# The Tokamak -

# An Imperfect Frame of Reference?

K.H. Schmitter

IPP 4/201 März 1981

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This report covers the contents of a lecture presented at the
INTERNATIONAL SCHOOL OF "FUSION REACTOR TECHNOLOGY"
(E. Majorana - C.S.C. - Erice)
5th Course: Unconventional Approaches to Fusion
Erice-Trapani-Sicily: 16-25 March 1981.

*Die nachstehende Arbeit wurde im Rahmen des Vertrages zwischen dem Max-Planck-Institut für Plasmaphysik und der Europäischen Atomgemeinschaft über die Zusammenarbeit auf dem Gebiete der Plasmaphysik durchgeführt.*

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Abstract

It is attempted to assess the suitability of tokamaks for fusion power plants on the basis of existing design studies by reference to the reality of energy production in fission power plants. A definition of suitability criteria and a discussion of their relation to the most important features of power plants are followed by a comparative treatment. For example, the mean volumetric net electric power density in the nuclear islands of tokamak power plant designs is only 2,5 to 4 E of the value common today in light water reactor nuclear islands. In addition, configuration problems, auxiliary power requirements and energy payback time are discussed and taken into account in the assessment.

# THE TOKAMAK - AN IMPERFECT FRAME OF REFERENCE?

## 1. INTRODUCTION

With respect to plasma physics the tokamak is probably the most successful principle of magnetic plasma confinement. The question whether the tokamak is basically suitable as a reactor concept is being more and more frequently asked. Opinions on this differ. It is attempted here to identify and assess some of the problems expected in the event of introducing the physical principle of the tokamak into power plant technology.

Essential tools for investigating power plant suitability are design and parameter studies of fusion power plants. Several detailed designs based on the tokamak principle are available. These designs are incomplete in keeping with the state of the art; there are, for example, no realistic concepts for fuelling, impurity control and ash disposal. Their assessment, however, already allows critical questions of power plant suitability to be identified and sound statements to be made on the technical probability of whether the tokamak principle is suitable for power plants. This will be demonstrated with some simple examples by relating them to the realities of energy production.

## 2. THE MOST IMPORTANT SYSTEM CRITERIA OF POWER PLANT SUITABILITY

The power plant suitability of an energy system is primarily assessed in terms of technical feasibility, operating properties, fuel supply and environmental impact.

Technical feasibility essentially depends on the possibility of technical implementation of rational production by industrial means, on production-costs or -energies, and on the availability of the necessary raw materials.

Operating properties include reliability, availability, maintainability, safety - as well as network compatibility. For example, power plants with reactors operating in the non-stationary mode without intermediate energy storage for bridging idling times are not network compatible. This is also the case when the minimum reactor size leads to output powers that are inadmissibly high for the network.

The various aspects of fuel supply such as fuel costs, fuel cost trends and fuel supply reliability, are largely uncritical for DT fusion, as is shown in a number of studies.

This does not apply to environmental impact, contrary to previously held views. The questions involved here are the tritium cycle, activation of reactor components and thermal impact on the environment.

Proliferation is not dealt with in this context.

## 3. RELATION BETWEEN POWER PLANT SUITABILITY AND THE MOST IMPORTANT NUCLEAR POWER PLANT PROPERTIES

The extent to which suitability criteria depend on the most important fusion power plant properties can be roughly estimated from present knowledge (Fig. 1).

- <u>Feasibility</u>

Feasibility is strongly related to the net power density of the nuclear island and the system complexity. As a rule in energy technology, increased complexity of the technical concept must lead to at least a corresponding increase of the net power density. Feasibility is also related to the circulating power and the circulating power losses of the system (internal energy consumption).

The following example serves to illustrate the relation between feasibility and configuration: a modular configuration composed of many identical or similar elements and/or groups of components can be industrially manufactured by more rational means than one consisting of few large and complicated elements, to be partly fabricated on site or in-situ or many small, non-interchangeable ones.

| SUITABILITY CRITERIA \ POWER PLANT PROPERTIES | REACTOR CONFIGURATION | NET POWER DENSITY OF THE NUCLEAR ISLAND | SYSTEM COMPLEXITY | CIRCULATING POWER | FUEL CYCLE |
|---|---|---|---|---|---|
| FEASIBILITY (techn. & econ.) | +++ | +++ | +++ | +++ | +++ |
| OPERATING PROPERTIES (availibility, safety, reliability, maintainability) | +++ | ++ | +++ | ++ | +++ |
| FUEL SUPPLY | | | | + | +++ |
| ENVIRONMENTAL IMPACT | ++ | ++ | | +++ | +++ |

+++ STRONG ++ MODERATE + WEAK

Fig. 1

- Operating properties

are essentially governed by the configuration and the complexity of the technical concept. Hence the operating properties thus very strongly depend on both. The inherent system safety is - among others - related to the power density of the nuclear island (emergency cooling), and the circulating power system will have an impact on the plant-availibility, -maintainability and -reliability.

- Environmental impact

Besides the thermal environmental impact by wasted heat production in the thermal cycle and dissipation in the internal power circulation system, the power density and the configuration will also have an influence on the environmentally relevant parameters fuel inventory and activation.

