Engineering Problems of a Thermonuclear Fusion Reactor

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Thermonuclear fusion is one of the very few major options for covering the future energy needs. It is a challenge to develop the necessary physics and technology so that this source of energy becomes available for the benefit of mankind. As follows from Fig. 1, thermonuclear fusion exploits the differences in normalized masses between light elements and releases energy by fusing light nuclei into heavier ones. Up to 1% of the mass can be released – 10 times that available from fission.

It would be of the highest benefit if a fusion process could be utilized which involved only stable elements and would yield the reaction power via the kinetic energy of the stable nuclei produced. Radioactivity would then be completely absent. Such reactions do exist but, unfortunately, the one having by far the highest reaction cross-section does not belong to this category. In view of the immense difficulties, however, which have to be mastered before the harnessing of fusion power becomes practicable, we are obliged to utilize the largest available cross-section, at least for the first generation of fusion reactors:

$$D + T \rightarrow {}^4He + n + 17.6\ MeV$$

There is enough deuterium available in nature but no tritium because this isotope of hydrogen decays by beta emission with a time constant of only 12.1 years. In addition, one of the two reaction products is a neutron carrying 80% of the reaction energy.

Essentially, these are the two points from which most of the engineering difficulties start. Tritium has to be bred from lithium using the reactions:

$${}^6Li + n \rightarrow {}^4He + T + 4.8\ MeV$$

$${}^7Li + n \rightarrow {}^4He + T + n - 2.5\ MeV$$

In order to utilize the fusion neutrons for this purpose the breeding has to be done in a blanket surrounding the reaction chamber. At the same time, this blanket has the task of converting the kinetic energy of the fusion neutrons into useful heat. The ${}^6Li$-reaction even contributes to the power output of a fusion reactor, whereas the secondary neutron of the ${}^7Li$-reaction makes it possible in principle to achieve tritium breeding ratios larger than one, without excessive use of extra neutron multipliers.

It is worth reiterating that complications introduced by this class of property are not generic to fusion in general. They are a consequence of the particular fusion reaction selected, *i.e.* they are the penalty one has to pay for taking advantage of the relatively large cross-section of the DT reaction.

One difficulty generic to fusion arises from the fact that even the large DT reaction cross-section is by orders of magnitude smaller than the hydrogen atomic dimensions being of the order of the proton dimension. This unavoidably brings the repulsive and long-range property of the electric charge of the nuclei into play, and Fig. 2 gives an impression of how much more frequent coulomb collisions are with respect to fusion processes. From this figure, it becomes immediately apparent that fusion by colliding beams of energetic particles has no chance since the momentum of the beam particles is lost too fast for a sufficient number of fusion reactions to occur. One rather has to use a medium in which the frequent coulomb collisions, at least on the average, do not lead to a loss of energy of the energetic fusion candidates. Such a medium is a plasma of appropriate temperature, say 10 keV.

Under typical conditions of magnetic confinement, upon which method this paper is concentrated, the integrated path length of an average particle is of the order of the diameter of the Earth before it undergoes a fusion collision. This gives an impression on how large the reflection coefficient has to be at the ends of linear devices, or how small the leakage of particles has to be out of toroidal devices if the reacting plasma is to be confined within devices of reasonable dimensions. Under these conditions the fusion power density is proportional to the square of the density of the reaction partners ($D : T = 1 : 1$) times a function of their temperature. Magnetic confinement concepts are characterized by the maximum plasma pressure they are able to confine, i.e. by the product of plasma density times plasma temperature. If this product is kept constant, the particular energy dependence of the DT reaction yields the maximum fusion power density for plasma temperatures around 13 keV.

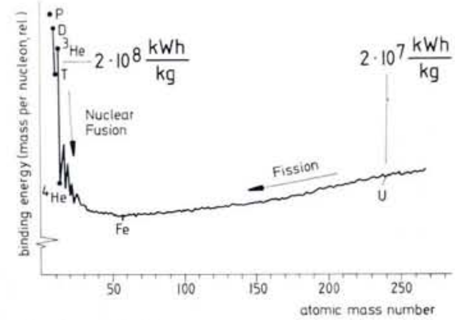
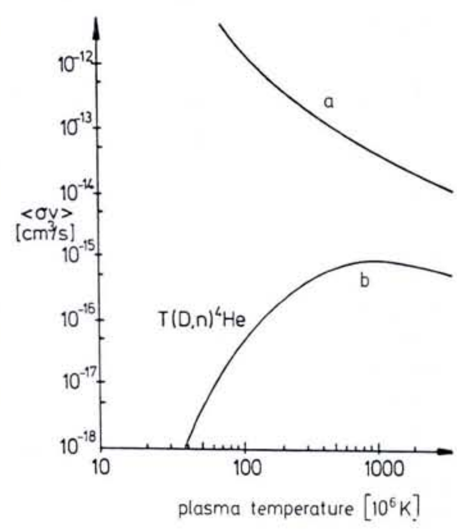


Fig. 1 — Binding energy per nucleon vs. atomic mass number.

