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STATUS OF THE U.S. STELLARATOR REACTOR STUDY*

J. F. Lyon

Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831, USA

K. Gulec

University of Tennessee, Knoxville, Tennessee, 37996, USA

R. L. Miller

Los Alamos National Laboratory, Los Alamos, New Mexico, 87545, USA

L. El-Guebaly

University of Wisconsin, Madison, Wisconsin, 63706, USA

representing the U.S. Stellarator Reactor Study Group

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ABSTRACT

Stellarators have significant operational advantages over tokamaks as ignited steady-state reactors. This scoping study, which uses an integrated cost-minimization code that incorporates costing and reactor component models self-consistently with a 1-D energy transport calculation, shows that a torsatron reactor could also be competitive with a tokamak reactor. The projected cost of electricity (COE) estimated using the ARIES costing algorithms is 62.5 mill/kW(e)h in constant 1992 dollars for a 1-GW(e) Compact Torsatron reactor reference case. The COE is relatively insensitive (<10% variation) over a wide range of assumptions including variations in the maximum field allowed on the coils, the coil elongation, the shape of the density profile, the beta limit, the confinement multiplier, and the presence of a large loss region for alpha particles. The largest variations in the COE occur for variations in the electrical power output demanded and the plasma-coil separation ratio.

1. INTRODUCTION

Stellarators would have significant operational advantages as ignited steady-state reactors because they do not require a net plasma current (and the continuous power recirculated to the plasma to drive it). The magnetic field is created by currents flowing solely in external coils, resulting in inherently steady-state, disruption-free magnetic configurations with relaxed constraints on the plasma parameters and profiles and a wide range of magnetic configurations available for optimization and control. The absence of both dangerous disruptions and continuous current drive power recirculated to the plasma eases the design of the first wall, blanket, and shield. The larger aspect ratio may allow access from the inboard side for easier maintenance. The type of stellarator configuration (torsatron) used as an example here can also have helical divertors outside the windings to reduce the power density on the divertor plates and, at the expense of a reduction in alpha-particle heating [1], a near-perpendicular loss region to eliminate helium ash accumulation.

Although stellarators have the potential for leading to a better reactor, they lag behind tokamaks in their development because of the greater resources that have been devoted to tokamaks and the wider range of possible stellarator configurations. Stellarator reactor studies are

Fig.1. Top and side views of the helical windings and the last closed flux surface for the CT6 configuration.

Fig.2 A vertical cross section of the helical windings and magnetic flux surfaces for CT6 at the beginning of a field period.

those developed in the ARIES and PULSAR tokamak reactor studies. However, there is no blanket directly under the helical windings on the inboard side of the torus in order to reduce R_0 . The inboard half of the helical winding with its side shields typically covers 15–20% of the first wall area. The loss in the global tritium breeding can be compensated by increasing the thickness of the blanket elsewhere and/or increasing the beryllium fraction. The thickness of the blanket elsewhere is increased here to 80 cm (versus 35 cm on the inboard side and 60 cm on the outboard side for ARIES–IV). The radial build is shown in Fig. 3. The components shown are (1) the superconducting winding pack, (2) the 5-cm-thick cryostat, (3) the 10-cm-thick vacuum vessel, (4) a lead shield, (5) a $\text{TiH}_{1.8}$ shield, (6) a SiC shield, (7) a $\text{Li}_2\text{O/SiC/Be/He}$ blanket module, and (8) the 1-cm-thick first wall. There are also gaps between the various components: 5 cm between the cryostat and the vacuum vessel, 2 cm between the shield and the vacuum vessel, 2 cm between the shield and the blanket, and 5 cm scrapeoff between the plasma edge and the first wall. The blanket and shield components, developed during the ARIES–IV and PULSAR studies, were chosen for their low activation properties. The $\text{TiH}_{1.8}$ shield makes the shielding effectiveness of the ceramic shield comparable to that of a metallic shield. The blanket is assumed to have a global tritium breeding ratio of 1.12 and an energy multiplication factor of 1.3. The shield has a thickness of 100 cm under the helical windings on the inboard side of the torus and a thickness of 75 cm elsewhere.

The coils, like those in ARIES use Nb_3Sn superconducting cables with a partially load-bearing CuNb stabilizer. The superconducting winding pack, which includes the supercon-

SI units. Stellarators and tokamaks have similar energy confinement time scaling, indicating that the underlying physics may be dominated by common toroidal plasma physics rather than coil-geometry-specific effects.

