong effort is made on heating, and also on technology (superconductivity, Tritium laboratory, Fusion Materials Testing, etc.).

2.3.3 U S S R

Their programme involves a very large number of professionals (about 3.000) and covers almost the whole spectrum of fusion science and and technology. It is centred on Tokamaks, but does not include any very large device. Their largest Tokamak T15, which will be ready in about 1984, is a superconducting device rather similar to the French proposal, Torus II. A specific aspect of the Soviet programme is the very high quality of their contributions in theoretical and experimental plasma physics.

2.3.4 Japan

The fusion reactor development programme is being carried out as one of the national projects. It started late (in the early seventies) but is characterized by an extremely fast growth (fig. 8), a strong involvement in industry, and a concrete and important collaboration with the U.S.A. It involves:

- Reactor core plasma development: special emphasis of this project is part of JAERI's Tokamak programme. The major device is JT-60, a large Tokamak of the JET generation which is planned to become operational in 1984. The programme is supplemented by strong university programmes, among which, the Heliotron-stellarator, inertial confinement and open systems with tandem mirrors have priority. To give an example, inertial confinement (lasers, electron beams, light-ion beams) is studied at a level almost competitive with the U.S.A. in a special "Institute of Laser Engineering" at Osaka University whose present annual budget (salaries excluded) is of the order of 10 MEUA. There is also a significant effort on basic research.
- Fusion reactor technology development: the emphasis is put on superconducting magnet, tritium, structural and blanket materials, neutronics and reactor design.

2.3.5 China

The fusion effort in China started at about the same time as in Europe. Up to now, its scientific outcome has not been well known, but it is not qualitatively comparable with that of the above-mentioned programmes. At present, the chinese effort seems to involve about 700 professionals working in 4 centres. It is diversified: Tokamaks, mirrors, pinches, plasma focus, laser fusion, reactor conceptual studies, fusion technology, heating techniques. Their largest device, the Tokamak HL-1, is a machine similar to the French TFR, and is scheduled to go into operation in 1981.

2.4 STATE OF THE ART

2.4.1 Tokamaks

- A synthetic overview of the present situation of Tokamaks for each of the four large programmes is given in Fig. 9 in which the major Tokamaks (existing, being built or planned) are represented in chronological order.

- In most machines, the target value for the plasma current I_p was not or has not yet been reached (Fig. 10). Routine conditions for working are often only 60 to 80 % of the maximum current experimentally achieved which has been reported also on Fig. 10.

- Record values of plasma parameters obtained so far by pushing the devices to their maximum reasonable performance are shown in Fig. 7; they have not been obtained simultaneously on the same device. Significant progress has also been made in improving mean-β values: 4 % has been recently attained, which is within a factor 2 of the figure needed for a reactor. Sustainment times of 3 s. have already been achieved but it will be up to the next generation of superconducting devices (T-15 and TORE-SUPRA, being built and planned) to extend this figure closer to the reactor target of 100 s. (30 s. on TORE-SUPRA).

- The good behaviour of plasmas in the collisionless regime, despite the fear that the so-called trapped ion mode could damage the confinement, was one of the recent major achievement in Tokamaks. Nevertheless, the overall plasma behaviour is not yet well understood and confinement times appear much shorter than expected from simple classical theories. Of the various empirical laws proposed so far, the so-called Alcator scaling law ($\tau \sim na^2$, already mentioned) is the better accepted. However, recent observations on Alcator C seem to point out a saturation in τ when n exceeds some finite value.

- The favourable behaviour of plasmas at higher density gives importance to the limitation coming from an another scaling law known as Murakami scaling, indicating that in ohmically heated discharges the maximum achievable density is proportional to B_t/R; disruptions occur when increasing the density over this limit. This scaling law is unfavourable to large devices. There is some theoretical indication that it could be overcome when intense auxiliary heating is applied to the plasma; the expected plasma density in JET (about 10^{14} cm^{-3}) relies on such considerations. Further information should soon be available.

- Some positive achievements obtained during these last few years are of particular significance: the ion temperature of 7 keV obtained on PLT, the high $\bar{\beta}$-value (larger than 4%) obtained in several experiments without occurrence of instabilities, and the capability of most devices to produce now hot plasmas with very low impurity content. The latter has been achieved mainly by a correct conditioning of the vacuum chamber and by using low-Z material for the limiter; in some experiments this was also due to the successful operation of divertors which proved to screen efficiently the influx of heavy impurities coming from the wall and permitted ASDEX to attain the record duration of 3 s. with a clean plasma. Nevertheless the exact mechanisms governing impurity dynamics are not yet clear at all. Therefore the problems connected with the exhaust of helium, for example, remain completely open.

- The long discharge durations achieved without appearance of high impurity concentrations give an additional reason to look for ways of sustaining currents for continuously working Tokamaks. Positive results in this direction have already been obtained experimentally with neutral injection on DITE. Some indications exist also that Lower Hybrid waves could sustain currents.

- Despite the danger represented by disruptions when increasing the total current, a great deal of work has been done particularly on the n and T profiles with a view of achieving low q-values in stable conditions. This problem has been widely addressed in order to understand what exact mechanism causes the disruptions destroying the plasma column. Stable q-values smaller than two at the limiter have been achieved when the plasma discharge evolution was carefully programmed.

  Some preliminary work has been done on elongated plasma cross-sections. Whereas no definite answer is yet available on the benefit to be expected, it has already been proved on several devices that this shaping, as also the plasma position, can be adequately controlled by the outer coils system.

- The fact that up to now most Tokamaks have been ohmically heated did not allow to establish clearly the role of temperature in the confinement scaling law. Both this point and the Murakami limitation should be clarified in the coming year when large supplementary heating power will be applied. The scaling laws situation, summarized in a recent report (EUR-CEA-FC-1034), is rather uncertain at present. The coming generation of Tokamaks (JET, etc...) will try to achieve conditions of the "near-reactor" regime. This means a large extrapolation from the present day Tokamaks, while the extrapolation from JET to the reactor will be relatively modest, at least in so far as the dimensions are concerned, (JET plasma cross-section dimensions are similar to those of INTOR).

- Each of the large programmes (except USSR) has undertaken the construction of a device whose plasma size approaches that in a reactor. Besides JET (described elsewhere), two major devices of similar magnitude (although with somewhat smaller dimensions) will come into operation in the next few years: the american TFTR (Princeton) and the Japanese JT-60.

  The objectives of TFTR (R_o = 2.65m, a = 1.1m, B_t = 5.2T, I_p = 2.5MA) are to study the physics of reactor grade plasmas ($n\tau = 10^{13}-10^{14} cm^{-3}$, T = 5-10 keV, DT power $\sim$ 1 $W.cm^{-3}$) and to advance the engineering base (high power neutral beams, safe reliable systems for tritium handling and remote maintenance, techniques for dealing with high pulsed thermal loads on the vacuum vessel). The first ohmic discharge is expected for the end of 1981, 3 neutral beam lines (18 MW - 0.5 s.) will be operational in June 1982, and the full DT diagnostics in December 1982. The "extended performance" version (TFM = TFTR Flexibility Modification) with 4 beam lines (40 MW, 1.5 s.) is expected for June 1983, while the machine should be ready for DT operation in January 1984.

  The Japanese JT 60, with a somewhat larger aspect ratio than JET and TFTR (R_o = 3m, a = 1m, B_t = 5T, I_p = 3.0MA), and a circular plasma cross-section (like TRTR), will be the only large Tokamak having a divertor. It is expected to be completed by 84 and is planned to work, with hydrogen only, to give a better understanding of heating and confinement in plasmas of large dimensions.

2.4.2 Heating

This is probably the field which has experienced the most impressive changes in the last few years.

2.4.2.1 Ohmic heating. This method cannot bring plasmas to thermonuclear temperatures, as the electrical resistivity of the plasma decreases rapidly when the temperature increases. A great deal of work has therefore been undertaken in the development of complementary heating methods.

2.4.2.2 Neutral Injection heating. This is the most highly developed method. It was applied for the first time on a Tokamak at Culham in 1975 at a low power level; now multimegawatt systems are installed on several devices. In 1978, NI brought the plasma ion temperature in PLT to the record value of 7 keV. Because of the very successful results obtained so far, NI is the main heating method for devices like TFTR or JET. Units of 2-3 MW in the 40-80 keV energy range are now available (Fig. 12), with a pulse duration limited by technical problems to a fraction of second. Multimegawatt units at accelerating voltage up to 100-160 keV are bein developed, with durations between a few seconds and 30 sec, possibly with direct conversion. The physics of neutral beam-plasma interaction is well known and numerical simulation is generally in very good agreement with experimental results.

The main drawbacks of NI are:

- The decrease in efficiency at the higher acceleration voltages required for penetration in larger plasmas.
- The large additional influx of neutral gas entering the vacuum chamber via the beamline during long pulse operation, and the increase in radiation due to charge exchange between beam and plasma impurities.

- The technical complexity of the system and the large space occupied in the immediate vicinity of the Tokamak.
- The difficulty of screening the system against neutrons when used on an active plasma.
- The high cost according to present estimates.

The first of these drawbacks could be overcome using either negative ion sources or direct conversion energy recycling, both concepts being studied presently. Negative ions should allow to reach much higher beam energy with acceptable efficiency.

2.4.2.3 RF heating methods now benefit from increased confidence in their performance, though they were for long considered as a back-up for neutral injection heating. Although less developed, they do not suffer from disadvantages similar to those outlined above: one can expect RF to be cheaper, to have a higher efficiency and to be easier to screen against neutrons. Fig. 12 shows that experimental results in the MW range are expected in the near future.

- ICRH

The most well developed, the most successful and probably the cheapest RF method is ICRH (Ion Cyclotron Resonant Heating). This is presently applied at the 500 kW level and should reach 3 MW in 1981. The present use of ICRH leads to an increase of the impurity content of the plasma which is not well understood. The achievement in FAR of an "all-metal" coupling structure was a major advance in view of the applicability to the reactor; anyhow the use of an internal antenna is considered a drawback.

- ECRH

Despite the fact that high power, long duration, ECRH (Electron Cyclotron Resonant Heating) sources are presently not available from industry in the right frequency range, ECRH seems to be one to the most serious contenders for future applications. Its energy deposition is expected to be very localized and good coupling should be easy to achieve.

. LH

Considered as one of the most promising RF methods for some years, LH (Lower Hybrid heating) has recently met with some difficulties, principally concerning the accessibility of the central part of large hot plasmas. A potential advantage of LH is its expected capacity to sustain plasma current (as can Neutral Injection) by means of travelling waves.

. AW

In principle, Alfven-Wave heating could be very cheap but the potentialities of this method for heating or for sustaining the plasma current still remain to be proved. The Swiss Tokamak TCA and the US Pretext Experiment should fill this gap in the near future.

. Other heating methods are less developed, have proved to be inadequate or are of limited application.

2.4.3 Technology

Traditionally, the term "fusion reactor technology" is applied to the industrial art of fabricating those pieces of fusion reactor equipment, which are not required or are not being used in present fusion devices. Consequently, the development and fabrication of components for present Tokamaks, such as power supplies, resistive coils, vacuum vessels, pumps, neutral beam injectors, RF sources etc. are, in particular in Europe, conducted and financed in the frame of the physics rather than of the technology programme. Five major reactor technology R & D areas have been identified specifically for the Tokamak line: tritium technology, superconducting magnet technology, first-wall and structural alloys development, blanket technology, and remote maintenance technology.

2.4.3.1 Tritium technology. For a fusion reactor, tritium inventories of 6-12 Kg and feed rates of 2-6 Kg/day have been anticipated; the corresponding figures for INTOR are about 2-4 Kg inventory and 0.9-1.3 Kg/day feed rate. Tritium production and handling for military purposes is well developed in the nuclear weapons countries : France, UK, USA and USSR. This expertise, however, is classified and probably does not cover all the technology required for fusion. New developments specifically related to fusion, and the testing of tritium systems for fusion, is most advanced in the US, which undertook the construction of the Tritium Systems Test Assembly (TSTA) at Los Alamos, scheduled for operation by 1982. The main objectives of TSTA are to demonstrate the fuel cycle, to develop and test environmental and personnel protection systems (particularly under abnormal and emergency conditions), to develop, test and qualify equipment for tritium service, and to demonstrate the reliability of components. TSTA will operate at 150 g (1.5×10^6 Ci) tritium inventory and 375 g/day throughput. Similar projects have been announced by Japan (Tritium Experimental Test Facility, operation scheduled for 1985), and by the Soviet Unions (Vacuum-Tritium Test Stand). A 1978 proposal to build a Tritium Laboratory in Europe was rejected by the Council of Ministers.

2.4.3.2 Superconducting magnets. The technology of NbTi based superconducting magnets is fairly well developed in Europe, Japan and the US. In the collaborative Large Coil Project a toroidal array of 6 D-shaped superconducting coils with a bore of 2.5×3.5 m^2 and maximum field of 8 Tesla will be jointly tested in the Oak Ridge National Laboratory Large Coil Test Facility, now under construction. The test programme includes electrical, mechanical and cryogenic performance of the toroidal array and its interaction with additional pulsed fields, corresponding to the poloidal field configuration of Tokamaks. Three coils will be provided by the US and one each by the European Community, Japan and Switzerland. The full LCP test programme at ORNL is scheduled for late 1982. It is expected that the LCP experience will provide the base for the construction of toroidal field coils of the size of INTOR, at 8 Tesla maximum field.

Nb_3Sn based superconductors are being developed worldwide: one of 3 US-coils for LCP will use this superconducting material; the Soviet Union has built and operated a small Tokamak T-7 (1 m major radius) with Nb_3Sn toroidal field coils. The LCP partners are planning an upgraded experiment at 12 Tesla field, using A15 type conductors, most likely Nb_3Sn. The US is also making a large effort in 12 Tesla field technology for mirror machines.

Pulsed (poloidal) field coil development is at rather low level, the only effort worth mentioning is taking place in the US (Los Alamos). Industrial involvement is strongest in the US, where 8 major companies are capable of designing and fabricating large coils.

2.4.3.3 Materials Research and Alloys development. The effort is largest in the US, which has the only 14-MeV neutron source of high enough intensity to investigate radiation damage in small samples up to an integrated dose of about 10^{19} n cm^{-2} (RTNS-II). The US is also constructing the Fusion Materials Irradiation Test Facility (FMIT) at Hanford, scheduled for operation in 1984. This Li (D, n) stripping source will have an experiment volume of 5-10 cm^3 at a maximum flux of 10^{15} n cm^{-2} sec and about 500 cm^3 at a flux of 10^{14} n cm^{-2} sec. No similar projects exist in Europe, Japan or the USSR. All over the world, accelerators and fission reactors are being used to simulate the radiation damage expected in the fusion environment. Correlation procedures between these radiation sources and the fusion environment using FMIT will be developed in the US.

2.4.3.4 Blanket technology. The bulk of the worldwide effort mainly consists of conceptual design studies and measurement of nuclear and physico-chemical data for breeding materials. A few nuclear measurements on blanket mock-ups (breeding ratio, neutron spectra) have been reported from several laboratories (KFA-Jülich, KfK-Karlsruhe in Europe, JAERI in Japan, LASL in the US), but no major technical development has yet been started.

2.4.3.5 Remote handling. The state of the art is given by the remote systems developments for JET and for TFTR. Industrial capabilities are well developed in Europe, Japan and the US.

2.4.4 Alternative lines in Toroidal confinement

Many possible approaches have been explored. Some of them, have emerged as possible alternatives to Tokamaks and have developed as such.

2.4.4.1 Stellarators (and related devices like Torsatrons). Essential contributions were made recently in Europe by Wendelstein VII-A at Garching and Cleo at Culham (discontinued in 1980) and others are expected very soon from the large Heliotron E (Japan) which has recently started operation. Experiments in Europe have shown that, with significant plasma parameters, net current free operation can be achieved by neutral injection. Under these conditions, there are no disruptions, MHD fluctuations have disappeared, plasma turbulence is greatly reduced and the overall confinement properties are improved. Theoretical investigations on advanced Stellarator configurations indicate that further improvement of the confinement properties should be possible.

A significant effort is undertaken in USSR, where a new device Uragan 3 (Karkhov), comparable in dimension with the Stellarator L2 (Lebedev) with a field strength about twice as high, is at the end of the construction phase.

Conversely, Stellarators have practically disappeared from the US programme except for a small device in Wisconsin.

2.4.4.2 Reversed Field Pinches. A strong programme has been pioneered in the European Community starting with ZETA at Harwell in the 1960s and now with the joint Culham-Padua programme. The basic properties of the RFP - stability at high beta, effective ohmic heating - have been established experimentally and theoretically. Many theoretical predictions on self-reversal (the natural generation of the Reversed Field configuration) and on stability at high beta have been confirmed on relatively low temperature plasmas with temperatures approximately 10-100 eV and beta approximately 10-50 %. Important new results on confinement and heating have been obtained at Padua and also in Japan on the TPE-IR device with plasma lifetimes approximately 1 ms and temperatures approximately 100 eV with beta greater than 10 %. Somewhat larger new machines at Culham and Los Alamos are shortly expected to yield results. A collaborative programme with the Community and the USA on the RFP has been established covering existing experiments and the design of the proposed, large RFX machine (see paragraph 3.4.1.5).

2.4.4.3 Elmo Bumpy Torus. This second alternative in the US Magnetic Confinement Fusion Programme (after Mirrors) is developed (EBT-3, Oak Ridge) in close collaboration with Japan (NBT, Nagoya). In these devices, stability is achieved by means of relativistic electron rings obtained by ECRH and regularly distributed around the torus. Low density ($n \sim 10^{12}\ cm^{-3}$) moderately hot ($T \sim$ 100-500 eV) clean plasmas have exhibited relatively long confinement times (up to 8 ms) in quasi steady state operation (10^4 s). A more ambitious device is now being built, EBT-P ($R \simeq$ 4.5m, $a \simeq$ 0.2m, $B_t \leqslant$ 21 kG) which should provide, after 1985, a proof-of-principle as to the feasibility of an EBT reactor concept.

2.4.4.4 Compact Toroids. This line covers at least 3 slightly different approaches: the Reversed Field Mirrors (Livermore), the Field Reversed θ-Pinches (Kurchatov, Los Alamos, Washington University) and the Spheromak for which a first limited experimental test has started (Princeton). None of those approaches have reached the necessary level for a sound assessment of their reactor potential.

