SYSTEM STUDIES OF COPPER- AND SUPERCONDUCTING-COIL PILOT PLANTS*

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ABSTRACT

The TETRA systems code is used to examine devices with both normal copper and superconducting coils as vehicles for steady-state production of fusion power in a Pilot Plant. If the constraints of plasma ignition and net electrical power production are dropped, such devices are much smaller and less expensive than ITER-like devices. For wall loads near 0.5 MW/m^2 with nominal ITER physics guidelines, devices with copper coils have major radii R near 2 m and direct costs near 1×10^9 \$, while devices with superconducting coils have R = 4.1 m and costs of 2.4×10^9 \$. However, the copper-coil devices have the burden of hundreds of megawatts of resistive power losses. All cases tend towards high aspect ratio (A > 4), high fields, and low current. The situation improves for the superconducting-coil cases if higher beta limits are permissible, whereas the copper-coil cases see less benefit from higher beta limits.

I. INTRODUCTION

The mission of the Pilot Plant[1] is to produce a steady-state source of high-grade heat in a fusion "power plant" configuration at the lowest possible capital cost. To this end, the primary plasma performance constraint is attainment of a steady-state wall load. Plasma ignition and net electrical power production are not necessary. Once these constraints are dropped, the door opens to the possibility of devices that are much smaller and less expensive than devices like the International Thermonuclear Experimental Reactor (ITER). We examine devices with both normal copper coils and superconducting coils as vehicles for producing steady-state fusion power for a Pilot Plant.

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II. MODELING

We use the TETRA systems code[2] to perform the trade-offs discussed here. This code performs a constrained optimization for a prescribed problem. The quantities allowed to vary and the constraints employed are listed in Table I. All cases are optimized using minimum direct cost. [Here "direct cost" refers only to the hardware and engineering costs; no installation, R&D, mock-up, conceptual design, contingency, or other indirect cost (e.g., G&A) is included. Including these costs can double the direct

TABLE I
Variables and constraints used in Pilot Plant optimization runs

Variables
Major radius
Aspect ratio,[a] $2.5 \le A \le 5$
Field on axis
T_e, > 7 keV
n_e
H factor,[b] ≤ 2 (ITER-Power scaling)
J_{TFC}, ≤ 1.5 kA/cm^2
Edge q, ≥ 3
TF coil thickness
OH coil thickness
Injection power
Constraints
ITER-Power H factor[b] ≤ 2 for plasma power balance
Volt-second capability > startup requirement
$\beta < \beta_{limit}$ (Troyon scaling, g = 3)
Wall load > input[c]

[a]Range in existing experiments.
[b]The H factor is a multiplier on the ITER-89 Power law energy confinement scaling. A value of 2 rougly corresponds to H-mode confinement.
[c]Main performance requirement.

cost.] This problem formulation basically asks for the minimum-cost steady-state tokamak that achieves a prescribed wall load and satisfies the ITER confinement and beta limits. There is no limit on the energy multiplication value Q.

The ITER global physics modeling[3] is used for both the copper-coil and superconducting-coil cases. One important difference from standard ITER rules is the neglect of the divertor heat load. We leave it unconstrained, since all long-pulse tokamaks have problematic divertor heat loads, and assume that this issue will be addressed outside the Pilot Plant arena. For superconducting-coil cases we use the superconducting modeling assumptions developed for ITER.[4] We have incorporated some simplified algorithms for the copper-coil modeling and costing, which are described in the appendix. Assumptions about the device configuration, operational scenario, and costing are also described in the appendix.

III. RESULTS

III.A. Nominal Beta and Confinement

We first find the cost sensitivity to the primary plasma performance criterion: neutron wall load. Figure 1 shows minimum-cost Pilot Plant parameters vs wall load for copper-coil and superconducting-coil cases. Table II lists some device parameters at the 0.5- and 1.0-MW/m^2 wall load points.

