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A REVIEW OF TOKAMAK POWER REACTOR AND BLANKET DESIGNS IN THE UNITED STATES

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ABSTRACT

The last major conceptual design study of a tokamak power reactor in the United States was STARFIRE [1] which was carried out in 1979-1980. Since that time U.S. studies have concentrated on engineering test reactors, demonstration reactors, parametric systems studies, scoping studies, and studies of selected critical issues such as pulsed vs. steady-state operation and blanket requirements. During this period, there have been many advancements in tokamak physics and reactor technology, and there has also been a recognition that it is desirable to improve the tokamak concept as a commercial power reactor candidate. During 1984-1985 several organizations [2] participated in the Tokamak Power Systems Study (TPSS) with the objective of developing ideas for improving the tokamak as a power reactor. Also, the U.S. completed a comprehensive Blanket Comparison and Selection Study [21] which formed the basis for further studies on improved blankets for fusion reactors.

1. INTRODUCTION

This paper presents an overview of recent work in the U.S. on tokamak power reactors and blanket design. The next section discusses some basic goals and trends in tokamak reactor studies in the U.S. This is followed by sections on major tokamak configuration options, RF current drive and the second stability regime, impurity control, and blanket options. Conclusions are summarized in the last section.

2. GOALS FOR TOKAMAK REACTORS

The basic goal for commercial tokamak reactors is to produce a product (e.g., electricity, fissile fuels, synthetic fuels, process heat, etc.) at a competitive cost compared to alternative means of producing that same product, and one which produces that product with desirable safety and environmental features. If fusion science and technology were well developed, then one could estimate with confidence the cost of various products (e.g., capital costs and cost of energy for electricity producing fusion plants) and quantify the associated safety/environmental risks. However, since fusion is in an early stage of development, we cannot accurately estimate the cost and safety risks of fusion reactors. Rather, we must identify the basic features which, if incorporated into a fusion reactor concept, would substantially improve the probability of developing a superior product in terms of economics and safety/environmental impact.

Some important characteristics for an attractive tokamak reactor include the following:

- A range of fusion power output per reactor. This is important in order to provide flexibility in terms of siting, utility grid size (and growth patterns), and matching to various applications of fusion energy.

Minimum tokamak reactor fusion power levels of a few hundred megawatts should be a goal, without undue economies of scale penalties compared to larger unit sizes.

- Reduced reactor capital and unit capital costs. It is not necessary for fusion to compete on a capital cost or unit capital cost [e.g., $/kW(e)] basis with all potential competitors, particularly those which have substantial fuel costs (which are essentially nil for fusion). However, there will obviously be upper bounds regarding captial costs so that it will be possible to achieve competitive product costs. In addition, it will be necessary to minimize the "at risk" capital funds. This can be accomplished by minimizing the reactor output (as noted above) and the unit capital cost. Features which will reduce unit capital costs include:

  -- increasing the mass power density of the reactor, which is defined as the electrical output of the reactor divided by its mass (similar definitions can be developed for non-electrical applications). A target value is about 100 kW(e)/tonne. Higher values of the mass power density will provide insurance against uncertain reactor component unit costs. Reductions in reactor mass will be impacted the most by reductions in the mass of the magnets (achieved largely by higher β without higher plasma currents) and shield (achieved largely by reducing the number and size of penetrations).

  -- increasing the overall efficiency of the power conversion system. This can be accomplished by increasing the efficiency of the system which converts fusion power into the desired product and by reducing the power requirements for fusion support systems. Important figures of merit for this area include the following.

		Target Value
Q	= average fusion power out/input power to sustain plasma	> 20
M	= blanket energy multiplication	$M \cdot \eta_t > 0.5$
η_t	= gross thermal efficiency	

  -- reducing reactor support subsystems. Not only is it desirable to minimize the required capacity of support systems (e.g., the power requirements of plasma heating systems or control systems) but it is also desirable to accomplish more than one function with a given subsystem (e.g., using one rf system to both heat and drive the current in the plasma).

