Overview of the U.S. Stellarator Reactor Study

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ABSTRACT

This study, which uses a cost-minimization code that incorporates the ARIES costing and reactor component models with a 1-D energy transport calculation, shows that a torsatron reactor could be competitive with a tokamak reactor.

I. 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 type of stellarator (a torsatron) used as an example in this paper 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, a near-perpendicular loss region to prevent helium ash accumulation.

Although stellarators have the potential for leading to a better reactor, they lag behind tokamaks in their development. The United States is starting a multiyear multi-institutional stellarator reactor study whose purpose is to "identify and assess the feasibility of critical issues and their consequences for development of the stellarator concept as a steady-state fusion reactor." The activities during the first year are focusing on physics optimization and selection of a stellarator coil configuration for more detailed engineering design evaluation in 1994. Four stellarator coil configurations are being studied: torsatrons with helical windings; torsatrons with modular coils; helias, a modular stellarator; and a helias-like heliac coil configuration. This paper summarizes the first phase of the work on the systems optimization studies.

II. APPROACH AND ASSUMPTIONS

The recent extensive Advanced Reactor Innovation and Evaluation Studies (ARIES) [1-3] have explored improving the attractiveness of tokamak reactors. Our study applies the

ARIES costing and component assumptions to optimization of stellarator reactors, which allows more accurate comparison of different stellarator and tokamak reactor configurations. The models for the detailed component geometry and reactor performance constraints are specialized in this paper for the example of a particular torsatron reactor.

A. Reference Stellarator Configuration

The reference stellarator configuration chosen for these studies is a Compact Torsatron [4] with six toroidal field periods (CT6). It is not an "optimum" stellarator configuration, only one of a family of torsatrons obtained by maximizing the average radius of the last closed magnetic surface subject to constraints that maximized the beta limit. A torsatron is a stellarator with two continuous helical windings having currents in the same direction and no separate toroidal field (TF) coils. The helical winding geometry and the last closed flux surface for the CT6 configuration are shown in Fig. 1. The two helical windings are characterized by a coil aspect ratio A_c

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

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$\equiv R_0/a_c = 2.5$, where R_0 and a_c are the major radius and minor radius of the helical winding. The last closed magnetic surface is characterized by an average plasma aspect ratio $A_p \equiv R_0/a_p = 3.87$ and ellipticity $\kappa = 2.1$. Here a_p is the average radius of the noncircular (and nonaxisymmetric) last closed surface. Although helical windings are used in this paper for simplicity, these coils can be modularized to have one (or two) coils per toroidal field period (for example, the "symmetron") or five to ten coils per field period (as in a modular stellarator).

B. Reactor Component Assumptions

The study assumes the same level of technology development ("achievable in 20 years") as in the ARIES-IV tokamak reactor study [3]. The compositions, average mass densities, and unit costs for the first-wall/blanket assembly, the shield, and the superconducting magnets are the same but there is no blanket directly under the helical windings on the inboard side of the torus in order to reduce R_0 . 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 to 80 cm (vs 35 cm on the inboard side and 60 cm on the outboard side for ARIES-IV). The blanket and shield components were chosen for their low activation properties. The shield has lead, $\text{TiH}_{1.8}$, and SiC components and the blanket module consists of Li_2O , SiC , and Be with helium cooling. The $\text{TiH}_{1.8}$ shield makes the shielding effectiveness of the ceramic shield comparable to that of a metallic shield. 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 Nb_3Sn superconducting winding pack is assumed to have a transverse (mainly toroidal) elongation $k = h/w$, where h and w are half the transverse width and half the radial depth of the helical winding pack, respectively.