We have now all the elements to define a fusion power reactor based on the concept of toroidal magnetic confinement. (In Europe all the preference is given to toroidal magnetic confinement as compared to linear mirror confinement.) In a toroidal configuration, magnetic fields primarily produced by currents in superconducting coils, confine a burning plasma of a temperature of 10-15 keV and the highest reasonable density (to achieve high power density). The plasma is surrounded by a blanket for breeding the tritium fuel component and converting the neutron energy into heat. An illustration of the major engineering difficulties connected with fusion reactors can best be achieved by inspecting the INTOR concept [1, 2, 3].

The INTOR concept is being developed by the INTOR Workshop which is an international, collaborative activity comprising Euratom, Japan, USA, and the USSR, running under the auspices of the IAEA, and concentrated on defining and designing the next step device in fusion programmes. INTOR is conceived to demonstrate the availability of reactor relevant burning-plasma physics and to serve as a test bed for developing and testing reactor technology. With these

Fig. 2 — $<\sigma v>$ vs. plasma temperature for elastic Coulomb collisions (a) and for DT fusion processes (b).



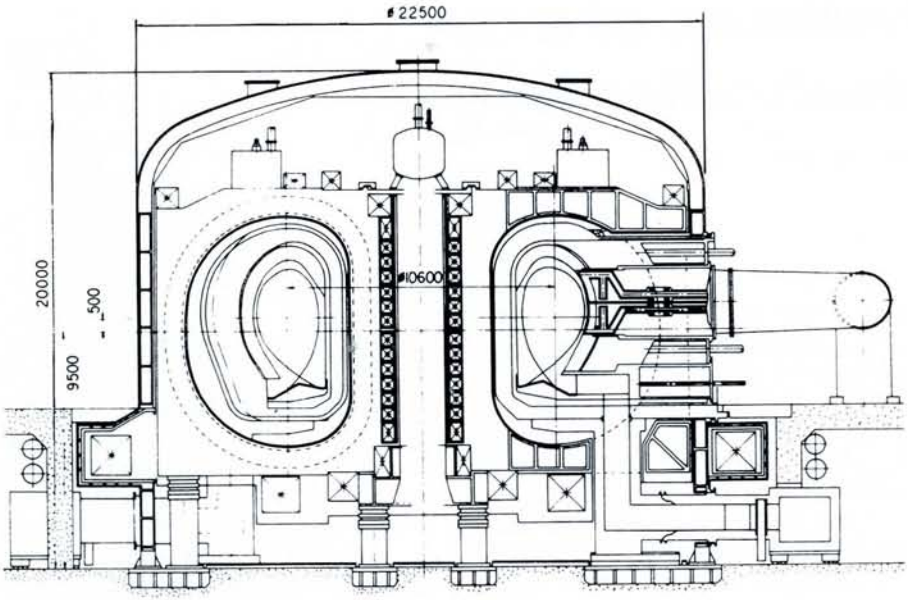


Fig. 3 — INTOR concept as of Phase II-A of the INTOR Workshop (artist's impression). The toroidal plasma is surrounded by a blanket and shield and these in turn by coils generating the toroidal magnetic field. Outside these coils ⊠ indicate further sets of coils exciting the necessary poloidal magnetic field. At the right-hand side an RF antenna system is indicated which has the task of heating the plasma to ignition temperatures.

aims INTOR has to involve all essential reactor components and is thus particularly suited to identify all of the engineering problems.

Fig. 3 is a display of the INTOR design concept at the end of Phase IIA, Part I of the INTOR Workshop. The innermost part of the device is the plasma doughnut which is to produce the fusion power. The plasma is confined by a strong toroidally oriented magnetic field which is twisted around a "magnetic axis", the centre line of the plasma cross-section. This way nested toroidal magnetic surfaces are formed which make particle and energy transport from surface to surface (i.e. from the inner part of the plasma towards its boundary) difficult, such that a rather efficient confinement of plasma density and energy is achieved. In order to maintain the burning state of the plasma in steady state, the remaining and unavoidable power losses resulting from particle collisions, radiation, and convection have to be balanced by the 20% fraction of fusion power carried by 3.5 MeV alpha particles. The quantitative achievement of this balance determines the plasma dimensions, parameters, and the strength of the magnetic field. Some key parameters of INTOR are listed in Table 1.

