1. EXECUTIVE SUMMARY

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1. EXECUTIVE SUMMARY

1.1. INTRODUCTION

During the past several years the ARIES Team has investigated the feasibility and potential features of tokamak fusion power plants [1–4]. Building upon this research, the Starlite project was initiated to develop the goals and requirements of fusion demonstration and commercial power plants and to assess the potential of tokamaks to meet these goals and objectives. In addition, the research has aimed at identifying both the trade-offs that lead to the optimal regime of operation for a tokamak power plant and the critical plasma physics and technology issues.

The Starlite study was conducted following modern product development procedures. The Fusion Demo should demonstrate to utility customers and the general public that fusion power is a secure, safe, licensable, and environmentally attractive power source that is ready for commercialization at an economically superior total cost. To provide a basis for the design development, inputs from the end-users were solicited and analyzed to yield a Demo mission statement and a set of top-level requirements (Secs. 1.2 and 2). If these requirements are fulfilled, the fusion Demo power plant should exhibit all of the attributes required for fusion to be a commercial success as a power plant. Special emphasis was placed on potential economics (Sec. 3) and safety and licensing (Sec. 4) issues that will be faced by fusion power.

This report includes the Starlite assessments of physics and engineering options for tokamak fusion demonstration and commercial power plants. A fusion power plant which is limited to the use of currently-proven physics and technologies is unlikely to be sufficiently attractive to a utility unless fuel shortages and regulatory restrictions are far more crippling to competing energy sources than currently anticipated. In that case, the task of choosing an appropriate set of engineering technologies today involves trade-offs between attractiveness and technical risk. Therefore, the Starlite assessment was aimed at investigating (1) the attractiveness (*i.e.,* the potential to satisfy the requirements and goals), and (2) the feasibility *e.g., critical issues* and credibility (*e.g.,* degree of extrapolation required from present data base). Only through a careful balance between attractiveness and feasibility the most promising options can be identified.

Five different tokamak physics regimes of operation were considered: (1) steady-state operation in the first-stability regime, *e.g.,* ARIES-I [1] , (2) pulsed-plasma tokamak

operation, *e.g.,* Pulsar [4], (3) steady-state operation in the second-stability regime, *e.g.,* ARIES-II and ARIES-IV [3], (4) steady-state operation with reversed-shear profile, and (5) low-aspect ratio tokamaks (spherical tokamaks). The extent of the plasma physics data base as well as performance as a power plant were considered (Secs. 1.3 and 5). In parallel, several options for engineering design (*e.g.,* choice of structural material, coolant, breeder for first wall and blanket) have been developed and assessed (Sec. 1.4 and Sec.6). Because of lack of a self-consistent power-plant design, reversed-shear and spherical tokamak options were explored in more detail (Secs. 7 and 8).

1.2. TOP-LEVEL REQUIREMENTS

Based on interaction and advice from U.S. electric utilities and industry, a set of criteria for fusion power is derived (Sec. 2). A similar set of criteria has been developed by the EPRI fusion working group [6]. These criteria and associated top-level requirements and goals (Table 1.2-I) can be divided into three general categories: (1) cost, (2) safety and environmental features, and (3) reliability, maintainability, and availability.

Top-level requirements and goals for cost of electricity (COE) were adopted for the Starlite project based on the estimated cost of competitive sources of electricity at the time of introduction of fusion in the marketplace (Sec. 3). These requirements and goals for COE also are in line with projections of future power plant costs based on energy forecasting models [7]. Safety and environmental requirements are included to circumvent the difficulties experienced by fission and, to some degree, will be faced by fossil fuels in the future. Fusion should be easy to license by the national and local regulating agencies, and be able to gain public acceptance. Fusion power plants should only generate low-level waste (*i.e.,* waste storage time less than a few hundred years). Realization of the full safety and environmental potential of fusion will also help fusion to achieve a cost advantage over other sources of electricity. Fusion power plants can be designed to achieve these criteria only through the use of low-activation material and care in design. However, these requirements result in stringent constraints on the sub-system choices and design. Lastly, it should be demonstrated that Demo and commercial power plants can achieve the necessary degree of reliability. Conceptual design studies can partially address this issue (*e.g.,* by including maintenance considerations in the design). However, this criterion, to a large degree, must be addressed in the development path of fusion power. Today's experiments, by their charter, are not intended to provide detailed engineering data to support the design, construction, and operation of a power plant.

Table 1.2-I.
Top-Level Requirements and Goals
for Commercial and Demonstration Fusion Power Plants

Element	Demo	Commercial
Must use technologies to be employed in commercial power plant	Yes	Yes
Net electric output must be greater than	75% commercial	N/A
Cost of electricity (COE) must be competitive (in 1992 mill/kWh)	80 (Goal)	65 (Goal)
	90 (Reqmt)	80 (Reqmt)
No evacuation plan required for any credible accident:		
Total dose at site boundary	< 1 rem	< 1 rem
Generate no rad-waste greater than	Class C	Class C
Must demonstrate public day-to-day activity is not disturbed	Yes	Yes
Must not expose workers to a higher risk than other power plants	Yes	Yes
Must demonstrate robotic maintenance of power core	Yes	Yes
Must demonstrate routine operation with less than		
(x) unscheduled shutdowns per year including disruptions	1	1/10
Demonstrate a closed tritium fuel cycle	Yes	Yes
Must demonstrate operation at partial load conditions at	50%	50%

1.3. ASSESSMENT OF TOKAMAK REGIMES

1.3.1. First-Stability Steady-State Regime

In the late 1980's, operation at high bootstrap current fraction as the approach to steady-state operation was proposed by ARIES [1] and SSTR [8] studies simultaneously and independently. In order to reduce the current-drive power, the plasma current is reduced while the bootstrap fraction is maximized. In the first-stability regime, this can be accomplished by operating with a moderately high plasma aspect ratio ($A \equiv 1/\epsilon \sim 4.5$) and low plasma current ($I_p \sim 10$ MA) at a relatively high poloidal beta ($\epsilon\beta_p \sim 0.6$). Detailed MHD and current drive analysis have showed that the maximum bootstrap fraction in this class is about ∼70% with most of the driven current located near the magnetic axis, leading to current-drive powers of about ∼100 MW delivered to the plasma.