2.4.4.5 Basic plasma physics studies of direct relevance for fusion are performed in several places, on small specific devices. The Superconducting Levitron (Culham) is a typical example: owing to the simple field topology, it is possible to study cross-field transport in well-defined conditions and thus to contribute actively to the physics of confinement.

2.4.5 Mirrors

- Owing to their simplicity, mirror devices have been favoured for a long time. But the low expected power gain factor Q (ratio of fusion power to input power) and the instabilities affecting these devices led to a continuous decrease in interest for this type of system.

- A positive change in attitude towards mirrors as alternative line came with the successful operation of 2XII-B line at Livermore (US), a large mirror device (L = 2m, B = 9-18 kG) stabilized by a new type of non-axisymmetric coils - the ying yang coils - and in which 5 MW of neutral injection allowed the sustainment of high quality, high-β

plasma ($n \sim 10^{14} cm^{-3}$, $T_i \sim 10$ keV). Furthermore, the use of gas boxes at the ends of the machine stabilized the loss-cone instability.

- The concept of Tandem Mirrors makes use of the ambipolar potential produced by two identical mirror machines to plug electrostatically the two ends of a solenoïd. Gamma 6 (Japan) and TMX (Livermore, the largest mirror device in the world) have already proved the positive effect of this ambipolar potential. A large device AMBAL (Novosibirsk), similar to TMX, should be soon (1981) providing more experimental information.

  A new version of TMX (TMX-V) will allow to study the improvement brought by the so-called "thermal barrier". According to recent calculations, this concept leads to Q-values as high as 10 in which case the simplicity of the mirror arrangement and the fact that it is potentially a steady-state machine could be truly compatible with reactor requirements.

- Another possible development retaining the simple mirror arrangement aims at producing a configuration with reversed fields inside the mirror trap thus achieving a closed configuration. This is presently studied on the 2XII-B device (renamed Beta 2) in which the 5 MW neutral beams are injected on a preformed field-reversed plasma ring created by a plasma gun.

- MFTF (Mirror Fusion Test Facility, Livermore) is being built as the next step in the US mirror programme. Initially planned as a single cell mirror plasma, it could be redesigned (MFTP-B) as a tandem mirror depending upon the results which will be obtained with TMX-V and Beta-2 resepctively.

2.4.6 Inertial confinement

Considerable progress is being made in powerful laser systems and in electron or light ion beam generators, and is likely to result in a rapid improvement of their performances. Recently, very encouraging results have been obtained: symmetrical implosions of pellets by means of laser irradiation have led to high compression ratios (about 100 times liquid density) of DT plasmas, and to appreciable rates of thermonuclear reactions (almost 10^{11} neutrons per shot). Scientific breakeven (fusion energy equal to energy delivered to the pellet) could be reached in the mid-eighties. More specifically:

Lasers

Laser sytems for plasma research are operating in many laboratories throughout the world. They are commonly providing output powers of the order of 1 TW during pulses of 0.1 to 3 nsec, at a wavelength between 0.5 and 10 microns.

Two multibeam systems are operating now in the USA at a power level above 10 TW: SHIVA (neodimium) at Livermore and HELIOS (CO_2) at Los Alamos. Pellet implosions at full power are performed systematically. Similar systems are under construction in Japan (Osaka University) and the USSR (Lebedev Institute).

100 TW - 100 kJ systems are under construction (NOVA, neodimium, 137 million dollars, Livermore; ANTARES, CO_2, Los Alamos) or planned (Osaka). Their aim is scientific breakeven.

For reactors, 100 TW - 1 MJ -high efficiency (10 %) -high repetition rate ($>$ 1 Hz) systems would probably be necessary.

Research on advanced lasers (short wavelength, high efficiency, high power) is widespread.

Electron and light ion beams

Significant progress has been made on beam-focalization and transport and on efficient high-intensity proton beam generation, allowing the design of large multibeam systems. EBFA, a 20-beam 30-TW device, is now operational at Sandia (USA). A similar device is under construction at the Kurchatov Institute (Moscow). Several smaller systems are in operation, in particular at Osaka. Light ions seem to offer better prospects than electrons, because of easier beam transport and energy absorption in the pellet. Both approaches lead to high efficiency and low cost devices. A new initiative, to magnetically implode small cylindrical liners at the megajoule level using standard electron beam generator technology, is now being followed at the Kurchatov Institute, and would have the advantage of an improved efficiency.

Heavy ions. Exploratory work, undertaken in several countries, seems to show that this "driver", together with short wavelength lasers and light ions, could be capable of being scaled economically to the beam energy and repetitive operation requirements of fusion reactors.

Pellet design and fabrication. High-gain pellets have complex multi-layer structures whose design is based on extensive numerical calculations and whose fabrication relies on very sophisticated advanced techniques. Both the codes, the designs and the techniques are partly classified.

Reactor conceptual design. Several groups are actively working in this field. It is foreseen that the VIth Technical Committee meeting on Inertial Fusion organized by the IAEA will be devoted to reactor concepts.

III PRESENT SITUATION IN THE E.C.

3.1 Organization

Under the Euratom Treaty, the Community research programme in the field of controlled thermonuclear fusion is adopted by the Council of Ministers, for periods not exceeding five years. In accordance with the Decision of the Council, the programme is part of a long term cooperative project embracing all the work carried out in the member States in the field of controlled thermonuclear fusion. It is designed to lead in due course to the joint construction of prototypes with a view to their industrial production and marketing.

In order to provide continuity, the so-called 'sliding programme' concept was introduced in 1976: in principle after three years of implementation of each five year programme, a new five year plan is adopted, which overlaps the last two years of the preceding period. This system allows the evolving scientific and technical situation and also the effects of inflation on costs to be taken into account.

The programme is implemented by means of Contracts of Association between Euratom and the organizations within the member States which are active in the field, and by the JET Joint Undertaking. A small part of the programme of the Joint Research Centre at Ispra is also dedicated to the fusion field. Several Association Contracts cover, by means of subcontracts or other arrangements, R & D work executed in other laboratories.

3.1.1 Associations

The first contract was made with the French Commissariat à l'Energie Atomique in 1959. The location of the associated laboratories is shown in Fig. 11, with the dates on which they joined. A certain level of cooperation already existed with the Culham laboratory in the United Kingdom in 1971 and 1972; an Association Contract was signed as soon as the U.K. became a member of the European Community.

In 1976 and 1978, at the request of two non-community countries, Sweden and Switzerland, agreements were concluded which provided the basis for the subsequent signing of contracts of association, these being practically identical with those between Euratom and the institutions of the member States.

Each contract provides for the financial participation of Euratom in the general running of the laboratories (operations, personnel, administration, etc.) to a uniform level of about 25%. In addition Euratom can contribute to a level of about 45% ("preferential support") in the capital investments necessary to build experimental devices large compared to the resources of the laboratories undertaking them, and which according to the Consultative bodies, have a scientific value of interest to the whole Community Programme.
Experiments supported in this way are available for use by all the other Associations. This system of financing, introduced in 1971 with the aim of stimulating the development and concentration of the programme, principally on the Tokamak line, has turned out to be a very effective means of coordination.

Each Association is managed by a Steering Committee, made up of a small number of representatives of Euratom and the associated institutions. Such a committee is responsible mainly for the Association programme and its budget. For the programme as a whole, there is a consultative and coordinating structure described in 3.1.4.

3.1.2 Joint European Torus

For this project, which is part of the Community Programme, a Joint Undertaking, in the sense of Chapter V of the Euratom treaty, was set up, by Decision of the Council, in May 1978. The members of the JET Joint Undertaking are: Euratom, all its associated partners in the frame of the fusion programme and Ireland and Luxembourg (which have no Contracts of Association). The responsibilities for the Project are vested in the JET Council and the Director of the Project. Each member of the Undertaking has two representatives on the JET Council. This body assumes the responsibility for steering the Joint Undertaking. It takes the decisions fundamental to the implementation of the project, exercises general control over the operation and reports back to the members. It is responsible, among other things, for assuring the collaboration between the associated laboratories and the Joint Undertaking in the operation of the Project, notably the establishment in good time of working rules and practices associated with the use and exploitation of JET. The JET Council is assisted by the JET Executive Committee and may seek the advice of a JET Scientific Council.

The expenditure of the Joint Undertaking is borne by Euratom at 80% and the United Kingdom Atomic Energy Authority (UKAEA) at 10%. The remaining 10% is shared between all Members, other than Euratom, having Contracts of Association with Euratom, in proportion to the Euratom financial participation in the total costs of the Associations.

The Project Team is formed in part by personnel put at the disposal of the Undertaking by the associated institutions other than UKAEA, or from other organizations, and in part by staff made available by the UKAEA (host organization). The first are recruited by Euratom as temporary agents: 150 such positions are foreseen in the Council Decision. The others stay as employees of the UKAEA. It is probable that the total number of staff at the end of the construction phase will exceed the figure of 320 originally estimated. Each member having a Contract of Association with Euratom shall undertake to re-employ the staff whom it placed at the disposal of the Project and who were recruited by the Commission for temporary posts, as soon as the work of such staff on the Project has been completed.

3.1.3 Joint Research Centre

The Ispra Laboratory of the Euratom Joint Research Center (JRC) is conducting research in some specific domains of fusion technology: system studies, blanket,materials, safety and environment. This work, financed at 100% by Euratom, was first introduced in 1977, by a Council Decision concerning the JRC programme. From the scientific and technical point of view, it is coordinated with the other fusion activities of the Community by the Commission's Directorate for the Fusion Programme, and in the frame of the consultative system outlined below.

3.1.4 The Consultative and Coordinating Structure

This structure has progressively been set up and has evolved according to the needs of the programme. It is now in process of re-organization. In the past it comprised:

- The Consultative Committee for Fusion (CCF), which was formed from responsible officials of Governments participating in the programme, including Sweden and Switzerland, at the level of responsibility for nuclear and energy research. Its task was to advise the Commission on the implementation and development of the programme, including the JET project; the changes of direction that might appear necessary; the preparation of future programmes; the determination of the total volume of fusion research activities in the European framework; the increasing concentration and integration of the work carried out in member States and the coordination at Community level of national planning.
- The Liaison Group (GdL), comprising leading scientists from the fusion Associations and from Euratom, had the task of the orientation and scientific coordination of the programme and of giving scientific opinions on proposals for new experiments submitted by the Associations with a view to receiving Community funding at the preferential level of 45%. It was assisted by Advisory Groups specialising in the main areas of programme activity (e.g. Tokamak, Heating and Injection, Alternative Lines).
- The Committee of Directors (CoD), whose members were the directors of the associated laboratories, the director of JET, and the director of the Fusion Programme at the Commission, had responsibility for the coordination in the implementation of the programme. In 1979, the CoD established a Technology Sub-Committee which, assisted by Expert Groups for different technological disciplines, was to give advice to the CoD on the technology programme.

A reorganization of the consultative and coordinating structure of the programme was recommended by the CCF in 1979 and, following a December 1980 Council Decision, will be operative from the beginning of 1981. The JET Project is not affected. The reorganization consists essentially of combining the roles of the three above-mentioned committees into a single one, the Consultative Committee for the Fusion Programme (CCFP). It should consist of three members from the Commission and for each member-State or other State participating fully in the Fusion Programme, three members appointed by the Government of that State. Each delegation will include, preferably, a member coming from a Government department and a member coming from the scientific or technical Community. The tasks of the CCFP include all those of the old committees; it will be able to create sub-committees and delegate to them some of these tasks.

3.2 VOLUME OF THE PRESENT FIVE-YEAR PROGRAMME

3.2.1 Personnel

The total number of professional physicists and engineers working in the fusion programme approaches 1000. This includes:

- about 850 professionals working in the associated laboratories of the member States, Sweden and Switzerland. A rough indication of the personnel in each Association is given in figure 11. These staff numbers include about 60 Euratom officials working in these laboratories.
- about 115 professional staff working at present in the JET Team. About one third of these come from the UKAEA and about two thirds from the Associated laboratories or other sources.
- about 30 scientific staff working in the frame of the technology programme of the Joint Research Centre at Ispra.
- finally about 10 Euratom staff working in the Fusion Directorate of the Commission in Brussels.

The total number of personnel, including non-qualified support manpower is more difficult to define, because in several associations certain services are used jointly with other laboratories which are not concerned with fusion. A rough estimate leads to an overall figure of about 3500.

In order to improve the use of the available human resources by making temporary exchanges of staff easier between the various laboratories working in the fusion field, Euratom and its Associates have implemented a multilateral agreement called the "Mobility Contract". This provides for Euratom to undertake travel and other expenses incurred in these secondments, while basic salaries continue to be paid by the "parent" laboratory. In 1979, for example, the scheme allowed the detachment between laboratories of about 50 research staff, for periods ranging from a few weeks to several months.

It can finally be noted that an agreement between the Community and Spain has been concluded this year which allows the extension of applicability of the Mobility Contract to this country, in which a relatively small activity in the field of fusion is presently in course of development.

3.2.2 Financial Volume

In round figures, the financial commitment devoted by the Community and its Associates to fusion research for the five year period 1979-83 will be of the order of 1000 Million European Units of Account (1000 MEUA)*.

- The expenses which can be funded, following the Council Decision of March 1980 concerning the 1979-83 programme are shown in Table I.

Table I

	Total volume MEUA	Euratom Participation %	MEUA
General expenses of the Associations	496	25	124
Investments covered by preferential support	130	45	58,5
JET	181	80	145
Staff mobility, management and administration	8	100	8
TOTAL:	815		335,5

The cost of construction of the JET Project in basic performance, was estimated at January 1979 prices at about 200** MEUA (20 MEUA had already been spent before the beginning of the five year programme). The supplementary funds necessary to cope mainly with the effects of inflation in 1979 and 1980, were estimated at 63** MEUA, at the beginning of 1980. The corresponding supplementary

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* The EUA is defined on the basis of a basket of currencies and its value consequently fluctuates. The average value in October 1980 was 1.385 dollars.

** These figures do not include the funds necessary for the preparation of JET for its extended performance, neither for its operation during the year 1983. The preliminary estimate by the Director of JET concerning the cost of construction in extended performance during the period 1982-86 and for operation during the period 1983-86 is about 310 MEUA at the price level of January 1981. A rough estimate of the total cost of JET, up to the end of 1986, at the price level of January 1981, would be at present of the order of 600 MEUA.

Community appropriation has already been requested from the Council. However the real effects of inflation in 1980 will probably be higher than estimated and will be known exactly at the beginning of 1981.

To the total in Table I should also be added the expenses of the Associations in the third States (Sweden and Switzerland) which are estimated, for the period 1979-1983, to be around 35 MEUA.

Finally, in the frame of the programme of the Joint Research Center for the four years 1980-83, about 26 MEUA will be devoted to activities in the area of fusion technology.

- The present distribution of financial effort among the Associations is illustrated in Table II, where the estimated total expenditure (in terms of payments) of each Association for the three years period 1979-1981 is given in MEUA.

Table II

Association	Estimated expenditure 79/81 MEUA	Percentage of sub-total Community
EUR-IPP	120.3	31.6
EUR-KFA	45.4	11.9
EUR-CEA	78.7	20.7
EUR-UKAEA	68.5	18
EUR-CNEN	32.7	8.6
EUR-CNR	3.8	1
EUR-FOM	19.9	5.2
EUR-EB	9	2.3
EUR-RISØ	2.7	0.7
Subtotal Community	381	100
EUR-NSBESD	5.7	
EUR-Switzerland	13.2	
Grand Total	399.9	

- The yearly evolution of the overall financial volume of the Associations programme (Sweden and Switzerland excluded) and of JET commitments budget is shown in Table III in MEUA.

Table III

Year	Associations programme MEUA	JET (commitments) MEUA
1976	82	3
1977	87	1
1978	105	16
1979	111	66,2
1980	128	62,1
1981	145	76,6
1982	160	115
1983	162	110

The figures concerning the years 1980 to 1983 are indicative estimates. The cost of management and staff mobility in the budget of the Associations is included. As far as the 5 years 1979-1983 are concerned, a comparison between Table III and Table I shows that the figures in Table I will probably have to be revised. In the frame of the 'sliding programme' system, already mentioned, a proposal for a revision intended to replace the 1979-83 programme by a new five year programme, starting in January 1982, must be submitted to the Council of Ministers before mid 1981. In the same proposal the funds necessary for the construction of JET in extended performance and for its operation up to the end of 1986 will be requested.

- The average rate of increase of the overall financial volume of the Associations programme over the years 1976 to 1980 has been 12% per year. The average rate of increase of the consumer price index in the Community over the same period has been 10.4% per year. The small difference is probably meaningless, but it is comparable with the effect on salaries of ageing of staff, whose number remained roughly constant.

Summing up one can state that the resources devoted to the Associations programme remain constant both in terms of real money and of manpower. The overall increase of resources devoted to the fusion programme is due to JET.

3.3 AIMS OF THE PRESENT EUROPEAN PROGRAMME

In the framework of the European strategy described in 2.3.1, the main objectives of the current five-year programme (1979-1983) are:

- to complete the construction of JET and to begin its operation. Although the Council Decision covers formally only the construction of JET in "basic performance", preparation of both the operation of JET and its extension to "extended performance" is included in this objective;

- to accumulate enough knowledge, both in physics and in technology, to be able to define the post-JET machine during the five-year period so that construction could start during the following plan.

Another objective of the programme is to assess up to which point other magnetic confinement schemes (Reversed Field Pinch, Stellarator, ...) are real alternatives to Tokamaks or are bringing essential contributions to the understanding of Tokamak physics.

In the field of inertial confinement, the objective is limited to keeping in touch with progress made elsewhere, whilst making some significant contributions to some key problems of light-matter interaction.

3.4 CONTENT OF THE PROGRAMME

3.4.1 Magnetic confinement

3.4.1.1 Tokamaks

The European programme experienced a dramatic change when promising results came from the Russian Tokamak experiment T-3 in 1968. The danger for the programme was that the main laboratories might start several modest-size projects comparable to T-3. This would have meant duplication while the gap with respect to the USSR would not have been filled.

Fortunately, this was avoided and the activities were reoriented around a few main devices of appreciable size (Table IV), with a distribution of tasks between the associated laboratories. It is difficult to judge to what extent this distribution has resulted from overall planning or from the fact that each laboratory, interacting with all the others within the Groupe de Liaison, chose to develop its own researches in areas where there was little ongoing activity. Both these processes were probably important, but the fact is that the distribution did occur.