One of the major engineering problems arises from the interaction between the plasma and the surrounding material (first wall). Whereas the 14 MeV fusion neutrons have a rather large penetration length and thus lead to volumetric heating, the plasma is carried to the walls by energetic particles and radiation leading to surface heating and sputtering. Under standard conditions the sputtering was seen to be so large that a sacrificial extra thickness of the wall of 1-2 cm had to be provided to achieve a sufficient lifetime of the first wall — at least from the sputtering point of view. Such a thick wall, however, experienced unacceptable thermal stresses arising from the pulsed operation connected with tokamak devices. Moreover, the large amount of sputtered material had a significant probability of diffusing into the plasma core and quenching the burn by excessive heat losses via impurity radiation. This problem was recognized already very early and led to a screening of all available materials for high sputtering resistance. But the potential of all these measures is only a gradual reduction of sputtering at best.

A much more efficient method would be to establish high density, low temperature plasma edge conditions such that the energy of the impinging particles was below the sputtering threshold. Divertor discharges, which aim at connecting the outermost magnetic field lines with remote divertor chambers, offer the highest chance for establishing such conditions. As follows from divertor experiments like ASDEX [4]) and from the accompagnying theoretical investigations there is now a large confidence that the necessary boundary conditions can be established, so that the sputtering problem seems now to be somewhat alleviated.

Thus the main damage of the first wall is arising from the high energy (14 MeV), high neutron flux which will be around 2-5 MW/m² depending on concept. Structural parts which are exposed to this high irradiation should withstand a neutron fluence of at least 10-20 MWa/m² before replacement becomes necessary. The materials available, stainless and ferritic steels, are developed for fission purposes and thus for other neutron spectra. The high energy neutrons present in fusion reactors cause much higher helium production rates than are known from typical fission applications so that the properties of these materials under fusion conditions have to be evaluated. Such programmes have been started but they lack a powerful 14 MeV neutron source. Due to the long lead-time especially tailored materials can only be developed in the long run, whereas for the next step one has to select from amongst those available.

As is also apparent from Fig. 3, three independent systems of coils are necessary for generating the tokamak confining magnetic field. For reasons of low power consumption all the coils are conceived as being superconducting. A set of large planar coils (TF-coils) generate the toroidal magnetic field which provides the largest fraction of the magnetic field energy. The ohmic heating (OH) transformer coils excite a loop voltage and thus a plasma current whose magnetic field generates the necessary twist about the magnetic axis of the toroidal magnetic field. A third coil circuit (PF-coils) provides a poloidal magnetic field necessary to balance the hoop force of the plasma current and generate the shape of the magnetic field required for plasma equilibrium and stability. Thus the plasma current is an essential ingredient of the tokamak concept. During the pulse it has to be raised from zero to the full burn value and then

Table 1:
INTOR major parameters, Phase IIA, Part I

Major radius, R(m)	5.2
Plasma radius, a(m)	1.2
Elongation, k	1.6
Burn time, (s)	100/200 *
Duty cycle, (%)	70/80 *
Average beta, $<\beta>$ (%)	5.6
Plasma current, I(MA)	6.4
D, T-density, $<n_i>$ (m^{-3})	1.4×10^{20}
Ion temp., $<T_i>$ (keV)	10
Toroidal field, B_t(T)	5.5
Plasma heating (MW)	50 ICRF
DT thermal power, P_{th}(MW)	620
Fluence goal (MWa/m²)	3
Neutr. wall load, P_n(MW)/m²)	1.3

* early/later stages of operation

to be reduced again. It is for this reason that the three magnetic circuits have to be excited and controlled separately and cannot be combined into one single coil system. This is only possible if the *total* confining magnetic field were produced by external coil currents as it is the case for stellarators. By the same token stellarators also have the inherent potential for steady state operation.

Resulting from the short decay length of multi-pole poloidal fields there is a high incentive to arrange the PF-coils close to the plasma surface. This, however, is not really possible since the coils have to be protected against excessive neutron irradiation. Moreover, located that close to the plasma, these coils would have to be interlinked with the TF-coils, making any assembly and maintenance scheme an extremely difficult one. Therefore, there is practically no alternative to arranging the PF coils outside the TF coils and accepting PF coil currents one order of magnitude higher than needed otherwise.