This mode of operation, however, leads to a low value of plasma $\beta \simeq 2\%$ because the important parameter $\epsilon\beta_p$ (which determines the bootstrap fraction) is related to achievable plasma β through

$$(\epsilon\beta_p)\,(\beta/\epsilon) \;\leq\; (\beta_N/20)^2\, S \tag{1.3-1}$$

where $S = (1+\kappa^2)/2$ is the plasma shape factor, κ is the plasma elongation, and β_N is defined by $\beta \leq \beta_N(I_p/B_T a_p)$. For a conventional first stability configuration, with optimally shaped current and pressure distributions and with sufficient triangularity in the cross-sectional shape, this equation must be obeyed with a Troyon coefficient of $\beta_N \simeq 3.5$. If a high f_{BS} is desired for steady state operation (such as ARIES-I), one can reduce the current or raise q_*, resulting in a lower β. In order to optimize the equilibrium and bootstrap current profile, these discharges operate with moderately elevated central q_0. For values of $q_0 \simeq 1.3$, the discharges are stable to kink modes without a conducting wall. Higher plasma β values can be obtained by further increasing q_0 but these discharges would require a conducting wall nearby and plasma rotation to stabilize the kink modes.

There is ample experimental data base for this regime, however, operation in discharges with durations longer than the current diffusion time is needed.

1.3.2. First-Stability Pulsed-Plasma Regime

Because the recirculating power for inductive current drive is small, it is argued that pulsed-tokamak power plants avoid the problem of non-inductive current drive: the

constraint on β_p is removed and plasma β can be higher. However, inductive current-drive also imposes certain constraints on the plasma operating regime. First, because large and expensive poloidal-field coils are needed to supply the inductive flux, these devices also tend to optimize at moderately high aspect ratio, low current, and moderately high bootstrap current fraction. Second, because the loop voltage is constant across the plasma, the current-density (induced and bootstrap) and pressure profiles are set for a given pair of density and temperature profiles. In principle, because the current-density profiles cannot be tailored, both the normalized β value and bootstrap fraction are limited ($\beta_N \sim 3.0$, $f_{BS} \lesssim 40\%$). As a result, the plasma β for an optimum pulsed-plasma operation is only marginally ($\sim$25%) larger than that of a steady-state plasma.

In addition, many critical engineering issues have to be resolved. For example, cyclic fatigue and eddy-currents induced in the structural support of the toroidal-field (TF) coils becomes major issues limiting the maximum toroidal field (*e.g.,* the superconducting-coil technology that yields a maximum field of 16 T in a steady-state power plant leads to a maximum field of about 13 T in a pulsed-plasma system). This decrease in the toroidal-field strength more than offsets the gains in plasma β values for a pulsed device, *i.e.,* even though the plasma β is higher in a pulsed device, the fusion power density is lower. Another key issue is the energy storage as the heat-transport system cannot tolerate rapid changes in the thermal power of the power plant during recharge of the OH transformer. Conventional designs for energy storage lead to large and expensive systems. An innovative, sensible-heat storage system in the outer shield was developed for the Pulsar design [4] which has removed the cost of energy-storage as a barrier to pulsed-tokamak power plant.

As a whole, even assuming the same reliability and unit cost for components, pulsed power plants are substantially more costly than steady-state ones. On the other hand, this regime of operation is supported by the main body of tokamak experimental data.

1.3.3. Second-Stability Steady-State Regime

As seen from Eq. (1.3-1), the only way to increase the plasma β and still have a large bootstrap current fraction is to increase the value of β_N. It may be possible to violate the Troyon limit if certain conditions are met such as through elevated central safety factor ($q_0 \geq 2$) [9] as in ARIES-II/IV designs, reversed magnetic shear [10,11], or at very low aspect ratios [12]. In addition to these requirements, stabilization of the external kink mode in these regimes requires a close fitting conducting wall and sufficient plasma rotation [13].

Second-stability operation (*i.e.,* operation at elevated values of central safety factor) allows for higher bootstrap current fraction than first-stability operation. While bootstrap current can become as large as the total plasma current (or larger), the bootstrap current-density profile is different than that needed for equilibrium and stability. As a result, current should be driven both on the axis and at the plasma edge while cancellation of part of the bootstrap current in the middle of the plasma is required. Essentially, the optimum configuration has a bootstrap current fraction of ∼90% . Compared to the steady-state, first-stability optimum regime, the current-drive power is somewhat smaller and the plasma β is about twice as large (a conducting wall for stabilization of kink modes is required). The critical physics issues for this regime of operation include startup and access to the second-stable operating point, as well as complexity of the current drive system. In addition, there is very little experimental evidence for this mode of operation.

1.3.4. Reversed-Shear Steady-State Regime

Reversed-shear plasma operation combines the best features of steady-state, first- and second-stability modes. There is a much better match between the bootstrap current-density profile and that needed for equilibrium and stability and, therefore, higher values of plasma β can be achieved with moderate current-drive power and less complicated current-drive system. The primary characteristics of a reversed-shear plasma are a hollow current-density profile, a non-monotonic safety-factor (q) profile, and relatively peaked pressure profiles. The hollow current-density profile gives rise to a safety-factor profile, which initially decreases from its value at the plasma center to a minimum value, and then rises to its value at the plasma edge. It is the initial decrease in the q-value away from the magnetic axis that provides the negative magnetic shear, which is responsible for stability to $n \rightarrow \infty$ ballooning modes. It also appears that in this regime, the plasma transport is suppressed and the a more peaked pressure profile that is consistent with the high β and high bootstrap-current fraction can be sustained. There is ample theoretical research on this regime and some experimental data base is available. An extensive experimental exploration of this regime of operation is currently on-going.