## 4. CONFIGURATION AND COMPLEXITY

The configuration and complexity mainly influence the feasibility and the operating behaviour. Compared with the linear configuration, the toroidal one is adverse to servicing. With a view to sufficient availability the necessary servicing therefore has to be restricted to those components subject to natural wear and tear, which essentially means the first wall and the underlying structural components. Most of the other critical reactor components require an extremely high degree of reliability. This applies particularly to all concatenated sub-systems that cannot be dismantled, such as the toroidal magnet coils. It was found, that individual damaged toroidal field coils of a tokamak reactor cannot be replaced.The reactor suitability of the tokamak is therefore, among others, conditional on managing to build superconducting torus magnets with a reliability corresponding to that required for reactor pressure vessels. It is not yet possible to state whether this condition can ever be met. The additional outlay for quality control on the production side and quality assurance would very likely influence feasibility.

- Impurity control and exhaust (Fig. 2)

In view of their own power requirements tokamak reactors have to be operated quasi steady state with long burn times. Otherwise the mean energy gain would be too small and the circulating power of the system too high. Short intervals and long burn times are also necessary with respect to the thermal conversion efficiency. For instance, during the idle time, the temperature in the cooling cycle drops, thus reducing the thermal efficiency of the power plant. The alternating thermal load leads to fatigue of the structural material and hence to reduction of the lifetime. The material lifetime and burn time are thus correlated. As regards efficiency and circulating power a duty cycle of 95 % should be aimed at. With a realistic idle time of approximately 50 s the lower limit of the burn time would be approximately 1000 s. Quasi-stationary operation with impurity control and continuous exhaust is therefore essential. Furthermore, the thermal loading of the first wall by charged particles and at radiation should be kept as low as possible, not only in view of surface effects but also to increase its neutron load capacity and life (wall temperature, wall temperature gradient).

The integration of magnetic divertors for impurity control, exhaust and $\alpha$-particle energy extraction proves (from reactor design studies) to be extremely difficult:
Magnetic divertors will always impair access for repair and maintainance of the first wall and the structure behind it.

IMPURITY CONTROL AND EXHAUST

A) BUNDLE DIVERTOR

PROBLEMS:
- NEUTRON SHIELDING
- MECH. FORCES
- PARTICLE COLLECTOR ($F \approx 15\ M^2/GW_{TH}$)

B) MULTIPOLE DIVERTOR

PROBLEMS:
- REPAIR AND MAINTENANCE
- ENERGY REQUIRED AND STRAY FIELD
- PARTICLE COLLECTOR

C) COLD PLASMA MANTEL

PROBLEMS:
- BASIC PRINCIPLES NOT YET UNDERSTOOD (EXPERIMENTAL)
- EXTRACTION OF PLASMA- AND $\alpha$-PARTICLE ENERGY THROUGH THE FIRST WALL (WALL LIFETIME!).
- SELECTIVE HELIUM PUMPING WITH HIGH PUMPING VELOCITY $v \geq 3.5 \times 10^{20}$ HELIUM ATOMS / $GW_{TH}$

Fig. 2

The bundle divertor acting as toroidal field divertor being the type least adverse to servicing requirements. The bundle divertor concept, however, requires its field coils to be installed close to the plasma and, as far as can be seen, does not allow adequate coil shielding for superconducting windings owing to lack of space. On the other hand, with non-shielded normal conducting field coils the circulating power of the reactor would probably be unduly increased. Other problems still confronting the bundle divertor are the accommodation of the collector surface areas. Other outstanding problems are the very high magnetic forces acting on the divertor coils and the short lifetimes to be expected as a result of radiation damage to the insulation material.

Existing tokamak fusion reactor designs with magnetic divertors have therefore exclusively used poloidal field- or multipole divertors. But using versions with multipoles installed inside, as in ASDEX, it has not yet been possible to find a feasible solution compatible with reactor access-, manifacture- and service requirements.

Poloidal field divertors with external multipoles would appreciably simplify access and manufacture, but greatly increase operating and manufacturing costs. We shall return to this point later. Poloidal field divertors will also have a problem with the load capacity of the collectors.

In the most recent reactor studies a way out of this dilemma was sought by applying the cold plasma mantle method of impurity control. Unfortunately, the experimental basis is not yet adequate for technological problem identification. But there is one thing that can already be stated, namely that removal of the plasma- and alpha particle energies in the case of the cold plasma mantle has to be performed very probably via the first wall, this having the previously mentioned negative effects on its neutronic load capacity and lifetime.

Resumé: TECHNICALLY CREDIBLE IDEAS FOR IMPURITY CONTROL, WALL PROTECTION, ALPHA PARTICLE ENERGY REMOVAL AND EXHAUST IN COMMERCIAL TOKAMAK REACTORS ARE NOT AVAILABLE.

- Superconducting toroidal magnet

The superconducting magnet system would probably be the most expensive part of a tokamak reactor. The typical target data for developing the toroidal magnet are shown in Fig. 3.

SUPERCONDUCTING TOROIDAL MAGNET

TYPICAL TARGET DATA:

| | |
|---|---|
| FIELD ON TORUS AXIS: | $\lesssim 6$ T |
| MAX. FIELD AT CONDUCTOR: | $\sim 10 \div 12$ T |
| TRANSIENT EXTERNAL FIELDS: | $\Delta B \sim 1$ T; $\dot{B} \sim 0.2$ T/s |
| FULL CRYOGENIC STABILIZATION | |
| BORE SURFACE AREA: | $\sim 150$ $M^2$ |
| STORED ENERGY: | $\sim \geq 10$ $GJ/GW_{EL}$ |
| FIELD RIPPLE: | $< 1$ % SS |

QUESTIONS:

WHICH SUPERCONDUCTOR? (NBTI OR A 15)

WHICH COOLING? (HE II OR HE $I_{3K}$)

WHICH STABILIZER? ( AL OR CU)

WHAT ACTION AGAINST EFFECT OF TRANSIENT POLOIDAL FIELDS?

