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# Reactor requirements

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Reactor requirements can be put in a clearer perspective if they are seen together with those of the preceding steps, i.e. JET, NET and DEMO. Extrapolation from JET is moderate in plasma size, considerable in machine size and large in plasma performance and machine endurance. Extrapolation from the present understanding of tokamak plasma physics, to be developed further both in JET and elsewhere and, from technologies that we plan to develop for NET, can lead to a viable reactor, but with uncertainties regarding economic performance. Although overall economic predictions for fusion may be premature, consideration of capital costs at least indicates the desirable direction and incentive for further progress in plasma physics and technology. Among physics issues the plasma power density (which is directly related to operational limits on beta, plasma density and plasma current), the power and particle loads on the walls of the device as well as the plasma exhaust requirements, and the prospects for steady-state operation, are of primary importance. Technologically the most severe requirements are on operational reliability, lifetime of plasma-facing components, and remote handling.

## 1. Introduction

The presently envisaged strategy towards the commercialization of fusion in Europe is a three-stage process of development beyond the JET experiment. NET (Toschi 1985), an engineering test facility with reactor-relevant plasma operation, would be used to gain experience of long-burn-time ignited plasma physics and to develop technical solutions to the problems of power generation and tritium breeding. This would be followed by DEMO, a test device for the long-term technical feasibility, safety and reliability of the designs developed on NET, which should eventually provide the database on which the economic power generation potential of tokamak fusion could be judged. A prototype commercial-sized reactor (PCSR) would then be built to demonstrate to the utilities that fusion was a desirable and viable alternative to its competitors.

To be commercially viable against this background, fusion, like any future energy source, has to meet certain basic requirements. From an economic standpoint it must generate power at a cost that is comparable to that of its competitors, in a device of a unit size and availability acceptable to the utility networks. Environmentally, it must be safe in operation and produce manageable short-term and fewer long-term waste disposal problems than its competitors. The last aspect is not considered here.

## 2. Relation between physics and technological constraints

The understanding of the interdependence between plasma physics, technology and engineering requirements is a prerequisite to identifying the most promising route towards the optimization of reactor performance. The main performance parameters depending on plasma physics assumptions are the achievable fusion power density in the plasma, the confinement capability or 'ignition margin' (i.e. the ratio of $\alpha$-particle power to loss power extrapolated to the operating point), the power and the particle load on the walls, and the burn pulse length.

### (a) *Power density*

The achievable fusion power density, $p_F$, in the range of the optimum working temperature of 10–15 keV, is proportional to $(nT)^2$, where $n$ is the plasma fuel particle density and $T$ the plasma temperature, or alternatively $p_F = f\beta^2B^4$. Disregarding issues related to the presence of ions different from deuterium and tritium, here $f$ is the fusion power density ratio, which is related to temperature and density profiles and is favoured by peaked profiles, $B$ is the magnetic field on the plasma axis and $\beta$, the ratio between the plasma pressure ($\propto nT$) and the magnetic pressure ($\propto B^2$) is constrained by plasma stability. The maximum percentage $\beta$ as derived from the stability analysis against ballooning and kink modes globally scales as $\beta = (gI/aB)$, where $I$ is the plasma current in megaamps, $a$ is the plasma minor radius and $g = 3$–$4$. Present experimental evidence on operational limits on $\beta$ is consistent with this equation.

For a given operating temperature and magnetic field the $\beta$-limit amounts to a density limit too. The plasma density $n$, however, may have a more stringent limit as a consequence of a thermal instability at the plasma edge; such a limit can be expressed as $n/(10^{19}\ \mathrm{m^{-3}}) \leqslant (MB/Rq_I)$, where $R$ is the plasma major radius, $q_I$ is the plasma safety factor related to current, and $M$ is a parameter whose experimental value varies from 10 to 30.

As $p_F \propto fg^2B^4a^2/q_I^2R^2$, high-power density is favoured by high field and low aspect ratio, $R/a$; but both are soon constrained by stress limits in the magnetic structure. There is also a lower bound on the safety factor, which corresponds to $q_I \approx 2$, related to the appearance of disruptions. Therefore substantial improvements in plasma power density can only be obtained by suitable plasma shaping to increase $I/aB$ and possibly by an increase in the $g$ factor. Again, shaping of the plasma as well as control of the current profile are suggested as means of increasing $g$, but the potential of this scheme is uncertain and the benefits might not compensate for the demanding engineering requirements that arise, for instance, in realizing extreme magnetic configurations.