Analysis has showed that to zeroth order, the cost of the device is independent of the plasma aspect ratio in the range of $A \sim 3 - 4$ (lower plasma β at the higher A is compensated by higher toroidal-field strength on axis and lower current-drive power). In the Starlite study, an aspect ratio of $A = 4$ was chosen because of engineering considerations. At this aspect ratio and with a plasma current of ∼11 MA, the maximum theoretical plasma β is ∼5.5% and the bootstrap-current fraction is ∼90%. Non-inductive current

drive is required to supplement bootstrap current on axis (*e.g.,* fast wave), at the plasma edge (*e.g.,* lower hybrid), and in mid plasma. While several candidate current-drive option exists for driving the current in mid-plasma (*e.g.,* high-frequency fast wave, mode conversion), the data base for these current-drive schemes is very small.

1.3.5. Low-Aspect-Ratio (Spherical) Tokamak

Operation at a low plasma aspect ratio (low-aspect ratio or spherical tokamaks) is another approach to achieving high plasma β and a high bootstrap current fraction. Unfortunately, the low aspect ratio rules out the use of superconducting toroidal-field coils as there is not enough space for a thick shield in the inboard. Therefore, a low-aspect-ratio tokamak requires equilibria with a very high plasma β (in order to minimize the Joule losses in the normal-conducting center-post). In addition, because the plasma current is very large ($\sim$30 to 40 MA) in a low-aspect-ratio tokamak, a very high bootstrap current which is also precisely aligned to the equilibrium current profile is essential (to minimize current-drive power). Detailed current-drive analysis has shown that even for a bootstrap-current fraction of 95%, the current drive power can easily exceeds several hundreds of MW because the mismatch is usually at the plasma edge and mid-plasma where current-drive systems are inefficient.

Since the power requirements associated with the edge current drive turned out to be so large, we concentrated on a series of equilibria that require only central current drive. We imposed the constraint that the surface averaged parallel current density be a linear function of the poloidal flux ψ interior to the point where the linear profile is tangent to the bootstrap current profile and equal to the bootstrap current outside this point (Figure 1.3-1). This constraint completely specifies the profiles and ensures that the first derivatives of equilibrium quantities are continuous everywhere. An extensive scan of MHD equilibria led to an equilibrium with $A = 1.25$, $\beta \simeq 40\%$ and over 99% of the plasma current self-driven (current-drive power is $\lesssim$10 MW). This equilibrium has an elongation of 3.0, which will certainly necessitate a vertical position control system. It is noted that these cases would require a conducting wall to stabilize the external kink mode.

Using the case noted above, preliminary system studies and engineering analyses were conducted. A major conclusion of the system studies was that, even for these cases with low current-drive power requirements, there still was significant recirculating power (*i.e.,* recirculating power fraction $\sim$0.6) associated with the resistive losses in the copper center post and return legs of the toroidal field coils. Because of the high-recirculating power

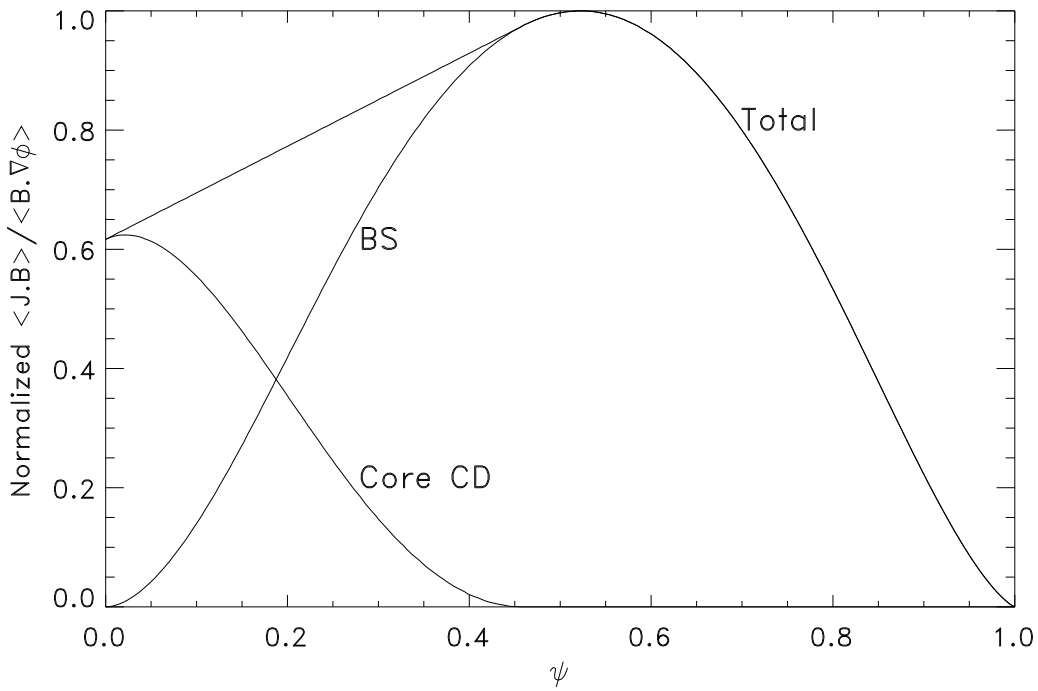

Figure 1.3-1. Parallel currents as functions of poloidal flux ψ for a high-bootstrap-fraction spherical tokamak equilibrium which does not require edge current drive.

fraction, the low-aspect ratio tokamak is quite sensitive to choice of plasma β, design of the center-post, and detail analysis of the current drive requirements. Other studies suggest lower values of recirculating power fraction in a low-aspect ratio power plants.