PROBLEMS:

- MANUFACTURING TECHNIQUE. (QUALITY ASSURANCE, QUALITY CONTROL)
- RELIABILITY
- RADIATION DAMAGE TO INSULATION MATERIALS AT LOW TEMPERATURES.
- POSSIBLE ANNEALING OF RADIATION DAMAGE TO STABILIZATION MATERIAL.

Fig. 3

The point "transient external fields" is a particularly difficult problem. These transient fields originate in the poloidal field system.

In some present-day reactor concepts, whose toroidal magnets are assembled from very few widely spaced coils (8 to 12) to facilitate blanket maintainance, the field ripple at the plasma surface is no longer smoothed by appropriate (costly) enlargement of the coil bore, but by normal conducting compensating coils near the plasma in the gaps between the toroidal coils.

An important question under discussion is the choice of superconducting material. On the one hand, there is the ductile niobium-titanium compound. Industry has a good command of the technique of producing multifilament conductors made of this material, but in order to attain the foregoing field values, the material has to be operated at lower temperatures than that of liquid helium ($T \leq 2$ K). The alternative is the A15 conductor, essentially $Nb_3Sn$ at present. This brittle material is less amenable to processing and industry is not yet very well versed in manufacturing conductor material.

The next question concerns magnet cooling, the point at issue being roughly: "Helium I or superfluid helium II?" Discussion is in progress. A closer look at the pro's and con's would exceed the scope of this survey.

Another question is the stabilizer: "Copper or aluminium?" Then, there is the choice of coil structural material: "Austenitic steel or high-grade aluminium alloys?" The last item on the list concerns the action to be taken against the effects of transient fields, such as shielding at low temperatures and/or development of special magnetic field transparent superconductors.

The on-site fabrication of such huge superconducting coils involves large technical problems, particularly as regards quality assurance. As already mentioned before, it is assumed that the toroidal field coils in tokamaks cannot be replaced, and it is an open question, if the extremely high reliability therefore required can ever be achieved.

Resumé: TOROIDAL MAGNETS FOR TOKAMAK REACTORS ARE FEASIBLE. BUT IT IS NOT SURE, IF THE RELIABILITY REQUIREMENTS CAN BE MET ECONOMICALLY.

- Poloidal field system with magnetic divertor

The technical problems of the poloidal field system depend on the configuration concept of the equilibrium field coils and the multipole divertor winding. A basic distinction is made between two types of configuration:

a) Both windings are located inside the toroidal field (Fig. 4)

This configuration affords the advantage of having the lowest excitation energy requirements, this being of great importance for those systems operated in the non-stationary mode, and, furthermore, the effect on the toroidal field magnet is relatively small. Disadvantages are:

POLOIDAL FIELD SYSTEM

WITH:

a) EQUILIBRIUM FIELD COILS

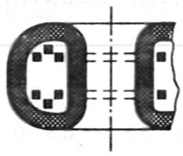
b) TRANSFORMER WINDING

c) POSSIBLY MULTIPOLE DIVERTOR WINDING

d) CONTROL COILS

TWO POTENTIAL BASIC CONFIGURATIONS:

A) COILS a) AND c) INSIDE THE TOROIDAL FIELD.



ADVANTAGE: SMALLEST ENERGY REQUIREMENT AND EFFECT ON TF.

DISADVANTAGE: COIL SEGMENTATION OR IN SITU FABRICATION, REPAIR AND MAINTENANCE EXTREMLY DIFFICULT.

Fig. 4

- the difficulty involved in manufacture (these coils have, so to speak, to be wound into the toroidal magnet on the building site),

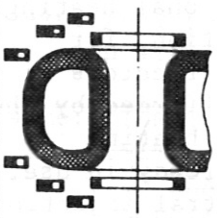
- the trouble involved in repair work, and the negative effect on the maintainance compatibility of the whole reactor.

b) No poloidal field windings inside the toroidal magnet (Fig. 5)

This configuration facilitates production and repair of windings and involves least obstruction to maintainance of the magnet blanket and first wall. But this entails the following disadvantage:

- The magnetic energy of the poloidal field system is much larger than that in the configuration with internal windings. The excitation power can then hardly be taken directly from the grid, thus making it necessary to insert large energy storage systems.

B) COILS a) AND c) OUTSIDE THE TOROIDAL FIELD.



ADVANTAGE: REPAIR AND MAINTENANCE REQUIREMENTS CAN BE MET.

DISADVANTAGE: MAGNETIC FIELD ENERGY MUCH LARGER AND PERHAPS PROHIBITIVELY HIGH STARTING POWER REQUIREMENT.

Fig. 5

- Large stray field production
- On site fabrication of the coils might be required.

Resumé: POLOIDAL FIELD SYSTEMS FOR COMMERCIAL FUSION REACTORS WITH MULTIPOLE DIVERTOR COILS ARE NOT YET FEASIBLE.