In the case of NET the plasma power density is anticipated to be 1.5 MW $\mathrm{m^{-3}}$ with $M \approx 30$ and $q_I = 2.1$. The suitability of this value for producing an economic reactor is considered in §5.

### (b) *Confinement*

The 'ignition margin', for $T = 10$–$15$ keV, is proportional to the product $n\tau T$ ($\tau$ is the energy confinement time). This is subject to a minimum requirement, namely that ignition must actually be reached. Conversely, if the ignition barrier is overcome a new requirement appears, namely that the plasma confinement must be degraded in a controllable way to permit a steady-state discharge. Because of the large uncertainties still present in the confinement properties of tokamak plasmas, reaching ignition in the next generation machines is one of the critical questions for the parameter choice of the device. However, provided that plasma confinement time increases with device dimensions and/or plasma current, this is no longer a concern in still larger devices. Such a scaling, is, for example, the ASDEX H-régime scaling (Gruber 1984) given by $\tau = (0.1\,RI)$, which is presently adopted as the reference scaling for

NET. A temperature clamping effect, if it allows the temperature to be increased in a larger size of device, plays a similar role. On the other hand, if the confinement time degrades continuously with the heating power (Goldston 1984) the tokamak approach to a reactor could be jeopardized.

Controlled confinement degradation is necessary as a means of controlling the plasma burn temperature in a reactor. If there is a soft $\beta$ limit or if a temperature clamping effect occurs, the reactor parameters can be chosen such that active burn temperature control during the burn pulse is not necessary. Otherwise, plasma confinement as well as the thermal stability of the plasma equilibrium at the operation point would have to be feedback controlled. This is a demanding problem. Various methods have been proposed including variable magnetic field ripple, plasma compression–decompression and the injection of impurities. All of these systems would affect the engineering design of a reactor.

### (*c*) *Particle and power load at the divertor target*

For controlling particle and power exhaust a poloidal divertor appears to be the most promising. A simple analysis indicates that the power load $p$ at the divertor target scales as $P \sim Q_{\perp}/R\Delta$, where $Q_{\perp}$ is the non-radiated plasma power flowing into the scrape-off layer, in the NET reference options assumed to be about two-thirds of the $\alpha$-particle heating power, and $\Delta$ is the characteristic power decay length in the scrape-off layer, which is related to cross-field heat transport in the scrape-off layer.

The peak power load on the NET divertor plates reaches, in an optimized geometry, 5–7 MW m$^{-2}$, causing severe erosion in addition to thermomechanical stresses. The erosion due to physical and chemical sputtering is very sensitive to plasma parameters and composition near the plates; for tungsten at 1800 K, neglecting both chemical sputtering and redeposition, the estimated peak erosion is as much as 15 mm during the lifetime of NET (*ca.* 1 MW a m$^{-2}$ fluence).

In devices of larger power output, such as a reactor, the technical requirements are more demanding because, with otherwise fixed conditions, the peak power load on the divertor plates increases linearly with plasma size. This tendency requires the development of methods to control the radial energy transport in the scrape-off layer to widen the power deposition profile.

### (*d*) *Burn length*

The burn length, for conventional tokamak operation, is determined by the transformer capability. Non-inductive current drive, in principle, could permit steady-state operation. However, under burn conditions in a reactor, large power demands are anticipated for this scheme (see, for example, Wégrowe 1985) so that it may not be viable for an efficient reactor. The RF power in megawatts required ($P$) to drive a current $I$ can be expressed as $P \approx 3 \times 10^{-20}\, RnI$. Preliminary calculations show an increase in specific capital cost (see §5) of 40% and a halving of overall plant efficiency when a current drive system capable of producing 0.05 A W$^{-1}$ is used.