C. Transport Assumptions

Different scalings for the energy confinement time τ_E fit present stellarator data. We choose the Lackner-Gottardi scaling [5],

$$\tau_E^{\text{LG}} = 0.17 R_0 a_p^2 n^{0.6} B_0^{0.8} P^{-0.6} \tau^{0.4}, \quad (1)$$

which fits both tokamak and stellarator data, with a confinement improvement factor H' similar to that in tokamaks. Here B_0 is the on-axis field, n is the line-averaged electron density (in 10^{20} m^{-3}), P is the absorbed heating power (in MW), and $\tau (= 1/q)$, where q is the tokamak safety factor) is the rotational transform. All other quantities are in SI units. The maximum density in stellarators is not determined by a disruption limit as in tokamaks. We assume that the line-average density is constrained to <1.5 times the value proposed by Sudo [6],

$$n_{\text{max}} = 0.25 (P B_0 / R_0 a_p^2)^{1/2}. \quad (2)$$

Transport model. One-dimensional heat transport equations are solved for the ion and electron temperatures $T_i(\rho)$ and $T_e(\rho)$ with a heat diffusivity $\chi(\rho) = \chi_0/(1 - \alpha\rho^2)$. The constants χ_0

and α are chosen to give an energy confinement time $\tau_E = H' \tau_E^{\text{LG}}$; the reference values are $H' = 2$ and $\alpha = 0.9$, for a factor of 10 increase 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, $n = n_0[(0.46)(1 - \rho^2) + 0.54]$.

Alpha-particle losses. Because the relatively large helical ripple in torsatrons can lead to a near-perpendicular loss region for energetic particles, we assume that all 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 $\sim 40\%$. 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 $W \approx 30T_e \sim 0.3\text{--}1 \text{ MeV}$, below which they rapidly scatter into the loss region [7]. The loss of alpha-particle heating is offset by increased fusion power because the loss region prevents accumulation of helium ash and the dilution of the deuterium-tritium (D-T) fuel. If $n_{\text{He}}/n_e \sim 10\%$ with no loss region and $n_{\text{O}}/n_e = 1\%$, $n_{\text{DT}}/n_e = 0.72$ with no loss region and 0.92 with a 40% loss region, so $P_{\text{fusion}}(\text{no loss region})/P_{\text{fusion}}(\text{with loss region}) = 0.61$. In addition, the loss region allows a factor of 1.28 higher beta value for the fuel ions and reduces the bremsstrahlung radiation losses by 11%. Usually the positive and negative effects balance each other.

D. Reactor Optimization

The quantity minimized in the reactor optimization is the projected cost of electricity (COE). The masses of the reactor components are used to calculate the cost of the reactor core, using the cost models established for the ARIES studies [2,3]. As in the ARIES studies, the costs assume "learning curve" credits of $\sim 50\%$ associated with a "tenth-of-a-kind" reactor.

The optimization variables are those related to the device size (R_0), the plasma parameters (volume-averaged density $\langle n \rangle$, density-averaged temperature $\langle T \rangle$), and the magnetic field (B_0 , h , and w). The reference constraints usually chosen for the optimization are net electric power $P_E = 1 \text{ GW}$, volume-average beta $\langle \beta \rangle \leq 5\%$, $n \leq 1.5n_{\text{max}}$, and a set of helical winding constraints ($k \leq k_{\text{max}} = 3$, the maximum magnetic field on the superconductor $B_{\text{max}} \leq 16 \text{ T}$, and limits on the current density in the winding pack, $j \leq j_{\text{max}} = 50 \text{ MA}\cdot\text{m}^{-2}$). However, the optimum values of $\langle \beta \rangle$, B_{max} , and j are not necessarily these upper limiting values. The optimization always leads to $k = k_{\text{max}}$ (which reduces w , and hence R_0 and the COE) and to $n = 1.5n_{\text{max}}$.