Concentrating first on the cases with nominal beta limit ($g_{Troyon} = 3$), we see that the copper-coil cases attain a prescribed wall load at lower cost and smaller size than do the superconducting-coil cases. The copper-coil devices have lower plasma current, fusion power, and injection power, all of which help to reduce the cost (as does the smaller size). The advantage of the copper coils is due in part to the fact that the distance between the coil and the plasma is smaller with copper coils than with superconducting coils—which helps to reduce the device size. The copper coils also have the advantage of slightly higher field levels at the coils, but the magnitude of this advantage is sensitive to our assumption of a maximum copper-coil current density of 1.5 kA/cm^2. Increasing this current density could further reduce the size of the copper-coil cases, at the expense of higher resistive power losses—which are already at the 300-MW level near wall loads of 0.5 MW/m^2. Finally, we note that the copper coils derive a larger fraction of the wall load from beam target fusion events than from thermal fusions (see below). For the larger superconducting-coil devices the fusion power originating from beam-thermal events has a relatively smaller impact on the wall load. All of these effects nonlinearly combined and tend to favor the copper-coil cases.

Because of the small size and low plasma current levels of the copper-coil cases, these cases have low energy confinement times and are generally confinement limited (i.e., the confinement H factor is at the upper bound of 2). The injection powers are determined by the needs for plasma power balance and wall load (see above) rather than by current drive considerations, and these cases tend to have low energy multiplications (Q near 1). On the other hand, injection powers for the larger superconducting-coil devices are dominated by current drive considerations (note that their H factors are below 2). The energy multiplication Q for the superconducting-coil cases tends to be higher, near 3. As noted, in the lower-Q copper-coil cases a larger fraction of the fusion power originates from beam-thermal events (25% to 30% for copper coils vs 10% to 15% for superconducting coils).

Table II also lists the annual operating costs for electric power [toroidal field (TF) coils + injection power]. Here the copper coils have a penalty of \$50 million to \$100 million per year (assuming a 50% availability).

Note that all cases have high divertor heat loads (the divertor heat load was not constrained). The divertor problem must be solved through some innovative concept if any steady-state tokamak is to be viable. Finally, we mention that all cases tend towards high aspect ratio ($A > 4$). Compared to ITER[5] (direct cost near 3.8×10^9 \$, and major radius of 6 m), all of these cases are smaller and less expensive.

III.B. Enhanced Beta Limits

Figure 1 also shows cases that assume increased beta limits (using a Troyon factor $g = 4$ instead of the nominal $g = 3$ T•m/MA). The superconducting-coil cases show a large reduction of cost and size with higher g, whereas the copper coils—which were closer to the confinement limit at $g = 3$—show only a slight cost and size decrease. Higher allowable beta reduces the plasma current level. For the copper-coil case, this requires an increase in the injection power to counteract the degradation in confinement.

Allowing enhanced confinement while using nominal beta limits does not benefit the superconducting-coil case (see above) and provides only a slight benefit for the copper-coil cases (which are also strongly beta limited).

IV. SUMMARY

Small, driven tokamaks are examined as a means to attain a steady-state fusion power source for a fusion Pilot Plant[1] at low capital cost. With no requirement for ignition, energy multiplication, or net electrical generation, devices that are smaller and less expensive than ITER-like devices, which aim for plasma ignition, become possible.