A possible alternative use of non-inductive current drive is for maintaining only the current in a quasi-steady state. This requires only moderate power, the plasma current being non-inductively sustained at low plasma density and temperature during transformer recharging. Although such a scheme would have the potential to reduce the cyclic load on the magnet system, this is not the case for the thermomechanical loads on the in-vessel components. To judge whether such an operation scenario is attractive for a reactor, the cost saving due to reduced fatigue would have to be quantified.

## 3. NET parameters and objectives

The device parameters which form the basis for the predesign of NET (NET team 1985) are the result of extensive optimization studies aiming at a minimum cost device for given performance objectives (Borrass 1985*a*). The optimization studies have shown that engineering constraints (e.g. access to the in-vessel components, space for blanket and shield, stress and strain limits) establish among the machine parameters a strong correlation that is independent of the physics scaling. Therefore the parameter set is largely determined by fixing one parameter of the machine, for instance plasma current or minor radius.

A reference set of parameters has been chosen for NET but the details of the plasma configuration (whether single-null or double-null) require further study. The main parameters for the double-null variant are shown in table 1 and a sketch of the device in figure 1.

Table 1. Major parameters of NET double-null reference option

| | | | |
|---|---|---|---|
| plasma major radius | 5.2 m | beta scaling factor, $g$ | 3.5 |
| plasma minor radius | 1.35 m | total beta | 5.6 % |
| plasma elongation | 2.2 | DT beta | 4.2 % |
| field on axis | 5.0 T | poloidal beta | 1.4 |
| toroidal field ripple at outboard | | fusion power | 600 MW |
| plasma edge | $\pm 1.2$ % | fusion power density | 1.6 MW |
| plasma current | 11 MA | neutron wall loading | 1 $MWm^{-3}$ |
| safety factor | 2.1 | ignition margin | 2.9 $MWm^{-2}$ |
| average burn temperature | 10 keV | burn time | 600 s |
| average electron density | $1.6 \times 10^{20}$ $m^{-3}$ | additional heating power (absorbed | 50 MW |
| Murakami parameter, $M/q_I$ | 16 | by plasma) | |

As far as possible, NET will adopt reactor-relevant technical solutions to design problems and will thus incorporate superconducting magnets in the basic machine. Its test programme will be aimed mainly at identifying the most appropriate designs for the generation of heat at temperatures suitable for power generation, while maintaining an adequate tritium breeding ratio, and it will do this by testing out a number of blanket concepts. Because this will be the first opportunity to test materials and components in an integrated way in a real fusion reactor environment, considerable evolution of the design concepts is expected and allowed for in the testing programme of NET. NET will also be used to assess the ability to extract tritium at the required rate to refuel the plasma, while keeping a low tritium inventory (plant safety relevant), and to demonstrate the ability to remotely maintain a reactor within an acceptable timescale (plant availability relevant).

To achieve its objectives, NET will operate in stages. Initially emphasis will be on maximizing the physics capability of the apparatus. The basic machine (i.e. the apparatus without blanket) could operate with a higher plasma current in a larger plasma so that the ignition margin as calculated with the ASDEX H-régime scaling would increase by about 50 %. A larger plasma size also offers greater opportunity for plasma shape optimization. The operation will continue