- Fueling (Fig. 6)

In a DT fusion reactor
3.6 grammes of deuterium and
5.4 grammes of tritium
are burned per gigawatt-hour. In quasi-stationary ignited operation the fuel throughput can be more than ten times as high as the quantity actually burned. For reasons of physics, refuelling of a tokamak during the burn process may not be possible at all or at least not exclusively from the boundary. Refuelling must then possibly be done with pellets consisting of a frozen DT mixture which have to be accelerated to velocities of $10^5$ to $10^6$ cm/s, so that they can penetrate deep enough into the dense and hot plasma before their mass is completely abblated. By means of gas dynamic acceleration it is possible to attain velocities of about $10^5$ cm/s. If higher velocities should be required, one of the as yet untested ideas such as laser acceleration would have to be looked-at more closely. Another problem with pellet injection will be the repetition rate required. This depends on the maximum permissible pellet mass ($\sim$ 1 % of the plasma mass to avoid disruption), the number of injectors, the reactor power and the burn-up. It is estimated that a repetition rate of about 50 Hz will be required.

Resumé: FUELING OF TOKAMAKS IS NOT SOLVED. A SATISFACTORY SOLUTION FOR REACTORS CANNOT BE PREDICTED OR PROMISED.

PLASMA SYSTEM:

FUELLING

REQUIREMENTS:

| | |
|---|---|
| TRITIUM | 5,4 G / GWH |
| DEUTERIUM | 3,6 G / GWH |

METHODS:

A) PELLET INJECTION ($v = 10^5 \div 10^6$ CM/S)

PROBLEMS:
- IF $v \gtrsim 5 \times 10^5$ CM/S REQUIRED
- INJECTION FREQUENCY

B) GAS PUFFING (POSS. IN CONJUNCTION WITH A)

IGNITION HEATING

IGNITION POWER REQUIREMENT: $\sim 50$ MW / $GW_{TH}$ FOR 1 S OR LONGER

IGNITION HEATING METHOD:

A) NEUTRAL INJECTION

PROBLEMS:
- NEUTRALIZATION E > 300 KEV
- RELIABILITY
- NEUTRON STREAMING
- ACTIVATION
- SHIELDING

B) HIGH-FREQUENCY HEATING

PROBLEMS:
- IMPURITY PRODUCTION
- COUPLERS
- GENERATION f > 20 GHz

Fig. 6

- Ignition heating (Fig. 6)

Additional heating up to ignition, for commercial reactors as well, is afforded by the additional heating methods already in use, namely neutral injection and high-frequency heating. Neutral injection would require injectors for energies of over 300 keV, and so use would have to be made of negative ion sources, the technique of which has not yet been developed.

The problems entailed in reactor operation, namely activation and contamination by tritium, shielding, remote handling for maintenance, repair and replacement, and the high degree of reliability required characterize the further outlay necessary for development.

A decisive problem that might seal the fate of neutral injection as a contestant for additional heating is neutron streaming: the unhampered outflow of fusion neutrons from the interior of the reactor through the beam ducts penetrating the blanket has to be avoided at all costs, even if the neutral injectors were to be incorporated in the shielding. This is necessary not only because of the neutron effects in the beam duct region but also because of the scattered neutron effects in the adjacent superconducting toroidal coils. Each beam duct would have to be encased in an approximately 1 m thick shield, and the resulting large coil spacing would necessitate additional trimming coils for smoothing the toroidal field ripple.

The method of high-frequency additional heating compared with neutral injection seems to entail fewer problems although (or perhaps because) it has not yet been so fully studied and tested experimentally.

In addition to the physical questions of wave-plasma interactions and wave coupling, the coupling technologies and the generator technology for very high frequency heating methodes still need further development. In the former case the transmissible power, or, to be more precise, the attainable antenna power density and in the latter case the generator power level are still unsatisfactory. It is expected, however, that the present keen interest in HF heating will soon yield worthwhile results.

Resumé: IGNITION IN TOKAMAKS WILL PROBABLY BE POSSIBLE BY NI-HEATING! BUT NI IS PROBABLY NO WAY FOR REACTOR HEATING. RF HEATING IS TECHNICALLY MORE PROMISING. A SATISFACTORY SOLUTION FOR COMMERCIAL TOKAMAK REACTOR HEATING CAN NOT YET BE SAFELY PREDICTED.

## 5. POWER DENSITY

The mean volumetric power density of a DT fusion reactor depends on the plasma power density ($\sim \beta^2 B^4$) and the sum of the volumes of reactor systems located outside the plasma, such as blanket and shielding system, magnet system, start-up heating system, fuel supply system, divertor system, vacuum system, etc. Its value is limited by the neutron load capacity and lifetime of the first wall and blanket structures. These depend not only on the material properties, but also on the reactor design and operation parameters.

First wall and blanket components of DT reactors must be periodically replaced before their lifetimes expire. Scheduled maintenance, requiring extraordinarily complicated remote operations, will - under realistic assumptions - also have a significant impact on the plant economy. In order to achieve competitive overall availability ($\geq$ 80 %), long lifetimes of the first wall and blanket structural materials must be achieved. Considering scheduled and unscheduled outages as well, a tokamak first wall lifetime of at least 7 years is assumed to be the lower limit for reaching competitive availabilities with any maintenance concept.