First Generation

- The achievement of high factors of performance has been a working aim of the laboratory at Fontenay-aux-Roses: the TFR Tokamak was the best experiment of its type in the world, in the period 1974-1976. Its level of performance has been continuously increased notably by making its system of auxiliary heating more and more powerful. Moreover, it is provided with a particularly comprehensive system of diagnostic equipment. This allows it to continue to make high level contributions to the understanding of Tokamak physics and plasma heating, especially concerning R.F. heating at the ion cyclotron frequency.
- The PULSATOR Tokamak at Garching was equally during the mid-1970's, able to make its contribution to this understanding. It allowed, on the one hand, the rapid acquisition of the necessary expertise for the exploitation of Tokamaks by means of the appropriate diagnostics systems, and on the other to reach relatively high levels of plasma density. PULSATOR has been practically shut down since 1979.

TABLE IV - MAIN EUROPEAN TOKAMAKS

First operation	Device	R_{cm}	a_{cm}	B_{kG}	I_{kA}	T_{sec}
	First generation					
(73)	PULSATOR (Garching)	70	12	28	95	0,2
(73)	TFR (Fontenay)	100	20/24	60	400/600	1
(77)	FT (Frascati)	83	21	100	1000	1
(73)	DITE (Culham)	117	27	28/35	250/350	0,5
	Second generation					
(80)	ASDEX (Garching)	164	40	28	500	5
(81)	TEXTOR (Jülich)	175	50	20/26	500/650	3
	Third generation					
(83)	JET	296	125/210	28/35	3800/4800	5/13
(85 ?)	TORE SUPRA	215	70	45	1700	30
(...)	ZEPHYR (Garching)	135	50	91	3700	6,5
(...)	FTU (Frascati)	90	30	80	1600	1

R = major plasma radius

a = minor plasma radius

B = toroidal field on plasma axis

I = maximum plasma current (design value)

T = maximum pulse duration at maximum performance (design value).

- The possibilities of extending the development of the Tokamak line in the direction of high magnetic fields has been the task of the laboratory at Frascati where the Tokamak FT has reached, with fields of the order of 60 kgauss, a product of density and confinement time which puts it among the highest levels recorded up to the present time. The recent addition of intense R.F. heating (0.5 to 1 MW) at the lower hybrid frequency should allow the determination of the scaling laws which connect the confinement time of a plasma and its temperature. Routine operation at 80 kG is planned for 1981.

- The DITE Tokamak at the Culham Laboratory, with its system of auxiliary heating by tangential neutral injection, has allowed, together with TFR, increasing confidence and understanding on this method of heating. This Tokamak has also been the first to be provided with a divertor (non axisymmetric type), a device which should allow a reduction in the accumulation of impurities in the discharge. Experimental evidence of beam driven currents has recently been obtained. An upgrading of DITE, of limited magnitude but of significant potentialities, is foreseen in the very near future. It will include improvements of the toroidal magnetic field strength, of the divertor, and of the neutral injection system. This should allow DITE to continue in the coming years to give important contributions to Tokamak physics.

- Smaller Tokamaks have been built over the years, for specific purposes. They are listed in Table V. Some of them are coming close to the end of their active life.

TABLE V.

Experiment	Site	Main objectives
PETULA	Grenoble	RF heating (low frequencies, and lower hybrid frequency)
ERASMUS	Brussels	RF heating (ion cyclotron frequency)
TCA	Lausanne	RF heating (Alfven waves)
WEGA	Grenoble	RF heating (lower hybrid frequency)
TORTURE	Jutphaas	Turbulent heating
THOR	Milano	RF heating (electron cyclotron frequency)
RINGBOOG	Jutphaas	Cold mantle between plasma and first wall
DANTE	Risø	Pellet ablation for refuelling
TOSCA	Culham	High beta, RF heating (electron cyclotron frequency)

Second Generation

This group consists of Tokamaks now coming into operation (Table IV), which are devoted mostly to studying the very serious problem of impurities. The situation in the impurities area has been substantially improved during the course of the last few years, due to the wall conditioning methods developed notably at Jülich and Garching and now generally employed throughout the Community, in certain cases with the active cooperation of these two laboratories. The problem still remains a serious one and is attacked on two fronts in the Community programme: first by the trials of an axisymmetric divertor (ASDEX) and secondly by systematic plasma-wall interaction studies (TEXTOR).

- ASDEX is a Tokamak equipped with an axisymmetric divertor. It has been in operation since the beginning of 1980 and has already demonstrated the possibility to obtain very pure plasmas by the joint use of divertor action and gettering techniques. In order to reach high temperatures, and high β-values powerful auxiliary heating is being prepared in collaboration with the Euratom-CEA Association: 2.5 MW of neutral injection, and 2 MW of R.F. at the lower hybrid frequency. ASDEX will also be used to study different methods of refuelling.
- TEXTOR is a Tokamak which is built specially to facilitate the study of plasma wall interactions. It is in final stages of completion and will be operational in summer 1981. Extended performance of the device, now in preparation, will include an increased magnetic field (26 kG), alternative wall structure and boundary configurations and powerful auxiliary heating both with neutral (2.6 MW) injection and R.F. ($\leqslant$3 MW). This latter heating system, at the ion cyclotron frequency, will be prepared and operated by the Euratom-Etat Belge Association.

Third Generation

In addition to JET (see paragraph 3.4.1.2), it includes other Tokamaks (Table IV) which are now in the design or definition phase, and whose construction is not yet finally decided. They could become operational in the mid-eighties and are listed below in decreasing order of project maturity:

- TORE SUPRA - this machine, working with hydrogen or deuterium, is planned to make contributions both in physics and in technology.

It would give experience in constructing and operating a superducting Tokamak (cooled with superfluid helium) and it would permit studies of R.F. heating (ion cyclotron and/or lower hybrid) at high power levels (up to 10 MW in long pulses), of the dynamics of impurities in absence of divertor, and of long pulses (30 seconds). The design of the machine is completed and extensive tests of superconducting coils have been successfully performed. The construction could start in 1981. The cost of TORE SUPRA including basic diagnostics equipments but excluding heating systems, is evaluated at about 40 MEUA.

- ZEPHYR*- the aims of this experiment are to investigate α-particle heating, to ignite a DT plasma, to keep the DT plasma burning for many energy confinement times and to investigate a tolerable shut-down procedure. The basic idea is to avoid the use of superconducting coils, so that, using high magnetic fields (91 kG on the magnetic axis), an experiment much smaller than FED or INTOR could give earlier important information on a burning plasma, and at a lower cost. The project is now at the end of the definition phase for a version relying upon neutral injection and adiabatic heating. Design and construction would take about 6 years. The cost of the device would be in the 120-150 MEUA bracket.

- FTU - the Frascati Tokamak Upgrade project consists essentially of a new load assembly for FT with greatly improved access, so that powerful R.F. heating (8 MW of lower hybrid) could be applied. FTU would work in deuterium and use FT power supplies and auxiliaries (cooling, vacuum, etc.). The expected value of $n\tau$ in FTU is in the range $0.7 - 2.10^{14}$ cm^{-3} .s., to be compared to the value of 4.10^{13} expected from FT and that already obtained ($1.5.10^{13}$) with this latter device. A plasma temperature of about 7 keV could be reached. It could then be possible to study R.F. heating close to the thermonuclear regime, as well as the scaling laws of energy confinement time with temperature in the same regime. At present, only a preliminary design of FTU has been performed. Design and construction would take 4 years. Capital cost would be of the order of 18 MEUA.

- Outside the system of Associations, CNEN is financing (about 0.1 MEUA) the pre-assessment of the feasibility, and the predesign of a very high field compact Tokamak working with DT and aiming at ignition. The study group is due to present its conclusions in Spring 1981.

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* Budgetary decisions concerning the future of this project are being taken these days.

3.4.1.2 Joint European Torus

Objectives

JET is a large experimental Tokamak. The objectives are:

- To determine the scaling of plasma behaviour as plasma parameters approach the reactor range.
- To study the plasma-wall interactions in these conditions.
- To study plasma-heating.
- To study alpha-particle production, confinement and consequent plasma heating.

The first objective is concerned mainly with two questions, the quality of thermal insulation that can be achieved between the plasma and the wall as reflected in the energy containment time τ , and the effectiveness with which the magnetic field is used as reflected in the ratio β between the plasma pressure p and the magnetic field pressure. The physics determining these two parameters is not understood in present Tokamaks. It clearly involves the non-linear consequences of plasma instabilities, and since there are a large number of dimensionless numbers involved such as β , the ratio of Larmor radius to plasma radius, the ratio of the mean free path to the plasma dimensions etc, it is not possible to scale results from present apparatus to larger devices with any confidence. The construction of JET is therefore a recognition of the fact that only through the operation of a device of reactor scale can we obtain definitive information on the plasma performance.

The second objective is related mainly to the mechanisms of impurity release from the wall, the subsequent motion of these impurities in the plasma and the additional radiation loss that they produce. The JET results in this field will be important but less directly related to an ultimate reactor than the confinement and $\overline{\beta}$ data. This is because a reactor will need special provisions to channel the escaping plasma out of the torus, for example a magnetic divertor. In JET the wall loading is smaller and the pulse length shorter than in a reactor so that according to present estimates it should work satisfactorily without a divertor.

To provide a divertor for JET and to maintain the plasma physics performance would have meant a substantial increase in cost. If ultimately it proves necessary to provide a magnetic divertor inside the JET vacuum vessel then the performance will be so degraded as to make the achievement of near reactor plasmas impossible.

The third objective is related to the fact that additional heating over and above that produced by the current flow through the plasma is needed to take a Tokamak plasma to temperatures of thermonuclear interest. Several methods are available and partially tested on present machines. The most advanced is that of neutral injection whereby beams of high energy (40-160 keV) neutral atoms are injected into the plasma. This method is relatively well understood and its efficiency is estimated to be very low for a JET scale plasma. Therefore it will be important to evaluate the alternatives of radio-frequency heating at various resonant frequencies. Here the physics of wave propagation and absorption is rather uncertain so that the evaluation must be done on reactor-scale plasmas.

The fourth objective is related to the fact that as reactor conditions are approached in a deuterium-tritium plasma so the power produced in alpha-particles begins to play a significant role in the energy balance while the presence of a substantial population of non-thermal alpha-particles may affect the plasma stability. Note that this objective does not require that the alpha-power exceeds the losses (ignition) but only that it should be significant, i.e. 20-30% of the losses.

Parameters

To achieve these objectives the JET Design Team proposed in 1975 the parameters for the apparatus which were accepted by the Partners in the Project and which have remained sensibly unchanged since. They are:

Plasma major radius	3.0 m
Maximum horizontal plasma radius	1.25 m
Maximum vertical plasma radius	2.10 m
Toroidal field strength on plasma axis	3.5 T
Plasma current	4.8 MA
Flat-top time for toroidal field	
basic performance	5 seconds
extended performance	13 seconds
High grade additional heating power	
basic performance	10 MW
extended performance	25 MW

These parameters were arrived at by a process of iteration, balancing cost against risk, engineering stresses against physics requirements etc. Recent advances in Tokamak physics since the parameters were frozen have in general supported the choice made and indeed they suggest that the performance of JET will be better than was originally anticipated. The novel features of the JET parameters are the use of a large physical size system with relatively low magnetic field strength (and hence low stresses), a D-shaped cross-section for the plasma and a very low aspect ratio, i.e. the ratio between the major and minor radii of the plasma. The large D-shaped cross-section for the plasma is almost identical with that chosen in many studies of post-JET tritium burning machines such as INTOR, Culham Tokamak Reactor, etc.

JET is physically larger than any other Tokamak under construction and has the potential to proceed further towards the goal of ignition than any other device in the world.

Basic Performance Machine

It was recognized at the outset (R-5)* that to achieve JET objectives would require the order of 20-30 MW of additional heating for 5-10 seconds. At the time of the Project approval the highest additional power that had been used on an experiment was of the order a few hundred kilowatts. Therefore the equipment was staged, and initial approval was for the 'basic performance' version with only 10 MW of additional heating, no tritium handling or remote handling equipment and limited power supplies which permit the full toroidal magnetic field to be maintained for only 5 seconds. In fact it now seems likely that due to budgetary limitations the additional heating in the 'basic performance' machine will even be limited to 5 MW of neutral injection.

Extended Performance

In the extended performance version the power supplies are increased so that the full potential of the toroidal and poloidal field systems can be exploited, tritium and remote-handling equipment is provided and, perhaps most important, the additional heating power is increased to 25 MW.

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* The JET Project, Design Proposal for the Joint European Torus, EUR 5516e.

Performance Predictions

The discussion of performance can be focused by first considering the requirements for ignition although this should not be taken to imply that ignition is a go or no-go criterion of success in JET. On very simple grounds the additional heating power P and $\bar{\beta}$ values required for ignition in JET can be shown to be

$$P = \frac{70}{\tau^2} \quad \text{Megawatts}$$

$$\bar{\beta} = \frac{12}{\tau} \quad \%$$

where τ is the energy containment time in seconds. Since the maximum permissible $\bar{\beta}$ in JET is estimated theoretically to be 6% this means requirements of $\tau > 2$ seconds and $P \sim 18$ MW. This calculation assumes that the heating power is deposited exactly where it is needed. This will not be so in practice hence the real requirement is P = 25-30 MW.

Detailed performance estimates have been made for JET using a computer code which treats a variety of loss mechanisms such as charge exchange, impurity radiation, thermal conduction etc., in a self-consistent way. A typical set of such predictions is shown in Table VI.

TABLE VI

	$\hat{T}$ (keV)	$\bar{T}$ (keV)	$\bar{n}$ ($\times 10^{20}$ m^{-3})	$\bar{\beta}$ (%)	τ (s)	$\hat{Y}$
Ohmic heating only	1.8	0.6	0.26	0.1	0.5	-
Ohmic heating plus 10 MW additional heating	7.0	2.3	0.44	0.6	0.5	-
Ohmic heating plus 25 MW additional heating, hydrogen plasma	12	4.0	0.75	2.0	0.5	-
As above but with deuterium-tritium plasma	21	4.2	1.0	2.4	0.5	0.6

where $\hat{T}$ = peak ion temperature, $\bar{T}$ = average ion temperature, $\bar{n}$ is the average density and $\hat{Y}$ is the ratio of alpha-particle power to the losses in the central core of the plasma. Evidently the JET objective of substantial alpha-power is achieved in the final phase according to these results although full ignition is not.

It should be emphasized that these calculations involve assumptions about electron thermal conductivity and impurity effects that are very poorly based when extrapolating to JET. With more favourable assumptions than those used for Table VI, ignition is readily obtainable while with less favourable ones the performance may be so poor as not to justify the operation in a D-T mixture at all.

Scientific Programme and Timescales

The scientific programme is based on four phases; these phases are to some extent determined by budgetary limitations and the speed with which installations for powerful additional heating can be developed and constructed. The phases and anticipated dates are:

Phase I	Ohmic heating only D-shaped plasmas in hydrogen Currents up to at least 3 MA	Begins in 1983 apparatus is complete
Phase 2	Ohmic heating plus 10 MW additional heating. Mainly hydrogen plasmas. Assess effects of heating and the impurity problem.	Starts in 1984
Phase 3	Ohmic heating plus 25 MW additional heating and long pulse field capability. Hydrogen and deuterium plasmas. Currents to 5 MA. Assess full performance potential of JET	Starts in 1985
Phase 4	Ohmic heating plus 25 MW additional heating. Study alpha-particle effects. Extensive use of neutron diagnostics.	Starts in 1987.

Remote handling will be required to deal with components inside the torus during Phase 3 operation in deuterium. The exact mixture of heating methods to be used in Phase 3 is still not finally decided. The decision will be delayed as long as possible, consistent with the programme in order to allow information from present machines to be taken into account. At present it seems likely that the mixture will consist of 10 MW of neutral injection and 15 MW of ion cyclotron heating together with a 5 MW lower hybrid resonance system, the latter to be used for assessment purposes only.

The decision to go to Phase 4 will be a very serious one because it involves the (irreversible) activation of the apparatus. This decision will have to be taken at the time in the light of the results of Phase 3. Only when good performance has been achieved and when going to tritium operation is judged to be more useful than other research possibilities, will the Project proceed to Phase 4. Nevertheless it is necessary to prepare the tritium and full remote handling equipment ready for 1987 so that the option to go to D-T plasmas will be available.

Role of the Associations

It has always been anticipated (R-5) that the JET programme would be a collaborative venture between the Associations and the Project itself. Thus the JET budget and staffing do not permit the work to be done entirely in-house. The role of the Associations is particularly important in three areas:

(i) the development of the additional heating systems;
(ii) the development, construction and exploitation of diagnostic systems;
(iii) the scientific exploitation of the machine.

Collaboration on (i) and (ii) has now started in a serious way while discussions on (iii) are just beginning. For diagnostics, difficulties and delays have already occurred due to the problem of finding a mutually satisfactory arrangement for these collaborations. Associations have to devote financial and staff resources to JET work. The incentives offered for this by the present arrangements seem to be inadequate.

In the exploitation phase it is anticipated that 50% of the experimental physicists will come on long-term attachment from the Associations (1-2 years typically) while the remaining 50% will be in-house. The expansion of the scientific effort on the Project will have to come mainly from the Associations, whether it be in-house or the attached effort. This is partly because of the JET statutes, but primarily because staff with the required knowledge and experience can only be found in the Associated Laboratories.

3.4.1.3 Heating

- The problem of auxiliary heating for a Tokamak is equally important and difficult as that of confinement. The Commission and its associates, being aware of this fact, have for several years devoted a significant proportion of their efforts to this problem, and have paid particular attention to the distribution of development work in this field among the associated laboratories.

- The development of neutral injectors has until now been concentrated in the laboratories of Fontenay-aux-Roses and Culham. These laboratories have not only built the injectors for their own machines (TFR and DITE), but have also assisted the other laboratories and, in certain cases, have contributed complete injection systems (WENDELSTEIN, ASDEX). The development of the injection system for JET has also been entrusted to them (beam lines of 1.25 MW of neutrals, first in hydrogen at 80 kV for the operation in basic performance, and then in deuterium at 160 kV for extended performance) under the supervision of a "JET Neutral Injection Steering Committee" whose members represent JET, the Commission, Culham and Fontenay. The Jülich laboratory has recently initiated some work on long-pulse neutral injection required for its research programme in TEXTOR.