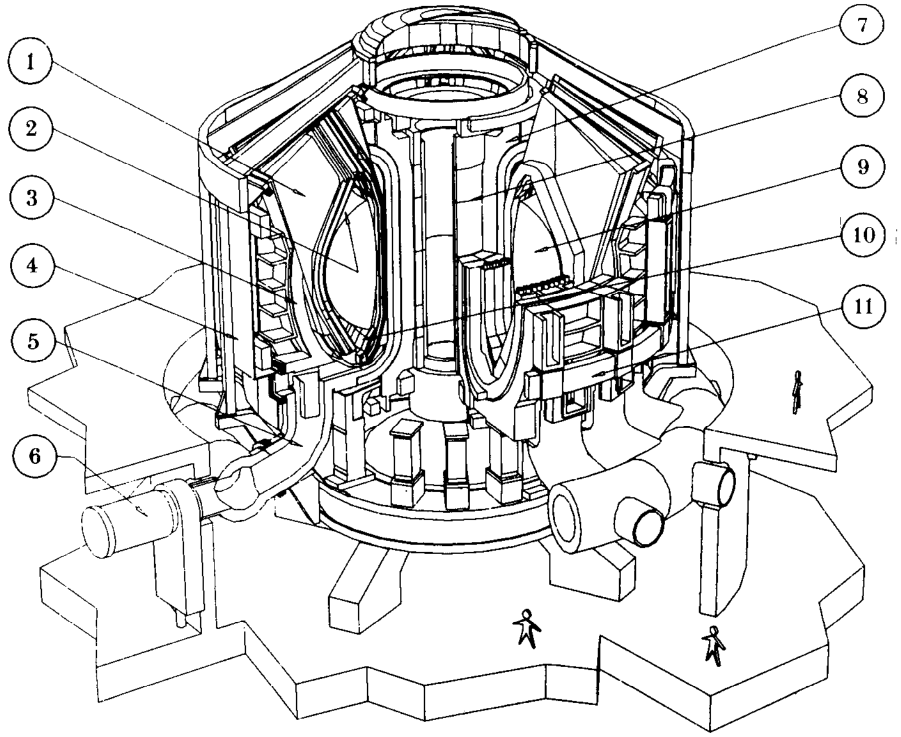


FIGURE 1. Sketch of NET-DN variant, showing (1) blanket, (2) first wall, (3) shield, (4) cryostat, (5) plasma exhaust duct, (6) vacuum pump, (7) toroidal-field coil, (8) inner poloidal-field coil, (9) plasma, (10) divertor plates, (11) outer poloidal-field coil.

through three further stages with increasing blanket coverage, each stage separated by a one-year shut-down for major refurbishing. In each stage the machine will reach an integral operating time of about 100 days. Ten days continuous operation is, according to the present analysis, needed to assess the main blanket performance (neutronics, tritium breeding and recovery, tritium permeation and inventory, energy extraction) at a specific operating condition (e.g. coolant temperature). With the above programme it will be possible to test two design iterations of a blanket concept and to test different concepts in parallel.

The NET operation will require about 26 kg of tritium to be supplied mainly from an external source. To maintain flexibility in staging the installation of blanket segments, only about one third of the tritium burned in the reactor will be expected to be bred in the blanket.

At the end of the operation the machine will have accumulated $10^5$ cycles and about 400 days of integrated operating time corresponding to 0.8 MW a $m^{-2}$ of neutron fluence. At this point the selection of workable contenders for all DEMO components will be possible and thus NET will have satisfied its objectives. Because DEMO will be closer in size and in power loading to the final commercial reactor conditions it is considered more relevant to carry out the long term endurance testing of components in the early phases of DEMO operation.

However, should the machine be operating with reasonable availability and breeding adequate tritium at the scheduled end of its operation, it could turn out to be desirable to continue using it for testing beyond the above objectives. For this reason in the design of NET the shielding is sufficient to protect the coils for an integrated operating time of 1000 days.

## 4. Beyond NET

Although in the technological area there is a stronger link between what is developed on NET and what is needed in DEMO, whereas between JET and NET there is a stronger link in plasma physics development, JET itself will give useful technological input to NET, particularly in the areas of first wall protection against non-standard operating conditions (e.g. disruptions), plasma heating, vacuum pumping and remote handling.

The likely extrapolation factors of the major parameters of JET and the subsequent steps in the programme are shown in table 2. In defining the parameters of DEMO and PCSR the

Table 2. Extrapolation factors from JET to future devices

| | JET | NET | DEMO | PCSR |
|---|---|---|---|---|
| plasma minor radius | 1.0 | 1.2 | 1.6 | 2.2 |
| plasma major radius | 1.0 | 1.9 | 2.9 | 3.4 |
| plasma current | 1.0 | 2.1 | 2.7 | 3.8 |
| pulse length | 0.01 | 1.0 | 5.0 | 5.0+ |
| total thermal power | — | 1.0 | 3.0 | 5.7 |
| neutron wall loading | — | 1.0 | 1.4 | 1.8 |
| neutron fluence (first wall) | — | 1.0 | 5.0 | 10.0 |

same physics scaling as for NET has been used. The PCSR has a total power sent out of 1200 MW(e). As the main purpose of DEMO is the testing of in-vessel components, the highest possible wall loading would be desirable. Thus, DEMO is chosen here to be the minimum cost device having 80% of the wall loading of the PCSR, and is therefore not the cheapest device at its electrical power output level.