- Neutron wall loading and power density

If the net electric power density $p_{el}$ of the tokamak reactor is defined as the quotient of the plant net electric power $P_{el}$ and the volume $V_c$ which is bounded by the surface of a cylinder enclosing the toroidal tokamak magnet

$$p_{el} = P_{el}/V_c$$

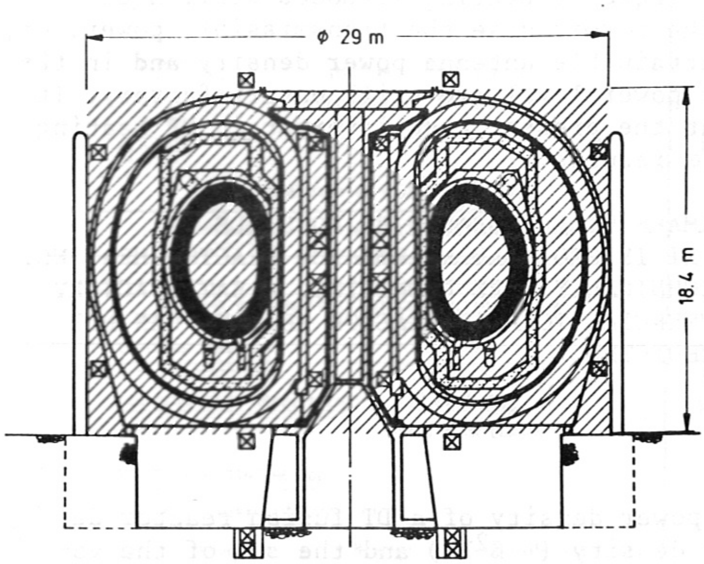



Fig. 7

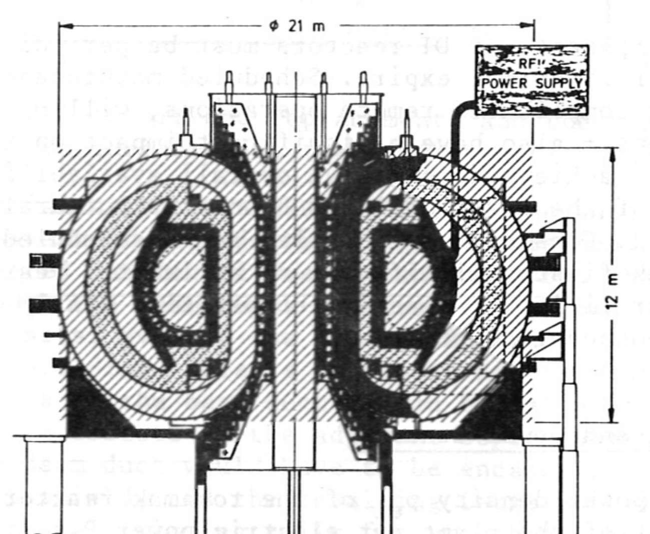



Fig. 8

one obtains for STARFIRE (Fig. 7) and NUWMAK (Fig. 8) the values in Fig.9.

Figure 9 also gives the neutron wall loadings $p_w$ and for NUWMAK the value of the power per unit weight $p_{el}^*$ of the nuclear islands. The reference value taken for the power density is that prevailing in the pressure vessel of pressurized water reactors (PWR). The structure of a PWR is less complex than that of a DT tokamak reactor would be and the materials required for its construction will, with all probability, entail lower specific energy costs than tokamak materials. In addition, the reference volumes chosen here for the tokamak reactors do not include essential subsystems of the nuclear island (e.g. start-up heating, fuel injection, selective vacuum pumps) because too little is as yet known about these. Power density comparisons made on this basis should therefore hardly lead to a pessimistic assessment of the economic chances of the tokamak as a power reactor principle.

| | | STARFIRE | NUWMAK | PWR | |
|---|---|---|---|---|---|
| net electric power | $P_{el}$ | 1.150 | 660 | 1.240 | $MW_{el}$ |
| volume of the nuclear island | $V_c$ | 12.150 | 4.160 | 310 | $m^3$ |
| neutron wall loading | $p_{w_0}$ | 350 | 400 | - | $W/cm^2$ |
| volumetric power density | $p_{el_0}$ | 0,095 | 0,159 | 4 | $W/cm^3$ |
| power density per unit weight | $p^*_{el_0}$ | - | 0,086 | 1,5 | $W/g$ |
| net electric power for $p_{el_0} = 4$ | | 48.600 | 16.640 | 1.240 | $MW_{el}$ |

$$\boxed{p_{el} = P_{el} / V_c}$$

$$\boxed{\begin{aligned} p_{el} &= k_1 \cdot p_w \\ k_1 &= \frac{P_{el}}{V_c \cdot p_{w_0}} \end{aligned}}$$

Fig. 9

<u>Resumé:</u>

IT CAN BE DRAWN FROM RECENT REACTOR DESIGN STUDIES, THAT THE MEAN VOLUMETRIC NET ELECTRIC POWER DENSITY IN TOKAMAK-REACTORS WOULD ONLY BE 2.5 to 4 % OF THE VALUE COMMON TODAY IN LIGHT WATER REACTORS AND THAT A TOKAMAK-REACTOR WOULD REQUIRE ABOUT 12 kg OF CONSTRUCTION MATERIAL PER $kW_{el}$ TO BE BUILT, OR A FACTOR OF 17 MORE THAN FOR THE LIGHT WATER REACTOR.