Unlike the LHD scaling, the Lackner-Gottardi scaling and the gyro-reduced Bohm scaling are dimensionally correct; i.e., they are expressible in terms of dimensionless plasma parameters. In addition, they have the same functional dependence on the reactor parameters R_0 , B_0 , n , and P , differing only by an aspect-ratio-dependent coefficient; for the torsatron reactor examples studied in this paper, $\tau_E^{\text{LG}} = 0.95\tau_E^{\text{gtB}}$. Coincidentally, the Lackner-Gottardi and LHD scalings also give nearly the same value of τ_E for typical reactor cases, even though they have different functional dependences on the reactor parameters. For this paper we choose Lackner-Gottardi scaling with a confinement improvement factor H' similar to the H-mode confinement improvement factor for tokamaks. Evidence from experiments and theoretical arguments support such a confinement improvement. However, the improvement with the square root of the ion mass used in tokamak scaling is not assumed in our study, and τ is evaluated at a normalized radius $\rho = r/a_p = 2/3$, rather than at the plasma edge; reversing either of these assumptions would improve the confinement time by a factor of 1.2–1.3.

The maximum plasma density in stellarators is not determined by a disruption limit as in tokamaks. Sudo et al. [9] have proposed a maximum line-average density,

$$n_{\text{max}} = 0.25(PB_0/R_0a_p^2)^{1/2}, \quad (4)$$

based on Heliotron E data. Densities a factor of 1.3 higher than this value have been observed in ATF. For this study, we assume that the line-average density is constrained to less than 1.5 times this value.

2.4. Transport Model

One-dimensional (1-D) heat transport equations for the ions and electrons are solved for the ion and electron temperatures $T_i(\rho)$ and $T_e(\rho)$ using a heat diffusivity $\chi(\rho) = \chi_0/(1 - \alpha\rho^\gamma)$ where the constants χ_0 , α , and γ are chosen to give an energy confinement time $\tau_E = H'\tau_E^{\text{LG}}$. The reference assumptions are $H' = 2$, $\alpha = 0.9$, and $\gamma = 2$, for a factor of 10 variation in χ from the center to the edge of the plasma. The internal heat sources and sinks include alpha-particle heating, bremsstrahlung and synchrotron radiation, and electron-ion Coulomb collisions. Impurities are modeled by 1% oxygen, giving $n_{\text{DT}}/n_e = 0.92$ and $Z_{\text{eff}} = 1.56$. The density profile used is the same as that assumed in the ARIES studies, $n = (n_0 - n_s)(1 - \rho^2)^{\alpha_n} + n_s$; the reference values are $\alpha_n = 1.0$ and $n_s/n_0 = 0.538$.

2.4.1. Alpha-Particle Losses

Because the relatively large helical ripple in torsatrons, combined with symmetry-breaking toroidal effects, can lead to a near-perpendicular loss region for energetic particles, we assume that all helically and toroidally trapped alpha particles are lost and calculate the additional energy lost by pitch-angle scattering into the loss region during the slowing-down process. The combined loss can reduce the effective alpha-particle heating by up to ~40%. Most of the alpha particles are not born in the loss region and transfer their energy to the background plasma until they slow down to an energy $W \approx 30T_e \sim 0.3\text{--}1$ MeV, below which they rapidly scatter into the loss region. This loss can produce an intense flux of energetic alpha particles on the divertor plates.

winding or modular coil determines the minimum size for a stellarator reactor because the ratio $A_\Delta \equiv R_0/\Delta$ is a constant for a given stellarator coil configuration ($A_\Delta = 6.37$ for CT6). The minimum value possible for R_0 is thus $A_\Delta d$ because d must be $\leq \Delta$. For the thicknesses of the components in Fig. 3, $d = 1.3 \text{ m} + w$, so $R_0 > 8.28 \text{ m} + 6.37w$. For six field periods, $w = 0.456(B_0 R_0 / jk)^{1/2}$ where j is in MA/m². For $k = 3$, $j = 50 \text{ MA/m}^2$, and typical values for B_0 (3–4 T) and R_0 (9–10 m), $w = 0.19\text{--}0.24 \text{ m}$, so $R_0 \approx 9.5\text{--}9.8 \text{ m}$.