  The above-mentioned activity concerns the conventional method of neutral injection: fast neutral particles are obtained from fast positive ions, through charge-exchange. The charge exchange cross-section goes down rapidly when the velocity of the particles increases, therefore such a method, which is quite appropriate for present Tokamaks, would have an extremely low efficiency (order of 10%) for large devices of reactor size in which only very fast neutrals can reach the centre of the discharge. Neutral injection based upon neutralization of negative ion beams could, at least in principle, be much more efficient; preliminary experimental investigations along this line are conducted in a joint programme between the EUR-CEA and EUR-NSBESD (Sweden) Associations.

- In the domain of high frequency heating (see Table V), the two most advanced methods (ion cyclotron and lower hybrid) are being developed essentially in France, the former on the Tokamak TFR at Fontenay and the latter on the smaller Tokamaks PETULA and WEGA at Grenoble; the ERM (Brussels) is also contributing to the study of ion cyclotron heating on the small Tokamak ERASMUS. WEGA is a cooperative experiment between three Associated laboratories: Grenoble, Garching and ERM. The WEGA team will contribute to preparing and exploiting the heating system of ASDEX (Garching). Similarly, the ERM team is taking full responsibility for the design, procurement and exploitation of a powerful heating system (R.F. generators of 3 MW) for TEXTOR of the EUR-KFA Association. Lower Hybrid is also the method which has been chosen by Frascati for their FT Tokamak (heating system ready early 1981). JET has placed 4 study contracts with industry to estimate the cost of high power R.F. heating systems; the result is essentially that Lower Hybrid is roughly speaking as expensive as neutral injection, whereas Ion Cyclotron (which has the technical drawback of requiring the presence of antennas inside the plasma vessel) is much cheaper; this result is not necessarily meaningful for the future, as the present cost estimates include development of high power tubes in the case of Lower Hybrid.

Two other RF heating methods are being explored: Alfven waves on the new Tokamak TCA which is just entering operation at Lausanne and electron cyclotron resonance heating on the two small Tokamaks TOSCA (Culham) and THOR (Milan).

- Turbulent heating is studied on TORTURE, a small Tokamak at Jutphaas.

3.4.1.4 Diagnostics

It was not until the end of the sixties that reliable plasma diagnostic methods began to become available. From then on, it became progressively obvious that the understanding of plasma behaviour in fusion devices required simultaneously the detailed knowledge of many different plasma parameters, some of which are extremely difficult to measure. The volume of information to be handled also implies the use of data acquisition systems of increasing complexity. For instance, an analysis of the mechanisms of plasma "disruptions", which limit the density obtainable in Tokamaks, has been made possible on TFR by the quality and diversity of the diagnostics which had been previously developed by the TFR group. In fact, in the Associations, a large fraction of the staff is permanently occupied developing new diagnostics, or improving and automating known methods so that they can be used reliably on large experiments. The technical difficulty and the high development cost of most of the diagnostics lead to a specialization, not only of the physicists involved, but also of the laboratories. Collaboration between laboratories becomes then a necessity, and is in fact widespread. For instance, German specialists came with their equipment from Garching and also from Jûlich to study plasma-wall interaction on the above-mentioned TFR; a practical result is the reduction of impurities in this machine.

- Most of the expertise in advanced plasma diagnostics being concentrated in the Associations, JET has to rely almost exclusively on the associated laboratories for the development of its own diagnostics. Practically, for each if its 27 diagnostics, JET has placed study contracts with groups of experts in the Associations; on the basis of the reports of these groups, it has been possible to make an assessment of the problems to be encountered and a cost estimate (about 40 MEUA at current prices). All these diagnostics should be tritium compatible.

For the construction of these diagnostics by the Associations, it was agreed to proceed as follows:

Phase I: completion of design (JET Article 14 contracts, which means that the financial load is on the Association, with a 45% contribution from the Commission) - about 1 MEUA is already committed.

Phase II: procurement, assembly, testing (contracts paid 100% by JET).

Phase III: installation and commissioning.

Phase IV: Exploitation on JET.

Some difficulties and delays arose in connection with phase I. Arrangements for phases III and IV are still to be discussed. During phase IV, it can be anticipated that about 40 physicists will come from the Associations and constitute, together with 40 JET physicists, the experimental teams working on the machine, on the basis of shared management.

3.4.1.5 Alternative lines

The following table gives an idea of the present Community effort on the alternative lines:

		Professionals
Reversed Field Pinch	Culham	34
	Padua	11
Stellarator	Garching	33
Screw pinch	Jutphaas	17
EXTRAP	Stockholm	4
		99

Reversed Field Pinch (RFP)

After the pioneering work with ZETA (Harwell,) research on RFP continued mainly at Culham (where an important theoretical work by TAYLOR gave a strong impetus to this line), Padua and Los Alamos. At present these 3 laboratories each possess a relatively small device and plan to build and operate a large machine in common, RFX; the engineering parameters of these machines are the following:

Table VI

	Eta Beta II Padua	HBTX IA Culham	ZT 40 Los Alamos	RFX Culham
major radius (cm)	65	80	114	180
minor radius (cm)	12.5	26	20	60
peak current (MA)	0.2-0.3	0.4	0.15-0.6	2
current decay time (ms)	0.05-1	1-5		120
present status	operational	near completion	operational	design nearly completed

Compared with Tokamaks, it is clear that the RFP is at least one generation behind and the physics less well understood. In particular there is little information on scaling laws, although a comparison between Eta Beta II and TPE-IR (Japan) suggests a favourable improvement of confinement time with current and radius. Research has now reached the stage where a large apparatus (RFX) is necessary to make progress. The essential objective of RFX is to find out whether or not the RFP can provide good confinement of high temperature ($\sim$ 1 KeV), high β (10-20 %), ohmically heated plasmas. The essential motivation of building RFX is the hope that RFP might lead to a less complex and cheaper reactor than the Tokamak.

The design studies of RFX are essentially complete. The hardware cost is estimated at about 36 MEUA and the construction time at about 5 years.

The collaborative arrangements between the participants (UKAEA-EUR and CNR-EUR Associations, Department of Energy-USA) are the following : Culham would build the load assembly and some additional equipment (25-30 MEUA), Padua (2-3 MEUA) and Los Alamos ($\sim$ 6 MEUA) would provide part of the power supplies including inductive storage. The experiment, to be located at Culham, would be a UKAEA-EURATOM Association project, with U.S. and Italian participation. EURATOM would finance at 45 % the hardware expenditures of Culham and Padua. Final approval of EURATOM, U.S. authorities and U.K. Government is being sought.

Stellarator

After terminating the programme of the CLEO Stellarator at Culham in May 1980, the Stellarator work in the European Community was concentrated in the IPP Garching. There, an experiment of significant overall size, WENDELSTEIN VII, with R = 2 m and B $\lesssim$ 4 T is in operation in its W VII-A version which is an L = 2 helical field with 5 field periods on the torus and a limiter radius of 17 cm yielding an average plasma radius of 10 cm. Neutral injection with 4 x 300 kW entering the torus is available.

Pure Stellarator operation (no ohmic current) was established in this machine yielding confinement properties noticeably improved with respect to those of Tokamaks. This is also true for those classes of particles for which theory suggests somewhat higher losses in a Stellarator field. The presently running experiments aim at investigating these effects in detail. A change in the injection angle might be an additional tool in this respect.

Theroretical studies on further optimizing Stellarator configurations are giving intermediate results at present, and a number of points have originated from this work which need experimental verification. Among them are the roles of shear and magnetic well, and the importance of the field ripple arising from the discreteness of the coils in a modular coil system which would no longer use separate helical windings. Such a verification would require an upgrading of the WENDELSTEIN VII-A experiment, which is compatible with the lay-out of the machine. This would include an increase of the minor radius by at least a factor of two (needed for protecting the plasma core from wall effects), and the introduction of R.F. heating for producing a non-ohmic heated target plasma.

Screw Pinch

The Screw pinch is a kind of pulsed Tokamak in which the toroidal field and the plasma current are induced simultaneously. The fast rising magnetic field compresses (and heats) the bulk of the plasma towards the centre of the vacuum chamber, leaving behind a low-density plasma which is formed by ionization of left-over neutral atoms. This leads to a configuration which has been shown to be very favourable for stability at high β and $q \sim 1$. It offers a simple and cheap method to obtain high-β plasmas of interest also for the Tokamak line of research. Reactor perspectives for the screw pinch by itself are presently not stressed although they can not be excluded.

In the EURATOM-FOM Association, two devices have been operated: SPICA and SPICA IV. In SPICA, maximum betas of 20 % have been reached for grossly stable plasmas. The temperature is around 50 eV and the operation time ($\simeq$ 65 μs) is limited by line radiation from oxygen impurity. In the small device SPICA IV, the possibility to create screw-pinch configurations of elongated cross-section (allowing for a higher β) has been demonstrated. A larger device, SPICA II, is under construction. In the first step, it will be a modification of the load assembly of SPICA into one with a minor cross-section having an elongation of 2.3. This will allow to reach stable plasmas with beta values of up to 40 % and temperatures of about 200 eV, well above the radiation barrier for oxygen. In a second step, enlargement of the power supply will allow a power crowbar and hence increase the pulse time considerably. SPICA II will yield experimental information relevant for Tokamaks in the high-beta regime. Operation of SPICA II is foreseen for the years 1982 till 1986.

EXTRAP

EXTRAP is a toroidal Z-pinch stabilized by an externally imposed transverse magnetic field, created by currents flowing in a set of toroidal external ring-shaped conductors. The concept has similarities with that of Compact Toroids. It is claimed that it could lead to an attractive high-β reactor system which would not require auxiliary heating. Experiments with a small linear configuration have confirmed theoretical expectations on stability. A toroidal device of limited size, EXTRAP-T1, has been designed; preferential support is being sought; the required investment is 0.26 MEUA. This research is conducted at the Royal Institute of Technology (Stockholm) of the EURATOM-NSBESD Association.

3.4.1.6 Theory and Computing

Out of the 1000 professionals of the European Fusion Programme, about 150 people are working on theoretical problems and/or developing computing codes. They have played a full part in the general develop- of the theory of high temperature plasmas.

A major activity is the study of the plasma response to small perturbations, i.e. linear stability theory. One approximation is to treat the plasma as an infinitely conducting fluid, the so-called ideal magnetohydrodynamic approximation. In this field MERCIER (Fontenay) has derived a generalised criterion for localised instability in toroidal systems, WESSON and SYKES (Culham) used numerical methods to study the evolution of three-dimensional instabilities in Tokamak plasmas, TROYON, GRUBER and KERNER (Lausanne and Garching) developed a computer code ERATO which solves the same problem more generally by a variational method. These last two methods become numerically impractical for short wavelength modes; a semi-analytic solution to this problem has been developed by TAYLOR, HASTIE and CONNOR (Culham). Early analytic investigations of the effect of finite aspect ratio and shaped cross-sections on stability were made by LAVAL, PELLAT et al. (Fontenay).

The inclusion of finite electrical resistivity into the plasma model introduces new modes of instability, the most important being tearing modes in which magnetic field lines are broken and reconnected. Computational studies of these modes have been made in Culham, Garching and Fontenay. In certain conditions the evolution of resistive instabilities will cause the plasma to move towards a state of lower energy. An important example has been demonstrated by TAYLOR (Culham) in which the fields in a reversed field pinch take up a more stable distribution after some turbulence.

Significant contributions have been made to the theory of diffusive transport in toroidal systems. Notable is the demonstration by PFIRSCH and SCHLÜTER (Garching) of the increase in classical transport by the toroidal nature of the plasma. STRINGER (Culham) highlighted the importance of plasma rotation about the minor axis and the establishment of stable equilibria with increased diffusion losses. The possibility of a steady state toroidal device with the plasma current largely driven by the cross-field diffusion was demonstrated by BICKERTON, CONNOR and TAYLOR (Culham).

In the field of computational physics, pioneering work was done by ROBERTS and HAIN (Culham and Garching) in the development of codes to model one-dimensional time-dependent plasma behaviour. This early work has flowered into a major activity in most of the Associated Laboratories. Important improvements have been made in the treatment of neutral particles, HUGHES (Culham), while the influence of a cool and radiating plasma boundary in reducing impurity influx has been shown by WATKINS and GIBSON (JET). Contributions to the modelling of impurity behaviour have been made by MERCIER and WERKOFF (Fontenay). Similarly CORDEY (Culham) has calculated the details of fast ion distribution due to neutral injection and compared the results with experiment. DÜCHS (Garching) has developed and extensively applied such codes to the modelling of existing experiments. In particular he has collaborated closely with workers in the Princeton Laboratory in developing an understanding of their experimental results.

Much theoretical work has also been done on the theory of radio frequency heating. Notable is the invention of the grill launcher for lower hybrid waves by LALLIA (Grenoble) and the development of a detailed theory for its operation (BRAMBILLA, Grenoble).

This discussion of theoretical work in Europe is only illustrative. Many other contributions have been made in the above fields and in others such as micro-instabilities and diagnostics. Suffice it to say that the European contribution to plasma theory has been of a very high standard.

All this activity requires the use of computers with large fast memories and high speed central processing units. Each of the associated laboratories has access to one or more facilities of various computing power. The best equipment is in Garching where a CRAY 1 computer has been recently integrated in a powerful, fusion dedicated, computing system.

Some cooperation between the associations, based on the exchange of codes or subroutines and on secondments of staff has been in progress for several years. Means of improving cooperation, by the standardisation and documentation of codes and by the adoption of compatible programming languages have been discussed within the "Ad Hoc Study Group on Fusion Computing". This group has also recommended to investigate the possibility of establishing computer links between some of the associated laboratories and with Livermore in the US. The computing system of this laboratory, also equipped with a GRAY 1 computer, is the centre of the very powerful fusion dedicated computer network of the American programme, to which all the fusion laboratories in the US are connected. The possibility of a direct link via satellite between Livermore and Garching (or Culham) has been discussed with American experts and might be the object of a Euratom-DOE agreement for cooperation.

3.4.1.7 Post-JET studies

The JET generation of experiments (JET, TFTR, T-15, JT-60) will, according to present assessments, lead to plasma parameters already close to those of future fusion reactors. It is therefore urgent to put more emphasis also on technology development, particularly on development and testing of first wall and blanket concepts, on plasma engineering and materials testing, etc.. An essential element in this respect is the design, construction, and operation of an engineering test reactor which would allow to do the above-mentioned R & D in a relevant environment and would simultaneously serve as a focal point for the fusion programme. First results of such a device are needed before a demonstration reactor (DEMO) can be designed successfully, and, therefore, there is general agreement that its objectives should comprise (i) the demonstration of DEMO plasma physics requirements, (ii) the demonstration of technologies needed for intrinsic parts of DEMO, like superconducting magnets and remote handling technology, (iii) service as a test facility for blanket concepts and components for relevant tritium production methods and tritium handling, for materials and for other technology development, and (iv) a demonstration of the reliable operation of a reactor.

Reactor studies

The essential features of DEMO or commercial reactors have been derived from system studies, including conceptual reactor designs such as the FINTOR study by JRC Ispra, the MARK II conceptual design study at Culham, or the Compact Tokamak conceptual design study at Jülich. While these studies quite obviously extrapolate from present physics and technology performances, they contribute largely to the credibility of the engineering feasibility of fusion power stations and give indications in which areas there is a need for development. It should be mentioned also that some non-Tokamak systems, like Reversed Field Pinch reactors, Screw Pinch reactors or Laser Fusion reactors, have already been studied in some detail.

Some years ago, the TIGER (Tokamak Installation for Generating Electricity R) Study Group was set up in the Culham Laboratory to answer the question "Is it now sensible to plan to build an electricity generating Tokamak as the next major step, starting construction sometime in the period 1985-1990 ?" The broad objectives of such an experiment were to be:

(a) to show continuous generation of net electricity;
(b) to provide, if possible, radiation life-time tests of vital components;
(c) to show that D-T fusion in toroidal magnetic geometry holds out some prospect of ultimate economic generation.

The answer of the study group to the question was "No". In their judgement an intermediate step is necessary between the JET and TIGER stages. Moreover, the study group thought that at least one new medium-sized machine would be needed to study long-pulse physics questions before embarking on the next major step after JET.

INTOR

The INTOR project, which runs under the auspices of the IAEA (see § 3.5.2), has the four objectives mentioned in the first paragraph above. It is already in its second phase (definition of the project), and consists of an intense cooperative effort by the four partners having major fusion programmes, EURATOM, Japan, USA and USSR. The way of working is the following: the work is defined by a Vienna workshop (6 members per partner) and then carried out in home bases; every partner deals with all the questions; at the next workshop session, the different solutions are compared and discussed with the aim of selecting the optimum solution for each individual problem. This way there is stepwise progress and up to now the project has been very successful. The present phase will produce in July 1981 a single conceptual design of INTOR as well as all additional material needed for the decision to go the next phase. This way of working in the INTOR project guarantees that all partners maintain their full independence because, if the cooperation were to cease at a certain moment, each existing home base (which at present involves of the order of 100 professionals per partner, most of them part-time) could without interruption continue with the full spectrum of work. This allowed Europe to concentrate all its "next step" effort on INTOR without excessive risk. Once the design phase of INTOR is reached, however, a combined design team is considered to be necessary; at this moment, independence would be lost. This transition could occur about mid-1982. The consequences of the decision to go to the design phase should be well considered.

The European home base consists of the NET team and the Euratom INTOR delegates; it is much smaller than those of our partners. The Euratom contributions to INTOR were good and influential during the data base assessment phase, they are still good in physics but are insufficient in conceptual design, mechanical engineering and tritium questions due to lack of manpower.

NET

The Next European Tokamak group has been set up in November 1978 by the Committee of Directors under the name European Post-JET Definition Group. It consists of a small number of physicists, engineers and draughtsmen based mostly at Ispra. The CoD has in the past insisted, sometimes against the wishes of the Commission, that support for INTOR should be the top priority of the NET group. As a consequence, there have been no conceptual design studies of a European Tokamak of post-JET generation. The situation has now changed and an effort will be made to strengthen and re-orient the NET group, so that it can start a specific activity.

