CONF-940630--12

11th TOPICAL MEETING ON THE TECHNOLOGY OF FUSION ENERGY
June 19-23, 1994

PROSPECTS FOR TOROIDAL FUSION REACTORS*

John Sheffield, Fusion Energy Division
John Galambos, Computing Applications Division
Oak Ridge National Laboratory

*Research sponsored by the Office of Fusion Energy, Department of Energy, under contract DE-ACO5-84OR21400 with the Martin Marietta Energy Systems, Inc.

PROSPECTS FOR TOROIDAL FUSION REACTORS

John Sheffield
Oak Ridge National Laboratory
P. O. Box 2009
Oak Ridge, Tennessee 37831-8070
(615)574-0988

John D. Galambos
Oak Ridge National Laboratory
P. O. Box 2009
Oak Ridge, Tennessee 37831-8218
(615)576-5482

ABSTRACT

Work on the International Thermonuclear Experimental Reactor (ITER) tokamak has refined understanding of the realities of a deuterium-tritium (D-T) burning magnetic fusion reactor. An ITER-like tokamak reactor using ITER costs and performance would lead to a cost of electricity (COE) of about 130 mills/kWh. Advanced tokamak physics to be tested in the Toroidal Physics Experiment (TPX), coupled with moderate components in engineering, technology, and unit costs, should lead to a COE comparable with best existing fission systems around 60 mills/kWh. However, a larger unit size, ~2000 $MW_{(e)}$, is favored for the fusion system. Alternative toroidal configurations to the conventional tokamak, such as the stellarator, reversed-field pinch, and field-reversed configuration, offer some potential advantage, but are less well developed, and have their own challenges.

I. INTRODUCTION

The work on ITER, in both the Conceptual Design Activity (CDA)[1] and, more recently, in the Engineering Design Activity,[2] has refined our understanding of the engineering and costing realities of a magnetic fusion reactor burning D-T. In this paper, we look first at the implications for tokamak reactor economics of using ITER-based technology, engineering, and costs. The costing model is that derived from generic reactor studies[3] and incorporated in the SUPERCODE.[4] In a second step, we consider the cost improvements that would come from incorporating advanced tokamak features,[5] which will be tested in TPX.[6] In addition, we allow for moderate improvement in unit costs and in technology, as discussed in the Advanced Reactor Innovation and Evaluation Studies (ARIES).[7] Third, we consider the "ultimate" cost of a D-T burning fusion reactor in which only the minimum of essential components (blanket, shield, tritium plant, heat removal, electricity production, etc.) is retained. Fourth, we show how fusion COEs might compare with fission COEs.[8,9] Finally, we comment on some alternative configurations to the tokamak that might offer advantages, albeit with different problems to overcome.

II. ITER AND TPX

The primary characteristics of ITER[2] and TPX[6] are given in Table 1. Important points to note are that while the nominal ITER fusion power is 1500 MW, it is capable of producing, including the exothermic blanket gain, more than 5000 MWth near the beta limit of $\beta_N \sim 3.5\%$ m-T/MA. Second, ITER can adopt many of the advanced operating scenarios to be tested in TPX, including steady-state operation with high bootstrap current, though it cannot take advantage of a higher plasma ellipticity.

Table 1. Primary characteristics of ITER and TPX

	ITER	TPX
Major radius (m)	8.10	2.25
Minor radius (m)	3.00	0.50
Elongation	1.6	2.0
Toroidal field (T)	5.7	4.0
Plasma current (MA)	24.0	2.0
Pulse length (s)	≳ 1000 (∞)	1000 (∞)
Fuel	D-T	D-D
Breeding blanket	Phase 2	No
Nominal fusion power (MW)	1500	

A principal need for high performance, true for any configuration, is to develop modes of operation and divertor systems that will lead to low helium and impurity levels in the plasma. A key driver of the present ITER design is the requirement to deal with a low rate of helium loss and 15 to 20% helium contamination of the plasma. In the worst case the fuel fraction is $n_{DT}/n_e \sim 0.5$, and for an attractive reactor it would be desirable to achieve $n_{DT}/n_e \sim 0.8$ to 0.9. The goal of TPX is to demonstrate improved operation with higher ellipticity than ITER, higher beta and confinement, and lower current in steady-state, while minimizing impurity contamination of the plasma. For the advanced tokamak reactor it is assumed that improved performance at high bootstrap current fraction ≳ 90%, low helium and impurity levels, and with optimum plasma shape and beta, can be developed.