Minimizing d (in particular, the radial depth of the winding pack and the thickness of the blanket and shield under the inboard half of the helical winding) and A_Δ is important in reducing the cost of torsatron reactors because the cost increases with R_0 . The smaller value of A_Δ obtained in Compact Torsatrons allows R_0 for this type of reactor to be a factor of ≈ 2 smaller than for modular stellarator reactors; typically $\approx 10 \text{ m}$ versus $\approx 20 \text{ m}$.

3. COMPARISONS WITH ARIES TOKAMAK REACTORS

3.1. Comparison of Plasma Parameters

The main plasma parameters obtained for the COE-optimized CT6 reference case are compared in Table I with those for ARIES-I', a first-stability tokamak reactor, and ARIES-IV, a second-stability tokamak reactor. The CT6 reference case is an updated version of an earlier CT6 reactor [11] based on a 0-D transport model and a much thinner blanket and shield under the inboard half of the helical windings. The ARIES-I' case is ARIES-I recalculated with the improved blanket and shield models developed in the ARIES-II and ARIES-IV studies. When ARIES-I was recalculated with the stellarator reactor code to benchmark our calculations and to

TABLE I
MAIN PLASMA PARAMETERS FOR REFERENCE CT6 TORSATRON
AND ARIES TOKAMAK REACTORS

	CT6	ARIES-I'	ARIES-IV
Major radius R_0 (m)	9.68	7.64	6.04
Average plasma radius a_p (m)	2.56	2.28	2.15
Plasma ellipticity κ	2.1	1.8	2.0
Plasma volume (m ³)	1260	710	500
Plasma current (MA)	0	10.9	6.6
Bootstrap current fraction	----	0.68	0.87
Edge safety factor q	1.05	4.5	12.2
Average electron density $\langle n \rangle$ (10 ²⁰ m ⁻³)	1.13	1.26	2.90
Average plasma temperature $\langle T \rangle$ (keV)	4.7	20	10
Central ion temperature (keV)	30.0	38.0	26.5
Volume-average toroidal beta (%)	3.2	1.9	3.4
Fraction of alpha-particle power lost	0.45	0	0.035
Fraction of plasma power radiated	0.25	0.50	0.23
Energy confinement time τ_E (s)	2.1	2.9	1.5
Plasma Q value	∞	18	30

plasma) and the cross section is much smaller (less total ampere-turns and higher average current density because of the lower magnetic field). The masses of the accompanying coil support structure and the VF coils are also much smaller for CT6. The primary reason for the lower current density in the ARIES cases is the larger amount of internal structure needed for the hoop force. The neutron wall loading is a factor of 1.6–2.1 smaller than that in the ARIES cases because of the larger CT6 wall area and the smaller required fusion power. These factors, plus the absence of disruptions, leads to lower replacement costs for the blanket. The fusion power, thermal power, and gross electric power are more for the ARIES cases because of the tokamak current drive requirements. The total mass of the CT6 fusion power core is 7,480 tonnes, versus 13,900 tonnes for ARIES-I' and 9,100 tonnes for ARIES-IV, resulting in a higher mass utilization efficiency and a lower COE for the CT6 case.

The total reactor equipment cost (whose elements are shown in Table III) for CT6 is approximately the same as that for ARIES-IV and 73% of that for ARIES-I'. The main differences are the larger cost for the CT6 blanket (because of the larger area) and the lower costs for the magnets (and the related primary support and structure) and the supplemental heating systems (only needed for plasma startup). The geometry-dependent fusion power core components (blanket and first wall, divertor, shields, magnets, vacuum vessel, and primary structure) are 71% of the \$1300-million total reactor plant equipment cost and 47% of the \$1950-million total direct cost for CT6; components that depend on the thermal and electrical power make up the remainder. The \$3650-million total capital cost for CT6 is 1.87 times the total direct cost because of financial charges that are proportional to the total direct cost. The difference in the COE for CT6 and ARIES-IV is due to the smaller blanket and first wall replacement costs for CT6 (longer lifetime because of the lower neutron flux). A large part of the 23% difference in the COE between CT6 and ARIES-I' is due to the 20% larger capital cost for ARIES-I'.