- Design simplification. In order to achieve a competitive product price, capital intensive energy sources like fusion must operate with a high capacity factor, typically $\gtrsim$ 70%. Probably the most important ingredient in achieving high capacity factors is minimizing design complexity. This will provide for less maintenance requirements and easier maintenance procedures when required, and, as noted above, reduced capital costs. Design complexity is difficult to quantify but one can identify several examples of design concepts which will reduce design complexity:

- -- steady-state operation
- -- first wall or limiters for impurity control
- -- reduced plasma shaping and control requirements
- -- combined plasma startup, heating and current drive system
- -- large duct, single pass liquid metal blankets
- -- mechanically integrated first wall/blanket/shield

- Enhanced safety and environmental features. Key items here include (1) achieving shallow land waste burial for all reactor components and (2) inherent safety, where the goal is to ensure that no major accident can occur that would have unacceptable public health or severe economic consequences. This should be achieved by passive features.

3. MAJOR CONFIGURATIONAL OPTIONS

A number of departures from conventional tokamak design have been studied in order to assess the benefits of new physics ideas on a practical reactor system. One of the major improvements in commercial tokamaks appears to be the possibility of operation at higher beta than is available to experimental machines. However, higher beta may be achieved via differing strategies, and an important task in this study was identifying the respective difficulties or penalties to be paid to achieve higher beta. Table I summarizes the key features of several designs for commercial reactors. Descriptions and comments on these options follow.

Operation in the second region of stability is theoretically possible via several routes. Plasma indentation to a bean shape [3] is perhaps the best known technique, however, our studies showed the large pusher coil to be difficult to shield and maintain in a reactor. Energetic particles can create pressure anisotropy [4] which might also ease the passage to higher stability. ECRH is an obvious means to achieve this goal, but the power requirements are presently unknown for a reactor. Toroidal plasma rotation appears to increase the outward shift of the magnetic axis, [5] increasing the magnetic well depth. This may also permit access to higher stability but was not studied in our current work.

The most desirable routes to second stability might arise through a careful control of the pressure and current density profiles in the plasma [6,7,8], achieved with ECRH for local heating/current drive and a fast wave/lower hybrid wave combination for bulk current drive. With this approach to the second stability regime the axial safety factor, q_0, may be raised to $q_0 \sim 1.5$-2.0. This increases stability and also reduces the total current, making steady-state current drive more attractive. Table I includes two reactors operating without indentation in the second stability region; they are characterized by large A ($\sim$ 5-6), low current ($I_\phi \sim$ 4-6 MA), and $\langle\beta\rangle = 0.25$. The reactor "SC" with superconducting toroidal coils features steady-state operation with fast-wave current drive, [10] which imposes a significant penalty for recirculating power. The reactor "CU" with normal coils pays a penalty for electric power to operate the resistive magnets.

In the first stability regime beta may be increased by vertically elongating the plasma,[7] and the ET [8] (elongated tokamak) of the table typifies this extreme, with κ = 9.4. A major concern with this reactor is shielding the equilibrium field (EFC) coils, which are located ~ 30 cm from the first wall to properly shape the plasma. Stability studies [11] also suggest crescent shaping and current density tailoring may be essential to achieve maximum beta (as high as ~ 0.04 I_ϕ(MA)/ a(m)B(T) in the first stability regime). The high current ($I_\phi \simeq$ 10 MA) and high density of ET argue against steady current drive, so the device is proposed to operate in a pulsed mode with a burn of ~10^3s.

An alternative approach to high first regime beta is simply going to the lowest possible A and simultaneously reducing the inboard blanket/shield dimension (necessitating normal TF coils) to keep the maximum toroidal field to a reasonable value.[12] The resulting "spherical tokamak" (ST in the table) has a very large current, $I_\phi \simeq$ 40 MA, which could only be generated by some highly efficient current drive technique such as F - Θ pumping, which is presently untested. Some studies [11] suggest ST will also require strong triangularity to achieve high beta.