III. COSTING

The COE is determined from the formula

$$COE \cong \frac{\text{Fixed charge rate x Capital cost + O\&M + Fuel}}{\text{Availability x Hours in a year x Net electric power}} + \text{Decommissioning.}^{3}$$

The fixed charge rate is the annual repayment (mortgage cost) on the money borrowed during construction. In constant dollars, the rate is ≃ 0.1. The capital cost includes interest charges during construction. Operations and maintenance (O&M) costs are expected to be similar to those of fission plants, though a different mix of skills is required. Fuel costs include deuterium, plus the annualized cost of the lithium breeding blankets used during the life of the plant. Decommissioning costs of 0.5 mill/kWh are assessed, similar to the practice for fission systems.

The availability is the most uncertain quantity; because of the lack of data on component and system reliability and maintenance requirements, a goal of 0.75 is set. The net electric power is given by

$$P_e = [0.14\,P_f + 0.8(1 + g_n)P_f]\,\eta_e - P_{Bop} - P_{aux} \quad (2).$$

P_f is the fusion power, g_n is the exothermic blanket gain, and η_e is the effective thermo-electric conversion efficiency. P_{Bop} is the power used in the balance of plant. P_{aux} is the auxiliary power (MW_e) used to sustain the plasma configuration and run the fusion reactor.

A self-consistent reactor design is obtained using the SUPERCODE.[4] The SUPERCODE systems code includes tokamak physics and engineering models, coupled through an optimization driver. In these calculations we utilize global plasma physics modeling typical of reactor studies and engineering/costing analyses that were developed to model the ITER-CDA device. We have also incorporated standard power reactor models,[3] and for all cases here, use the minimum COE as the optimization figure-of-merit. Table 2 lists some primary reactor modeling assumptions. The cost models are different from those used in the ARIES study,[10] because we normalize our cost scalings with the ITER-CDA design. For all cases shown, we employ global, volume-averaged transport models with profiles adjusted to match parabolic shapes for temperature and density. One area of our modeling that does differ from the usual reactor models is the incorporation of a fixed-boundary magnetohydrodynamic (MHD) equilibrium calculation that provides the relationship between the plasma current, the current profile, and the plasma geometry. The physics modeling includes constraints for impurity levels, power balance, beta limit, MHD requirements, current-drive, alpha particle confinement, and inductive volt-seconds. The engineering models include constraints for toroidal field (TF) coils, poloidal field coils, TF coil ripple, shielding, divertor build, injection power, and neutron wall loading.[11] The primary difference in these modeling assumptions, is the use of a lower helium ash concentration (5%) and slightly higher elongation (k = 2 at the 95% surface) for advances beyond the basic ITER layout. The parameters of reference reactor designs are given in Table 3.

Table 2. Assumptions used in the plant power balance, and COE calculation.

Plant Power Balance	
Thermal to electric efficiency	0.45[a]
Percentage of plasma thermal power converted to electricity	70%
Blanket energy gain	0.30
Current drive power efficiency, wall plug to plasma efficiency	72%
Costing assumptions	
Construction time (year)	6
Plant life (year)	30
Average capacity factor	75%
Indirect + contingency cost factor	46%[b]
Fixed charge rate[c] (F_{CRO})	0.0966
Effective cost of money (year^{-1})[d]	0.1135
Inflation rate (year^{-1})	0.05
Direct cost	10th of-a-kind[e]

[a]Assumes a high-temperature helium cooling system.
[b]Taken from the ITER CDA for indirect + contingency cost.
[c]Constant dollar.
[d]We input the fixed charge rate (FCRO) independent from this value. The cost of money is used only for estimating the capitalization factor (f_{cap0}).
[e]A 20% cost reduction is applied to the tokamak reactor plant equipment (corresponding to a 94% learning ratio for each doubling of the number of units).