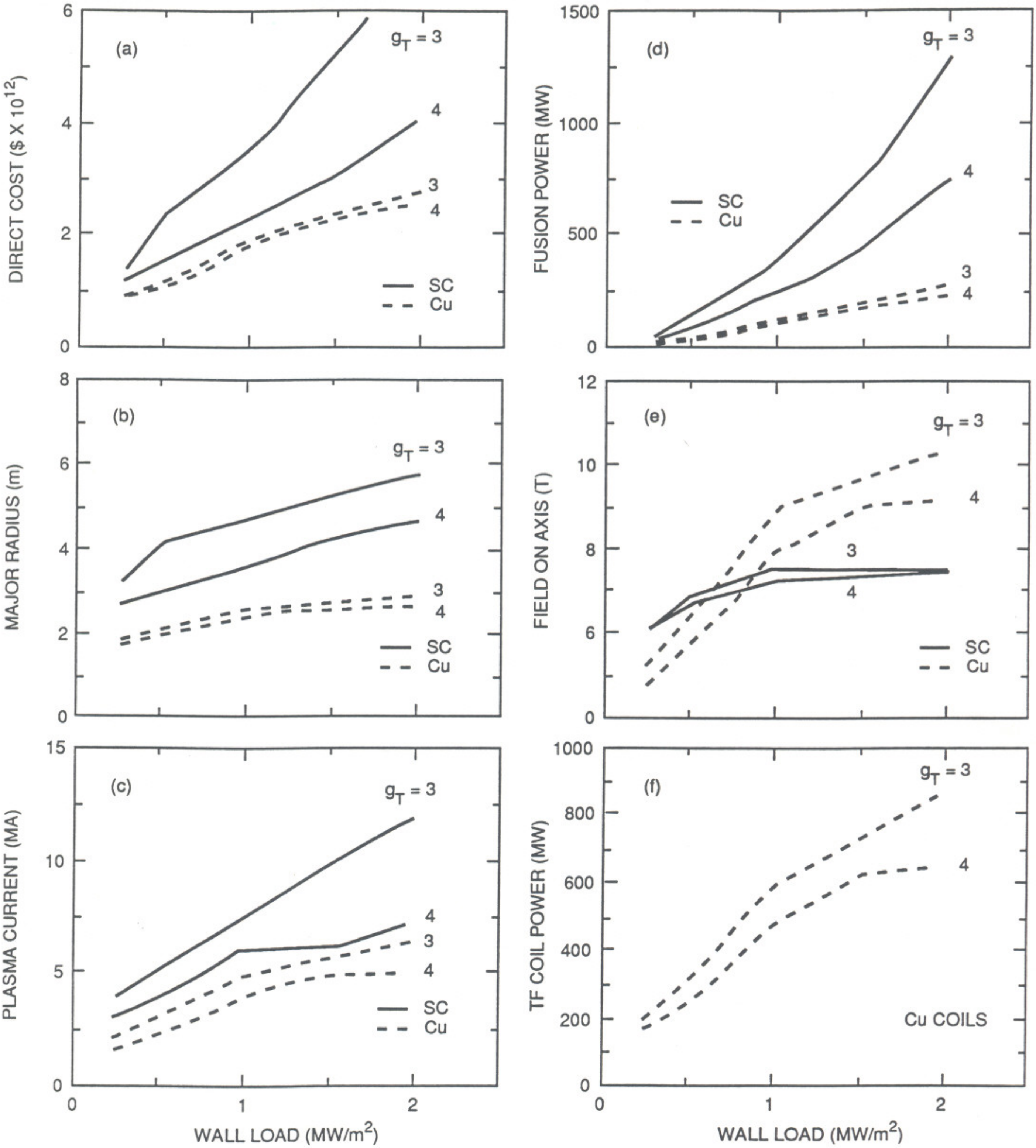

Fig. 1. Minimum-cost device parameters vs wall load for (a) direct cost (\$ $\times 10^9$), (b) major radius (m), (c) plasma current (MA), (d) fusion power (MW), (e) field on axis (T), and (f) resistive power in the TF coils (MW). Cases are shown for copper coils and superconducting coils, with beta limits of $g_{Troyon} = 3$ and 4.

TABLE II

Device parameters for minimum-cost copper-coil and superconducting-coil Pilot Plant devices with wall loads of 0.5 and 1 MW/m^2