4. DETAILED STUDY OF SECOND STABILITY REGIME WITH RF CURRENT DRIVE

Work at ANL has identified several areas in which the tokamak concept could be improved relative to the STARFIRE design. Many of these improvements derive from the possibility of operating in the second stability regime. In particular, plasmas with toroidal betas in excess of 20 percent have been found theoretically stable to ideal MHD modes in high aspect ratio tokamaks with quite modest toroidal currents. A tokamak reactor may be most attractive if it embodies the four following characteristics. First, high beta is essential to reducing cost, since considerable capital is invested in the large toroidal field magnets. Second, high aspect ratio (A ~5) will facilitate the use of efficient fast wave current drive and could simplify reactor maintenance operations. Third, low toroidal current ($\lesssim$ 5 MA) will greatly reduce the cost of the equilibrium field coil (EFC) system and attendant startup power supplies and will reduce the consequences of a plasma disruption. Fourth, steady-state operation is likely to reduce costs and increase reliability relative to pulsed operating cycles.

The key to achieving these reactor goals may well be rf current drive. At very high beta, tokamak equilibria require very broad, or even hollow, current density profiles, and it may be possible to tailor the current density in an rf-driven discharge by careful selection of the wave properties. Furthermore, a judicious choice of current density profiles can also result in small toroidal current; in addition to the advantages cited above, this helps by reducing the rf current drive power, which thus improves the reactor's power balance. A quantitative study of RF-current drive for this application is given in Ref. 10; we find from our preliminary work on this problem that a wide variety of equilibria can be created with the fast wave.

Other hallmarks of the second stability regime are that large aspect ratio and only a mildly shaped plasma cross section are needed to get high beta. The resulting reactor may have simplified toroidal field (TF) coils which are nearly circular and an equilibrium field (EF) coil system which is relatively small and easy to maintain.

Of course the greatest benefit of high beta is the promise of reduced toroidal magnetic field, since the TF coils are a major capital expense for a tokamak. At relatively low fields ($B_M \sim 6$ T) the coil current density can be fairly high so the TF coils can be quite compact. Other benefits of low B_M are related to using liquid-metal coolants. These improvements in blanket design and impurity control are discussed below.

5. IMPURITY CONTROL

The TPSS effort in impurity control examined innovative systems that offer substantial improvements to the cost, complexity and reliability of tokamak reactors. The concepts examined were lithium-cooled, self-pumped limiters. Two types of configurations were examined: a first wall/limiter and a slot limiter.

The self-pumped impurity control concept [14] uses vanadium or certain other materials to selectively trap impinging helium from the plasma, in-situ, on a continuously growing surface while providing for high recycling of hydrogen. No vacuum ducts or pumps are used (except for a very small startup system). Vanadium or other trapping material is added to the surface at a rate of 3-4 times the alpha particle production rate to allow continuous operation. Trapping material can be added by various means, e.g., by injecting pellets into the edge or scrapeoff plasma where material is ablated and transported to the trapping surface.

The maximum operating temperature for the limiter trapping surface is an important design parameter in determining the system lifetime. This is controlled primarily by thermal helium release. The desired net helium trapping fraction in the surface material is about 30 atomic %. Based on extrapolation from limited experimental data, e.g. [15-18], helium trapping will be possible up to ~ 0.7 of the melting temperature. We also estimate an acceptable tritium inventory in these materials even for multi-year operation.

The first wall/limiter design combines the functions of first wall and limiter into a single first wall structure. The wall conforms closely to the outermost plasma flux surface and can be shaped in various ways, e.g., for constant heat flux. Trapping material is added to the plasma scrapeoff or edge region where it is transported to the wall, along with hydrogen and helium ions from the plasma. The arriving trapping material deposits on the surface and serves to trap impinging helium. Hydrogen is recycled back to the plasma. The entire wall area (~ 300 m^2) is used for helium trapping. For a plasma with a neutron wall loading of 2.5 MW/m^2, a full 10 year impurity control system lifetime (the estimated blanket lifetime) is possible with a vanadium growth rate of 1.5 mm/yr for a total growth of 1.5 cm.

Plasma transport calculations [19,20] combined with erosion/redeposition calculations have been performed for the first wall/limiter. The effects of 100% recyling of hydrogen at the boundary, together with pellet fueling to make up for the DT burnup, appear to be acceptable. Both higher central temperatures and lower central densities (desirable for current drive) are predicted, as well as lower edge temperatures (desirable to minimize sputtering).

The other major issue of plasma contamination by sputtered and injected impurities appears to be rather similar to that of a pumped limiter. This is because the edge current of injected impurities is typically much less (~ 0.1) than the current of impurities arising from the normal sputtering process.