The COE is minimized when R_0 is minimized because the cost of the most expensive components, the constant-thickness blankets and shields, varies as R_0^2 . The minimum value for R_0 is set by the need for a certain distance between the plasma edge and the center of the helical winding for half the

thickness of the coil winding pack, w , plus the total thickness d of the plasma-wall separation, the first wall, the blanket, the shield, the vacuum vessel wall, and the cryostat. The space between the plasma edge and the helical winding on the inboard half of the torus is the determining distance. The minimum distance Δ between the edge of the plasma (the last closed flux surface) and the centerline of the helical winding determines the minimum size for a stellarator reactor because the ratio $A_\Delta \equiv R_0/\Delta$ is a constant for a given coil configuration ($A_\Delta = 6.37$ for CT6). The minimum value possible for R_0 is thus $A_\Delta(d + w)$ because Δ must be $\geq d + w$.

III. COMPARISONS WITH ARIES REACTORS

The plasma parameters obtained for the COE-optimized CT6 reference case are compared in Table I with those for the ARIES-I' and ARIES-IV [3] tokamak reactors. The larger major radius and plasma aspect ratio for CT6 lead to a plasma volume 2–2.8 times that of ARIES-I' and ARIES-IV, respectively, which allows a lower value of the density-averaged temperature $\langle T \rangle$ for the same power output; $\langle T \rangle$ is only 0.39–0.77 that for the ARIES cases. The larger ratio of central ion temperature to $\langle T \rangle$ is due to the fact that the temperature profile calculated for CT6 is more peaked than that assumed for the ARIES cases. The most noticeable difference is in the fraction of alpha-particle power lost and the plasma Q value. In most respects, the CT6 case is closer to ARIES-IV than to ARIES-I'.

Table II gives the main device parameters for CT6 and the ARIES-I' and ARIES-IV reactors. The field on axis for CT6 is one-half to two-thirds of that in the ARIES cases, but the maximum field on the helical windings is similar to that on the TF coils in ARIES-IV, as is the total stored magnetic energy and the mass of the VF coils. ARIES-I' has much larger values for these quantities. The mass of the CT6 helical winding is a factor of 1.6–3.1 lower than the mass of the ARIES TF coils because the CT6 coil perimeter is smaller (the winding is closer to the plasma) and the cross section is much smaller (similar or less total ampere-turns and higher average current density because of the lower magnetic field).

Table I
Main plasma parameters for 1-GW(e) CT6 torsatron and ARIES tokamak reactors

	CT6	ARIES I'	ARIES IV
Major radius R_0 (m)	10.03	7.64	6.04
Ave. plasma radius a_p (m)	2.66	2.28	2.15
Plasma volume (m^3)	1400	710	500
Plasma current (MA)	0	10.9	6.6
Ave. density $\langle n \rangle$ ($10^{20} m^{-3}$)	1.30	1.26	2.90
Ave. temperature $\langle T \rangle$ (keV)	7.7	20	10
Central T_i (keV)	24.6	38.0	26.5
Vol.-ave. toroidal beta (%)	3.1	1.9	3.4
Alpha-particle power lost	42%	0	3.5%
Plasma Q value	∞	18	30

Table II
Main device parameters for 1-GW(e) CT6 torsatron and ARIES tokamak reactors

	CT6	ARIES I'	ARIES IV
Toroidal field on axis (T)	5.0	10.6	7.6
$B_{\max}$ on coils (T)	16.0	19.1	15.9
Current density j ($MA \cdot m^{-2}$)	46.4	24.5	30.0
Mass of HF/TF coils (t)	1330	4180	2130
Mass of VF coils (t)	630	930	610
Current-drive power (MW)	0	115	68
Fusion power (MW)	1760	2040	2020
Neut. wall power ($MW \cdot m^{-2}$)	1.18	2.06	2.67
Recirculating power fraction	0.09	0.21	0.19
Power density [$kW(e)/tonne$]	113	72	111
COE [$mill/kW(e)h$] ^(a)	65.6	76.6	67.7

(a) In constant 1992 dollars

The neutron wall loading is a factor of 1.7–2.3 smaller than that in the ARIES cases because of the larger CT6 wall area and the smaller required fusion power. The fusion power, thermal power, gross electric power, and recirculating power fraction are more for the ARIES cases because of the tokamak's current drive requirement. The total mass of the CT6 fusion power core is 8,870 tonnes vs 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 difference in the COE for CT6 and ARIES-IV is due to the smaller blanket and first wall replacement costs for CT6. A large part of the 17% difference in the COE between CT6 and ARIES-I' is due to the 14% larger capital cost for ARIES-I'.