	Copper coils		Superconducting coils	
	0.5 MW/m^2	1.0 MW/m^2	0.5 MW/m^2	1.0 MW/m^2
Device				
Direct cost ($\$ \times 10^9$)	1.14	1.88	2.38	3.47
Major radius (m)	2.12	2.60	4.14	4.71
Aspect ratio	5.0[a]	5.0[a]	5.0[a]	4.8
Field on axis (T)	6.5	8.9	5.2	7.3
Plasma current (MA)	2.8	4.7	6.7	7.6
Q	0.94	1.5	1.8	3.0
Fluence (MW a/m^2)[b]	1.25	2.5	1.25	2.5
Power cost ($\$ \times 10^6$ per year)[c]	93	178	37	62
Plasma				
$H_{ITER\text{-}Power}$	1.7/1.5	2.0[a]/1.7	1.7/1.6	1.5/1.3
T_e (keV)	7.0	7.7	7.2	8.8
n_e (10^{20} m^{-3})	1.12	2.32	1.45	1.58
$P_{fusion}/P_{beam\text{-}fusion}$ (MW)	38/31	115/31	145/31	394/54
Beta (%)	3.0	3.0	2.8	3.2
P_{inj} (MW)	40	77	79	131
Volt-seconds (V•s)	19	39	69	110
Edge q	3.0[a]	3.0[a]	3.0[a]	3.0[a]
Z_{eff}	1.55	1.55	1.61	1.55
Bootstrap fraction	0.44	0.44	0.49	0.43
Beam energy[d] (MeV)	0.23	0.98	1.0	1.4
Divertor load (MW/m^2)	18	18	18	35
TF coil				
TF coil to plasma (m)	0.47	0.49	0.99	1.01
$P_{TF\ coil}$, in/out (MW)	138/177	260/340	—	—
Coil weight (kg)	973,000	1,850,000	5,600,000	9,900,000
$J_{inner\ leg}$ (MA/m^2)	1.5	1.5[a]	1.3	1.0
B_{max} (T)	11.0	14.6	13.1	13.7

[a] At a limit.
[b] Assumes 5 years of operation at 50% availability.
[c] Electric power for the injection power (at 50% efficiency) and TF coil resistive losses; assumes 50% availability; power cost = 54 mill/kWh (ORNL costs).
[d] Level determined to permit penetration to the plasma axis.

Both copper-coil and superconducting-coil cases are examined. With nominal physics guidelines, the copper-coil cases are smaller and less expensive than superconducting-coil cases. For wall loads near 0.5 MW/m^2, the copper-coil case has a major radius near 2 m and a direct cost near 1×10^9 \$, vs R = 4.1 m and a direct cost of 2.4×10^9 \$ for the superconducting-coil case. These devices are heavily driven (Q = 1 for copper coils and Q = 3 for superconducting coils), with low plasma current and high aspect ratio. The copper-coil cases have resistive coil losses near 300 MW at the 0.5-MW/m^2 level. Higher wall loads imply larger, more costly devices. If higher beta limits are possible, the size and cost of the superconducting-coil cases decrease sharply (25% to 35%).

APPENDIX: MODELING USED FOR PILOT PLANT STUDIES

Copper-Coil Models

The TETRA systems code used in these studies was developed to model superconducting-coil tokamaks. We have employed some simple assumptions to enable modeling of devices with copper coils. The overall TF coil current density is limited by the optimizer (to $\leq$1.5 kA/cm^2 here, for which steady-state water cooling is possible with present-day technology), and there are no copper-coil stress constraints. The outboard and horizontal TF coil legs are assumed to have an overall current density of 0.75 kA/cm^2. Resistive power losses are calculated on the basis of these current densities and the coil size, which is determined by the radial/vertical build models discussed below. Also, we have added the capability to allow for internal poloidal field (PF) coils (located inside the TF coils). This is a common feature of demountable plate coil copper devices that permits added operational flexibility.

General Assumptions

The following general assumptions are used in all cases.

- Plasma shape defined by the ITER assumptions of κ_x = 2.2 and δ_x = 0.35 (the lower triangularity allows room for the inner divertor).
- Inductive startup (the OH coil is ramped from 5 kA/cm^2 to 0 during startup and left off during the burn for the resistive copper-coil cases).
- Ripple at plasma outboard midplane $\leq$ 0.5% (peak to average).
- n_α/n_e = 0.10 (this is conservative for the low Q values of Pilot Plants).
- PF coils: internal for copper coils, external for superconducting coils.
- Double-null plasma configuration.
- Steady-state plasma operation.
- Neutral beam current drive (which for these cases means negative ion sources, since beam energies > 1 MeV are required for penetration).