The first wall/limiter system is a simple, integrated structure without leading edges. However, to ameliorate concerns about plasma contamination, a slot self-pumped limiter was developed. Typical design parameters are shown in Table II. The limiter consists of a front face, two leading edges, and a slot region. The limiter would be toroidally continuous and located at one poloidal location. Helium trapping is done on both sides of the slot region. Trapping material is added to the slot plasma where it is transported to the trapping surfaces. (Material added to the plasma in the slot is more likely to be kept out of the main plasma.) The slot limiter essentially separates the functions of heat removal and particle removal. Most of the transport heat is taken on the thin tantalum front face. Particle removal occurs in the low heat flux slot region. One can then select the trapping material for other than thermal properties. Typical choices are vanadium, nickel or iron.

The designs employ the desirable characteristics of high velocity lithium flow and reduced lengths of the heated sections in the high surface heat flux region. This results in reduced coolant residence time in the high heat flux region which is important for liquid metal systems. A key feature of both the first wall and slot limiter designs is the use of electrical insulators to reduce the MHD pressure drop to acceptable levels. The insulator is used on the three coolant channel surfaces adjacent to the plasma chamber (the wall facing the plasma is not insulated).

Self-pumping eliminates the need for most vacuum ducts, pumps, and the large associated penetration shielding. The elimination of penetration shielding results in an approximate cost savings of 10-100 M$ and weight savings of 700-7000 metric tons for TPSS and STARFIRE size reactors, respectively. The reduction of tritium recycle and refueling saves 25M$ in the fuel recovery system (independent of reactor size). The helium removal efficiency of a self-pumped limiter is estimated at ~ 10 - 25% which is higher than for a pumped limiter. Finally, the extended lifetime, integrated structure, and maintenance free operation of self-pumping may offer significant reliability/availability advantages.

6. BLANKET CONCEPTS

6.1 Summary of Blanket Comparison and Selection Study

The recent Blanket Comparison and Selection Study (BCSS) [21], which was a comprehensive U.S. evaluation of fusion reactor blanket design and the status of blanket technology, serves as an excellent basis for further development of blanket technology. This study provided an evaluation of over 130 blanket concepts for the reference case of electric power producing, DT fueled reactors in both tokamak and tandem mirror (TMR) configurations. Based on a specific set of reactor operating parameters, the current understanding of materials and blanket technology, and a uniform evaluation methodology developed as part of the study, a limited number of concepts were identified that offer the greatest potential for making fusion an attractive energy source. Based on the systematic and comprehensive evaluation performed, the leading concepts are indicated in Table III.

6.2 Recent U.S. Blanket Developments

The primary objectives of more recent blanket design efforts in the U.S. have been to improve the attractiveness of fusion energy. The major efforts in the last two years can be classified into the following areas: [22,23]

- Improvements in self-cooled liquid-metal concepts provided by reduced magnetic fields resulting from higher β as well as extensive use of electrically insulated walls.
- Expanded use of reduced activation materials, e.g., modified ferritic steels or vanadium alloys, and partial blanket replacement to minimize radioactive waste management requirements.
- Simplification of solid breeder concepts by innovative tritium recovery scenarios.
- Improved economic performance by incorporating substantial amounts of beryllium or other candidate materials as energy multipliers.
- Improvement in the energy conversion efficiency of helium-cooled designs by utilization of higher temperature strucutral materials such as vanadium.
- Innovative concepts that emphasize design simplicity which utilize an alternate breeder/coolant, viz., Flibe.

6.3 Self-Cooled Liquid Metal Blanket

The self-cooled, liquid-metal blanket concept provides several features: 1) design simplicity associated with utilization of the same fluid as both breeder and coolant, 2) most of the fusion energy is deposited directly in the coolant, and 3) the coolant also serves as the tritium recovery fluid. In a current study liquid lithium, as the breeder-coolant with a vanadium alloy structure, was developed as the reference self-cooled liquid-metal blanket concept.[22] A ferritic steel structure and a LiPb breeder were considered as backup options. The MHD effects associated with self-cooled liquid-metal blanket/first wall systems are substantially reduced by the lower magnetic fields resulting from higher β plasmas. Therefore, improved performance characteristics of self-cooled liquid-metal blanket concepts are achievable compared to the BCSS guidelines because of the relaxed design constraints.