If a linear relation between the volumetric power density and neutron wall load is assumed, extrapolation of the data from the two fusion reactor designs yields the curves shown in Fig. 10, where the dependence of the blanket volume on the power density is ignored (optimistic extrapolation!). Compacting the construction beyond a certain limit is achieved at the expense of complexity and availability. The upper compacting limit is characterized by the so-called "most compact" tokamak reactor ($A = 3$; $r_w = b = 1{,}75$ m), whose power refers to the sum of the net volumes of the plasma vessel ($A$, $r_w$) and the outer system ($b$), which comprise the blanket, shielding and magnet only.

From Fig. 10 it can be deduced that the power density of the PWR could only be attained in a tokamak reactor of NUWMAK-type design, if structural material is available, which permits a neutron wall loading of about 90 $MW/m^2$. For a lifetime of 7 years this corresponds to an integrated wall loading of 630 $MWa/m^2$.

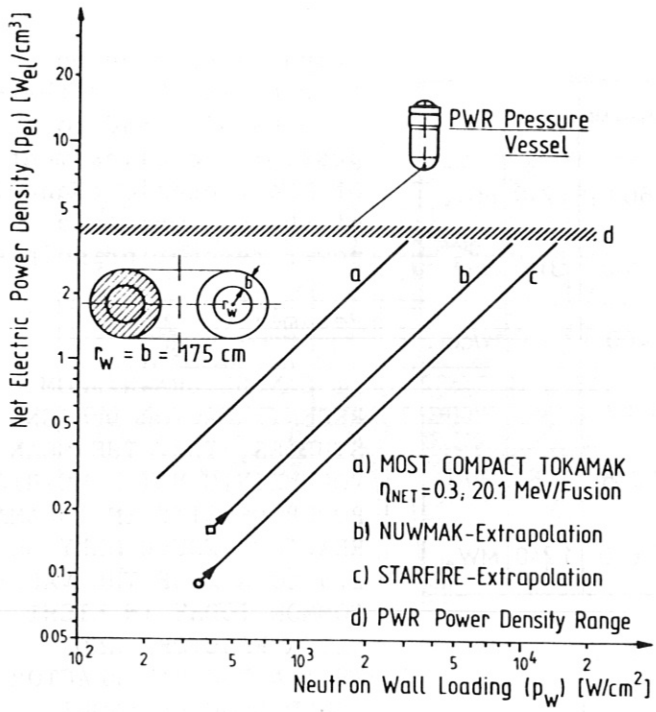



Fig. 10

Assuming also a linear relation between the power density per unit weight $p^*_{el}$ and the neutron wall loading the extrapolation of NUWMAK's $p^*_{el_o}$ = 0.086 W/g (Fig. 11) yields a somewhat lower breakeven value. The same net electric power per unit weight as in the PWR (1.32 W/g) is reached at a neutron wall loading of about 70 MW/m$^2$. The corresponding integrated neutron wall loading is 490 MWa/m$^2$.

The case is now considered where material with properties required to withstand these extraordinarely high wall loadings is available. The minimum mean ß-values than required for power density breakeven and the lower limit of the Tokamak reactor power can be taken from Fig. 12:

The overall power density of a DT-fusion reactor reaches its maximum for a = d, where d is the radial thickness of blanket plus shield plus toroidal magnet. Assuming 1.5 m $\leq$ a $\leq$ 2,5 m and breakeven neutron wall loadings between 50 MW/m$^2$, the corresponding ß-ranges are 0,37 $\leq$ ß $\leq$ 0,72 for a toroidal field on axis $B_o$ = 4 T or 0,16 $\leq$ ß $\leq$ 0,32 for $B_o$ = 6 T. The thermal power of these tokamaks would be in the ranges between 10 GW and 20 GW.

Resumé: THE ß-VALUES REQUIRED FOR COMPETITIVE TOKAMAK POWER DENSITIES VERY LIKELY CAN NOT BE ACHIEVED IN TOKAMAKS. IF ALL POWER DENSITY CONSTRAINTS WERE DISPENSED WITH, THE MINIMUM TOKAMAK REACTOR POWER WOULD BE TO LARGE, REGARDING NETWORK COMPATIBILITY.