TABLE III
COSTS OF REACTOR EQUIPMENT FOR REFERENCE CT6 TORSATRON
AND ARIES TOKAMAK REACTORS

Item	Costs ^(a) (M\$)		
	CT6	ARIES-I'	ARIES-IV
First wall, blanket, and reflector	245	105	87
Shields	495	516	407
Magnets	102	437	223
Supplemental heating systems	49	155	176
Primary structure and support	13	71	37
Reactor vacuum systems	66	62	53
Power supply, switching, storage	50	50	50
Impurity control	6	12	6
ECH breakdown system	0	4	4
Total reactor equipment	1026	1411	1041

(a) In constant 1992 dollars.

4.2. Sensitivity to Confinement Improvement

Figures 9 and 10 show the variation of the plasma and device parameters with H' . Improved energy confinement is reflected most strongly in the factor of 13 increase in $\langle\beta\rangle$, from 0.4% to 5.4%, mostly due to the factor of 7.9 decrease in B_0^2 and to a lesser extent to the factor of 2 decrease in the plasma volume (from 2450 m³ to 1200 m³), as H' increases from 0.5 to 2. The density falls slightly as H' increases because it is constrained to be less than 1.5 times the value given by Eq. (4) and the ratio $B_0/R_0 a_p^2$ decreases slightly as H' increases. The temperature increases with increasing H' to keep the fusion power constant. Because the scattered alpha-particle losses also increase with T_e , the product $\langle n \rangle \langle T \rangle$ must increase slightly with H' . Although the COE is approximately constant for H' between 1 and 2, a value closer to 2 is preferred because $B_{\max}$ decreases with H' ; $B_{\max}$ is 16 T at $H' = 1$ and 11.8 T at $H' = 2$.

Fig. 9. Dependence of the reactor plasma parameters on the confinement improvement factor H' .

Fig. 10. Dependence of the reactor device parameters on the confinement improvement factor H' .

Figure 10 shows a large decrease in B_0 and a smaller decrease in R_0 as H' is increased. The decrease in R_0 is due to the decrease in w . The mass of the fusion power core (FPC) decreases continuously from 24,040 tonnes at $H' = 0.5$ to 6,980 tonnes at $H' = 2$. The rapid increase in the mass with decreasing H' below $H' = 1$ is due to larger masses for the coil systems and for the blankets and shields ($\propto R_0^2$). The helical and VF coil masses increase with decreasing H' because of both the increasing coil lengths ($\propto R_0$) and the increasing coil cross sections [area $\propto B_0 R_0 / j$, where both B_0 and R_0 increase with decreasing H' and j decreases (from 49 MA/m² at $H' = 1$ to 19 MA/m² at $H' = 0.5$)].

Larger values for the COE are obtained if the ARIES temperature profile [$T(\rho) \propto (1 - \rho^2)^{1.1}$] is chosen in a 0-D optimization rather than the self-consistent temperature profiles calculated here. However, the COE decreases to the same values for $H' > 2.5$, as shown in Fig. 11.

cates that values more than twice this value should be attainable. If experiments were to indicate that this value is not attainable, then a slightly larger reactor size and higher COE would be required, as shown in Figs. 13 and 14. For example, if operation were limited to $\langle\beta\rangle = 2\%$, then R_0 would be 1.7% larger and the COE would be 2.3% higher than the reference CT6 values. Even lowering the $\langle\beta\rangle$ limit to 1.5% would only increase R_0 by 3.4% and the COE by 4.8%. Most of the variation in $\langle\beta\rangle$ is due to B_0 : B_0^2 increases by a factor of 4.4 as $\langle\beta\rangle$ decreases from 3.2% to 0.5%. For $\langle\beta\rangle \leq 1.5\%$, $B_{\max}$ is constrained at 16 T; w must increase (and hence R_0 and the COE) as the required HF current increases to produce the increased B_0 .

Fig. 13. Dependence of the COE and plasma parameters on the imposed beta limit.

Fig. 14. Dependence of reactor parameters on the imposed beta limit.

The other parameter variations have a smaller effect. Doubling the oxygen impurity fraction to 2% only increases the COE by 0.7%. Eliminating the loss region for fast particles reduces the fraction of alpha-particle power lost from 45% for the reference case to zero and increases the helium ash density from zero for the reference case to a density fraction that depends on the helium confinement time, which can be very long. The increased helium fraction leads to a smaller fuel ion fraction n_{DT}/n_e ; the fusion power varies as $(n_{DT}/n_e)^2$. Assumed helium fractions of 5% [$(n_{DT}/n_e)^2 = 0.79$] and 10% [$(n_{DT}/n_e)^2 = 0.61$] decrease the COE by 2.1% and 1.7%, respectively; no solution could be found for a helium density fraction $\geq 15\%$ [$(n_{DT}/n_e)^2 \leq 0.45$].