The total reactor equipment cost for CT6 is 9% more than that for ARIES-IV and 19% less than that for ARIES-I'. The main differences are the larger cost for the CT6 blanket and the lower costs for the magnets and the supplemental heating systems. The geometry-dependent fusion power core components are 74% of the \$1410-million total reactor plant equipment cost and 50% of the \$2070-million total direct cost for CT6. The total capital cost for CT6 is \$3860 million.

IV. SENSITIVITY TO TRANSPORT ASSUMPTIONS

Fig. 2 shows the radial profiles obtained for the electron and ion temperatures and the assumed (ARIES) density profile for the optimized CT6 reference case. The broad density profiles seen in stellarators lead to peaked temperature profiles. However, the pressure profile is relatively insensitive to variations in the density profile and there is only a 3% variation in the COE for parabolic density profiles as the ratio of the edge density to the central density is varied from 0.1 to 1. There is also only a 3% variation in the COE as the coefficient α in $\chi(\rho)$ is varied from $\alpha = 0.67$ [$\chi(1)/\chi(0) = 3$] to $\alpha = 0.967$ [$\chi(1)/\chi(0) = 30$].

Fig. 3 shows the variation of the plasma and device parameters with H' . Improved energy confinement is reflected most

Fig. 2 Electron and ion temperature and density profiles for the CT6 reference case.

Fig. 3 Dependence of the plasma and device parameters on the confinement improvement factor H' .

strongly in the factor of 6 increase in $\langle\beta\rangle$, from 0.76% to 4.6%, mostly due to the factor of 3.9 decrease in B_0^2 and to a lesser extent to the factor of 2 decrease in the plasma volume (from 2580 m³ to 1310 m³), as H' increases from 1 to 2.5. The temperature must increase with increasing H' to offset the decrease in the plasma volume and the increase in the scattered alpha-particle losses, which increase with T_e . The ratio $T_i(0)/\langle T \rangle$ is approximately independent of H' , indicating that the ion temperature profile does not change significantly with H' . The decrease in R_0 is due to the decrease in w . The mass of the fusion power core (FPC) decreases continuously from 26,480 tonnes at $H' = 1$ to 7,970 tonnes at $H' = 2.5$.

These results indicate that only a modest improvement in confinement is needed for an attractive torsatron reactor. We choose $H' = 2$ as a conservative target for stellarator confinement improvement. There is already some evidence for confinement improvement in stellarators due to H-mode-like operation and beta self-stabilization effects.

V. EFFECTS OF OTHER ASSUMPTIONS

The COE is relatively insensitive to the confinement improvement factor (and other parameter assumptions) because R_0 only changes significantly when there is a relatively large change in the value of w ; $R_0 = A_\Delta(d + w)$ where $A_\Delta = 6.37$ and $w \propto (B_0 R_0 / jk)^{1/2}$. For $k = 3$, $j = 50 \text{ MA}\cdot\text{m}^{-2}$, and typical values for B_0 (5 T) and R_0 (10 m), $w = 0.26 \text{ m}$, which is much less than the $d = 1.3\text{-m}$ thickness of the fixed components. The only way to reduce R_0 (and the COE) is by decreasing A_Δ (a different coil configuration), d (a thinner blanket and shield), or w (lower B_0 , which reduces τ_E and increases β , or higher j and k , which are subject to other constraints).