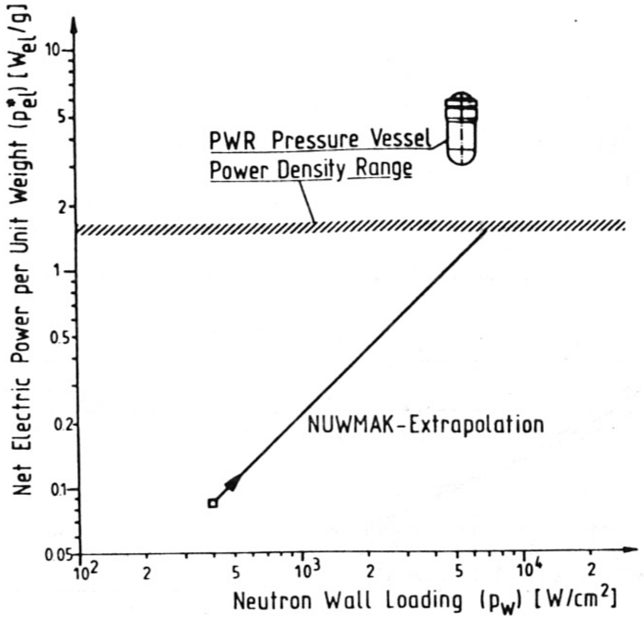



Fig. 11

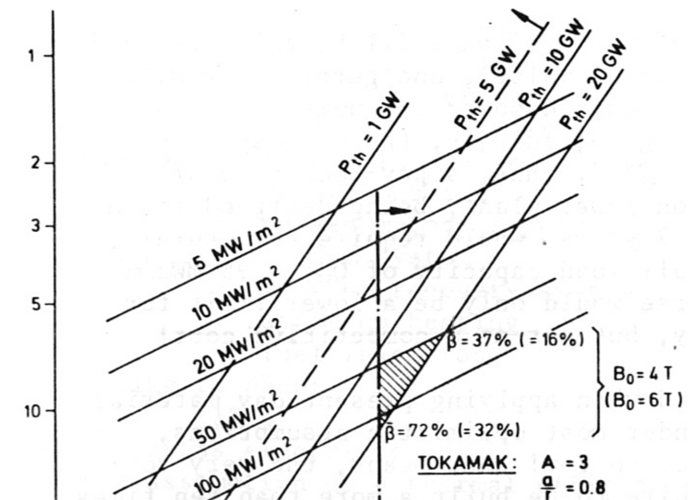



Fig. 12

– <u>Energy pay-back time</u>

The energy pay-back time $t_{pb}$ is the quotient of the energy required for producing a power plant and its net electric power output. This time might yield more meaningful criteria for assessing energy systems. Pay-back times for light water power plants between about 0.9 years and 2.3 years have been published. Pay-back time and power density are linked to each other. They may be linked in the following form:

$$t_{pb} = t_{pb_o} \cdot \{1+\alpha\cdot[(\frac{P_{el_o}}{P_{el}})^x - 1]\}$$

where:

a = the fraction of the nuclear island pay-back time to the total plant pay-back time of the reference REACTOR,

x reflects the ratio of average specific material energy cost of the tokamak reactor as compared to those of PWR ($x \geq 1$), unity has been chosen in the following as a conservative assumption.

Taking as reference values a mean pay-back

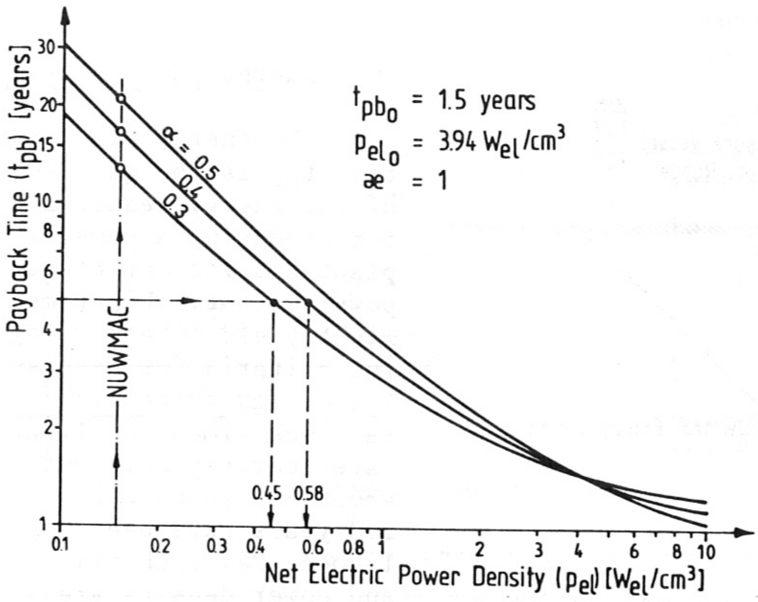



Fig. 13

time $t_{pb_o}$ of 1,5 years and a net electric power density of 3,96 W/cm$^3$ for PWR power plants, one can draw curves (Fig. 13) for the pay-back time of tokamak power plants as a function of the net electric power per unit volume.

Assuming that a pay-back time of 5 years can still be tolerated and $\alpha = 0{,}37$ (i.e. the present PWR-plant value), one gets: $p_{el} = 0.54$ W/cm$^3$ as a lower limit for the required power per unit volume of the tokamak reactor, and from Fig. 9, curve b, the corresponding neutron wall loading $p_W \geq 13.5$ MW/m$^2$. Thus, a pay-back time of 5 years for a NUWMAK-type fusion power plant, being designed for a first wall replacement time of 7 years, would require structural materials with an integrated wall load capacity of $Q_{wi} \geq 95$ MWa/m$^2$ in such a reactor. This of course would only be a lower limit for the production of useful energy, but not at a competitive cost!

The power density to be achieved when applying present day material (stainless steel!) would be, under most optimistic assumptions, more than one order of magnitude to low! That means, the very complex tokamak reactor would require to be built a more than ten times larger quantity of material of much higher quality than PWR of the same net electric output.

Resumé: THE CONSTRUCTION OF A TOKAMAK FUSION REACTOR PRODUCING USEFUL ENERGY AT A COMPETITIVE COST WITH MATERIALS NOW UNDER DISCUSSION IS, IN ALL PROBABILITY, NOT POSSIBLE.