3.4.2 Fusion Reactor Technology

The need for new technological developments for fusion was first established in Europe at the British Nuclear Engineering Society fusion reactor conference at Culham, 1969. For fusion reactor technology, five major areas have been identified (see § 2.4.5). Due to the fact that these technologies are not applied in present fusion devices, budgetary limitations of the programme inevitably translated fusion reactor technology R & D into second priority. The proposed budget 1979-83 for materials research was reduced by 50 % and the project of building a tritium test facility was postponed to the next programme revision in 1982, leaving only a small budget for preparatory work. Therefore only about 10 % of the present total budget is spent on technology research and substantial development takes place in only one area, namely superconducting magnet technology. The other fields are reduced to relatively small and scattered R & D activities, without any major project in the present programme. However, an effort is being made to build these activites into a base for future projects. In most cases these technological developments are conducted in non-associated fission research establishments under subcontracts by one of the associated plasma physics laboratories.

Superconducting magnets

The work in progress is entirely aimed at DC toroidal field coils. Essentially, it consists of 3 projects: the European participation in the Large Coil Project (LCP) at Oak Ridge National Laboratory, USA, under IEA Implementing Agreement; the superfluid-He cooled toroidal field coils for the TORE SUPRA Tokamak of CEA; and the 12 Tesla test facility SULTAN at the Swiss Institute for Nuclear Research (SIN), a joint project by CNEN Frascati, the Swiss Institute and the Dutch ECN-Petten establishment. The European coil for LCP will use a NbTi conductor and forced flow He-cooling at 3.8 K; the development is carried out by KfK-Karlsruhe, the construction by industry, and testing of the coil before shipping to ORNL again by KfK. The TF coil system developed for the TORE SUPRA experiment will use NbTi, bath-cooled by superfluid helium at 1.8 K, to produce 9 Tesla maximum field; the 18 coils will be circular, with a bore of about 2.2 m; qualifying tests have been concluded in 1980; the project TORE SUPRA is waiting for final approval. A-15 conductor development

(Nb_3Sn, Nb_3Al, V_3Ga) is in progress at CNEN/Frascati, ECN-Petten and KfK-Karlsruhe, with the aim of providing the technology for 12 Tesla maximum fields; the SULTAN test facility, scheduled for 12 Tesla operation by early 1984, will allow for testing conductors with forced flow cooling and small coils (1 m outer diameter).

Tritium

The development of tritium systems and handling capabilities for fusion is in a very preliminary stage. Most experimental activities in progress involve milligram quantities or no tritium at all (i.e. use hydrogen or deuterium instead). These activities include experiments on: exhaust purification by removal of impurities in hot metal beds (started 1980); tritium recovery from lithium by fractional distillation (thermodynamic feasibility demonstrated) ; containment, cleanup and safety systems at the 10^3 Ci or 0.1 g level (under construction) (all KFA-Jülich). However, KFA-Jülich has also undertaken the construction of a tritium storage facility for 10^5 Ci (scheduled for 1982) and the conceptual design of a tritium test facility at the 10^6 Ci level. These activities, if amplified, could lead to the development of the tritium systems for the Next Step. It should be noted, however, that proposals for such a development were rejected in the past by two member States, who claim that they have sufficient (but undisclosed) expertise to furnish the tritium systems against full payment. The tritium systems for the extended performance of JET are being studied under JET-contract by CEA and UKAEA. The anticipated inventory for JET is about 10 g, the feed rate about 8 g/day.

Materials R & D

Materials R & D is conducted in several laboratories[*], and consists mainly of radiation damage studies. Recently, also fatigue and combined effects are being addressed. Radiation damage is currently studied by charged particle irradiation of samples in accelerators. With this simulation technique, a high number of displacements per atom can be obtained in short times, but the effect of dpa rates much higher than produced by fusion neutrons is not well understood. Also, the penetration depth of charged particles is very small and volume damage is produced

(*) JRC-Ispra, KFA-Jülich/HMI-Berlin, AERE Harwell, SCK/CEN Mol, KfK-Karlsruhe, ECN-Petten, Studsvik (in decreasing order of man-years involved).

in very thin samples only, which makes the measurement of mechanical properties difficult, if not impossible. Helium damage is simulated by implantation of He, either prior to, or in a dual beam set-up, simultaneously with displacement damage. The correlation between these two techniques, which lead to significantly different results, as well as between implantation and He-generation by neutrons is again not well understood. In spite of these constraints, accelerators are considered a useful tool for the development of the basic understanding of radiation damage phenomena, even if they cannot provide reliable engineering data for materials behaviour under fusion reactor conditions.

Fission reactors with fluxes above 10^{14} n cm^{-2} sec can match fusion dpa rates, but helium generation rates are only approximated in alloys containing nickel (through the two-step reaction $^{58}Ni+n \rightarrow {}^{59}Ni$, $^{59}Ni+n \rightarrow {}^{56}Fe+{}^{4}He$). The BR-2 reactor at Mol and the HFR of the JRC at Petten are particularly well suited to produce the right ratio of dpa and He-generation rates in such alloys. Irradiation programmes in these reactors are under preparation.

14 MeV neutron sources of sufficient intensity for radiation damage studies do not exist and are not being developed in Europe, although the expertise to build them is potentially available. However, the construction cost of the most advanced type of source, based on the D-Li stripping reaction, would be of the order of 100 MEUA, exceeding by far the present 5 years budget for materials R & D. As an alternative to this high expenditure, one could envisage buying irradiation time on the Fusion Materials Irradiation Test Facility (FMIT) presently in construction in the US. When considering the use of different irradiation sources, one has to be aware of the fact that a realistic fusion radiation environment will only be provided by the fusion reactor itself.

In a coordinated effort, samples of a reference lot of stainless steel (316 L) will be distributed to the laboratories involved in radiation damage studies. In addition, these laboratories have also agreed to concentrate their efforts on a limited number of model alloys: SS 316 L, Ti-6, Al-4V, V-20 Ti (or TZM), the ferritic steel EM 12, and the manganese steel AMCR. In the frame of recent IEA Implementing Agreement, the European programme will also interact with the corresponding programmes of Canada, Japan, US and Switzerland.

Blanket technology

Blanket technology is quite a complex subject by itself, and the blanket also interacts with most of the other components of the fusion reactor. Conceptual blanket design studies have been conducted since 1970, and in particular the computational tools for studying the neutronic performance are fairly well developed. However, no reference concept has yet evolved and experimental activities are both scarce and sporadic. The experimental activities are: measurement of ^{7}Li nuclear data (AERE Harwell/University of Birmingham and JRC Geel); measurement of the local tritium breeding ratio and of neutron spectra in lithium and some lithium alloys, including in some cases neutron multipliers like Be and Pb (KFA-Jülich); systematic study of physico-chemical properties of Li-compounds (KFA-Jülich); and lithium corrosion studies in steels and vanadium alloys (SCK/CEN Mol, KfK Karlsruhe, JRC-Ispra). Tritium (deuterium) permeation in different blanket structural materials are being measured at the JRC-Ispra.

Remote maintenance

Remote maintenance techniques will be essential for any major D-T device and are considered particularly difficult for the Tokamak. Some tools are being developed for JET (e.g. remote cutting and rewelding of joints and seals), but these do not cover the full range of remote handling equipment believed to be necessary for the Next Step. Current activities in the field consist of paper studies made in the frame of the INTOR study.

3.4.3 Inertial confinement

The Council decision, stating that the fusion programme of the Community is a long term collaborative project embracing all work carried out on fusion within the member States (plus now Sweden and Switzerland), has been difficult to implement in the field of inertial confinement, because of political problems linked to possible military implications. The situation in Europe is the following :

- In the framework of the Associations, research is limited both in volume (13 MEUA for five years) and in objectives (study of light-matter interactions and transport phenomena, development of high-power lasers). It is at present conducted in four laboratories ;

. Frascati, where pioneering work on laser fusion (HOT ICE experiment) started in the mid-sixties but was discontinued in 1970, set up a new laser group in 1976. A 200 J neodimium glass laser is nearing completion. It will be used for the study of the hydrodynamic stability of ablation-driven implosions.

. Garching works on high power (1 TW) iodine laser development and on basic theoretical and experimental studies on light-plasma interaction and energy transport mechanisms.

. Göteborg and Brussels (Université Libre) have theoretical groups. One of these (Brussels) is collaborating closely with Garching.

. A fifth laboratory (Ecole Polytechnique), which is making basic theoretical and experimental studies on light-matter interaction, could join the Association system in the near future.

- Other laboratories, civilian (Rutherford, Frascati, ...) or not (Limeil, Aldermaston, ...) are following independent laser programmes which are not directly aiming at the production of controlled thermonuclear fusion energy.

- There is no activity on electron or light ion beam fusion, apart from that of a new and small group at the Kernforschungszentrum, Karlsruhe.

- In heavy ion fusion, there is one organized effort in Germany, where a four-year exploratory programme was started in 1979 with a funding of about 8 MEUA. Some conceptual work is also going on at the Rutherford Laboratory.

It is perhaps significant that in the field of inertial confinement - where there is little coordination at the European level - the Community, despite the large number of staff (of the order of 150 professionals) scattered throughout various disconnected laboratories, is playing only a modest role on the world scene.

3.5 INTERNATIONAL COOPERATION

3.5.1 International Energy Agency.

In the frame of the IEA (Paris), Euratom is the "leading organization" for cooperation in the field of fusion, which is the task of the Fusion Power Coordinating Committee (FPCC). The Implementing Agreements concluded in this framework are signed by Euratom for itself and on behalf of its associates in the Community fusion programme. This also allows the Euratom-CEA association to participate. There are at present three such Agreements, in which the US is our main partner.

The subject of the first Agreement, concluded in 1977, is the study of plasma-wall interaction in the TEXTOR device, whose construction is now being completed in JÜLICH. This project is conducted and financed by the EUR-KFA association, with 45 % Euratom support. The agreement provides for a participation of experts, mainly from the US and Japan, in the construction and in the operation of the device.

The second, concerning the development of superconducting magnets for fusion, was also concluded in 1977. It provides for the assembly in Oak Ridge (US) of a toroidal array of six superconducting coils of large dimensions. Three of these coils are being built in the USA, two in Europe (one at Karlsruhe, with Euratom preferential support, the other in Switzerland) and one in Japan. Each partner undertakes the expenses for the coil he supplies and will participate in the tests on the toroidal assembly. The cost of one coil is of the order of 6 million dollars.

The third was concluded this year and deals with the studies of radiation damage in fusion materials. It provides in particular for the participation of European specialists in the construction at Hanford (US) of the Fusion Materials Irradiation Test (FMIT) facility. The cost for the construction of this facility, which will be operational by 1984, is of the order of 100 million dollars and is financed by the US. Moreover in the framework of this agreement a broad radiation damage programme including joint experiments, development of correlation procedures for different sources, and establishment of a common pool of data should be implemented.

In the frame of the IEA regular exchanges of information on the large projects (JET, TFTR and JT 60), have taken place over the last five years and are complemented by a series of workshops on specific problems of these devices.

3.5.2 International Atomic Energy Agency

In the frame of the IAEA (Vienna), the International Fusion Research Council (IFRC) is the advisory body to the Director General of the Agency. The IAEA organizes the biennial International conference on

Plasma Physics and Controlled Nuclear Fusion, edits the review "Nuclear Fusion" and promotes other initiatives such as an annual conference on large Tokamaks.

The problem of how to proceed in the phase after JET-TFTR-JT 60 was set at a meeting in 1977 on international cooperation on fusion at the MIT, where the building of the next device collectively under the aegis of either the IAEA or the IEA was explictly considered as a possibility, in 1978, following a proposal by the Soviet delegation at the IAEA and a recommendation of the IFRC, the INTOR study was undertaken, under the auspices of this Agency. According to the initial recommendation of the IFRC, the INTOR project should proceed in five phase : data base assessment, definition, design, construction and operation, each phase being followed by a decision whether to start the next one. For the first two phases, a series of "workshops" has been organized in Vienna, starting at the end of 1978, in which each of the four large fusion programmes takes part with a delegation formed by a small number of experts. Each delegation benefits from a home support of different strength. The data-base-assessment phase was concluded at the end of 1979 with the publication of a detailed report. The definition phase is now in progress. The further developments of this venture are uncertain.

3.5.3 Bilateral contacts

It has been agreed between Euratom and the US Department of Energy to establish a systematic exchange of information and possibly of staff in the field of alternative lines, where the two programmes are complementary. The development of neutral injection lines has been also the subject of an exchange of information and some cooperative intiatives in this field are envisaged.

IV MEDIUM-TERM EUROPEAN PLANS

4.1 MEDIUM-TERM OBJECTIVES

- Exploratory long term planning studies, conducted in the framework of the Community programme as well as in other parts of the world, and based on the quite reasonable assumption that a demonstration reactor (DEMO) would be of the Tokamak type, have indicated a choice of possible strategies to reach DEMO. These strategies lead to different time schedules and different total expenditures depending mainly upon the amount of risk they involve (as an order of magnitude, about 25 years and 15.000 MEUA could be necessary to have DEMO in operation). But all strategies, even if they diverge at later times, have a common trunk (see § 3.4.1.7): the major machine(s) to be built after JET (or its foreign equivalents) will be a deuterium-tritium burner(s) and should demonstrate the scientific and technical feasibility of DEMO. Thus, all world fusion programmes are now in a position to fix an intermediate aim for their medium-term efforts both in physics and in technology: the Next Step(s).

  A Next Step is a large Tokamak which should:

  - operate with D-T;
  - aim at long-pulse burn and possible ignition of the plasma;
  - demonstrate on a reactor scale the "intrinsic" technologies, i.e. tritium, superconducting magnets, and remote handling technologies;
  - provide for engineering testing of the breeding blanket and for studies of the first wall, of structural alloys and other important reactor technologies.

  Confidence that such a machine can be built (possibly with an unefficient heating system, with a low $\bar{\beta}$, with a low wall loading, etc...) is based on the extrapolation of present knowledge in physics (the necessary experimental check of the scaling laws will be provided by machines of the JET generation) and on the assessment (which has been made in particular by the INTOR group) that the required technologies can be developed in due time. Estimates of the construction cost of the Next Step indicate a figure of the order of 1 to 2 BEUA; from a technical point of view, the machine could be in operation as soon as 1990.

The number of steps between the JET and DEMO generations (only one, the Next Step, or more ?), as well as the usefulness to maintain a multiplicity of devices within the Next Step generation (only one INTOR for the whole world, or one for each large programme, ... ?) are open questions which will be at least partly elucidated with the definition of the detailed objectives of the Next Step(s).

- In Europe, the precise formulation of the strategy towards the Next Step and of the programme to implement this strategy will rely upon the recommendations of the Panel. At present, the European plans are based on the following tentative assumptions and objectives:

  . the most urgent task of the fusion programme is to establish the conditions for, and demonstration of D-T ignition (or near-ignition) in a toroidal magnetic confinement system;

  . this task will be partially and perhaps even totally fulfilled by JET, whose programme should thus be conducted with vigour and determination by the whole European fusion community;

  . the Community must continue to actively explore and hopefully launch the next major step after JET, a Tokamak which aims to establish the scientific and technical feasibility of DEMO;

  . both our present uncertainties in the detailed physics of toroidal confinement and our ignorance of the ways to arrive at economic net power production suggest to carry out studies of alternative systems;

  . international collaboration will continue to be followed both in the planning and execution of fusion research.

- In the USA, the national policy has been recently defined by the "Magnetic Fusion Energy Engineering Act" approved by President Carter on 7 October 1980. This act is based on the recommendations (June 1980) of a Fusion Review Panel, chaired by Dr. BUCHSBAUM; its main article (Section 2) specifies:

"It is therefore declared to be the policy of the United States and the purpose of this Act to accelerate the national effort in research, development, and demonstration activities related to magnetic fusion energy systems. Further, it is declared to be the policy of the United States and the purpose of this Act that the objectives of such program shall be:

(1) to promote an orderly transition from the current research and development program through commercial development;

(2) to establish a national goal of demonstrating the engineering feasibility of magnetic fusion by the early 1990's;

(3) to achieve at the earliest practicable time, but not later than the year 1990, operation of a magnetic fusion engineering device based on the best available confinement concept;

(4) to establish as a national goal the operation of a magnetic fusion demonstration plant at the turn of the twenty-first century;

....

(7) to continue international cooperation in magnetic fusion research for the benefit of all nations;

............"

- In Japan, a review committee is evaluating the national programme after JT-60 (the Japanese JET); its final report is due to be submitted to the Nuclear Fusion Council of the Japan Atomic Energy Commission at the beginning of 1981.

- IN USSR, to our knowledge, INTOR is the focal point of their medium-term objectives.

4.2 MEDIUM-TERM STRATEGY

4.2.1 Tokamaks

- For the European Community, the fundamental questions concerning the Next Step are: when ? with whom ? A final answer is neither possible nor necessary now. We propose to undertake immediately the definition (1981) of a European Next Step (Next European Tokamak), followed by a phase of conceptual design of NET (1982-83). This would have the following advantages:

  - a future independent European option is preserved;
  - a strong focus is provided for the European fusion programme, in particular for technology development;
  - European industry will be involved from the very beginning;
  - the European position with respect to international cooperation will be strengthened.

The implementation of such a proposal, which is compatible with, and even useful for the participation of Europe to INTOR in the early phases, would require:

- considerable strengthening of the NET team;
- acceleration of the NET-relevant technology programme;
- setting-up an adequate steering system for the NET group and the fusion technology programme together;
- appropriate funding.

These questions are addressed in paragraph 4.3.

By the end of the conceptual design phase, a decision would have to be taken whether to proceed, immediately or not, with engineering design and construction, alone or in international cooperation. Next programme revision could be made to coincide in time with this decision.

- The strategy mentioned above implies that JET receives full support, as most major results from JET-extended performance should be available before starting the actual construction of NET (around 1988, as the engineering design could last from 1984 to 1987).

- It implies also a continuous assessment of both the experimental programme of the Tokamaks and the development of supporting activities (heating, diagnostics, etc.) in order to meet in due time the requirements of JET and NET, and to insure the compatibility (staff and money) of the various elements of the overall programme. This last point is of particular importance when considering the construction of Tokamaks of the third generation (see Table IV) as each new machine will absorb a large team of specialists (order of 100 professionals) for a long period (order of 10 years).