5.3. Other Coil Configurations

Two other coil configurations were studied to determine the sensitivity of the COE to the coil parameters: a nine-field-period Compact Torsatron (CT9) and a twelve-field-period Compact Torsatron (CT12) based on the ATF coil set. The relevant coil parameters are $A_p = 3.77$, $A_c = 2.50$, and $A_\Delta = 6.37$ for CT6; $A_p = 4.66$, $A_c = 3.24$, and $A_\Delta = 8.60$ for CT9; and $A_p = 7.78$, $A_c = 4.49$, and $A_\Delta = 9.50$ for CT12. For the same assumptions as for the reference CT6 case, the COE for CT12 is 64 mill/kW(e)h. Although R_0 is 46% larger, the COE is only 2.5% higher than for the reference CT6 case because the area of the plasma surface from which the blanket and shield volumes are scaled is 2.7% less than for the CT6 case (due to the much lower value for A_p). For CT9, the COE is 69.1 mill/kW(e)h, 10.5% higher than for the reference CT6 case. Although the value of A_p is larger than that for CT6, it is not large enough to

material changes from SiC/TiH_{1.8}/Pb to SiC/B₄C. The thickness of the shield elsewhere increases from 0.75 m to 1.05 m and the material remains SiC. The blankets are unchanged. The net effect is to increase the COE for the reference case from 62.5 mill/kW(e)h to 72 mill/kW(e)h. The COE again varies slowly with H' : from 75.3 mill/kW(e)h at $H' = 1$ to 70.3 mill/kW(e)h at $H' = 2.1$.

6. CONCLUSIONS AND DISCUSSION

Stellarators have significant operational advantages over tokamaks as ignited steady-state reactors. This scoping study shows that a torsatron reactor could also be competitive with a tokamak reactor for a range of assumptions. The COE for the reference CT6 case, 62.5 mill/kW(e)h in constant 1992 dollars, allows relaxing different assumptions and constraints while still keeping the COE competitive. The CT6 torsatron configuration examined in this paper is not an "optimum" stellarator configuration for the reactor application; it is only one of the family of Compact Torsatron configurations that were obtained by maximizing the average radius of the last closed magnetic surface, subject to MHD constraints that maximized the $\langle\beta\rangle$ limit. While this is desirable for an experiment, it tends to minimize the critical distance Δ between the plasma edge and the center of the coil winding pack, rather than maximizing it as needed for the reactor application. The COE could be reduced significantly if the ratio R_0/Δ could be decreased.

Although results to date are encouraging, further work on optimization of the coil configuration is needed in a number of areas to define an improved reference case: (1) improved low-aspect-ratio torsatron configurations with feasible modular coils that preserve the open helical structure of torsatrons and allow more room between the plasma edge and the center of the coils, which would reduce the size (and cost) of the reactor; (2) 3-D neutronics calculations to determine the required thickness of the side shields for the inboard coils, which determines the maximum allowable elongation for the coils and hence their minimum radial depth; (3) shaping of the coil cross section, as is done in ATF and LHD, to allow more room at the sides of the coil for neutron shielding; (4) 3-D calculation of the maximum field on modular coils, which determines the maximum current density and stress in these coils; (5) optimization of the divertor geometry and its impact on the blanket design; and (6) 3-D calculations of the beta limits and ripple-induced transport for modular coil stellarators with optimized magnetic configurations and optimized plasma profiles. The emphasis on finding a modular-coil version of the torsatron is due to concerns about fabrication and repair of large helical windings. Also, more may be gained by looking at other stellarator reactor candidates.

As stellarators approach the reactor collisionality regime, ripple-induced losses of thermal particles and compensating electric field effects should become more important. The simple one-dimensional transport model used in this paper for comparison with the ARIES tokamak reactor studies can be replaced with a more refined 1-D stellarator transport model that calculates self-consistent radial profiles of the ion and electron temperatures and densities and the ambipolar radial electric field.

Further improvement in the outlook for torsatron reactors, and for stellarator reactors in general, depends on establishing the physics basis needed for reactor extrapolation and on further optimizing the stellarator coil configuration. Present stellarators [12] can develop much of

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