This particular arrangement of the PF coils relative to the TF coils has even further engineering consequences in that their magnetic interaction leads to a substantial twist about the axis of rotation of the device. This twist has to be balanced by a massive structure which must be broken up into a large number of mutually insulated sub-structures in order to limit the occurrence of eddy currents and to allow the time-varying poloidal magnetic fields to penetrate to the plasma on a fast enough time-scale. There are engineering solutions to this problem but they are only just compatible with presently available technology.

Moreover, the large distance between plasma and TF coils allows the generation of only rather open divertors, whereas present experiments in ASDEX have used a rather stiff and closed divertor configuration. In order to extend the data base to the open divertor and at the same time check the viability of the INTOR-type configuration, ASDEX-upgrade is being built at the IPP Garching (see Fig. 4). Results from this device are expected from 1988 onwards.

Other engineering difficulties arise from the introduction of a blanket and a shield between plasma and coils. As already said, these components are necessary to breed tritium, to convert the neutron energy into heat and to reduce the primary fusion neutron flux by six orders of magnitude or so before it arrives at the superconducting coils. This shielding factor is necessary for three reasons:

(i) Excessive heat loads in the coils have to be avoided to limit the refrigeration power.

(ii) Too high radiation damage of the superconductor and its stabilizer and

(iii) of the insulating material would lead to an early degradation of the coil performance.

Frequent replacement of the coils, however, is both much too difficult and too expensive. Thus the coils are assumed semi-permanent in the INTOR concept. All these three points require shielding factors of about equal magnitude.

It would ease the problems considerably if for the first device of this kind one could dispense with at least tritium breeding. But in view of the large quantities of tritium to be burned, which can reach several kilograms per annum for INTOR (about one 1 kg/day for reactors) this is not possible, particularly for Europe with its limited capability for tritium production. Thus, any INTOR-like device has to breed quite a fraction of the tritium it consumes; and this has to be done without affecting the availability of the basic machine. The dilemma is that this blanket has to be designed before INTOR can contribute to blanket development.

There are a large number of breeder materials from which a selection has to be made. They include liquid metals like pure lithium-lead, or ceramics like lithium-oxide, -aluminate, -silicate, etc. Liquid metals are susceptible to large, design-dependent MHD forces arising from the liquid metal flowing through a magnetic field or from time-dependent magnetic fields acting on the electrically conducting liquid. Ceramics do not show this effect. Their difficulties are connected with the high operating temperature needed for tritium extraction and the not too wide temperature window before sintering sets in. All of the materials show considerable corrosion problems so that a choice is difficult to make at present. One has to wait for results of the screening and development programmes which have been initiated in order to produce the data base needed for the up-coming decisions.

There is a red thread connecting the above examples: many of the problems of tokamak reactor technology are resulting from the pulsed nature of the tokamak operation which is a consequence of the OH current being inductively driven. Steady state operation would alleviate these problems to quite an extent. There is a considerable effort going on in the fusion programmes to convert also the tokamak into a steady state machine by driving the plasma current by other means, preferentially by RF waves [5]). All the studies done so far have indicated, however, that the power consumption connected with current drive is too high to be acceptable. In fact, this power might even exceed the RF-power needed for reactor start-up and is needed in a CW fashion. Repetitive recharging of the transformer at lowered plasma densities, each time when it ran out of volt-seconds, might have a higher chance as far as power consumption is

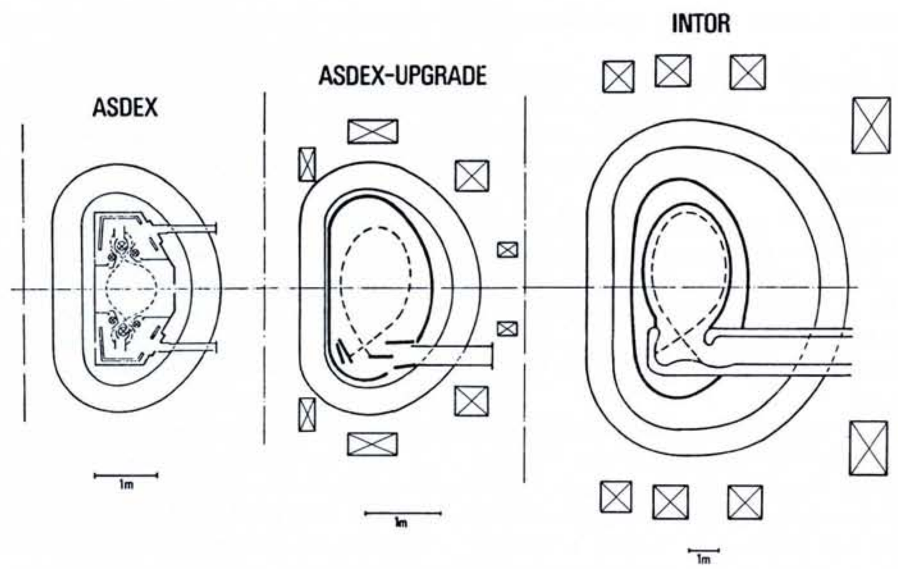


Fig. 4 — Comparison between the Divertor Experiments ASDEX and ASDEX-Upgrade and INTOR which is relying on a functioning divertor. Note the big difference in scale.