The spherical tokamak offers unique features and is currently pursued vigorously in the physics community. However, the level of analysis has not reached the stage needed to make sound assessment of this concept. Many critical issues have to be resolved such as: the sensitivity of the current drive power requirements to variations in plasma profiles, center-post design, start-up, design of in-vessel components which are subject to high heat and particle loads, *etc.* The true potential of a low-aspect ratio tokamak power plant can only be assessed after a comprehensive design study.

1.3.6. Systems Assessment of Tokamak Physics Regimes

A system assessment of the above five tokamak plasma regimes has been performed. In order to provide a common basis for comparing the cost of power plants, a similar blanket and shield concept has been used. The design uses vanadium alloy as structural

material of first wall, blanket, and divertor and liquid lithium as the breeder which is a modification of ARIES-II fusion core design. It is not clear if such a design can be utilized in a spherical tokamak (most probably, the performance of the thermal power system would be lower because of the higher wall loading and presence of water as center-post coolant). In addition, we have utilized a maintenance scheme for radial sector removal (similar to those of ARIES-IV design [3]). This integrated sector arrangement eliminates in-vessel maintenance operations and provides a very sturdy continuous structure able to withstand large loads. The penalty is the increased size of the TF coils needed to allow adequate space for sector removal. Lastly, in order to minimize unscheduled interruption of plant operation, all designs operate at 90% of maximum theoretical β in order to avoid plasma disruption. The major parameters of the five Starlite candidates are shown in Table 1.3-I.

In addition to economic performance, an assessment of maturity of each concept was made. To do so, several figures of merits were identified (Sec. 5). The MHD figures of merits are β/ϵ and $\epsilon\beta_p$ as they indicate progress toward high β discharges with a high bootstrap current fraction. The current-drive figure of merit is defined as $\gamma_B = (\overline{n_e}/10^{20})I_pR/P_{CD}$ which is the conventional measure of current drive efficiency but with the total plasma current appearing to account for bootstrap effects. The energy confinement figures of merit are chosen to be $\beta\tau_E/a^2$ as a function of $(\beta/\epsilon)^2$ to indicate progress towards discharges with high energy confinement time and high plasma β. Figure 1.3-2 shows the existing tokamak data base in two dimensional parameter spaces of these figures of merit. The five Starlite candidates are also shown.

The first-stability pulsed-plasma and steady-state regimes are closest to the present data base. Of course these regimes should be demonstrated at long-pulse discharges with burning plasmas. On the other hand, the economic performance of pulsed-plasma operation is poor. First-stability steady-state did not achieve the economic requirements for the Starlite project. High-field TF coils can improve the attractiveness of this regime of operation. The second-stability regime has better economic performance but the experimental data base for this regime is very small. The data base for the spherical tokamaks is not mature and, in addition, many critical issues remain. Detailed design studies are needed before the true potential of the spherical tokamak power plant can be assessed. The reversed-shear mode of operation offers the best economic performance. The data base for this regime, while small, is growing rapidly. Based on the superior economic performance and the growing experimental and theoretical data base, the Starlite project has chosen reversed-shear as the reference plasma operation regime for the ARIES-RS design [5].

Table 1.3-I.
Parameters for the Five Starlite Power Plant Candidates

	FS	PU	RS	SS[(a)]	LAR
Plasma aspect ratio, $A = R/a_p$	4.0	4.0	4.0	4.0	1.25
Major radius, R(m)	7.96	8.68	5.04	6.40	5.00
Plasma minor radius, a_p(m)	1.99	2.17	1.26	1.60	4.00
Plasma elongation, κ_X	1.81	1.80	1.99	2.03	3.40
Plasma triangularity δ_X	0.71	0.50	0.69	0.67	0.55
Cylindrical safety factor q_*	3.77	2.40	2.37	4.60	3.54
Central Safety factor, q_0	1.3	0.7	2.8	2.0	2.9
Stability parameter, $\epsilon\beta_p$	0.54	0.32	0.56	1.22	1.14
Normalized beta, $\beta_N(\%)$	2.88	2.70	4.76	5.28	6.42
Beta parameter, $\beta/\epsilon(\%)$	8.12	10.00	21.24	12.17	45.36
Confinement ratio, τ^*_{He}/τ_E	10.0	10.0	10.0	10.0	10.0
Ignition parameter, $\beta\tau_E/a_p^2$(%s/m^2)	0.78	2.18	4.17	1.35	11.05
ITER-89P scaling multiplier, H	1.71	2.38	2.40	2.47	3.02
Normalized confinement multiplier, H/q_*	0.45	0.99	1.01	0.54	0.85
Plasma current, I_p(MA)	12.6	15.0	10.3	7.72	40.1
Bootstrap-current fraction, f_{BS}	0.57	0.34	0.89	>1	0.997[(b)]
Current-drive efficiency, $\gamma_B(10^{20}$A/W m$^2)$	0.56	NA	2.02	0.49	34.2
Current-drive power to plasma, P_{CD}(MW)	236.6	0	64.7	199.1	7.3
On-axis toroidal field, B_T(T)	8.99	7.46	7.35	8.37	1.77
Peak field at TF coil, B_{TF}(T)	16.0	13.1	15.7	15.9	14.8
Normalized heat flux, P_{HEAT}/R(MW/m)	71.2	29.5	71.3	89.0	124.2
Recirculating power fraction, $(1/Q_E)$	0.29	0.06	0.13	0.33	0.63
COE (mill/kWh)	99.7	130.2	69.7	92.6	116.0

(a) This design is not optimized to the lowest COE.