A. Coil Assumptions

The most critical coil parameters (j_{max} , k_{max} , and B_{max}) were varied over representative ranges to test the sensitivity of the reactor parameters to these choices. Increasing any of these decreases w and hence R_0 ($\Delta R_0 = A_\Delta \Delta w$). Decreasing the current density from $50 \text{ MA}\cdot\text{m}^{-2}$ to $30 \text{ MA}\cdot\text{m}^{-2}$ increased R_0 by 0.5 m and the COE by 5 mill/kW(e)h. The COE is more sensitive to changes in the maximum transverse coil elongation; the COE is 81 mill/kW(e)h for $k_{\text{max}} = 1$ (where $B_{\text{max}} = 16 \text{ T}$ and $R_0 = 12.3 \text{ m}$) and 63.4 mill/kW(e)h for $k_{\text{max}} = 4.5$ (where $B_{\text{max}} = 14.7 \text{ T}$ and $R_0 = 9.6 \text{ m}$). We choose $k_{\text{max}} = 3$ (vs 2.6 for ARIES-I) as a compromise; higher values of k_{max} yield a smaller coil depth that leads to a smaller R_0 and a lower COE, while lower values yield more room for blankets between the helical windings on the inboard side.

Lower values for B_{max} are possible with an increased COE as shown in Fig. 4. The COE increases with decreasing B_{max} because R_0 increases from 10 m at $B_{\text{max}} = 16 \text{ T}$ to 12.8 m at $B_{\text{max}} = 8 \text{ T}$. The value for j has the largest variation over this range (from $46.4 \text{ MA}\cdot\text{m}^{-2}$ to $7.8 \text{ MA}\cdot\text{m}^{-2}$) because the value of B_{max} depends on the perimeter of the helical winding cross section; lower B_{max} leads to a higher value for w (and R_0) and hence to a lower j for a given B_0 ($j \propto 1/w^2$). The values of the

Fig. 4 Variation of reactor parameters with the maximum field on the superconducting winding pack.

other parameters are relatively independent of the limiting value for $B_{\max}$; B_0 and $\langle\beta\rangle$ change by $<11\%$ as $B_{\max}$ varies from 8 T to 16 T.

B. Other Parameter Assumptions

The physics parameters were varied to test the sensitivity to these choices. The value of $\langle\beta\rangle$ obtained for the reference case is a modest 3.1%. Values of $\langle\beta\rangle$ of 2.1% have been obtained and theory indicates that values more than twice this value should be attainable. If operation were limited to $\langle\beta\rangle = 2\%$, then R_0 would be 5% larger and the COE would be 8.4% higher than the reference CT6 values, as shown in Fig. 5. Even lowering the $\langle\beta\rangle$ limit to 1.5% would only increase R_0 by 9.1% and the COE by 17%. Most of the variation in $\langle\beta\rangle$ is due to B_0 : B_0^2 increases by a factor of 3 as $\langle\beta\rangle$ decreases from 3.1% to 0.7%. The plasma volume increases (from 1400 m³ at $\langle\beta\rangle = 3.1\%$ to 2720 m³ at $\langle\beta\rangle = 0.7\%$), so $\langle T\rangle$ decreases to give the same fusion power; the plasma density is approximately constant. The value of $B_{\max}$ is constrained at 16 T; w must increase (and R_0 and the COE) as the required HF current increases to produce the increased B_0 . The optimum values (for a minimum COE) for $\langle\beta\rangle$ and H' are related and are not necessarily the limiting values; here $\langle\beta\rangle \propto (H')^2$.

Eliminating the loss region reduces the fraction of alpha-particle power lost from 42% to zero but increases the helium ash density, which leads to a smaller fuel ion fraction $n_D T/n_e$; the fusion power varies as $(n_D T/n_e)^2$. Assumed helium fractions of 5% [$(n_D T/n_e)^2 = 0.79$] and 10% [$(n_D T/n_e)^2 = 0.61$] decrease the COE by 3.5% and 2.9%, respectively.

The other parameter variations have a smaller effect. Doubling the oxygen impurity fraction to 2% only increases the COE by 4.6%. Removing the density limit also has a modest effect; although $\langle n\rangle$ increases from $1.3 \times 10^{20} \text{ m}^{-3}$ to $2.4 \times 10^{20} \text{ m}^{-3}$ and $\langle T\rangle$ decreases from 7.7 keV to 4 keV, the COE only decreases by 4%.