Electrical insulators have been incorporated into the blanket to further reduce operating pressures. For the lower neutron wall loadings corresponding to reactor outputs of 600-800 MWe, a partially permanent blanket has been developed that could last the entire reactor lifetime. Only a relatively thin blanket section near the first wall needs to be routinely replaced, thus minimizing waste management. A high temperature shield, utilizing lithium coolant, maximizes energy recovery and eliminates the safety concern associated with a water-cooled shield. The vanadium-alloy structure potentially provides long lifetime and higher temperature operation. Placement of manifolds in the core of the higher aspect ratio torus should provide a simpler design, better maintenance access, improved safety, and reduced shield costs.

Because the design details depend on the neutron wall loading, a number of design options were explored. These options utilize either a reversed poloidal or a straight through poloidal flow concept. The structural material in all cases is vanadium alloy and the coolant is lithium. Designs with separate limiters can achieve a neutron wall loading capability of about 5 MW/m^2 with bare structural walls near the first wall and insulated laminated wall construction in regions of low fluence only. When laminated wall construction is used in the first wall coolant channels, the neutron wall loading capability exceeds 10 MW/m^2. Designs with first-wall, self-pumped limiters require laminated construction at the first wall coolant channels. Their neutron wall loading capability is about 5 MW/m^2, but the first wall lifetime is limited to 3 years for such wall loadings.

6.4 Solid Breeder Blanket

A review of the Li_2O/He/FS concept indicated that significant improvements would be realized in the areas of tritium breeding, blanket thickness, blanket energy multiplication, power-conversion efficiency, breeder temperature window, and geometrical integrity of the coolant and purge paths by using a neutron multiplier (Be), a higher temperature structural material (vanadium-based alloy), and a tube geometry.

The main thrust of the proposed design changes is to lower the cost of electricity by making major improvements to the thermal power generation and the power-conversion efficiency. By substituting a vanadium alloy (e.g. V-15Cr-5Ti) for the BCSS HT-9 structural material, coolant outlet temperatures can be increased from $\lesssim$ 540°C to $\gtrsim$ 650°C to optimize the gross power-conversion efficiency of the system. By mixing Be with the solid breeder, the higher energy multiplication (~ 1.3 for 30% Be-70% Li_2O vs. ~ 1.1 for the BCSS design) results in a higher power generation for a given neutron wall loading.

A special feature of this design is that the He coolant acts as both the convective medium for heat transport and tritium transport. Although the tubes are sealed, the V-alloy material is very permeable to hydrogen isotopes. Thus, the tritium generated within the breeder tube permeates the tube wall at very low tritium partial pressures. Similarly, the 7 vppm D_2 in the coolant permeates into the tube to enhance tritium desorption as DT from the solid breeder surfaces. The breeder tubes are arranged in a hexagonal array and are separated by grid spacers. The tube diameter and thickness are chosen to minimize the structure volume fraction for neutronics calculations subject to thermal-hydraulic and structural constraints. The tube length is chosen to allow a reasonable internal gas space (plenum) for helium pressure build-up while trying to minimize the length in the blanket thickness direction. The tube thickness, diameter and plenum-to-breeder length ratio result in an end-of-life plenum pressure of 6.4 MPa for the limiting design stress of 220 MPa for the cladding. For a neutron wall loading of 2.5 MW/m^2, this results in an effective lifetime of 2.8 full power years.

The design calls for cylindrical breeder pellets fabricated from a 30 vol.% Be and 70 vol.% Li_2O mixture to give a finished product which has 20% porosity. The porosity serves the dual purpose of providing pathways for tritium percolation from the breeder and accommodating some of the helium-induced swelling. The Be enhances tritium breeding and energy multiplication, as well as allowing a thinner blanket to be used. By mixing the Be with Li_2O (rather than incorporating it as a separate layer), the effective thermal conductivity of the breeder is enhanced (by a factor of two for the proposed volume fraction) which more than compensates for the increase in local heating rates due to the Be. From a thermochemical point of view, the Be, which has a higher affinity for oxygen than does the breeder or cladding, controls the oxygen activity within the breeder tube to levels acceptable for inhibiting vanadium oxidation and LiOT buildup in the breeder. As the upper temperature limit is based on LiOT mass transport concerns, the presence of the Be along with the sealed tube design minimize the siginficance of LiOT formation and transport. However, an unresolved issue is the compatibility of vanadium alloys with helium coolant for both the breeder cladding and the first wall.