(b) Includes diamagnetic current.

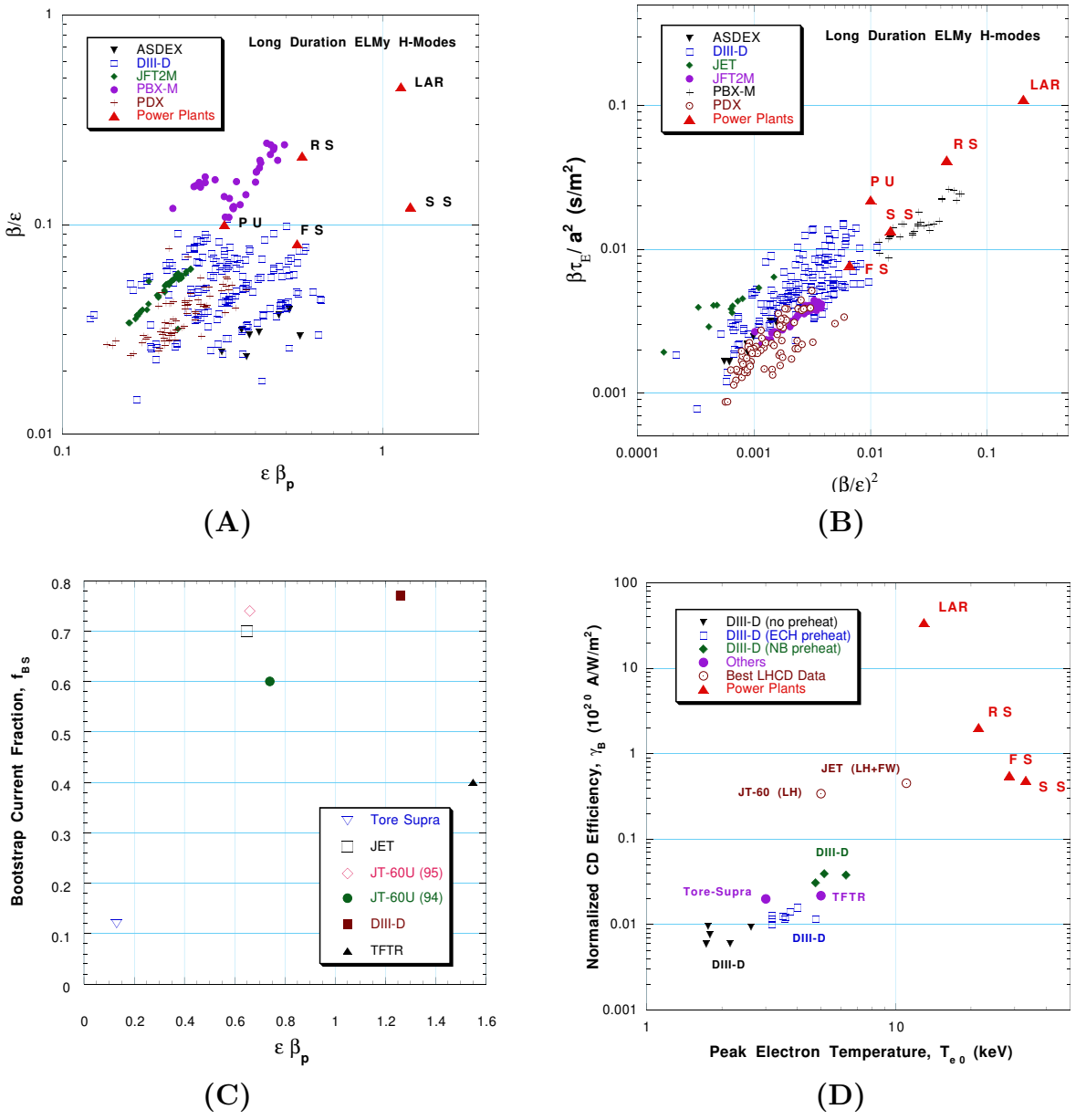

Figure 1.3-2. (A) MHD figures of merit for long-pulse ELMy, H-mode discharges in several existing devices; (B) Energy confinement figures of merit for a number existing tokamaks (Data provided by Stan Kaye, PPPL); (C) Large tokamak experiments which display substantial bootstrap fraction at high poloidal beta [Note that poloidal beta definitions vary somewhat; and plasma conditions (collisionality, density gradient, impurity content, *etc.*) also vary]; (D) Fast-wave current-drive experimental database on DIII-D [14], Tore Supra [15], and TFTR [16]; best data point for lower-hybrid current-drive on JT-60 [17] and for lower-hybrid and FWCD synergy on JET [18]. The five Starlite fusion power plant options (FS, PU, RS, SS, and LAR) are also shown.

1.4. ENGINEERING ASSESSMENT OF DESIGN OPTIONS

Through its successful operation, the U.S. Fusion Demo must be sufficiently convincing that a utility or independent power producer will choose to purchase one as its next electric generating plant. A fusion power plant which is limited to the use of currently-proven technologies is unlikely to be sufficiently attractive to a utility unless fuel shortages and regulatory restrictions are far more crippling to competing energy sources than currently anticipated. In that case, the task of choosing an appropriate set of engineering technologies today involves trade-offs between attractiveness and technical risk. The design space for an attractive D–T tokamak fusion power core is not unlimited; previous studies have shown that advanced low-activation ferritic steel, vanadium alloy, or SiC/SiC composites are the only candidates we have for the primary in-vessel structural material. An assessment of engineering design options has been performed using these three materials and the associated in-vessel component designs which are compatible with them.