There are clearly some considerable steps to be taken to move from JET to reactor conditions. The major ones are in pulse length and power handling, which NET will address, and in endurance, which DEMO will quantify. Table 2 supports the view that testing on a device of the scale of NET is representative of the essential scientific and technological aspects of fusion power reactors.

## 5. Sensitivity studies on reactor performance

The code developed for NET sensitivity studies has been adapted to reactor requirements and used to investigate the influence of various parameters such as unit size, beta level, safety factor $q$, stress level and profiles on the capital cost of the apparatus (Spears 1985). Typical results are shown in figure 2 and summarized in table 3. This figure describes the variation of specific capital cost (i.e. cost per unit power sent out) relative to a specific device (a 1200 MW(e) reactor with the plasma physics of table 1) as a function of $g$ and power sent out. The results are plotted against mass power density in the fusion power core, that is, the power sent out divided by the mass required for the torus (first wall, blanket, shield, etc.) and magnets (including structure). Also shown is wall loading, which increases strongly at high mass power density.

### (a) *Unit size*

The economy of a fusion reactor is particularly sensitive to unit size up to about 1200 MW(e) (from a 600 to a 1200 MW(e) unit the specific capital cost reduces by 36% at $g = 3.5$), the

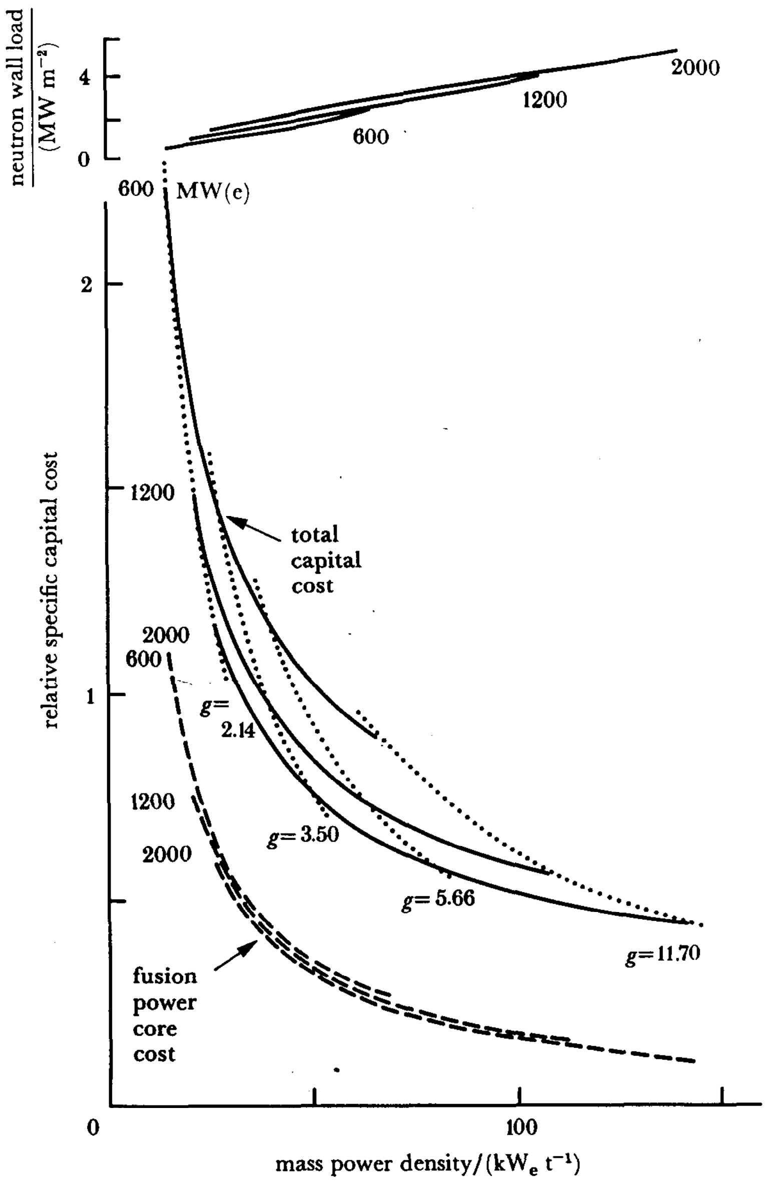


FIGURE 2. Correlation between capital cost, neutron wall loading and mass power density for minimum-cost devices at given $\beta$-level and power sent out.

gain being more pronounced at lower $g$ values. Beyond a 1200 MW(e) unit, the gain in capital cost is more modest and thus a reasonable unit size would be in line with recent fission reactor installations, at least in Europe (i.e. 1200–1500 MW(e)).