4.2.2 Technology

Two questions have a particular strategic importance: the supply of tritium for NET, and a decision concerning fusion materials technology.

Tritium supply

The estimates for INTOR may help appreciating the order of magnitude of the problem: the start-up inventory is 2-5 kg, and the consumption (which could be partially covered by breeding) is about 100 kg for 12 years of operation. There are in principle two solutions:

- to negotiate supplies with countries producing tritium on a large scale, either within the Community (France, United Kingdom) or outside (USA, USSR, potentially Canada);
- to build up a production capacity within the frame of the European Fusion Programme.

Given the very large investments and long lead-time required by the second solution, we propose to start exploring now the possibility of further negotiations, and to recommend that at least partial breeding be seriously envisaged in the design of NET.

Materials

Special consideration has to be given to materials research and alloys development for fusion. Apart from pulsed fatigue and radiation damage to insulators, no problem requiring a major R & D effort has been identified for INTOR. Therefore, materials research is mainly oriented towards the long-range perspective of the Demonstration Reactor. We propose, however, to maintain R & D activity in this field, both because of the importance of structure lifetime for the practical feasibility of the reactor, and because of the long lead-time for the development of a new alloy in the case this would be required. We do not envisage building in the near future a large neutron source but rather participating in the experimental exploitation of such devices built elsewhere (e.g. FMIT in the USA; cost 105 M$).

4.2.3 Alternative lines

Research in this field is both expensive (large machines are needed to reach the stage of "proof of principle"), risky (alternative lines are less understood than Tokamaks), and necessary (in case the Tokamak would encounter unforeseen major difficulties or would turn out to be economically unsatisfactory). Sharing the risks, by a repartition of the alternative lines between the large programmes through an active international cooperation, appears as a reasonable solution. We propose:

- to concentrate most of our efforts on the Reversed Field Pinch (the RFX project is in an advanced state of maturity), and possibly on Stellarators;

- to stand on our wait-and-see position concerning Mirrors (the situation would have to change drastically if Hybrids were retained as one of the aims of the European Fusion Programme);
- to maintain and strengthen slightly our small effort on light-matter interaction up to the mid-eighties, when break-even experiments should be performed elsewhere and when clear-cut decisions will be needed on inertial confinement; in the meantime efforts should be made to overcome the political difficulties that such decisions would raise;
- to strengthen international collaboration.

4.3 OUTLINE OF NEXT FIVE-YEAR PROGRAMME (1982-1986)

The present 5-year programme (1979-1983) will undergo revision in 1981, when according to the sliding programme principle, a new 5-year programme (1982-1986) will be proposed by the Commission. This programme proposal will be finalized after advice from the Panel. The following paragraph therefore gives only a broad outline, reflecting the present state of thinking.

4.3.1 NET and related technology

As most of fusion technology will be directed towards the preparation of NET, and will be developed outside the Associated laboratories, we propose to set up in 1981 a Steering Committee which, in the framework of the general programme, would be responsible for the management of the NET conceptual design team and of the NET related technology programme. At the occasion of next programme revision, which should coincide with the end of NET conceptual design, an autonomous structure (silimar to that of JET ?) would be proposed for NET and technology in case the Community would consider opportune to embark on the engineering design of NET.

NET design team. About 50 professionals and 2 years would be required for the conceptual design. This staff, working full time and coming only partly from the Associations, should be located in a single place, which could be either Ispra or one of the major fusion laboratories. The NET team should also constitute the home base for the INTOR European team.

NET related technology. A substantial acceleration of this part of the programme is needed. The main arguments will be:

- Superconducting magnets: LCP experimental phase; development of 12 Tesla technology (toroidal field coils of LCP or NET size); poloidal field coils development.
- Fuel cycle: development of exhaust processing techniques (removal of He and impurities, isotope separation), of containment and safety systems, of T-recovery from the blanket; construction of a tritium test facility for the above systems.
- Remote handling: according to needs defined by the NET design.
- Blanket: development of a reference design for NET and of alternative designs for testing in NET; development of nuclear out-of-pile and non-nuclear testing of blanket modules (according to the outcome of the blanket study conducted in 1981).
- Materials: study of fatigue life of NET first-wall material (stainless steel or other candidate alloys); irradiation behaviour of insulators.

4.3.2 JET

In the optics of NET, JET, whose plasma cross-section has comparable dimensions, should essentially provide a confirmation of the scaling laws. The programme of JET has been exhaustively described in § 3.4.1.2. A formal decision is required for the operation of JET and for the preparation of the extended performance. The major substantial decision to operate JET in tritium will have to be taken, in the light of the experimental results obtained in deuterium operation. Nevertheless, all the required provisions should be taken already from now, in order to avoid any delay whenever this step becomes feasible. Estimates of the JET budget for the period 1982-86 are given in § 4.3.5.

4.3.3 Associations

Their programme remains concentrated on Tokamak physics, on supporting activities for Tokamaks, on alternative lines, and on some contributions to technology. Support to JET and NET will be increased.

Tokamaks. Tokamaks of the first and second generation (see Table IV) as well as smaller devices of the Tokamak type (see Table V) or of other types (e.g. SPICA, page 56), will be exploited with improved diagnostics to widen our understanding of Tokamak physics. Some problems, like the physics of long pulse operation, cannot be studied on existing devices and could require the construction of new machines (third generation) of intermediate size.

Supporting activities. An increased emphasis will be put on heating, diagnostics, and plasma-wall interactions. Work for JET will receive special attention.

Alternative lines. As mentioned in § 4.2.3, efforts will be concentrated mostly on Reversed Field Pinches and possibly on Stellarators. A description of the proposed RFX-device is given in § 3.4.1.5 (page 54). It is proposed to devote 10-15% of the total resources to work on Alternative Lines.

Technology. There will be a continuation of the limited effort made directly in the Associated Laboratories (e.g. CEA on superconducting magnets). The bulk of technology work, which is subcontracted by the Associations to other laboratories, and which will strongly increase, should be coordinated by the NET Steering Committee (see § 4.3.1).

Support to JET and NET. Apart from in-house work for JET and NET, the Associations should delegate staff for participation to the exploitation of JET and for strengthening the NET team.

Preliminary evaluations indicate that the Associated Laboratories could work with a constant budget (in real money) and a constant or slightly decreasing staff.

4.3.4 Materials research for the reactor

Research on radiation damage to first wall and structural materials would include: scoping studies of model alloys on accelerators; irradiation of selected Ni-containing alloys in fission reactors; cooperation with the USA to establish a proper correlation between radiation damage observed with intense neutron sources and that expected from a real fusion environment.

4.3.5 Tentative budget

A preliminary assessment of the overall financial volume of the programme 1982-1986 described above leads to the following indicative figures expressed in MEUA at January 1981 prices:

TABLE VIII

NET	design team	(1)	15 MEUA
	technology	(2)	150 MEUA
JET		(3)	310 MEUA
Associations (Sweden and Switzerland excluded) (4)			670 MEUA
Materials			25 MEUA
Management and staff mobility			10 MEUA
Industrial developments (5)			15 MEUA
Total (6)			$\cong$ 1.200 MEUA

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(1) corresponds only to the phase of conceptual design (1982-1983); the expenses relative to a possible engineering design would be asked for at the next programme revision.

(2) the present technology budget, which is of the order of 50 MEUA, was incorporated in the general expenses of the Associations for the period 1979-83.

(3) see § 3.2.2.

(4) 670 $\cong$ 145 x 5 - 50 - 8, in which 145 is the budget of the Associations for 1981 (see Table III); the 1979-83 budget of the Associations included technology (50) and management and staff mobility (8); the 1982-86 budget of the Associations could be higher than 670 MEUA if the construction of several new medium-size machines were decided.

(5) this new line in the budget would be necessary to cover industrial developments of general interest for the fusion programme, which are not specifically requested by any participant; an example is the development of "gyrotrons" for Electron Cyclotron Resonance Heating (see § 2.4.2.3).

(6) This amount does not include Sweden, Switzerland and the J.R.C. In the proposal that the Commission will present to the Council in Summer 1981, the figures will be expressed at January 1982 prices.

- In this scheme, the Associations are working at a constant level of expenses (real money). JET is coming on top. New technology should come on top (the Associations have mostly physicists and conventional engineers, new technology is done by subcontracts with other laboratories having the right experts). NET design team could draw (only partly) on these 3 sources.

- Funding for the work in the Associations can follow the established scheme, i.e. 25% basic support and 45% preferential support for projects selected by the Commission. The funding of JET will also continue according to the established scheme, i.e. 80% paid by the Commission. The funding for NET (design and technology) should be supported by the Commission at a rate not smaller than 45%. Most of the technology development will be undertaken either by laboratories outside the system of Associations, or by industry (remote handling for instance).

4.3.6 Staff problems

Since the recruitment of junior professionals has been extremely limited in the recent years, the average age of the staff is increasing at a rate which approaches one year per year. Moreover, the staff is very homogeneous in age: most people are about 50. For a programme which should last a few more decades this ageing of the staff is a dangerous phenomenon. This problem is a very difficult one, as the staff policy is in the hands of many independent Institutions.

4.3.7 Political problems

As in the past, political difficulties (due to military implications) are expected for the implementation of two items of the programme: inertial confinement and tritium technology. We can wait a few years (see § 4.2.3) for a clarification on inertial confinement; we cannot wait for tritium technology which is essential for the Next Step. It should be noted that, because of the strong connection between tritium technology and environmental safety, military or foreign experience is of no help (unless fully transferred); a demonstration of safe operation of the tritium systems for NET is absolutely mandatory.

4.4 INTERNATIONAL COLLABORATION

Worldwide international collaboration is a well established feature of Fusion Research. At the beginning, this has been promoted by the universal interest for the objectives and made possible by the fundamental research character of the activity; later on by the increasing recognition of the difficulties and by the fact that small scale experiments disconnected from a large programme are of limited interest in Fusion Research. The existing situation is described in 3.5; we give in the following lines our ideas on its possible evolution and developments.

- Inside Europe, the collaboration between Euratom, Sweden and Switzerland has been useful for all the partners, and it should not face major problems in the near future. We have to implement the recently signed agreement with Spain for the exchange of personnel. A similar agreement could be concluded with Yougoslavia.

- In the frame of the IEA, for two of the already signed implementing agreements (LCP and TEXTOR) the construction of the facilities will be completed soon; we have to extend the scope of the agreements in order to cover the operation of these facilities and their possible upgrading.

  Concerning fusion materials we intend to give the widest possible content to the corresponding agreement in particular in two directions: repartitions of the problems, and mutual access to the irradiation facilities (fission reactors and neutron sources).

  The Commission has been requested by the IEA-FPCC to prepare, in collaboration with the European partners (UKAEA and CNR) and in contact with the other potential partners (at least USA), a draft implementing agreement covering the common construction and operation of RFX; the participation of the USA is very useful for the scientific and technical points of view and could be important also from the financial one (about 15% of the cost).

  Other agreements, e.g. in the field of plasma heating or for the construction of experiments of common interest, should be considered.

- In the frame of the IAEA, the major problem is INTOR. In the beginning EURATOM, in spite of the evident complexity and difficulty of

- the operation, has been a strong supporter of it for two reasons:

  . the worldwide participation made possible to envisage with limited risk and cost for each partner, "the maximum reasonable step beyond the present generation of experiments",

  . Europe was automatically in a central position and not only from the geographical point of view.

  The deterioration of the general political situation casts growing doubts on the future of INTOR.

  Our attitude should be: strong support and participation if a reasonable probability appears to proceed up to the construction of INTOR as a 4-partners venture. Otherwise avoid that the existence of a platonic intention prevent us to find other solutions for the next step. The actions proposed on the previous pages seem appropriate to this aim.

- The European Fusion Programme is the only one which has concluded no general agreement specific to Fusion with any of the other 3 large world programmes. We hope that this situation could be soon improved at least with the USA. Framework agreements should include as possible cooperative areas, among others:

  - Tokamak physics and technology
  - Alternative lines
  - Fusion reactor technology
  - Collaboration on design and construction of next step(s).

  A big problem for the operation of a European next step could be the supply of large quantities of Tritium; an agreement on Fusion with Canada could help. Preliminary contacts are already in progress.

- A unique and important characteristic of the European Fusion Programme is its global feature. Such a characteristic should be maintained not only in all our scientific and technical activities but also in the field of international collaboration, where framework agreements between the Community and the other world programmes should be preferred to bilateral arrangements between member States and third countries.