The requirements described in Sec. 2 provide goals for engineering components that are very challenging to meet. Safety and environmental requirements severely limit material choices. Performance requirements provide strong incentives to operate at high coolant temperature. Reliability requirements may be the most difficult to meet. Design solutions must be simple, incorporate adequate performance margins, and be tested fully.

For this assessment, detailed designs were not developed. Instead, various design options have been examined and their potential to meet the power plant requirements assessed. These options include material choices for the structure, breeder and coolant. In order to provide a framework for this assessment, the three primary in-vessel structural material classes are used to distinguish design classes. The design options using these materials are surveyed, paying special attention to the first wall, blanket, shield and divertor, and the support systems required to operate them. For each class of designs, their ability to meet power plant requirements are summarized and the key issues are assessed. Key issues are those uncertainties which, if not adequately resolved, would lead to an unattractive product which does not meet one or more of the top-level requirements.

1.4.1. Assessment of Ferritic Steel designs

Design Options. Conceptual designs using ferritic steel as the structural material have been proposed and developed for both solid and liquid breeder materials, as well as water, helium and liquid-metal coolants. For most breeder, multiplier and coolant materials

(except for Pb-Li), ferritic steels are compatible up to the strength-based temperature limit of the material (550 °C). The design combination which appears to be most appealing in the current world fusion R&D uses beryllium multiplier with solid breeder and is cooled by helium flowing through tubes. Reduced activation designs involve the use of Li_2O or Li_2TiO_3 solid breeders and Fe-9Cr-2WVTa ferritic steel.

Evaluation Highlights. The evaluation focused on the cost of electricity (COE), maintainability without high-risk exposure to workers, and waste disposal. Factors influencing the COE are the material and fabrication costs, the thermal efficiency achievable (limited by the maximum structural temperature allowed), the peak surface and volumetric heat loads allowed, and the lifetime. More expensive than austenitic steels, ferritic steels are estimated to cost 20%–25% of vanadium alloys. In addition, some ferritic steels (not those alloys under development for fusion) are design-code-qualified materials with an extensive database and industrial experience.

The 550 °C temperature limit restricts the thermal efficiency, and the peak surface heat load for a 5-mm wall is limited to ∼0.8 MW/m^2. This would restrict the peak neutron wall loading and, in some cases, result in a larger machine with higher capital costs. The surface heat flux limit also would make ferritic steels unattractive as a divertor structural material. It is desirable for a structural material to tolerate ≥1 MW/m^2 of surface heat load.

Lifetime limits depend on operating stresses and irradiation effects. By reducing the operating stresses, it is estimated that lifetimes of at least 12 MW-y/m^2 (3 full power years) can be achieved with ferritic steels. Preliminary cost analyses indicate that the required cost of electricity (≤80 mills/kWh) could be achieved with ferritic steel as a structural material because the lower material and fabrication costs would tend to partially off-set the cost associated with larger machine size and reduced thermal efficiency. These results, however, assume that a different, high-performance material, is used for the divertor.

Maintenance of any of the candidate structural materials would have to be performed remotely because of the high level of radioactivity and heat following shut-down. In terms of waste disposal, reduced-activation ferritic steels are better than austenitic steels and qualify as Class-C waste according to NRC guidelines.

Issues. The lower temperature limit for ferritic steels is not well established. Ferritic steels irradiated at temperatures ≤400 °C tend to embrittle, as indicated by a decreasing

tensile ductility and an increasing ductile-to-brittle-transition temperature (DBTT) with irradiation. Experiments in mixed-spectrum fission reactors result in DBTT values of about 250 °C under conditions of simultaneous helium production and neutron damage, while experiments at ≥365 °C in fast fission reactors with little helium production give DBTT <0 °C for some of these steels. The He production (≥10 appm-He/dpa) would be relatively high in a fusion power core. Without fully understanding the role of helium in raising the DBTT, it is difficult to extrapolate the fission reactor results to fusion power plant conditions. If the DBTT were to reach 250 °C, then the minimum operating temperature would have to be set at 285 °C, leaving a relatively narrow temperature window for design, and possible problems with start-up, shut-down and off-normal events during shut-down.

The maximum operating temperature of 550 °C limits the thermal conversion efficiency. In addition, use of ferritic steels in the divertor is a major concern. Lastly, ferromagnetic effects are important unresolved issues.

1.4.2. Assessment of V designs

Design Options. A key design issue associated with V blankets is the compatibility of V-alloy with either hydrogen or oxygen environments at higher temperatures (>300–400 °C). Vanadium-alloys have strong affinity for both hydrogen and oxygen, and become brittle with moderate hydrogen or oxygen concentration. Lithium is one of the few materials which has even stronger affinity to hydrogen and oxygen than vanadium [19]. Thus, it is natural to design the vanadium blanket with lithium acting as both coolant and breeding material [3, 20]. A self-cooled blanket simplifies the blanket design. However, electrical insulation between the coolant and metallic walls may be needed to reduce the MHD pressure drop, and the large lithium inventory causes safety concerns.

To reduce or eliminate lithium safety concern, a He-cooled lithium breeding blanket was examined in the BCSS project [21] and also studied for the Starlite study (Sec. 6). A closed-cycle gas turbine was proposed for power conversion in the latter design. The elimination of the steam cycle reduces the source term for both hydrogen and oxygen. A surface modification approach is proposed to protect the V-alloy, if necessary. Another design has been proposed using Li_2O as the breeding material to enhance even further the safety characteristics [22]. The use of SiC or Al_2O_3 for V-alloy protection was proposed, but not discussed in detail.