Table 3. ITER-like reactor parameters

	ITER - like physics $\beta_N < 3.5$ κ=1.6		Advanced physics $\beta_N < 6$ κ= 2.0 90% BS fraction
	1200 MW(e)	1800 MW(e)	2000 MW(e)
COE (mills/kWh)	132	102	63
Overnight capital cost (1993 billion$)	8.21	9.41	5.95
$f_{recirculate}$ (%)	28.5	25.6	11.7
Core mass (ktonnes)	43.9	49.1	24.4
MPD (kW_e/tonne)	27.3	36.7	79.6
Major radius (m)	8.1	8.1	6.50
Aspect ratio	2.55	2.39	1.95
Plasma current (MA)	22.0	26.3	13.3
Field on axis (T)	4.83	4.83	4.53
$B_{max\text{-}TF\ coil}$ (T)	11.4	12.3	11.3
q_{95}	3.0	3.0	4.0
Fusion power (MW)	2960	4290	4120
Injection power (MW)	216	279	32
Bootstrap fraction	0.37	0.36	0.90
Plasma energy gain, *Q*	13.7	15.4	1.31
ITER-89 P *H* factor	1.92	1.76	2.22
Total beta (%)	5.10	5.60	9.01
Neutron wall load (MW/m^2)	1.72	2.32	3.48

IV. COMPARISON OF FUSION AND FISSION COE

Analysis of the performance and costs of fission reactors is made regularly. The present best experience for ~1200 MW(e) pressurized-water reactors (PWRs) has been assessed by Delene,[8] and projections of best future performance for ~600 MW(e) reactors have been made by the U.S. Council for Energy Awareness.[9] A comparison of projected fusion costs with the fission experience and projections is given in Table 4.

The ITER-1200 is a 1200-MW(e) fusion reactor based directly on an ITER-like design with an intermediate fusion power run at the beta limit of β_N = 3.5% m-T/MA. The ITER-1800 is a similar design run at the beta limit. Increased costs for handling the higher power and more frequent component replacement are included. The ITER-2000 Advanced is a design made smaller through the use of advanced tokamak features and incorporating 20% lower unit costs. The fusion base reactor is a D-T system at 2000 MW(e) stripped to the minimum of components, that is, no magnets, and ignited.

It can be seen that advanced tokamak reactors have the potential to be competitive, though further gains will be needed for them to compete with the projected best fission systems. However, they would be of larger unit size than the fission systems. It should also be noted that there is a wide range of COE from existing U.S. fission reactors 50 to >100 mills/kWh, so there is no absolute guide on acceptable cost. Further, this analysis takes no account of the potential advantage of fusion reactors in safety and through the use of low activation materials leading to a much lower radioactive lifetime and waste impact.

Table 4. Comparison of fission and fusion

FY 1993 $ Fixed charge rate = 0.097. F_{AV} = mills/kWh

	Capital	O&M	Fuel + Decom.	Total
ITER-1200[a]	116	9.5	6.5	132
ITER-1800[a]	88	8	6	102
ITER-2000 Advanced[a,b,c]	50.5	7.5	5	63
PWR-best experience[d]	42.5	10	8	60.5
Fission-projected best[e]	28.5	9.5	8	46
Fusion base 2000[a,f]	≲28	≲7	~5	≲40

[a] 6 years construction time - basic ITER (K = 1.6 etc.), 10th of a kind costs.

[b] H-mode factor 2, Troyon factor 6, 5% helium, 90% bootstrap current, 20% cost reduction for fusion-related items.

[c] COE scales roughly as $P_{net}^{-0.5}$

[d] J. Delene, 1990, 1200-MW(e), 8-year construction lead time.

[e] U.S. Council for Energy Awareness, 1992, 600-MW(e), 5-year construction time, optimistic assumptions, and cheap uranium.

[f] Fusion base means all components except magnet systems.

V. ALTERNATIVE TOROIDAL CONFIGURATIONS

Important issues for a tokamak are demonstration of plasma disruption control to simplify the engineering and operation at moderately high-power density with a low level of power recirculated to the plasma. Alternative configurations, such as the very low aspect ratio tokamak (spherical torus), the stellarator, reversed-field pinch, and field-reversed configuration, are other interesting but less-developed routes to the realization of an alternative reactor.[12] Each features some improvements over the tokamak, but involves other issues in terms of reactor viability. Some key factors are listed in Table 5. The stellarator is disruption free and is the only inherently steady-state configuration, but must demonstrate reactor-relevant confinement at high temperatures and beta in an engineeringly acceptable coil configuration. The spherical torus and reversed-field pinch have high enough beta to permit the use of modest scale copper toroidal coils. They do not disrupt, in experiments, and would operate at higher power density. The spherical torus has good access for maintenance, but must demonstrate a plasma performance that leads to acceptable levels of recirculating power to the coils and plasma current drive. The reversed-field pinch must confirm the favorable confinement scaling at higher current and demonstrate an efficient current-drive scheme. The field-reversed configuration has the simplest coil configuration and the highest beta and power density. Its key issues are to demonstrate reactor-relevant confinement scaling and an efficient current-drive scheme. For all configurations, the development of a divertor system for impurity control and helium removal is crucial.