LIST OF ANNEXES

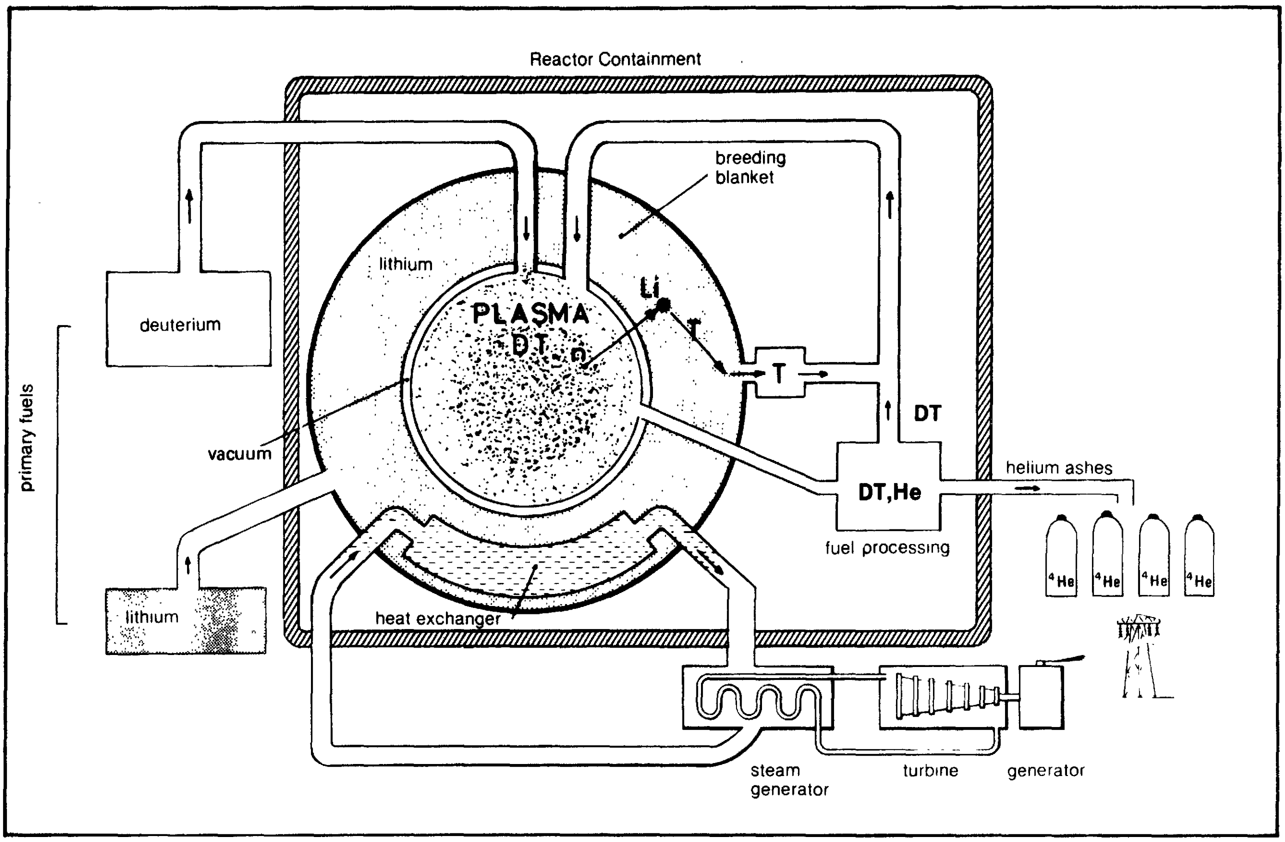


Fig. 1. Principle of a Fusion Reactor

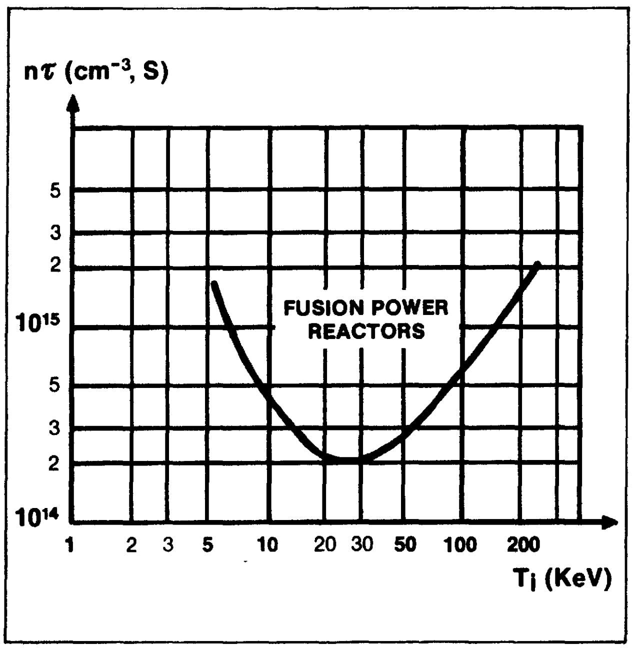


Fig. 2. Ignition curve for the D-T reaction

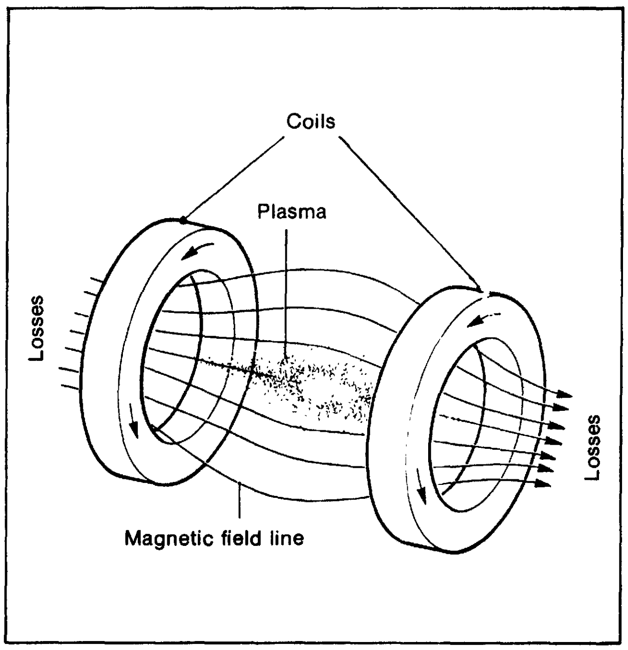


Fig. 3. Schematic of the magnetic mirror

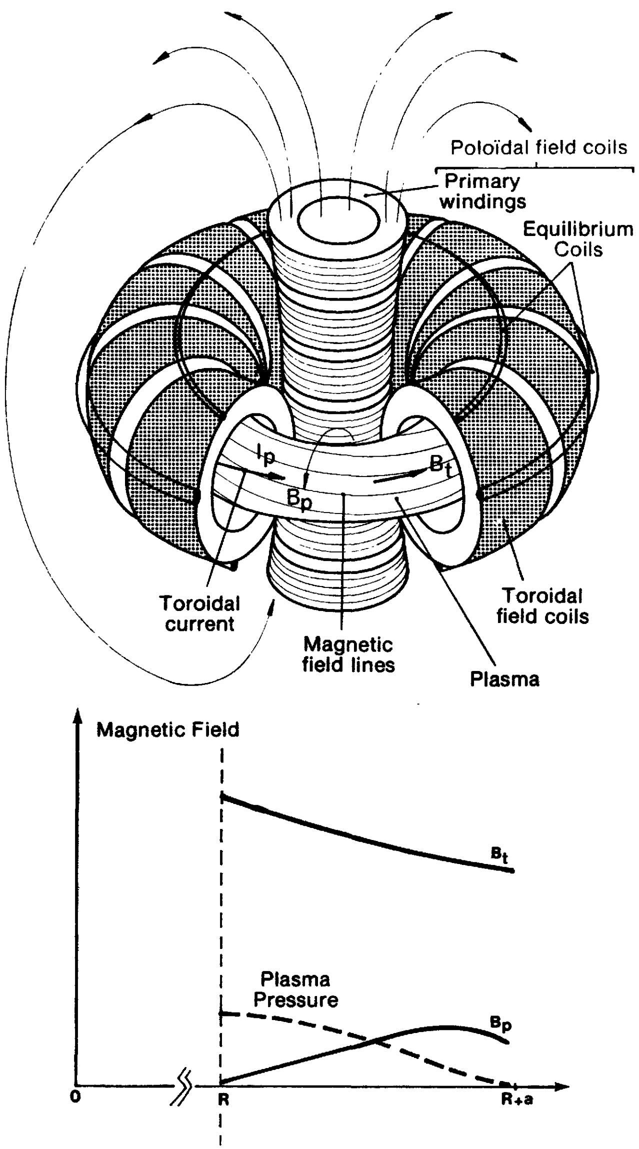


Fig. 4. Schematic of the Tokamak

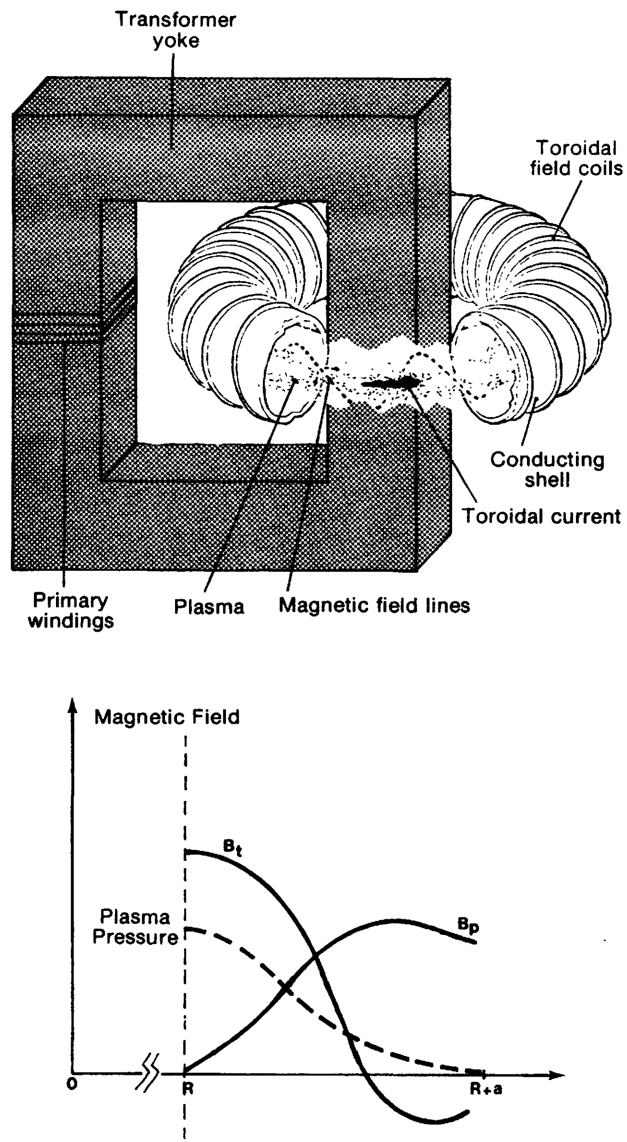


Fig. 5. Schematic of the Reverse Field Pinch

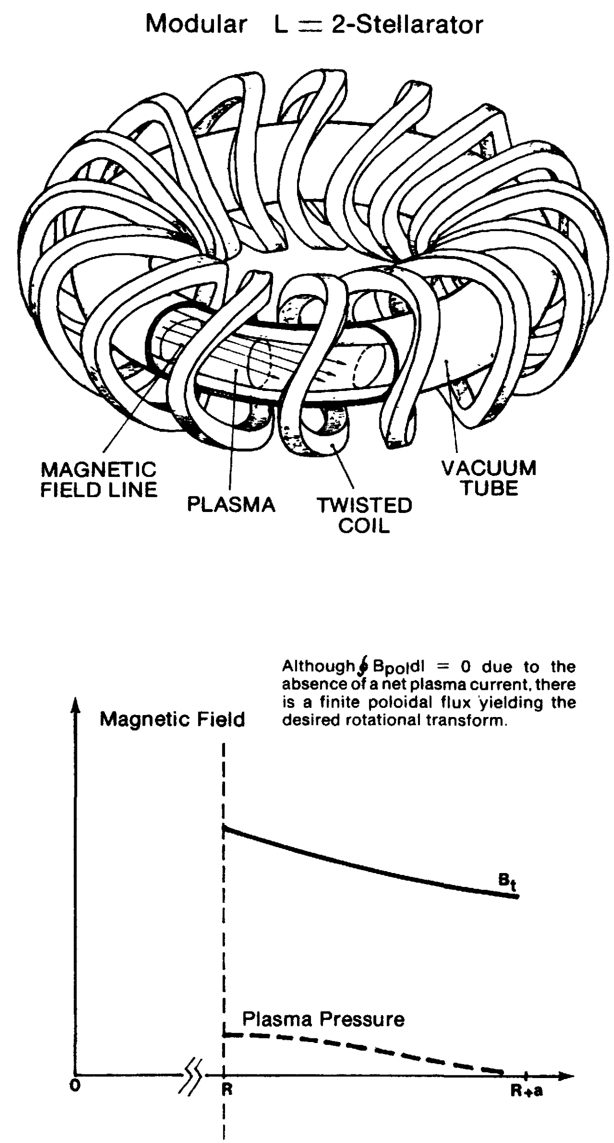


Fig. 6. Schematic of the Stellarator

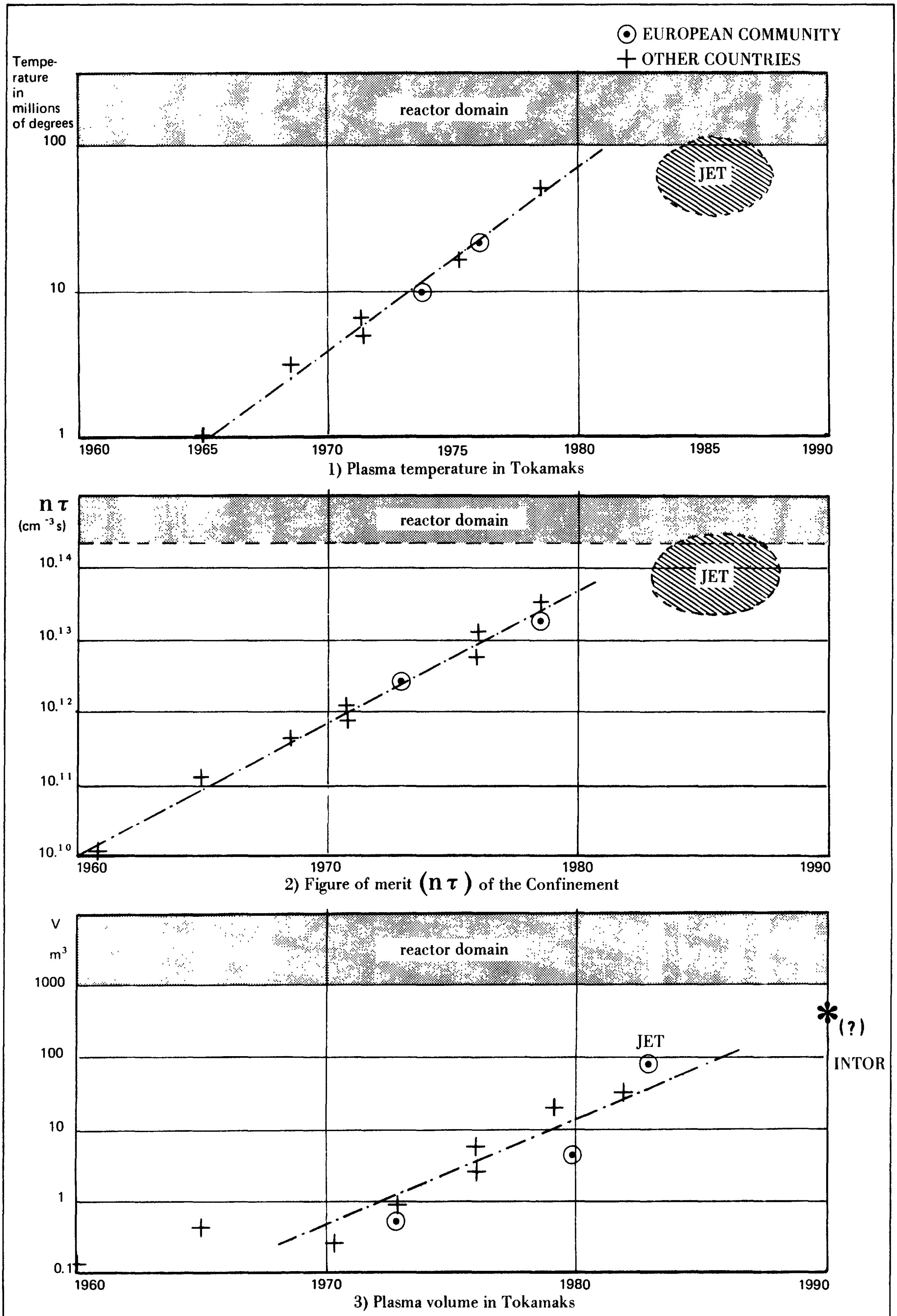


Fig. 7. Progess in Tokamaks

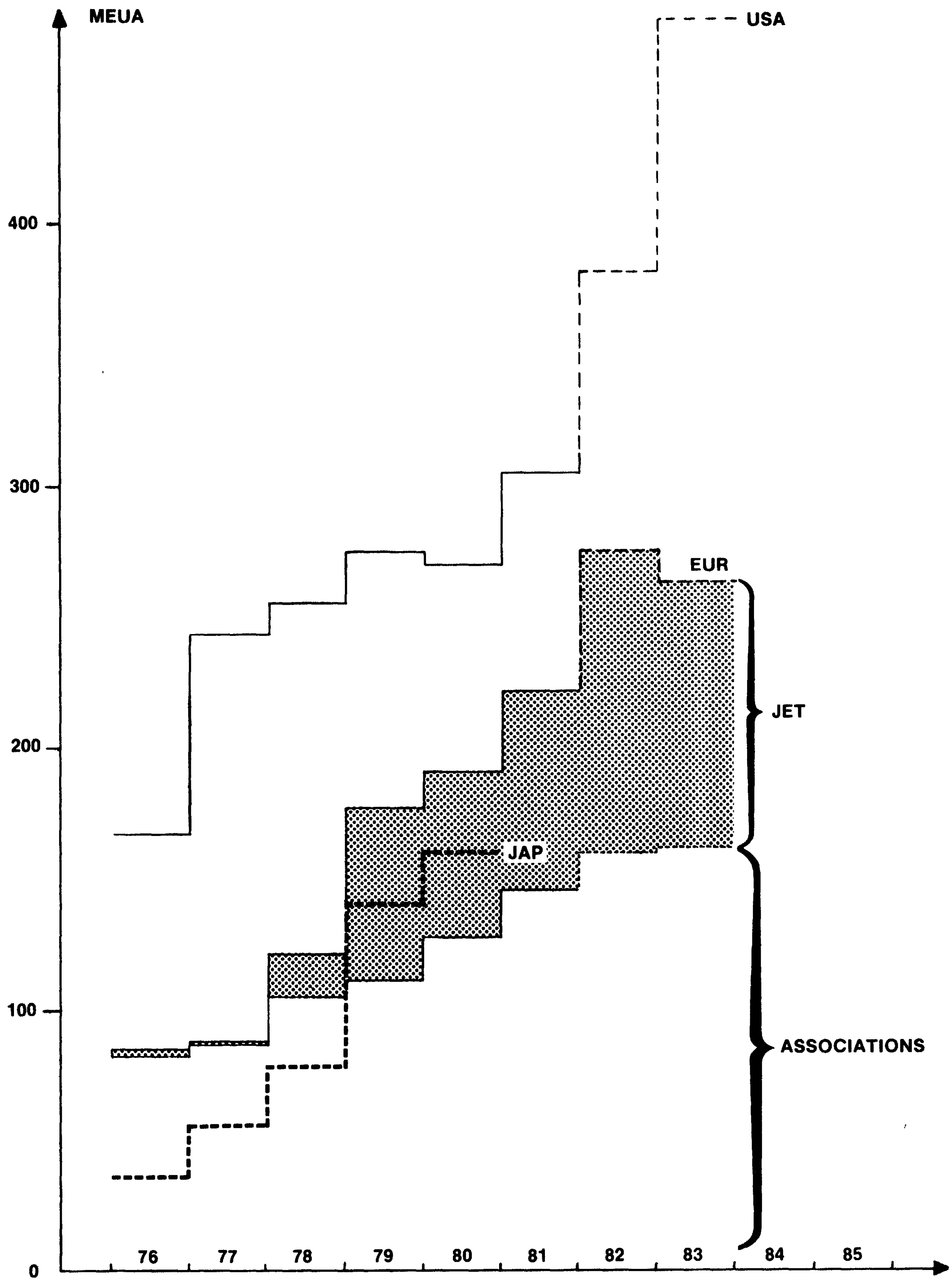
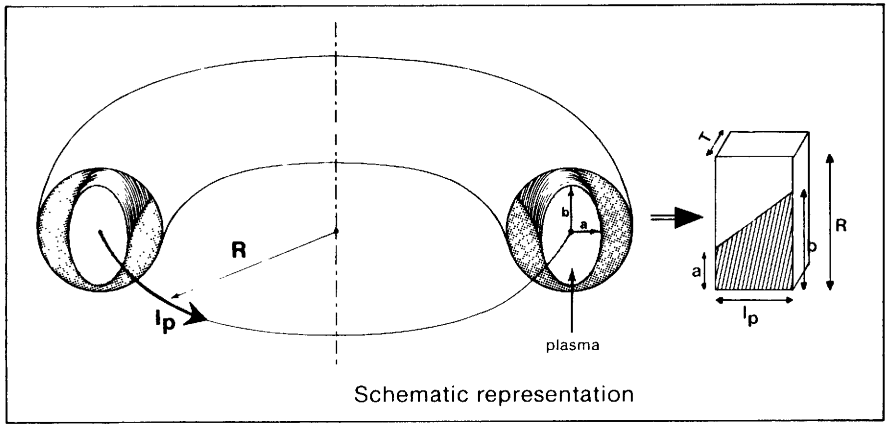


Fig. 8. Evolution of financial effort devoted to fusion in Million European Units of Accounts.
USA: magnetic confinement programme (including mirrors), 1 EUA = 1.3 U.S.$.
JAPAN: overall fusion programme (including inertial), 1 EUA = 270 yen.
EUR: overall fusion programme (excluding Sweden, Switzerland and JRC).



Major Tokamaks in the World (Target values)

- Each machine is represented by a parallelepiped whose:
  - width is proportional to the planned maximum plasma current Ip,
  - height is proportional to the major radius (R) of the device,
  - thickness is proportional to planned maximum plasma duration T.
- The intermediate line (horizontal for a circular plasma cross-section, tilted for an elongated cross-section) represents the minor radius (a) and the height (b) of the plasma cross-section.
- Shaded rectangles are machines in operation, white rectangles machines in construction, and dotted rectangles machines in the design phase.