### (*b*) *Value of g*

An increase in $g$ can be used either to increase the power density and/or to reduce the magnetic field, both leading to a capital cost saving. Figure 2 shows that the value $g = 3.5$ lies at about the knee of the curve of cost against mass power density; in fact a $g$ reduction of about 40 % induces a cost increase of about 50 % whereas a $g$ increase of 60 % induces a cost saving of 23 %. Reductions of $g$ below the reference value of 3.5 would therefore quite seriously affect reactor economics.

TABLE 3. PARAMETERS OF MINIMUM-COST DEVICES AT 1200 MW(e)

| | | | | |
|---|---|---|---|---|
| mass power density/(kW(e) $t^{-1}$) | 38 | 50 | 100 | 150 |
| beta-level factor, $g$ | 3.5 | 4.6 | 10.7 | 21.7 |
| useful beta (%) | 3.0 | 3.9 | 8.6 | 18.9 |
| relative specific capital cost | 1.0 | 0.85 | 0.6 | 0.50 |
| fusion power core cost fraction | 0.43 | 0.4 | 0.3 | 0.23 |
| mean neutron wall loading/(MW $m^{-2}$) | 1.8 | 2.2 | 3.8 | 5.0 |

Table 3 shows that a reduction of 40% in the specific capital cost (corresponding to a mass power density of 100 kW(e) $t^{-1}$) would require an increase of $g$ by a factor of 3 if power output is unchanged. The present status of physics investigation suggests that this may be a difficult target to achieve. Furthermore, these trends in cost may turn out to be weaker than is shown because a high wall loading is required for a device at the cost optimum design point at high $g$. This might increase capital cost, over that given here, as more robust and costly technical design solutions may be required to withstand higher heat loadings. It might also increase generating costs by lowering plant availability due to more frequent material replacement. Wall loading may be kept approximately constant by using an increased $g$ to reduce toroidal fields. The cost reduction achieved is, however, rather modest as only the magnet cost is affected, all other device dimensions remaining essentially unchanged.

Figure 2 and table 3 also show the contribution to the capital cost made by the fusion power core itself. For the reactor, the design of this region is speculative at present, and its specification is the purpose of the future development programme. However, for fusion to become competitive, it is essential to develop, alongside the best plasma physics available, the most cost-effective technical solutions possible. These two development programmes must therefore proceed hand in hand.

### (c) *The sensitivity to q, stress limit and burn duration*

The sensitivity of the cost of the reference point (i.e. 1200 MW, $g = 3.5$) to changes in some of the assumptions made is shown in table 4 for the most sensitive parameters. The sensitivity is defined as the relative change in the cost, divided by a given relative change in the parameter.

Three plasma physics parameters ($g, f$ and $q_I$) head the list. These parameters are not independent because the attainable values of $g$ and $q_I$ depend on the radial profiles of plasma density, temperature and current density in a way that will only be finally determined after extensive experimentation on reactor-level plasmas. These profiles are implicit in $f$, which, for a parabolic radial temperature profile with a flat density profile, has a value of 1.5 for a density weighted mean temperature of 10 keV. Stronger peaking of the pressure profile would enhance the value of $f$, but whether this is beneficial depends on there being no offsetting changes in $g$ and $q_I$.