VI. CONCLUSIONS

While the ITER experiment is large compared to the core of a fission reactor, it is on the path to a potentially competitive fusion reactor. The scale and cost of ITER are set in part by a conservative approach to making the first integrated demonstration of controlled fusion; and, in part, they reflect the present development of physics, technology, and engineering. Moderate improvements, expected to accrue from future developments, should lead to a more cost-effective fusion reactor. However, to be competitive with fission systems the unit size of the reactor will be greater ~2000 $MW_{(e)}$, rather than $\lesssim$1000 $MW_{(e)}$ for fission. Alternative toroidal configurations may offer further improvements. All of these fusion reactors have potential advantages in regard to safety, and all lower radioactive lifetime and waste impact. For all fusion systems, the development of a data base supporting the required availability is crucial. ITER and TPX will be important facilities in establishing this information.

ACKNOWLEDGMENTS

Research sponsored by the Office of Fusion Energy, Department of Energy, under contract DE-AC05-84OR21400 with the Martin Marietta Energy Systems, Inc.

Table 5. Reactor features

Area	Advanced tokamak	Advanced stellarator	Reversed-field pinch	Field-reversed configuration
Transport	OK $H \gtrsim 2$	OK at large size Need E-field in torsatron	Connor-Taylor or Carreras-Diamond ? Scaling	Need 10x better than present scaling
Helium removal	H ↑ ?	E-field effects? Direct losses	?	OK?
Divertor	ITER-like large radial distance	ITER-like? Needed?	?	Axial good
Steady State $R/a \gtrsim 3$ $R/a \lesssim 1.8$	Efficiency? Disruption? Efficiency?	Good	F-θ pumping?	Proton-driven in D-^3He?
Beta	$\lesssim$ 10%, OK, $R/a \geq 3$ Large for $R/a \lesssim 1.8$	$\lesssim$ 5 % OK Experiments needed	$\beta\theta \cong 0.2$	Good Tilt mode?

- Generic Needs
  - — A way to enhance helium losses
  - — Low after-heat, low activation materials

REFERENCES

1. *ITER Concept Definition*, IAEA, Vienna, IAEA/ITER/DS/3, Vol. 1 and 2 (1989).

2. P-H Rebut, "Overview of the ITER Engineering Design Activities," to appear in *Proceedings of the 15th Symposium on Fusion Engineering, Hyannis, Massachusetts, October 11-15, 1993.*

3. J. Sheffield, R. A. Dory, S. M. Cohn, J. G. Delene, L. Parsley, D.E.T.F. Ashby, and W. Reirsen, *Fusion Technology*, **9**, 199 (1986).

4. S. W. Haney et al., *Fusion Technology*, **21**, 1749 (1992).

5. G. L. Jackson et al., *Physics Review Letters*, **67**, 3098 (1991).

6. J. A. Schmidt, K. I. Thomassen, R. G. Goldston, G. H. Neilson, W. M. Nevins, J. C. Sinnis et al., *Journal of Fusion Energy*, **12**, 215 (1993).

7. F. R. Najmabadi, R. W. Conn, and the ARIES team, *Fusion Technology*, **19**, 783 (1991).

8. J. G. Delene, *Fusion Technology*, **19**, 897 (1991).

9. U.S. Council for Energy Awareness, 1992.

10. F. Najmabadi et al., "*The ARIES-I Tokamak Reactor Study*," UCLA-PPG-1323 (1991).

11. J. Galambos, L. J. Perkins, S. Haney, J. Mandrekas, "The Impact of Improved Physics on Commercial Tokamak Reactors," ORNL-TM-12483, Jan. (1994).

12. J. Sheffield, "The Physics of Magnetic Fusion Reactors," to be published in *Reviews of Modern Physics* (1994).