6.5 Flibe-Pool Reactor Concept

A new fusion reactor system concept has been developed based on a pool reactor configuration. The reactor concept is called IPFR, Integrated Pool Fusion Reactor.[24] The nuclear island, including only the first wall and the superconducting magnets, is submerged under a molten Flibe pool. The Flibe will fill the space between the first wall and the superconducting magnets and will provide the necessary magnet protection. Therefore, the Flibe serves the multiple functions of breeding, cooling, moderating, and shielding thus eliminating the requirement of a structural blanket, reflector, and shield. The required thickness of the Flibe is ~ 130 cm for superconducting magnet protection, which is acceptable. Therefore, the only structural layer remaining between the plasma chamber and the TF coil is the first wall.

A Flibe-to-Flibe intermediate heat exchanger (IHX) is also located in the pool, either at the center of the torus or at the outer edge, depending upon the availability of space. The IHX is needed for safety and tritium containment reasons. Since the working fluid is Flibe on both sides of the IHX, the IHX can continue to operate with small leaks. A pump(s) is also submerged in the pool to generate a Flibe flow upward around the first wall and downward through the IHX for the purpose of heat transport.

First wall cooling and tritium control are the two most critical issues of this concept. A detailed two-dimensional thermal-hydraulic calculation has been performed for the design, and the results indicate that up to 100 W/cm^2 first wall surface heating can be handled. The tritium problem appears to be manageable by using an intermediate loop.

The concept is rather new and has several potential features: By eliminating the blanket, shield, and the primary loop, the cost of the system can be significantly reduced. After draining of the Flibe, the first wall is exposed, and replacement of the first wall is more easily achieved. The amount of radioactive waste to be disposed is also reduced. A pool-type reactor is thought to be inherently safe. The steam generator is outside the reactor building and is radiation free, and thus hands-on maintenance is feasible.

7. CONCLUSIONS

In an effort to improve reactor economics, several ideas were examined to increase the reactor's mass power density. A key feature here is increasing the plasma β. Several concepts were examined which include higher β in the first stability regime via very low aspect ratios or highly elongated plasma shapes and access to the second stability region with and without bean shaping of the plasma cross-section. Both copper and superconducting coil concepts were considered for the first and second stability regimes.

Steady-state operation was examined for a number of cases. Both fast wave current drive and electron cyclotron heated tokamaks were examined for the second stability regime, while F-Θ pumping was considered for first stability regime reactors with very low aspect ratios. New ideas for impurity control were developed including the concept of a helium pumping first wall.

Based on this TPSS work, studies are continuing which emphasize superconducting devices in the second stability regime with $\beta \sim 15 \rightarrow 25\%$, aspect ratio ~ 6, elongation of $1.3 \rightarrow 1.6$, major radius $\sim 5 \rightarrow 6$ m, neutron wall loadings of $4 \rightarrow 5$ MW/m^2, maximum TF magnetic fields of $6 \rightarrow 8$ T, and net electrical powers of $600 \rightarrow 800$ MW(e).

The Blanket Comparison and Selection Study provided an evaluation of over 130 blanket concepts for the reference case of electric power producing, DT fueled reactors in both tokamak and tandem mirror (TMR) configurations.

The major efforts since the BCSS for improving fusion reactor blanket performance can be classified into the following areas: improvements in self-cooled liquid-metal concepts provided by reduced magnetic fields due to higher β operation and extensive use of electrically insulated walls; simplification of solid breeder concepts by innovative tritium recovery scenarios and improved economic performance by incorporating substantial amounts of beryllium as energy and neutron multipliers; expanded use of reduced activation materials, e.g., modified ferritic steels or vanadium alloys; partial blanket replacement to minimize radioactive waste management requirements; and innovative concepts that increase design simplicity and safety such as immersing the reactor in a Flibe pool.

In general, a number of new concepts have been developed which substantially improve the potential commercial attractiveness of tokamak power reactors.