Fig. 5 Dependence of the plasma and device parameters on the imposed beta limit.

C. Other Coil Configurations

Two other coil configurations were studied to determine the sensitivity of the COE to the coil parameters: Compact Torsatrons with nine field periods (CT9) and twelve field periods (CT12). For the same assumptions as for the reference CT6 case, the COE for CT12 is 66.8 mill/kW(e)h. Although R_0 is 44% larger, the COE is only 1.8% higher than for the reference CT6 case because the area of the plasma surface from which the blanket and shield volumes are scaled is 4.6% less than for the CT6 case (A_p is 7.78 for CT12 vs 3.77 for CT6). For CT9, the COE is 72.3 mill/kW(e)h, 10.2% higher than for the reference CT6 case. Although the value of A_p (= 4.66) is larger than that for CT6, it is not large enough to compensate for the 32% larger value of R_0 ; the plasma surface area is 35% larger than that for the CT6 reference case.

Alternative coil configurations could lead to significantly lower values for the COE. Fig. 6 shows the reduction in R_0 and the COE that could be obtained if the plasma-coil separation ratio A_Δ could be reduced for a fixed coil aspect ratio A_c . This is done in this calculation by placing the edge of the

Fig. 6 Dependence of the COE and the plasma aspect ratio on the plasma-wall separation ratio.

plasma at each of successively smaller flux surfaces, so that A_p increases as A_Δ decreases. However, this calculation avoids the issue of the diverted flux layer outside the last closed flux surface, which would have to be addressed.

D. Scaling with Electric Power Output

Fig. 7 shows the variation of the reactor parameters as P_E is increased from 0.6 GW(e) to 2 GW(e). Increasing the electric power output from 1 GW(e) to 1.5 GW(e) and 2 GW(e) would reduce the COE by 22% and 31%, respectively. Even the relatively modest increase in P_E from 1 GW(e) to 1.2 GW(e) would reduce the COE by 11%. The mass of the fusion power core is approximately independent of P_E because the reactor size does not change with P_E ; R_0 is 10.2 m for $P_E = 0.6$ GW(e) and 9.9 m for $P_E = 2$ GW(e). Essentially all the de-

Fig. 7 Dependence of the COE and other reactor parameters on the net electric power output.

crease in the COE with increasing P_E is due to the fact that the COE is inversely proportional to P_E . The additional power results from an increase in $\langle n \rangle$ [from $1.0 \times 10^{20} \text{ m}^{-3}$ for $P_E = 0.6$ GW(e) to $1.8 \times 10^{20} \text{ m}^{-3}$ for $P_E = 2$ GW(e)] and in $\langle T \rangle$ [from 7.3 keV for $P_E = 0.6$ GW(e) to 8 keV for $P_E = 2$ GW(e)]. This leads to the nearly linear increase in $\langle \beta \rangle$ with P_E seen in fig. 16; B_0 only decreases by 13% over this range.

VI. CONCLUSIONS AND DISCUSSION

This study shows that a torsatron reactor could be competitive with a tokamak reactor for a range of assumptions. The COE for the 1-GW(e) CT6 reference case, 65.6 mill/kW(e)h in constant 1992 dollars, allows relaxing different assumptions and constraints while still keeping the COE competitive. The COE is relatively insensitive (<10% variation) over a wide range of assumptions including variations in B_{max} , the shape of the density profile, the beta limit, the confinement multiplier, and the presence of a large loss region for alpha particles. The optimum values for $\langle \beta \rangle$ and H' are related and are not necessarily the limiting values; moderate values are sufficient for a competitive COE: $\langle \beta \rangle \geq 2\%$ and $H' \geq 1.5$. A competitive COE is obtained for $B_{\text{max}} \geq 10$ T. The largest variations in the COE occur for variations in the electrical power output and the plasma-coil separation ratio.

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