## 6. MAINTENANCE

The time required for maintenance and repair work governs the availability of the reactor. A power plant availability of 75 %, that is the mean value for fission power plants, is aimed at. In other words: For scheduled regular maintenance and for repair in the event of unforeseen defects the time available per annum would be 90 days.

Considering modul size, weight and complexity of the inner plasma systems, first wall and blanket and the fact that all maintenance and repair of these highly activated pieces has to be performed remote, it is obvious that 90 days would be short in conjunction with a lifetime of 7 years for the first wall. In recent studies it is therefore suggested to avoid every wall/blanket change-out during a reactor design lifetime of 30 years, because the change out procedures were assessed to be too difficult, too expensive and too time consuming and would have also a negative impact on the plant reliability. This would require structural material lifes about a factor of 4 higher than stated before.

## 7. AUXILIARY POWER CONSUMPTION AND OVERALL POWER PLANT EFFICIENCY

For the operation of PWR-plants about 5 ./. 7 % of the gross electric power is consumed by

- main coolant pumps
- feed water pumps
- cooling water pumps
- cooling tower operation.

A tokamak reactor would need additional auxiliary power mainly for

- magnet cooling
- pulsed magnet supply
- vacuum system
- Tritium recovery system
- start up heating
- fuel injection

amounting to about

12 %

of the gross electric power.

On the assumption of a thermal efficiency of 34 % (similar to PWR-plants), the overall tokamak power plant efficiency would be

$$\eta_{total} = 28\ \%$$

(compared with 33 % in present-day PWR-plants).

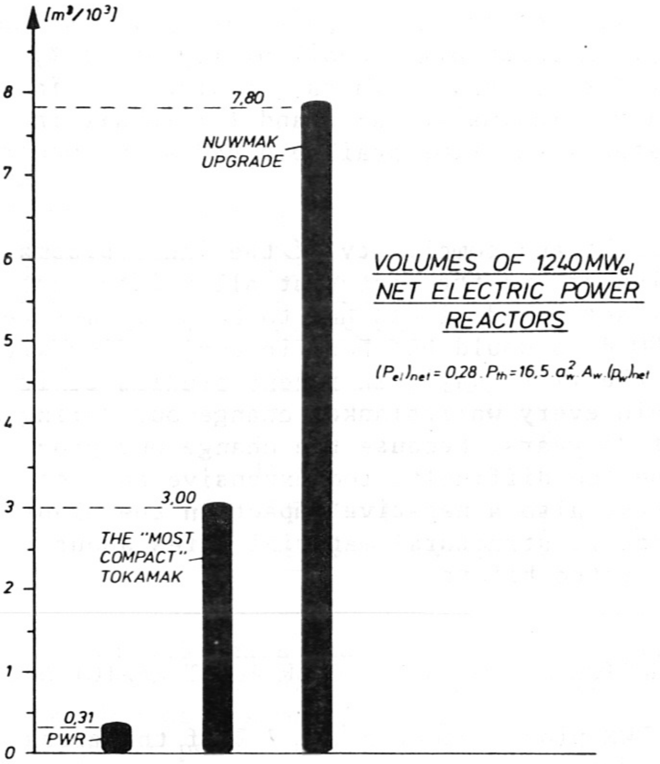



Fig. 14

## 8. SUMMARY

The mean volumetric net electric power density in nuclear islands of recent conceptual tokamak power plant designs would be 2.5 to 4 % of the value common today in the less complex structures of light water reactor nuclear islands (Fig. 14). Such tokamak reactors would require about 12 kg of construction material per $kW_{el}$ to be built, or a factor of 17 more than for the PWR. Considering the principally low overall reactor power density, owing to wall load constraint and ß-limitation, the tokamak system will very likely be a too complex concept for a commercially competitive pure DT fusion reactor.

Even if a sufficiently high power density could be achaeved the suitibility of the tokamak concept would nevertheless be poor by following reasons:

- the minimum reactor unit power would be too large, even for base load purposes;
- major components have to be fabricated on site or in situ;
- a satisfaction of the reliability and availability requirements of base load plants would entail extraordinarely high and very likely inacceptable cost for quality assurance and redundancy improvement;
- the poor maintenability and limited life of major components would cause long downtimes, already for scheduled maintenance;
- the overall net plant efficiency would be low.

## 9. CONCLUSIONS

This assessment yields the following criteria for choosing alternative lines:

The most important selection criterion should be the mean power density attainable. Economically competitive power densities require first of all plasma confinement systems with high mean ß limits ($\bar{\beta} > 15$ %). The power density is then limited by the neutron load capacity and lifetime of the first wall. It is therefore important to consider only those systems which in principle afford the possibility of minimizing the thermal and particle loads on the first wall and keeping the cyclic thermal and mechanical stresses small. Alternative concepts should, moreover, allow wall modul replacement as simply and hence as frequently as possible, so that the first wall life time requirements can possibly be shortened in favour of higher loads without reducing as a result the availability of the reactor to less than, for example, 75 %. Wall material in the form of circulating liquid metal would be particularly interesting for obtaining a high load capacity.

The power density problem is probably less critical for fusion hybrids operating as fuel factories.

The wall load constraint would disappear for neutron-free fuel cycles and the mean reactor power density attainable would only depend on the plasma power density, provided that the thermal and particle loads on the wall can be kept sufficiently low.

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This work was performed under the terms of the agreement of association between the Max-Planck-Institut für Plasmaphysik and EURATOM.