Evaluation Highlights. Vanadium-based design options have been evaluated relative to the requirements and in all cases the component designs either meet the requirements or are capable of meeting them, depending on the results of complete system integration which is yet to be performed.

Issues. For any fusion power core design, a key issue is associated with the development of the structural material. There is no intense 14-MeV neutron source available. Therefore, the material response to fusion neutrons can not be quantified. For a more advanced structural material, there is also a lack of industrial experience. Therefore, large scale fabrication and joining to other structural materials also provides cause for concern.

For the self-cooled liquid metal design, the key issue is to develop and assure long term reliability of the insulating coating. Only limited results in this area are available and some important effects, such as neutron irradiation and thermal cycling, have not been included in experiments. Tritium recovery from lithium to the design goal of $\sim$1 appm has yet to be demonstrated. For a blanket with lithium as the breeding material, safety is always a key concern. Lithium will react with water, air, concrete, nitrogen, *etc.* The power plant design should aim at minimizing the frequency and the consequence of the lithium reaction when an accident occurs.

1.4.3. Assessment of SiC Designs

Design Options. SiC composite material with a projected allowable temperature capability of $\geq$1000 °C was selected as the structural materials for ARIES-I and ARIES-IV blanket and divertor designs. These designs have been used as the basis for assessing the advantages and issues associated with a possible power plant based on SiC composites as the primary in-vessel structural material.

To take full advantage of the low-activation and high-temperature capability of SiC composite material, 5–10 MPa helium was used as the coolant. ARIES-I used Li_2ZrO_3 as the solid tritium breeder, Be metal sphere-pac as the neutron multiplier, and W as the divertor coating material. Due to the induced activation of Zr and W, the safety rating of this design was Level of Safety Assurance (LSA)-2. Based on new experimental results, Li_2O was selected as the solid breeder for the ARIES-IV designs. If the use of W can be avoided as the PFC coating material, then a completely passively safe design can be obtained, if Be chemical energy and toxicity concerns are eliminated.

The ARIES-II design used the toroidal flow, nested shell blanket configuration, where the SiC composite components can be made from smaller ceramic (about 1 m×1 m) parts. Reliability of this configuration depends on the successful development of ceramic joining techniques. The ARIES-IV reference design has a poloidal flow configuration. This relies on the successful development of the fabrication of large (>1 m wide, ∼1 m deep and >7 m in height components) leak-tight SiC composite components.

Evaluation Highlights. The ARIES-IV blanket design satisfies all of the neutronics and thermal hydraulic performance requirements. Except for the need of Be neutron multiplier (which was included due to uncertainty in neutronics data base and may not be needed), this design can have excellent safety characteristics. By increasing the helium pressure to about 12 MPa, and taking advantage of the improved recuperator performance a closed cycle gas turbine system at a helium outlet temperature of about 950 °C and a gross thermal efficiency of about 55% can be used. In the evaluation for the use of SiC-composite as structural material, there was no doubt of its projected benefits; the key question is whether its development schedule can meet the project schedule of a Demo given small resources devoted to this material in the U.S.

Issues. Key issues for the application of SiC composite are in the areas of material development, the behavior of composites in a fusion environment, the need for electrically-conducting components to stabilize the plasma, and the development of reliable plasma facing components (PFC).

Fundamental improvements in material irradiated properties, such as thermal conductivity, are needed, for example by using advanced SiC fiber and interface materials. Other material development issues include economic fabrication of large SiC-composite components, development of vacuum leak-tight components, the technique of brazing ceramic parts, and the development of the joining techniques of SiC-composite to metallic parts. Under advanced physics scenarios (*e.g.*, the reversed shear mode), close conducting structures are needed for plasma kink stabilization, and separate passive and active, toroidally connected and electrically conducting rings are needed for vertical stabilization. This will determine the amount of electrically-conducting (*e.g.,* metallic) elements that will be needed close to the plasma, thus impacting the safety rating and thermal performance of the plant. Various techniques of leveling the surface heat flux, such as the use of the radiative divertor approach is essential, since it dictates the feasibility of removing surface heat flux by helium coolant and the utilization of low activation material for the PFC surface and components.

1.4.4. Neutronics Assessment

A neutronics assessment of breeder and shield materials was performed. Among the 6 liquid breeders and 14 solid breeders considered in the analysis, lithium provides the highest breeding, followed by PbLi and then Li_2O. The most promising liquid metal (Li, $Li_{17}Pb_{83}$) and solid ceramic (Li_2O, Li_2TiO_3) breeders were selected for further analysis to illustrate the impact on breeding of the candidate structures (V, SiC, and FS) and neutron multipliers (Be, Be_2C, BeO, and Pb). The study concluded that both Li and $Li_{17}Pb_{83}$ have the potential for tritium self-sufficiency without a neutron multiplier. Solid breeders will most likely require a neutron multiplier to achieve net breeding in realistic designs.

All structural materials degrade the breeding and enhance the energy multiplication, except SiC, that degrades both. Vanadium has the least impact on the breeding, followed by stainless steel and SiC, except for Li, $Li_{17}Pb_{83}$. For ceramic breeders, all multipliers enhance breeding, except BeO, and the energy multiplication, except Pb.

The blanket supplies all the tritium needed for operation (0.2–0.3 kg/d). An external T supply is only needed to start and for a short time until steady state production of T is reached. To guarantee T self-sufficiency, the uncertainties in all design elements should be accounted for when estimating the breeding level for the various blanket options. The largest source of uncertainty is the basic nuclear data and the calculational model ($\sim$10%). The overall tritium breeding ratio (TBR) calculated with present-day data and models should exceed 1.1 to assure that the actual achievable TBR from the blanket after operation is $>$1.01. In order to enhance the breeding capability of the blanket, it is recommended to reduce the structure content, maximize the blanket coverage, and locate penetrations off mid-plane as much as practically possible. Other options, particularly for blankets with marginal breeding, include the use of neutron multipliers and/or enriching the lithium.