Build Assumptions

A key impact on the device outcome is the assumption on radial and vertical build components. We use the values listed in Tables A.I and A.II. For the superconducting-coil cases, they are the same as those used in the ITER-like device studies (except that scrape-off length is scaled here). Quantities identified as "variable" are not fixed but are iterated by the optimizer.

Cost modeling

The following simple scalings were introduced to the TETRA cost module for the purpose of modeling copper-coil devices and low-power operation:

TABLE A.I.
Radial build at midplane (in meters)

	Copper coils	Super-conducting coils
Hole in middle	Variable, >0.25	Variable, >0.25
TF coil[a]	Variable	
Gap	0.02	0.08
OH coil	Variable, >0.05	Variable
Dewar	—	0.07
Gap	0.02	—
Inboard shield/ blanket	0.30	0.765
First wall/VV	0.03	0.035
SOL	$0.1 \times a_{minor}$	$0.1 \times a_{minor}$
Plasma	Variable	Variable
SOL	$0.1 \times a_{minor}$	$0.1 \times a_{minor}$
First wall/VV	0.03	0.035
Outboard blanket	0.30	0.235
Outboard shield	0.25	1.0
Gap[b]	Variable, >1.0	Variable, >0.10

Note: VV = vacuum vessel, SOL = scrape-off layer.

[a]For the superconductor case, the OHC and TFC positions are reversed

[b]This gap is sized for ripple, but for the copper-coil case, a minimum 1-m distance is required to allow internal PF coils.

TABLE A.II.
Vertical build (in meters)

	Value
Plasma	Variable
Top scrape-off	Adequate to permit a 75-cm distance from X-point to strikepoint
Divertor structure	0.20
Shield	0.30 (0.60 for super-conducting-coil option)
Gap	0.15
TF coil	Variable

- Inner copper TF coil legs: 200 \$/kg (for plate-type coils; based on the ATF experience, but uncertain).
- Outer copper TF coil legs: 100 \$/kg (for plate-type coils; based on the ATF experience, but uncertain).
- Copper PF coils: 110 \$/kg.
- Magnet power supplies: 0.7 \$/W.
- Heat transfer system: 0.1×10^6 \$/MW (includes power for TF coils, PF coils, fusion power, injectors, and auxiliary equipment).

- Fuel and tritium systems: 7×10^6 \$/$(MW_{fusion})^{0.5}$ (includes the pellet injectors and tritium systems).

These are meant only to be rough cost scalings. It is hoped that work in progress on TPX will lead to more appropriate costing methods for these categories. We have also made the following changes in the fixed costs normally used in ITER-like device costing (for both copper-coil and superconducting-coil cases): lowering the maintenance costs from 125×10^6 \$ to 40×10^6 \$, lowering the I&C costs from 150×10^6 \$ to 60×10^6 \$, increasing the current drive cost from 3.3 \$/W to 5 \$/W, and increasing the divertor costs from 3×10^5 \$/m^2 to 1.5×10^6 \$/m^2. These changes are intended to better match the TPX methods and introduce the assumption that substantial remote maintenance R&D will have been done by the time this device is built.

REFERENCES

1. S. DEAN, C. BAKER, D. COHN, D. DAUTOVICH, W. ELLIS, and W. MORRISON, "Pilot Plant: An Affordable Step Towards Fusion Power," Fusion Power Associates, Gaithersburg, Maryland (Dec. 2, 1991).
2. R. L. REID et al., "ETR/ITER Systems Code," ORNL/FEDC-87/7, Oak Ridge National Laboratory (1987).
3. N. UCKAN, "ITER Physics Design Information Document," ITER-TN-PH-0-5, International Atomic Energy Agency, Vienna (November 1990).
4. MAGNET DESIGN UNIT, "ITER Magnet System," ITER Documentation Series, International Atomic Energy Agency, Vienna (1991).
5. "ITER Conceptual Design Report," ITER Documentation Series No. 18, International Atomic Energy Agency, Vienna (1991).