concerned but it requires a rather complicated operation scenario and might be rejected finally for this reason. Perhaps, then will come the time when the stellarator's inherent potential of DC operation is given preference if also its other properties have continued to develop favourably in the meantime.

One also has to remember that in all of the world fusion programmes, the development of fusion technologies was set aside initially until sufficient confidence had been developed that all the physics hurdles could be overcome. In view of the large extrapolations required this was a reasonable strategy, but it has created a situation where one now can only screen materials and technologies which have been developed for other purposes and check how they will perform under fusion conditions. It is highly unlikely that the optimum materials and technologies are already among the fully developed ones, considering that the fusion conditions are rather unique. This is not too bad a situation though, because it now allows orienting the various technology development paths with much broader and deeper knowledge on the required properties and achieving a much higher cost-effectiveness than it would have been possible earlier.

Design work for INTOR-like devices will continue to identify the engineering needs in greater and greater detail. The R and D programmes will provide answers to these engineering needs, and one has to be careful not to conclude that the state of the art reached up to now already represented the full potential of fusion technology. Nevertheless, in its deliberations, the INTOR team has stated that solutions have already been found to all engineering problems encountered, although admittedly some of these solutions are of a rather complicated nature. The construction and operation of an INTOR-like machine would therefore serve as an invaluable basis for the development of second generation technologies aimed at further improving fusion reactor engineering.

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4. — M. Keilhacker *et al.*, *Plasma Physics and Controlled Nuclear Fusion Research* (Proc. 10th Int. Conf. London 1984), Vol. I, IAEA Vienna (1985), 71.
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D.N. Stacey writes:

In my article, I referred to the Z^3 law derived by M. and Mme Bouchiat as a major step in stimulating the programme of research on parity nonconservation in atoms. In singling out this contribution for particular mention because of its importance I certainly did not intend to associate M. and Mme Bouchiat with early overestimates of PNC effects, and indeed the article goes on to discuss these in the context of bismuth. I am therefore glad that the Editor has helped to avoid any possibility that readers might be misled on this point by printing the above comment.

LETTER TO THE EDITOR

Parity Violation Predictions

Comments on the Review Article of D.N. Stacey

The uninformed reader may get the impression, from the review article by D.N. Stacey published in *Europhysics News* **16** (1985) 2 (February), that in the early work of C.C. and M.A. Bouchiat [1]) the parity violation effects in heavy atoms were grossly overestimated: "The effects are much smaller than were at first predicted to be". In reference 1, a detailed evaluation was given for the case of atomic caesium only. The results of the Paris experiments can be expressed in terms of the ratio of parity violating 6S-7S electric dipole amplitude $E_1^{p.v.}$ to the spin-independent transition dipole amplitude αE_0 induced by an electric field E_0. The predicted ratio was:

$E_1^{p.v.}/\alpha \approx 2.8 \times 10^{-4}$ V/cm

which is to be compared to the experimental ratio (deduced from reference 2)

$1.56 \pm 0.17 \pm 0.12 \times 10^{-4}$ V/cm.

The parity violation effect was indeed overestimated by a factor 1.8 but the order of magnitude was clearly the correct one. The p.v. electric dipole $E_1^{p.v.}$ is proportional to the weak charge of the nucleus Q_w which depends upon the weak mixing angle θ_w, the only free parameter in the Glashow-Weinberg-Salam electroweak theory. In our 1974 work, we used a preliminary value of θ_w coming from the very early neutrino experiments and got a value of Q_w of about - 100 while the most recent experimental analysis, involving radiative corrections, gives $Q_w = -68.6 \pm 0.3$. In this way, a substantial part of the discrepancy is accounted for. The remaining 20% reduction factor is very likely to be associated with many-body effects not included in our early evaluation. There exist now several calculations of p.v. in atomic caesium, involving a treatment of many-body effects; all the results lie within the experimental error bars [3]).

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M.A. and C. Bouchiat