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TABLE I

VARIOUS TPSS DESIGNS

	2nd Stability SC	2nd Stability CU	ET	ST
Plasma minor dimensions, a/b(m)	0.875/1.75	1.34/2.68	0.27/2.54	1.50/4.50
Plasma major toroidal radius, R_T(m)	5.25	6.72	2.73	2.70
Aspect ratio, A = R_T/a	6.0	5.0	10.0	1.8
Plasma volume, $V_p(m^3)$	150	476	46.6	358
Average plasma density, $\langle n\rangle(10^{20}/m^3)$	1.38	1.5	4.0	1.21
Average plasma temperature, $\langle T\rangle$(keV)	23	17	20	15
Plasma energy (GJ)	0.23	0.19	0.18	0.32
Toroidal field energy (GJ)	5.1	9.2	0.91	12
Net electric power P_E(MWe)	540	1,100	513	500
Total thermal power, P_{TH}(MWt)	1,559	3,007	1,768	2,047
Recirculating power fraction, $1/Q_E$	0.23	0.19	0.38	0.32
Thermal conversion, efficiency, η_{TH}	0.45	0.45	0.45	0.36
Net plant efficiency, η_p	0.35	0.37	0.29	0.24
Neutron first-wall loading, $I_w(MW/m^2)$	3.4	3.2	7.0	3.3
Plasma power density, $P_F/V_p(MW/m^3)$	8.5	5.26	38.0	4.7
Average beta, $\langle\beta\rangle$	0.25	0.25	0.315	0.291
Field at plasma, B_0(T)	3.83	2.51	4.12	2.51
Field at coil, B_c(T)	6.0	3.57	6.82	6.9
Plasma current, I_ϕ (MA)	4.0	6.1	10.1	39.8
Mass power density, (kWe/tonne)	110	90	115	79

TABLE II

SLOT SELF-PUMPED LIMITER - TYPICAL PARAMETERS

Parameter	Value
Location	Inboard or bottom
Shape	Flat or curved
Height	2 m
Front face area	60 m^2
Area of trapping surface(s)	120
Base material	Tantalum
Leading edge material	Tantalum
Trapping material	Ni,V, or Fe
Slot width	15 cm
Fraction of plasma outflux to slot	15%
Power to slot	~ 6 MW
Heat load - front face and leading edges[a]	1.5 MW/m^2
Heat load - trapping surfaces	< 0.1 MW/m^2
Slot plasma temperature	~ 10 eV
Helium removal efficiency	10%
Maximum trapping thickness	3 + 3 = 6 cm (Ni)
Operating life[a]	10 years

[a] For q_n = 2.5 MW/m^2, 75% availability.

TABLE III

RANKING AND DESIGN FEATURES FOR LEADING BLANKET CONCEPTS

Concept	Ranking	Design Features
Li/Li/V*	Overall top rated concept for tokamak, marginally superior for TMR.	• Advanced, low activation, high temperature structural alloy. • Inherent design simplicity of self-cooled concepts. • FW coolant flow parallel to B-field to facilitate heat transfer and to reduce MHD pressure. • Blanket serves as manifold to reduce MHD pressure. • Nitrogen reactor room environment to reduce chemical reactivity problem with Li.
LiPb/LiPb/V*	High ranking for TMR only.	• Advanced, low activation, high temperature structural alloy. • Inherent design simplicity of self-cooled concepts. • Simple coolant channel flow geometry in TMR configuration. • Wall thickness variation for coolant velocity control. • LiPb to He tritium recovery method.
Li_2O/He/FS*	Top rated solid breeder concept.	• Lobular, pressurized module with low-temperature helium cooling first wall. • Li_2O in plate form to optimize tritium breeding and accommodate swelling. • Low-pressure helium purge for tritium recovery. • Coolant manifold integral with blanket.
Li/He/FS*	Rates well below top three. Only marginally superior to other concepts. Unique feasibility issues.	• Lobular pressurized module with low temperature helium cooling first wall. • Low velocity Li flow for tritium recovery. • Use of barriers to facilitate tritium containment. • Nitrogen reactor room environment to reduce chemical reactivity problem with Li.

* Breeder material/coolant/structural material, V = vanadium alloy, FS = ferritic steel.