A list of top level requirements was compiled for the Demo shield (bulk, penetration, and biological). The bulk shield will provide lifetime protection for the vacuum vessel and magnets, must have low safety and environmental impact, and should be reliable, maintainable, and replaceable. The shield is a lifetime component and the blanket/reflector will provide lifetime protection for the bulk shield. As the bulk shield has major impact on the overall machine size and cost, it should be optimized for high performance, low cost, and minimal safety impact. Steel has the best shielding performance, followed by V and then SiC. General directions were identified for designing an efficient and cost-effective bulk shield. These include the use of steel structure for the outer layers of the

shield and employing higher performance materials in the inboard region and cheaper but efficient shielding materials in the less constrained outboard and divertor regions.

The safety features of breeders and structures will be a key factor in selecting materials. SiC composites possess the lowest activation and afterheat characteristics, followed by V and then FS (modified HT-9 or F82H alloys). All candidate structural materials would qualify for Class C low level waste according to NRC limits. Safety-related issues for the candidate materials are well documented in the literature and design solutions can be identified.

1.4.5. Assessment

Three candidate primary in-vessel structural materials – ferritic steel, vanadium alloy, and SiC composite – have been examined to determine if they can meet the requirements for an attractive power plant and to characterize the remaining technical issues.

SiC composite has a unique potential for safety and high performance, but the database requires substantial improvement in order to identify and develop a satisfactory material composition. Significant improvements are needed in the basic properties, and several key material issues such as joining and hermeticity must be resolved. It was judged premature to adopt such a material for Demo, which was assumed to have a relatively near-term schedule given small resources devoted to this material in the U.S.

Ferritic steel has the largest database, however, as with all materials, behavior after long-term operation in a complete fusion environment is highly uncertain. The features of ferritic steel which cause the greatest concern is its restricted temperature window, which limits the maximum achievable thermal conversion efficiency, and its loss of ductility. It is not clear how one would operate a high-power device with potentially brittle materials. Economic studies suggest that the lower cost of steel (as compared with vanadium alloy) does not completely offset the lower power density and lower efficiency. In addition, use of ferritic steels in the divertor is a major concern, and ferromagnetic effects are important unresolved issues.

Vanadium offers significant advantages in its high temperature and high thermal performance capability, and its low activation. The combination of V and Li appears particularly unique in material chemistry, offering good compatibility up to very high temperatures. The primary concerns with vanadium alloy are high cost and liquid metal MHD effects (including the need for and feasibility of insulating coatings). Minimization of vanadium in the shield and external systems is an important cost-reduction strategy.

The absence of an established industrial base is also an important consideration. Due to its greater potential for attractive commercial power plants and a development path which appears practical within a 25-year time-frame given the present U.S. R&D program, the combination of Li breeder/coolant and vanadium-alloy structure was chosen as the next design concept to undergo full system design and analysis in the ARIES-RS study.

1.5. SUMMARY AND CONCLUSIONS

The Starlite Project was initiated to investigate the mission, requirements and goals, features, and R&D needs of the Fusion Demonstration Power Plant. The Fusion Demo should demonstrate to utility customers and the general public that fusion power is a secure, safe, licensable, and environmentally attractive power source that is ready for commercialization at an economically superior total cost. To provide a basis for the design development, inputs from the end-users were solicited and analyzed to yield a Demo mission statement and a set of top-level requirements. If this mission statement is achieved and the requirements fulfilled, the fusion Demo power plant should exhibit all of the attributes required for fusion to be a commercial success as a power plant heat source.

An assessment of the five tokamak physics regimes of operation was made based on the economic performance as summarized in Table 1.3-I and based on the maturity of data base as summarized in Figure 1.3-2. The first-stability pulsed-plasma and steady-state regimes are closest to present data base. Of course these regimes should be demonstrated at long-pulse discharges with burning plasmas. On the other hand, the economic performance of pulsed-plasma operation is poor. First-stability steady-state did not achieve the economic requirements for the Starlite project. High-field TF coils can improve the attractiveness of this regime of operation. The second-stability regime has better economic performance but the experimental data base for this regime is very small. The data base for the spherical tokamaks is not mature and, in addition, many critical issues remain. Detailed design studies are needed before the true potential of spherical tokamak power plant can be assessed. The reversed-shear mode of operation offers the best economic performance. The data base for this regime, while small, is growing rapidly. Based on the superior economic performance and the growing experimental and theoretical data base, the Starlite project has chosen reversed-shear as the reference plasma operation regime for the ARIES-RS design [5].

SiC composite has a unique potential for safety and high performance, but the database requires substantial improvement in order to identify and develop a satisfac-

tory material composition. It was judged premature to adopt such a material for Demo, which was assumed to have a relatively near-term schedule given small resources devoted to this material in the U.S. Ferritic steel has the largest database, however, as with all materials, behavior after long-term operation in a complete fusion environment is highly uncertain. The features of ferritic steel which cause the greatest concern is its restricted temperature window, which limits the maximum achievable power density and thermal conversion efficiency, and its loss of ductility. Vanadium offers significant advantages in its high temperature and high thermal performance capability, and its low activation. The primary concerns with vanadium alloy are high cost and liquid metal MHD effects (including the need for and feasibility of insulating coatings), and the safety issues associated with the use of liquid lithium. The absence of an established industrial base is also an important consideration. Due to its greater potential for attractive commercial power plants and a development path which appears practical within a 25-year timeframe given the present U.S. R&D program, the combination of Li breeder/coolant and vanadium-alloy structure was chosen as the next design concept to undergo full system design and analysis in the ARIES-RS study.

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