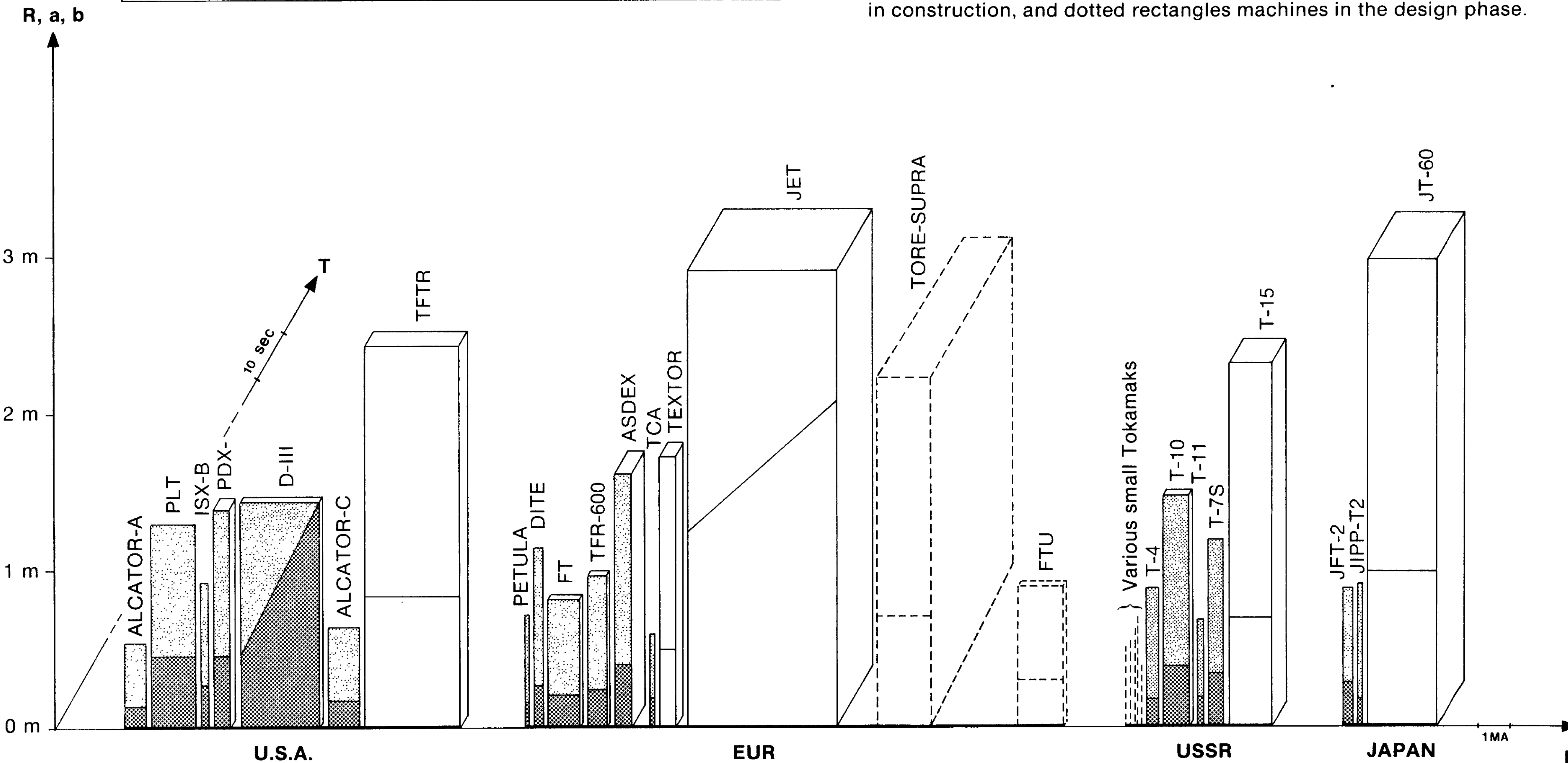
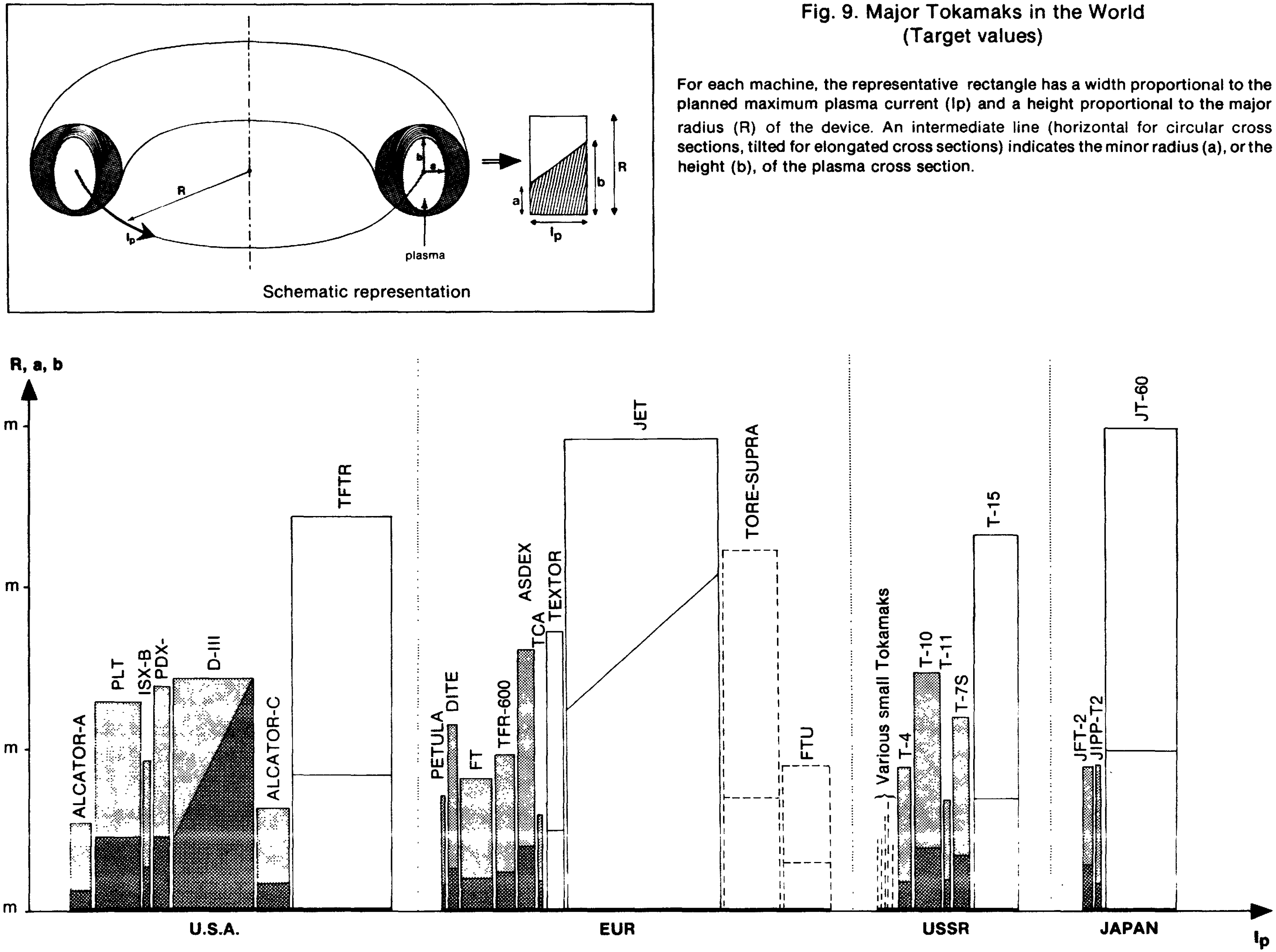


Fig. 9. Major Tokamaks in the World (Target values)

For each machine, the representative rectangle has a width proportional to the planned maximum plasma current (Ip) and a height proportional to the major radius (R) of the device. An intermediate line (horizontal for circular cross sections, tilted for elongated cross sections) indicates the minor radius (a), or the height (b), of the plasma cross section.



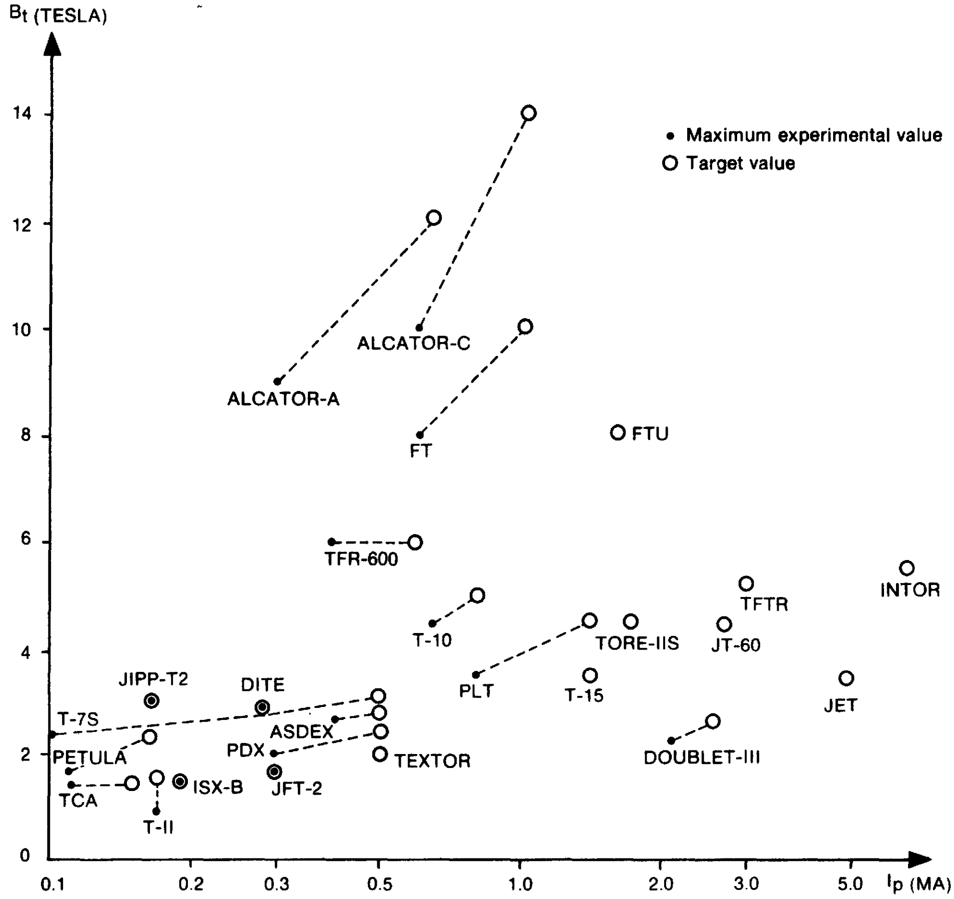


Fig. 10. Planned Performance of Tokamaks compared with actual performances. (Toroidal magnetic field B_t and plasma current I_p)

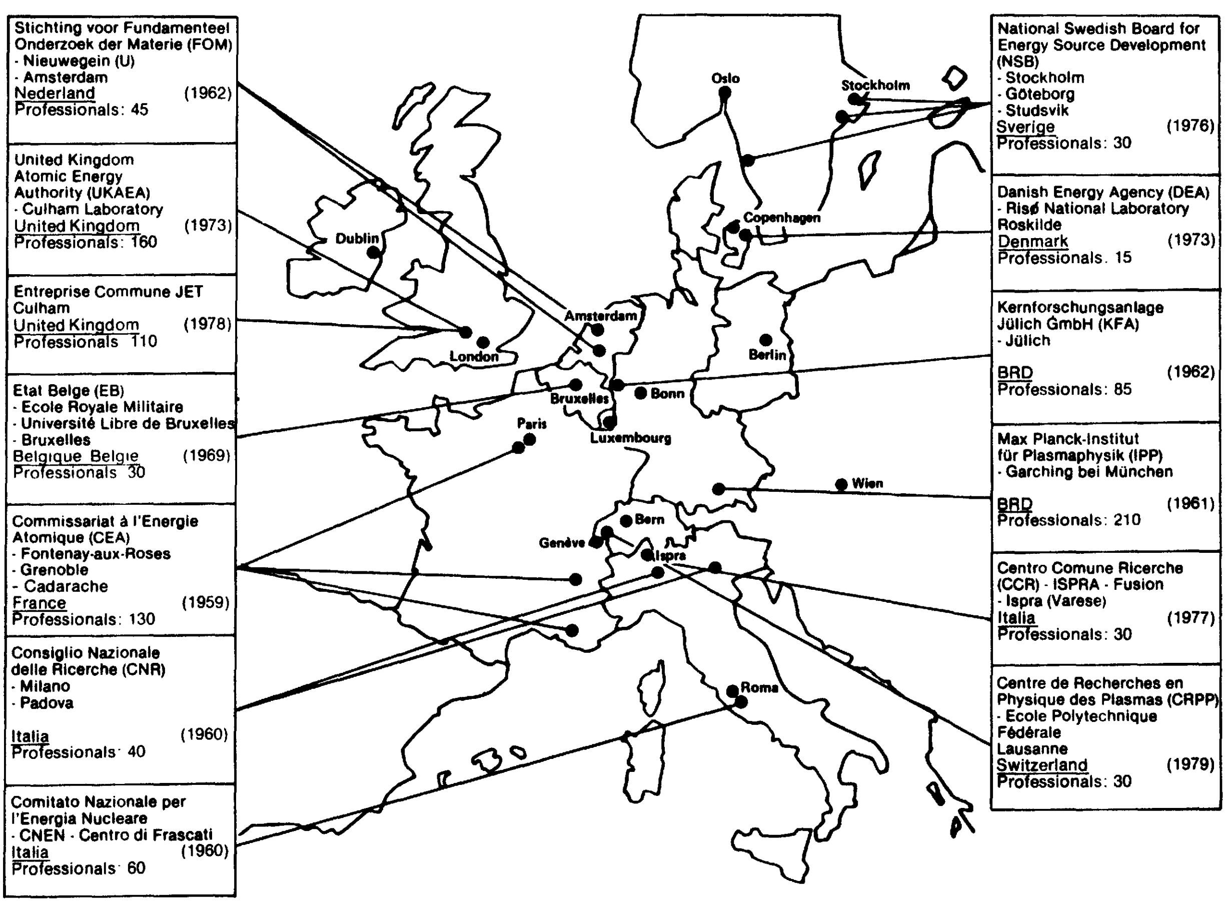


Fig. 11. Location of fusion laboratories in Europe

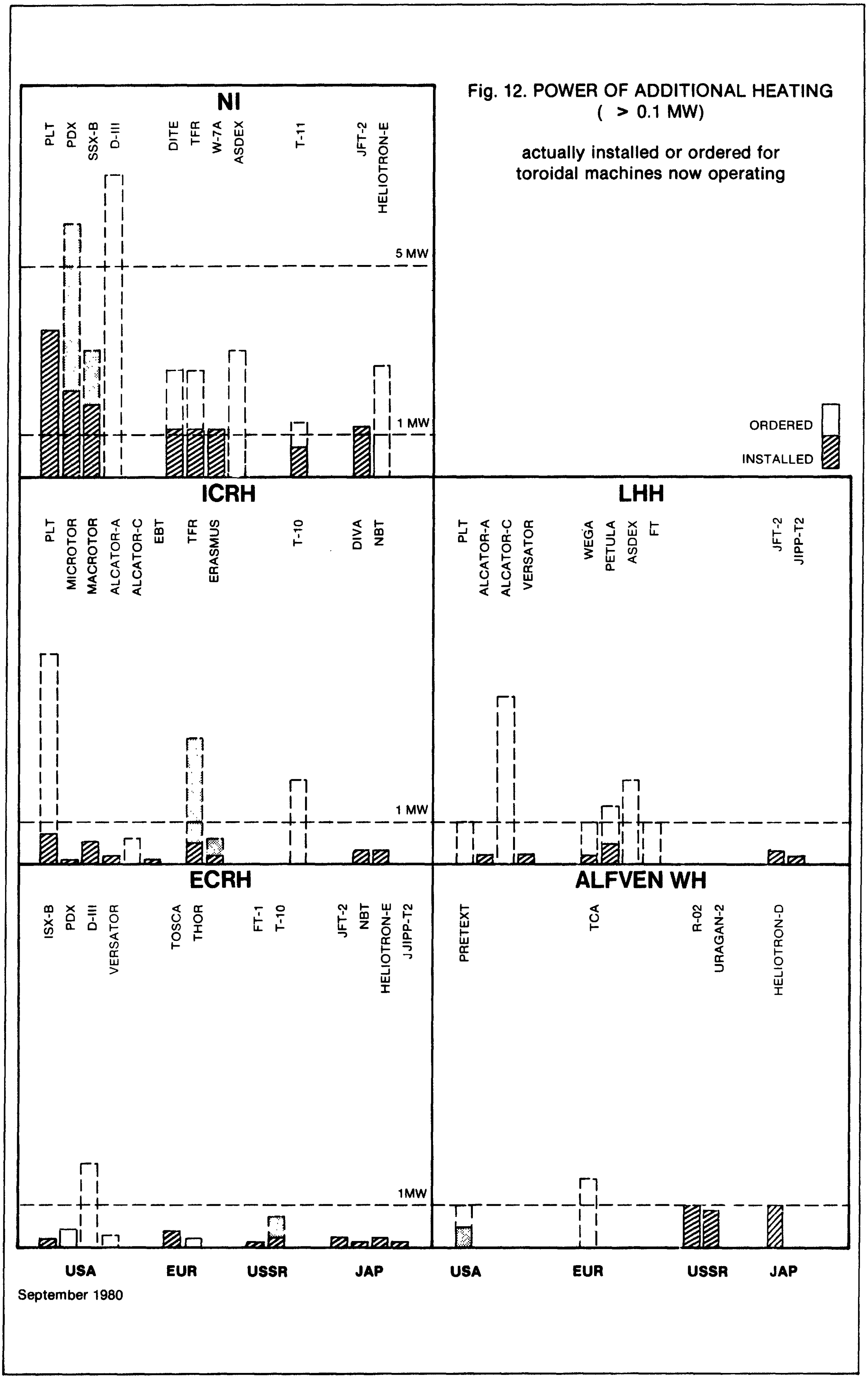


Fig. 12. POWER OF ADDITIONAL HEATING
(> 0.1 MW)
actually installed or ordered for
toroidal machines now operating
NI
PLT
PDX
SSX-B
D-III
DITE
TFR
W-7A
ASDEX
T-11
JFT-2
HELIOTRON-E
5 MW
1 MW
ORDERED
INSTALLED
ICRH
PLT
MICROTOR
MACROTOR
ALCATOR-A
ALCATOR-C
EBT
TFR
ERASMUS
T-10
DIVA
NBT
1 MW
LHH
PLT
ALCATOR-A
ALCATOR-C
VERSATOR
WEGA
PETULA
ASDEX
FT
JFT-2
JIPP-T2
ECRH
ISX-B
PDX
D-III
VERSATOR
TOSCA
THOR
FT-1
T-10
JFT-2
NBT
HELIOTRON-E
JJIPP-T2
1MW
ALFVEN WH
PRETEXT
TCA
R-02
URAGAN-2
HELIOTRON-D
USA
EUR
USSR
JAP
USA
EUR
USSR
JAP
September 1980