TABLE 4. SUMMARY OF MOST SENSITIVE PARAMETERS

| parameter | value | sensitivity |
|---|---|---|
| beta level, $g$ | 3.5 | −0.57 |
| fusion power density ratio, $f$ | 1.5 | −0.32 |
| safety factor, $q_I$ | 2.1 | 0.23 |
| TF coil stress level | 180 MN $m^{-2}$ | −0.14 |
| burn time (volt seconds) | 5000 s | 0.11 |
| blanket thickness | 0.55, 0.85 m | 0.11 |

It would appear that changes in $g$ and $q_I$ should be equally effective in increasing plasma power density and hence reducing cost, all else being equal. When $g$ is raised at constant $q_I$, $aB/I$ remains fixed for fixed aspect ratio, so reductions in $a$, $B$ and $I$ are allowed and all produce additive cost benefits. However, when $q_I$ is reduced at constant $g$, $aB/I$ must be reduced in line with $q_I$, so although reductions in $a$ and $B$ can be made, an increase in $I$ is necessary to keep power constant, resulting in a smaller overall cost benefit for $q_I$ reductions.

Stress levels in the toroidal-field coil are also quite important, though less than these physics parameters. The use of better quality materials in superconducting coil manufacture may ease this limit towards higher values, but superconducting material itself is strain sensitive and this provides a nearby limit. Similarly blanket thickness is not a major driver of costs within the likely range of uncertainty in its value. The values shown in table 4 are those inboard and outboard of the plasma respectively and are assumed to be 20 cm larger than typical blankets that might be employed on NET, to hopfully provide adequate tritium breeding in the power reactor. Of course, if suitable blankets cannot be designed to fit within this space restriction, more space must be provided as tritium breeding is an absolute necessity of the DT-based fusion devices considered here.

Considering only the impact of transformer capability and ignoring the effect of fatigue resistance on costs, sensitivity to burn time is also indicated in table 4. On this basis, the effect is fairly small. Here the reference burn time is 5000 s and 30 % of the plasma inductive flux is assumed to be required for 1000 s. For power reactors the longest possible burn times are needed to reduce cyclic mechanical fatigue induced failure of coils and their structures and cyclic thermal fatigue of torus systems such as the first wall. With the value chosen for burn time in the reactor studies reported here (i.e. 5000 s), just over $10^5$ cycles would be required over the operating life of the plant.

### (*d*) *Cost model*

The above analysis relies on the use of cost models, which, although representative of todays best estimates, obviously have uncertain credibility in absolute terms. The conclusions, however, are only affected by the relative cost values. The results of figure 2 are also intuitively appealing because they show a basic proportionality of the fusion-power core cost and the mass (hence the almost superimposed curves) with an additional dependence of total cost on core mass (i.e. volume, affecting building size), and on power sent out.

As has already been pointed out (Borass 1985*b*), the choice of various cost drivers (e.g. toroidal-field magnetic energy, toroidal-field coil outer radius, engineered volume) on their own as figures of merit results in somewhat different parameters at the optimum design point. However, the cost of all such devices is quite similar. This is to be expected because cost undoubtedly depends on all these figures of merit, plus others (e.g. peak field, power sent out) weighed together. Therefore, the inclusion of all cost drivers within a cost model gives deductions based on that model a higher credibility than any analysis based on the size of a single cost driver could produce.

## 6. Conclusions

NET, and the power-generating reactors to follow it, require considerable steps forward beyond JET in the areas of pulse length elongation, power handling capability and tritium breeding. The high apparent capital cost of fusion reactors provides a very strong incentive to aim in experiments for the highest beta-level within a given configuration, but this must be done without adding significant plant complexity or incurring offsetting costs by requiring expensive design solutions at high wall loading. Although a reactor based on the tokamak appears feasible, its economic viability will be in doubt until engineering design solutions have been further developed and tested under reactor conditions in NET. It is also hoped that JET, in performing its mission, will help identify ways of improving on the present perception of plasma physics to aid this process.

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## *Discussion*

D. C. Robinson (*Culham Laboratory, Abingdon, Oxfordshire*). In his presentation Dr Toschi described the double-null magnetic configuration and yet single-null configurations are being pursued on other tokamaks and possibly on JET. Could he comment on the advantages and disadvantages of a single-null configuration compared with a double null on NET?

R. Toschi. We in NET will keep both options open until we have studied in greater detail the double-null option, as we did for the single null. Double null offers advantages over single null because it is a symmetric configuration (e.g. plasma control, electromechanical loads) and has higher beta and a reduced capital cost but problems may arise from no pumping at the upper plates and from the removal of these plates.